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L-17-104

10 CFR 50, Appendix E

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

SUBJECT:
Beaver Valley Power Station, Unit Nos. 1 and 2
BV-1 Docket No. 50-334, License No. DPR-66
BV-2 Docket No. 50-412, License No. NPF-73
Response to Request for Additional Information Regarding a Request to Revise the Emergency Plan (CAC Nos. MF8448 AND MF8449)

By correspondence dated September 28, 2016 (Accession No. ML16277A194), FirstEnergy Nuclear Operating Company (FENOC) submitted a request to revise the current Beaver Valley Power Station Units 1 and 2 (BVPS) Emergency Plan emergency action level scheme to one based on Nuclear Energy Institute (NEI) 99-01, "Development of Emergency Action Levels for Non-Passive Reactors," Revision 6.

By correspondence dated April 10, 2017 (Accession No. ML17093A762), the Nuclear Regulatory Commission (NRC) requested additional information to complete its review. Attachment 1 provides FENOC's response to this request. As a result, the Emergency Action Level Technical Bases and Emergency Preparedness Plan, Section 1, Definitions documents have been updated and are enclosed. In addition, during development of the responses, FENOC identified additional changes needed to clarify the original September 28, 2016 submittal. Attachment 2 provides a summary of these changes. No changes were identified to the previously provided significant hazards or environmental considerations.

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

I declare under penalty of perjury that the foregoing is true and correct. Executed on May 20, 2017.

Sincerely,

Marty L. Richey

Attachments:
1. Response to April 10, 2017 Request for Additional Information
2. FENOC Identified Changes

Enclosures:
A. Emergency Action Level Technical Bases Document
B. Beaver Valley Power Station, Unit Nos. 1 and 2, Emergency Preparedness Plan, Section 1, Definitions

cc: NRC Region I Administrator
NRC Resident Inspector
NRC Project Manager
Director BRP/DEP
Site BRP/DEP Representative

By correspondence dated September 28, 2016, FirstEnergy Nuclear Operating Company (FENOC) submitted a license amendment request for Nuclear Regulatory Commission (NRC) review and approval. By correspondence dated April 10, 2017, NRC staff requested additional information to complete its review. The requested information is presented below in bold type, followed by the FENOC response. In order to maintain emphasis indicated by the NRC with bold type, those applicable parts are underlined.

BVPS-RAI-1

NEI 99-01, Revision 6, Section 4.7, "EAL/Threshold References to AOP [Abnormal Operating Procedures] and EOP [Emergency Operating Procedures] Setpoints/Criteria," states: "As reflected in the generic guidance, the criteria/values used in several EALs and fission product barrier thresholds may be drawn from a plant's AOPs and EOPs," and, "Developers should verify that appropriate administrative controls are in place to ensure that a subsequent change to an AOP or EOP is screened to determine if an evaluation pursuant to 10 CFR 50.54(q) is required."

Please explain what controls are in place at Beaver Valley to ensure that a subsequent change to an AOP or EOP is screened to determine if an evaluation pursuant to 10 CFR 50.54(q) is required.

Response:

The following has been added to Section 1.0, Purpose.

- Additionally, changes to plant AOPs and EOPs that may impact EAL bases shall be evaluated in accordance with the provisions of 10 CFR 50.54(q).

BVPS-RAI-2

NEI 99-01, Revision 6, Section 5.1, "General Considerations," states, in part: "For ICs [initiating conditions] and EALs that have a stipulated time duration (e.g., 15 minutes, 30 minutes, etc.), the Emergency Director should not wait until the applicable time has elapsed, but should declare the event as soon as it is determined that the condition has exceeded, or will likely exceed, the applicable time."

Beaver Valley proposed Section 3.1.7, "Emergency Action Levels with Embedded Time Requirements," states, in part: "Some EALs have embedded time requirements. Declaration must be made as soon as the Emergency Director recognizes that the conditions will not be successfully resolved within 15 minutes."

Some of the proposed Beaver Valley EALs have time requirements other than the specified 15 minutes, and the declarations should be made when the

decisionmaker recognizes that the EAL specific time requirement will be exceeded.

Please revise EAL Bases, Section 3.1.7, to be consistent with endorsed guidance or please provide justification for the difference.

Response:
Beaver Valley proposed Section 3.1.3, “Imminent Conditions”, contains the NEI 99-01, Revision 6, Section 5.1, General Considerations cited language; therefore Section 3.1.7 “Emergency Action Levels with Embedded Time Requirements” was deleted from the EAL Technical Bases Document.

BVPS-RAI-3
The proposed EAL Bases are inconsistent in the title of the Bases. Most are titled, “Basis,” while some are titled, “NEI 99-01 Basis” (e.g., Unit 1, CA3.2, and Unit 2, RG2.1, CA1.2, SU5.1, SU5.2, and SU7.1)

Please provide a justification for the different titles or revise the titles for consistency and clarity.

Response:
The following technical bases changes were made:

- Unit 1 EAL CA3.2 has been revised to delete “NEI 99-01” from the title of the basis.
- Unit 2 EALs RG2.1, CA1.2, SU5.1, SU5.2 and SU7.1 have been revised to delete “NEI 99-01” from the title of the basis.

BVPS-RAI-4
The proposed EALs RA1.3 and RA1.4 (Unit 1 only) Bases include: “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.”

These EALs are dependent upon sample analysis results and are not associated with effluent monitors. Please remove the indicated wording from the Bases for these EALs or provide a justification for its inclusion.

Response:
The cited statement was deleted from the following EAL basis:

- Unit 1, EALs RA1.3 and RA1.4
- Unit 2, EAL RA1.3

BVPS-RAI-5
The NEI 99-01, Revision 6, EAL AU2, developer notes, state, in part: "Specify the mode applicability of a particular indication if it is not available in all modes."

Please verify that the instruments listed in proposed EAL RU2.1 are available in all modes. If instruments are not available in all modes, specify the mode of applicability or provide a justification for the difference from endorsed guidance.

Response:
EAL RU2.1 (Unit 1 and 2) has been revised to specify the mode applicability for the indications not available in all modes.

BVPS-RAI-6
The proposed EALs RA2.1 (Unit 1 only) and RA2.2 Bases include the following statement: "Once sealed, damage to a loaded cask causing loss of the CONFINEMENT BOUNDARY is classified in accordance with IC E-HU1." The proposed EAL scheme does not include an IC E-HU1; it lists an EU1.1.

Please change the Basis to reflect the proposed EAL numbering scheme or provide justification for the difference.

Response:
EALs RA2.1 (Unit 1 only) and RA2.2 (Unit 1 and 2) have been revised to state the following, "Once sealed, damage to a loaded cask causing loss of the CONFINEMENT BOUNDARY is classified in accordance with EAL EU1.1."

BVPS-RAI-7
The proposed EALs CU1.2, CA1.2, CS1.3, and CG1.2 list only containment sumps as a location where increasing levels could indicate reactor coolant system (RCS) leakage.

Please verify that no other tanks (such as a component cooling water surge tank, refueling water storage tank, or reactor coolant drain tank) could capture and indicate RCS leakage and should be added to the EALs.

Response:
EALs CU1.2, CA1.2, CS1.3 and CG1.2 have been revised to include additional tanks that could capture and indicate RCS leakage.

<u>BVPS-RAI-8</u>
The proposed EAL CU3.1 (Unit 1 only) Basis states: "This IC addresses an UNPLANNED increase in RCS temperature above the Technical Specification cold shutdown temperature limit<u>, or the inability to determine RCS temperature and level</u>, and represents a potential degradation of the level of safety of the plant." (emphasis added)

This EAL is only associated with an increase in RCS temperature. The indicated phrase is more applicable to EAL CU3.2 and should be removed from this Basis. Please remove this from the Basis or provide justification for the difference.

<u>Response:</u>
EAL CU3.1 (Unit 1 only) basis was revised to delete the statement ",or the inability to determine RCS temperature and level," and statement "This IC addresses" was revised to read "This EAL addresses."

<u>BVPS-RAI-9</u>
The proposed EAL CU3.2 (Unit 2 only) Basis states: This IC addresses an <u>UNPLANNED increase in RCS temperature above the Technical Specification cold shutdown temperature limit</u>, or the inability to determine RCS temperature and level, and represents a potential degradation of the level of safety of the plant." [emphasis added]

This EAL is only associated with the loss of RCS temperature and level indications. The indicated phrase is more applicable to EAL CU3.1 and should be removed from this Basis. Please remove this from the Basis or provide justification for the difference.

<u>Response:</u>
EAL CU3.2 (Unit 2 only) basis was revised to delete the statement "an UNPLANNED increase in RCS temperature above the Technical Specification cold shutdown temperature limit, or" and statement "This IC addresses" was revised to read "This EAL addresses."

<u>BVPS-RAI-10</u>
The proposed EAL CU3.2 Basis includes: "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 only the loss of temperature and level indication; therefore, the above statement does not apply. Please remove this from the Basis or provide justification for the difference.

Response:
EAL CU3.2 (Unit 1 and 2) basis was revised to delete the statement "A momentary UNPLANNED excursion above the Technical Specification cold shutdown temperature limit when the heat removal function is available does not warrant a classification."

BVPS-RAI-11

The proposed EAL CU5.2 and CU5.3 Bases state that it is the cold condition equivalent of the hot condition EAL SU7.1. This is incorrect, as this EAL is the cold equivalent of the proposed hot condition EAL SU7.2.

Please correct the Basis or provide justification for the difference.

Response:
EALs CU5.2 and CU5.3 have been revised to reference their appropriate equivalent proposed hot condition EAL. As a result, EAL CU5.2 (Unit 1 and 2) basis was revised to the following statement, "This EAL is the cold condition equivalent of the hot condition EAL SU7.2." and EAL CU5.3 (Unit 1 and 2) basis was revised to the following statement, "This EAL is the cold condition equivalent of the hot condition EAL SU7.3."

BVPS-RAI-12
The proposed EALs CA2.1, SS1.1, SG1.1, and SG1.2 include the words, "AC [alternating current] power capability." [emphasis added]

Additionally, their Bases define it as, "…AC power source(s) is available to the emergency buses, whether the buses are powered from it or not." CU2.1 and SA1.1 Bases also include this definition. These differences from the endorsed guidance are not identified in Attachment 5, "Beaver Valley Power Station, Unit No. 1, NEI 99-01, Revision 6, EAL Comparison Matrix," and Attachment 6, "Beaver Valley Power Station, Unit No. 2, NEI 99-01, Revision 6, EAL Comparison Matrix."

The intent of this EAL is to ensure that an EAL is declared upon a total loss of AC power that compromises the performance of all systems requiring electric power for emergency core cooling, containment heat removal/pressure control, spent fuel heat removal, and the ultimate heat sink. This additional criteria could prevent the EAL from being declared in a condition where the AC power sources are available, but not able to be connected to the emergency buses. The NRC staff considers the addition of this criteria to the EALs and the definition in the Basis to be a deviation from endorsed guidance.

Please remove the reference to "capability" in EALs CA2.1, SS1.1, SG1.1, and SG1.2, and its definition in EALs CU2.1, CA2.1, SU1.1, SA1.1 (Unit 2 only), SS1.1, SG1.1, and SG1.2, from the Bases discussion, or explain how the addition of this

condition could not potentially delay or prevent classification of a loss of AC power to emergency buses.

<u>Response:</u>
The reference to "capability" was deleted from EALs CA2.1, SS1.1, SG1.1, and SG1.2 and from the basis discussion for EALs CU2.1, CA2.1, SU1.1, SS1.1, SG1.1, SG1.2 (Unit 1 and 2) and SA1.1 (Unit 2 only).

<u>BVPS-RAI-13</u>
The proposed EALs CA2.1, SS1.1, SG1.1, and SG1.2, and/or their Bases contain tables of AC power sources. A list of readily available power sources may lead to event declarations when mitigative strategies are effective in reestablishing emergency power to these buses. In other words, if a list of power sources is provided for these EALs, and those sources are unavailable, then an EAL decisionmaker would be compelled to declare events, even if mitigative strategies using other power sources are effective. It is not necessary to document these power sources for these EALs, as the EAL is not concerned with the power source as much as the power loss to the emergency bus. (See Emergency Preparedness Frequently Asked Question (EPFAQ) Number: 2015-015.)

Please remove the tables from these EALs or provide a justification for the difference from endorsed guidance.

<u>Response:</u>
The following technical bases changes were made:

- Tables 1C-2 and 2C-2 and references to Tables 1C-2 and 2C-2 were removed from the applicable basis for EAL CA2.1.
- Tables 1S-1 and 2S-1 and references to Tables 1S-1 and 2S-1 were removed from the applicable basis for EALs SS1.1, SG1.1, and SG1.2.

<u>BVPS-RAI-14</u>
RAI deleted based on similarity to BVPS-RAI-12.

<u>BVPS-RAI-15</u>
The proposed EAL CA3.2 includes, "RCS temperature cannot be monitored." This conditional statement is not in NEI-99-01, Revision 6, EAL CA3(2). This is incorrectly identified as a difference instead of a deviation in Attachment 5, "Beaver Valley Power Station, Unit No. 1, NEI 99-01, Revision 6, "EAL Comparison Matrix," and Attachment 6, "Beaver Valley Power Station, Unit No. 2, NEI 99-01, Revision 6, EAL Comparison Matrix." The addition of this criteria could cause a classification of the event to be different than what is provided in the generic scheme guidance.

Please remove the above statement or provide further justification for this deviation.

Response:
The proposed BVPS CA3.2 deviates from the endorsed NEI 99-01, Revision 6 guidance (NEI IC CA3 EAL 2) by the addition of a restriction limiting ALERT declaration due to a 10 pounds per square inch (psi) RCS pressure rise to those times when RCS temperature cannot be monitored. The intent of this is to reduce the likelihood of a small RCS heat up from an initial cold shutdown condition with the over pressure protection system (OPPS) in-service resulting in an ALERT declaration when the RCS temperature indication is available and temperature remains well below 200 degrees Fahrenheit (°F) throughout the event.

The NEI 99-01, Revision 6 CA3 (NEI designator) is the inability to keep the core cooled due to an unplanned loss of decay heat removal capability. An UNUSAL EVENT declaration is made if the RCS temperature exceeds 200 °F (NEI CU3 EAL 1) or if RCS temperature and level indication is lost for 15 minutes. The basis of NEI CU3 EAL 2 states that escalation to an ALERT would be by CA3 (NEI designator) if the temperature exceeds the temperature criteria.

The following example illustrates how a 10 psi pressure increase can occur while RCS temperature is below 200 degrees °F. The initial RCS temperature is 120 degrees °F, RCS pressure is 350 psig, the pressurizer is half full, and the OPPS is in-service. From this initial condition, an inadvertent RCS heatup could result in enough fluid thermal growth to raise the pressurizer level and compress the steam volume, resulting in a 10 psi RCS pressure increase, without the hot leg RCS temperature reaching 200 degrees °F. Therefore, as long as RCS temperature indication is available, the deviation is intended to reduce the risk of an ALERT declaration occurring when the 200 degree °F threshold has not yet been reached. The logic is similar to that of EAL CU3. That is, if the RCS temperature is above 200 degrees °F OR RCS temperature or level indication is lost, then the thresholds for event declaration have been met. This deviation is in alignment with the current BVPS EAL scheme which was reviewed and approved by the NRC (Accession No. ML12313A340).

Therefore, no changes to the proposed CA3.2 (Units 1 and 2) were needed.

BVPS-RAI-16
The proposed Unit 1 EALs CS1.3 and CG1.2 Bases include the statement: "The CRM [containment radiation monitor] threshold values have been established at 15R/hr…." The proposed Unit 2 EALs CS1.3 and CG1.2 Bases include the statement: "The CG7/CS7 CRM threshold values have been established at 15R/hr…." [emphasis added]

Please correct this typographical error or explain the significance of CG7/CS7.

Response:
EALs CS1.3 and CG1.2 (Unit 2 only) basis were revised to delete the phrase "CG7/CS7."

BVPS-RAI-17
The proposed EAL HU3.3 Basis includes the statement: "As used here, the term 'offsite' is meant to be areas external to the BVPS PROTECTED AREA." This definition is different from the definition for "offsite" in proposed Section 1, Definitions.

Please provide this information as a note to the EAL and the wallboard to prevent possible misclassification or provide justification for not including a note.

Response:
EAL HU3.3 (Unit 1 and 2) was revised to include the following note, "Note 14: As used here, the term 'offsite' is meant to be areas external to the BVPS PROTECTED AREA."

BVPS-RAI-18
The proposed EAL HA5.1 (Unit 2 only) Mode Applicability states: "Refers to Table 1H-2 for mode of applicability." The table provided in the EAL is Table 2H-2.

Please fix this typographical error.

Response:
Mode Applicability for EAL HA5.1 (Unit 2 only) was revised to state "Refer to Table 2H-2 for Mode Applicability".

BVPS-RAI-19
The proposed EAL HS1.1 Basis includes the statement: "This IC does not apply to a HOSTILE ACTION directed at an ISFSI [Independent Spent Fuel Storage Installation] PROTECTED AREA located outside the plant PROTECTED AREA; such an attack should be assessed using IC HA1."

This statement would not be applicable to Beaver Valley since the ISFSI is located within the plant protected area. Please remove this statement to avoid possible misclassification.

Response:
EAL HS1.1 (Unit 1 and 2) basis was revised to delete the following statement, "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"

<u>BVPS-RAI-20</u>
The proposed Unit 1 EAL SS2.1 Basis states: "1VM-BAT-1.2.3.4 should be used to validate the voltage for EAL declaration." This statement does not appear in the proposed Unit 1 EAL SG1.2 or in the Unit 2 EALs SS2.1 and SG1.2.

Please describe why this validation is required for EAL declaration for Unit 1 and why it would not be applicable to SG1.2.

<u>Response:</u>
The 1VM-BAT-1,2,3,4 is the calibrated equipment used to validate Unit 1 DC voltage for emergency declarations. The Unit 1 DC voltage indicator available in the Control Room is an annunciator; which is set above the EAL value. Upon annunciator indication, Operations procedures direct an operator to be dispatched to 1VM-BAT-1,2,3,4, to determine the DC voltage from the calibrated equipment. This EAL validation statement and applicable references have been added to Unit 1 SG1.2 and CU4.1 basis sections. In addition, to prevent possible misclassification Note 17 was created and added to Unit 1 EALs SS2.1, SG1.2, and CU4.1.

The Unit 2 DC voltage indicators in the Control Room are the calibrated equipment used for emergency declarations, and therefore does not need additional validation.

<u>BVPS-RAI-21</u>
The proposed EAL SA3.1, Table 1S-3, "Significant Transients," third bulleted transient, "Electrical load rejection > 25% electrical load," is not in alignment with the endorsed guidance, "Electrical load rejection > 25% <u>full</u> electrical load." [emphasis added]. Additionally, the proposed Basis for this EAL lists load rejections of greater than 25% <u>full</u> electrical load as a significant transient.

Please revise the EAL in alignment with the endorsed guidance or provide a justification for this difference.

<u>Response:</u>
The following technical bases changes were made:

- Unit 1 EAL SA3.1 Table 1S-3, Significant Transients, third bulleted transient was revised to state the following, "Electrical load rejection > 25% full electrical load".
- Unit 2 EAL SA3.1 Table 2S-3, Significant Transients, third bulleted transient was revised to state the following, "Electrical load rejection > 25% full electrical load".

<u>BVPS-RAI-22</u>
The proposed EAL SU5.3 Basis states, in part: "This EAL thus applies to leakage into the containment…" However, the EAL is only applicable to unisolable leakage from the RCS to a location outside containment.

To avoid a possible delay in classification due to confusion by the decision makers, please remove the phrase "into containment" or provide a justification for the statement as written.

Response:
EAL SU5.3 (Unit 1 and 2) basis was revised to delete the phrase, "into the containment, a secondary-side system (e.g., steam generator tube leakage) or".

BVPS-RAI-23
For the proposed EALs SU6.1, SU6.2, SA6.1, and SS6.1, a power level (greater than or equal to 5%) was added to the EALs. The intent of NEI 99-01, Revision 6, is to align the above EAL classifications with site-specific EOP criteria of a successful reactor shutdown. The consistency between EALs and EOPs would benefit the decisionmakers by providing consistent criteria. The power level provided in NEI 99-01, Revision 6, developer notes, is an example that represents a typical EOP indication for a generic power plant.

Please consider either using the same EOP reactor shutdown criteria in the EOPs or using wording similar to endorsed guidance.

Response:
The referenced Unit 1 and Unit 2 EALs (SU6.1, SU6.2, SA6.1, and SS6.1) were revised to include wording similar to endorsed guidance and the applicable basis sections were revised to delete references to the greater than or equal to 5% power.

BVPS-RAI-24
The proposed EALs SU7.2 and SU7.3 Bases state that these EALs are the hot condition equivalent of EAL CU5.1. SU7.2 and 7.3 are actually the hot equivalent of CU5.2 and 5.3, respectively.

Please revise the Bases to reference the correct cold condition EALs or delete these statements.

Response:
The following technical bases changes were made:

- EAL SU7.2 (Unit 1 and 2) basis was revised to the following statement, "This EAL is the hot condition equivalent of the cold condition EAL CU5.2."
- EAL SU7.3 (Unit 1 and 2) basis was revised to the following statement, "This EAL is the hot condition equivalent of the cold condition EAL CU5.3."

BVPS-RAI-25
The proposed EALs CA6.1 and SA9.1 Bases state: "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."

This statement is not in alignment with the endorsed guidance. Please revise the Bases to reflect the endorsed guidance or consider the following additional guidance.

Note – Additional guidance has been requested by NEI in Emergency Preparedness Frequently Asked Question (EPFAQ) EPFAQ 2016-002 related to this EAL. Please consider the guidance in the EPFAQ.

Response:

- CA6.1 and SA9.1 (Unit 1 and 2) basis sections have been revised to delete the following statement, “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.”

Changes identified in EPFAQ 2016-002 were reviewed and incorporated into the BVPS Emergency Action Level Technical Bases Document as follows:

- ICs CA6 and SA9 (Unit 1 and 2) have been revised to incorporate EPFAQ 2016-002 revised IC language.
- EALs CA6.1 and SA9.1 (Unit 1 and 2) have been revised to incorporate EPFAQ 2016-002 revised EAL language.
- The following note was added to CA6.1 and SA9.1 (Unit 1 and 2), “Note 15: If the affected SAFETY SYSTEM train was already inoperable or out of service before the hazardous event occurred, then this emergency classification is not warranted.”
- The following note was added to CA6.1 and SA9.1 (Unit 1 and 2), “Note 16: If the hazardous event only resulted in VISIBLE DAMAGE with no indications of degraded performance to at least one train of a SAFETY SYSTEM, then this emergency classification is not warranted.”
- Revised definition VISIBLE DAMAGE to align with EPFAQ 2016-002 definition.
- Revised CA6.1 and SA9.1 (Unit 1 and 2) basis sections to incorporate EPFAQ 2016-002 revised basis language.

BVPS-RAI-26
Category “F” Technical Basis includes the statement: “The FISSION PRODUCT BARRIER THRESHOLDS specified within a scheme reflect plant-specific DBNPS design and operating characteristics.” This is consistent for Units 1 and 2 EALs.

Please correct the typographical error or define the term “DBNPS.”

Response:
The following technical bases changes have been made:

- Attachment 1, Category “F” Technical basis has been revised to state the following, “The FISSION PRODUCT BARRIER THRESHOLDS specified within a scheme reflect plant-specific BVPS design and operating characteristics.”

- Attachment 3, Category "F" Technical basis has been revised to state the following, "The FISSION PRODUCT BARRIER THRESHOLDS specified within a scheme reflect plant-specific BVPS design and operating characteristics."

BVPS-RAI-27
The Unit 1 Fission Product Barrier Loss/Potential Loss Matrix and Bases, Table 1F-1, "Fission Product Barrier Threshold Matrix," is not centered on the page, and the "Category" column is missing (applies to both clean and marked up copies). Please verify the table is complete and readable.

Response:
There were no changes made to Table 1F-1. The table was verified to be complete and readable.

BVPS-RAI-28
NEI 99-01, Revision 6, "Fuel Clad Fission Barrier RCS Activity/Containment Radiation, Loss 3.B," Basis includes the statement: "Add this paragraph (or similar wording) to the Basis if the threshold includes a sample analysis component, 'It is recognized that sample collection and analysis of reactor coolant with highly elevated activity levels could require several hours to complete. Nonetheless, a sample-related threshold is included as a backup to other indications.' "

The proposed FC.C Loss Threshold 2 contains a sample analysis component.

Please revise the Basis to include the statement concerning the analysis component or justify this difference from endorsed guidance.

Response:
The following statement was added to the proposed FC.C Loss Threshold 2 (Unit 1 and 2), "It is recognized that sample collection and analysis of reactor coolant with highly elevated activity levels could require several hours to complete. Nonetheless, a sample-related threshold is included as a backup to other indications."

BVPS-RAI-29
Concerning the proposed Table 1F-2, "Containment Radiation - R/hr (RM-1RM-219A or B)," as it relates to RC.C Loss, Threshold 1, please address the following:

a. **The column labeled, "Time After S/D [Shutdown] (Hrs.)," contains entries for 2-8 hours and >16 hours.**

There are no expected values for the time period between 2-8 hours and >16 hours. Please revise the table to reflect expected values for the period 8-16 hours after shutdown (applies to all Categories utilizing Tables 1F-2 and 2F-2).

Response:
Tables 1F-2 and 2F-2 were revised to include expected values for the time period between >8-16 hours and >16-48 hours after shutdown. The table was revised from >16 hours to >16-48 hours to clarify the containment high range radiation monitor calculation range which is from time 0 hours to 48 hours. In addition, the tables were revised to include ">" in the "Time After S/D (Hrs.)" column to clarify the appropriate Time After S/D row.

b. RC [Reactor Coolant] Loss column has entries of 8 R/hr (Unit 2, Table 2F-2, 11 R/hr).

Please verify that this can be determined on the instruments, as the Basis states: "The detector range is approximately 1 to 1E8 R/hr (logarithmic scale)." Also, verify that at normal 100% power, the instruments read less than 8 R/hr (11 R/hr, Unit 2).

Response:
The containment high range radiation monitors were verified to read near the bottom of their scale (approximately 1 R/hr), and therefore are readable at less than 8 R/hr (11 R/hr, Unit 2) at normal power operation.

Additionally, during the review it was identified that the correct detector range for the Unit 1 Radiation Monitor, RM-1RM-219A/B, is 1 to 1E7, not 1 to 1E8. The Unit 1 bases was revised to read the following statement, "The detector range is approximately 1 to 1E7 R/hr." The Unit 2 bases was verified to have the correct detector range. The Unit 1 and Unit 2 basis sections were revised to delete the reference to logarithmic scale.

BVPS-RAI-30
The proposed FC.B Loss, Threshold 1, Basis for Unit 2 references the plant safety monitoring system for monitoring critical safety function status trees. Other fission product barrier threshold Bases reference the safety parameter display system for monitoring the critical safety function status trees.

Please revise the Bases to reflect the proper system for monitoring the Unit 2 critical safety function status trees.

Response:
The proposed FC.B Loss, Threshold 1 basis (Unit 2 only) has been revised to reference the safety parameter display system (SPDS) instead of the plant safety monitoring system (PSMS).

BVPS-RAI-31
The proposed RC.A RCS or Steam Generator (SG) Tube Leakage, Potential Loss, Threshold 2, states: "[CSFT [Critical Safety Function Tree] Integrity-RED Path

conditions met." This is not consistent with the Bases, which state: "CSFT RCS Integrity-Red Path." [emphasis added]

Please add "RCS" to the RCS or SG Tube Leakage, Potential Loss, Threshold 2, or provide a justification for the difference.

Response:
The following technical bases changes have been made:

- The proposed RC.A Potential Loss, Threshold 2 (Unit 1 and 2) has been revised to the following statement, "RCS Integrity-RED Path conditions met"
- Table 1F-1, Fission Product Barrier Threshold Matrix, RC.A Potential Loss, Threshold 2 (Unit 1 only) has been revised to the following statement, "RCS Integrity-RED Path conditions met"
- Table 2F-1, Fission Product Barrier Threshold Matrix, RC.A Potential Loss, Threshold 2 (Unit 2 only) has been revised to the following statement, "RCS Integrity-RED Path conditions met"

BVPS-RAI-32 (added following the clarification call)
BVPS EAL Bases, Section 3.2.6, contains the following example:

> **An ATWS [anticipated transient without scram] occurs and the high pressure ECCS [emergency core cooling system] systems fail to automatically start. RPV [reactor pressure vessel] 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.**

This example does not correspond to the example cited in Section 5.8 of NEI 99-01, Revision 6. Additionally, in the example, starting the high pressure ECCS system would essentially be criterion for loss of the RCS barrier, whether the reactor vessel level is restored or not. The loss of the RCS barrier would result in an alert declaration, whereas the ATWS is a notification of unusual event. The alert would be the correct classification.

Please revise the EAL Basis to reflect endorsed guidance, or provide justification for this difference.

Response:
The BVPS EAL bases, Section 3.2.6 has been revised to reflect the endorsed guidance.

The following FirstEnergy Nuclear Operating Company (FENOC) identified changes have been included in the updated technical bases document. For each change, the affected emergency action level (EAL) or area within the technical bases document is presented below in bold type, followed by a brief description and basis for the change.

EALs HU2.1, CA6.1, and SA9.1 (Unit 1 only)

<u>Description of Change:</u>
The BVPS seismic monitoring system was replaced subsequent to the September 28, 2016 submittal of the proposed EAL scheme change. The upgrade provided Unit 1 the same seismic capability independent of the Unit 2 seismic instrumentation. The only change is from Unit 2 to Unit 1 equipment.

- EAL HU2.1 (Unit 1 only) and associated bases have been revised to incorporate the new seismic instrumentation. The EAL has been revised to the following:
    "Seismic event **> OBE (> 0.06g)** as indicated by lit lamp on 1ER-CCC-1 Seismic Instrumentation Central Control Cabinet"
- EALs CA6.1 and SA9.1 basis sections have been revised to incorporate the new seismic instrumentation. The basis has been revised to the following:
    "Control Room alarm indication of an earthquake greater than OBE is indicated on the seismic monitoring system cabinet 1ER-CCC-1."

EAL HA1.1 (Unit 1 and 2)

<u>Description of Change:</u>
The following statement was deleted from HA1.1 basis section: "This includes any action directed against an ISFSI that is located outside the plant PROTECTED AREA." This statement is not applicable to Beaver Valley since the ISFSI is located within the plant protected area.

EAL RA2.2 (Unit 2 only)

<u>Description of the Change:</u>
The radiation monitors identified in the EAL have been revised to add their specific indication levels (High Alarm or Alert Alarm). This change aligns the EAL to the cited monitor's actual alarm limit.

Fission Product Barrier Loss/Potential Loss Matrix and Bases (Unit 1 and 2)

<u>Description of Change:</u>
Containment Barrier, Category A, RCS or SG Tube Leakage, ECLs resulting from primary-to-secondary leakage, was revised to read "Unusual Event per SU5.2." SU5.2 is the appropriate EAL for leakage greater than 25 gpm.

Enclosure A

L-17-104

Emergency Action Level Technical Bases Document

(460 Pages Follow)

SECTION 4

EMERGENCY CONDITIONS

[BVPS Units No. 1 & No. 2

EMERGENCY ACTION LEVEL (EAL)
BASES DOCUMENT]

TABLE OF CONTENTS

SECTION PAGE

1.0 PURPOSE

This document provides an explanation and rationale for each Emergency Action Level (EAL) included in the EAL Upgrade Project for Beaver Valley Power Station (BVPS). Decision-makers responsible for implementation of EPP-I-1a(b), "Recognition and Classification of Emergency Conditions," may use this document as a technical reference in support of EAL interpretation. This information may assist the Emergency Director in making classifications, particularly those involving judgment or multiple events. The basis information may also be useful in training and for explaining event classifications to off-site officials.

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

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

2.0 DISCUSSION

2.1 Background

EALs are the plant-specific indications, conditions or instrument readings that are utilized to classify emergency conditions defined in the BVPS Emergency Preparedness Plan (EPP).

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," (ref. 4.1.1), BVPS conducted an EAL implementation upgrade project that produced the EALs discussed herein.

2.2 Fission Product Barriers

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

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

The primary fission product barriers are:

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

B. Reactor Coolant System (RCS): The RCS Barrier includes the RCS primary side and its connections up to and including the pressurizer safety and relief valves, and other connections up to and including the primary isolation valves.

C. Containment (CT): The Containment Barrier includes the containment building and connections up to and including the outermost containment isolation valves. This barrier also includes the main steam, feedwater, and blowdown line extensions outside the containment building up to and including the outermost secondary side isolation valve. Containment Barrier thresholds are used as criteria for escalation of the ECL from Alert to a Site Area Emergency or a General Emergency.

2.3 Fission Product Barrier Classification Criteria

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

Alert:

Any Loss or any Potential Loss of either Fuel Clad or 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 BVPS EAL scheme includes the following features:

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

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

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

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

EAL Groups, Categories and Subcategories

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

The primary tool for determining the emergency classification level is the EAL Classification Matrix. The user of the EAL Classification Matrix may (but is not required to) consult the EAL Technical Bases Document in order to obtain additional information concerning the EALs under classification consideration. The user should consult Section 3.0 and Attachments 1 & 2 (Unit 1) or Attachments 3 & 4 (Unit 2) of this document for such information.

2.5 Technical Bases Information

EAL technical bases are provided in Attachment 1 and Attachment 3 for each EAL according to EAL group (Any, Hot, Cold), EAL category (R, C, H, S, E and F) and EAL subcategory. A summary explanation of each category and subcategory is given at the beginning of the technical bases discussions of the EALs included in the category. For each EAL, the following information is provided:

Category Letter & Title

Subcategory Number & Title

Initiating Condition (IC)

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

EAL Identifier (enclosed in rectangle)

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

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

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

3. Third character (number): Subcategory number within the given category. Subcategories are sequentially numbered beginning with the number one (1). If a category does not have a subcategory, this character is assigned the number one (1).
4. Fourth character (number): The numerical sequence of the EAL within the EAL subcategory. If the subcategory has only one EAL, it is given the number one (1).

Classification (enclosed in rectangle):

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

EAL (enclosed in rectangle)

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

Mode Applicability

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

Definitions:

If the EAL wording contains a defined term (i.e. capitalized word), the definition of the term is contained within the Emergency Plan Section 1, Definitions.

Basis:

The basis section provides a description of the rationale for the EAL as provided in NEI 99-01 Rev. 6 and plant-specific information that provides BVPS relevant information concerning the EAL.

BVPS Basis Reference(s):

Site-specific source documentation from which the EAL is derived.

2.6 Operating Mode Applicability (ref. 4.1.8)

Note: Refer to section 3.3.2 for guidance on event caused mode changes

Mode[(a)]	**Reactivity Condition, Keff**	**% Rated Thermal Power**[(a)]	**Average Coolant Temperature**
1) Power Operation	≥ 0.99	> 5%	N/A
2) Startup	≥ 0.99	≤ 5%	N/A
3) Hot Standby	< 0.99	N/A	≥ 350° F
4) Hot Shutdown[(b)]	< 0.99	N/A	350° F > T_{avg} > 200° F
5) Cold Shutdown[(b)]	< 0.99	N/A	≤ 200° F
6) Refueling	One or more reactor vessel head closure bolts less than fully tensioned.		
D) Defueled	All reactor fuel removed from reactor pressure vessel (full core off load during refueling or extended outage).		

(a) Excluding decay heat.

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

3.0 GUIDANCE ON MAKING EMERGENCY CLASSIFICATIONS

3.1 General Considerations

When making an emergency classification, the Emergency Director must consider all information having a bearing on the proper assessment of an Initiating Condition (IC). This includes the emergency action level (EAL) plus the associated Operating Mode Applicability, Notes, and the informing basis information. In the Recognition Category F matrices, EALs are based on loss or potential loss of FISSION PRODUCT BARRIER THRESHOLDS; the thresholds serve the same function as an EAL.

3.1.1 Classification Timeliness

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

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. The validation of indications should be completed in a manner that supports timely emergency declaration.

3.1.3 Imminent Conditions

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

3.1.4 Planned vs. Unplanned Events

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

3.1.5 Classification Based on Analysis

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

3.1.6 Emergency Director Judgment

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

3.2 Classification Methodology

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

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

3.2.1 Classification of Multiple Events and Conditions

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

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

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

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

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

3.2.2 Consideration of Mode Changes During Classification

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

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

3.2.3 Classification of Imminent Conditions

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

3.2.4 Emergency Classification Level Upgrading and Downgrading

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

As noted above, guidance concerning classification of rapidly escalating events or conditions is provided in RIS 2007-02, "Clarification of NRC Guidance for Emergency Notifications During Quickly Changing Events," (ref. 4.1.2).

3.2.5 Classification of Short-Lived Events

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

3.2.6 Classification of Transient Conditions

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

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

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

> An ATWS occurs and the auxiliary feedwater system fails to automatically start. Steam generator levels rapidly decrease and the plant enters an inadequate RCS heat removal condition (a potential loss of both the fuel clad and RCS barriers). If an operator manually starts the auxiliary feedwater system in accordance with an EOP step and clears the inadequate RCS heat removal condition prior to an emergency declaration, then the classification should be based on the ATWS only.

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

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

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

In these cases, no emergency declaration is warranted; however, the guidance contained in NUREG-1022, “Event Reporting Guidelines: 10CFR50.72 and 50.73,” (ref. 4.1.3) is applicable. Specifically, the event should be reported to the NRC in accordance with 10 CFR § 50.72 (ref. 4.1.4) within one hour of the discovery of the undeclared event or condition. The licensee should also notify appropriate State and local agencies in accordance with the agreed upon arrangements.

3.2.8 Retraction of an Emergency Declaration

Guidance on the retraction of an emergency declaration reported to the NRC is discussed in NUREG-1022, “Event Reporting Guidelines: 10CFR50.72 and 50.73,” (ref. 4.1.3).

4.0 REFERENCES

4.1 Developmental

4.1.1 NEI 99-01 Revision 6, Methodology for the Development of Emergency Action Levels for Non-Passive Reactors, ADAMS Accession Number ML13091A209

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

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

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

4.1.5 10 § CFR 50.73 License Event Report System

4.1.6 1/2-OM-53C.4A.100.1 Security Threat

4.1.7 1/2-ODC-2.02 BVPS ODCM: Gaseous Effluents Attachment Q: Gaseous Effluents

4.1.8 Technical Specifications Table 1.1-1 Modes

4.1.9 NOP-OP-1005 Shutdown Defense In Depth

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

4.1.11 1/2M-6.4.AP Reduced Inventory/Midloop Operation Checklist

4.2 Implementing

4.2.1 EPP-I-1a/b Recognition and Classification of Emergency Conditions

4.2.2 1/2 NEI 99-01 Rev. 6 to BVPS EAL Comparison Matrix

4.2.3 1/2 BVPS EAL Matrix

5.0 BVPS-TO-NEI 99-01 Rev. 6 EAL CROSS-REFERENCE

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

BVPS	NEI 99-01 Rev. 6	
EAL	IC	Example EAL
RU1.1	AU1	1
RU1.2	AU1	2
RU1.3	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

BVPS	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
CU5.2	CU5	2
CU5.3	CU5	3
CA1.1	CA1	1
CA1.2	CA1	2
CA2.1	CA2	1
CA3.1	CA3	1
CA3.2	CA3	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
HU1.2	HU1	2

BVPS	NEI 99-01 Rev. 6	
EAL	IC	Example EAL
HU1.3	HU1	3
HU2.1	HU2	1
HU3.1	HU3	1
HU3.2	HU3	2
HU3.3	HU3	3
HU3.4	HU3	4
HU4.1	HU4	1
HU4.2	HU4	2
HU4.3	HU4	3
HU4.4	HU4	4
HU7.1	HU7	1
HA1.1	HA1	1
HA1.2	HA1	2
HA5.1	HA5	1
HA6.1	HA6	1
HA7.1	HA7	1
HS1.1	HS1	1
HS6.1	HS6	1
HS7.1	HS7	1
HG7.1	HG7	1
SU1.1	SU1	1
SU3.1	SU2	1
SU4.1	SU3	1
SU4.2	SU3	2
SU5.1	SU4	1

BVPS	NEI 99-01 Rev. 6	
EAL	IC	Example EAL
SU5.2	SU4	2
SU5.3	SU4	3
SU6.1	SU5	1
SU6.2	SU5	2
SU7.1	SU6	1
SU7.2	SU6	2
SU7.3	SU6	3
SU8.1	SU7	1, 2
SA1.1	SA1	1
SA3.1	SA2	1
SA6.1	SA5	1
SA9.1	SA9	1
SS1.1	SS1	1
SS2.1	SS8	1
SS6.1	SS5	1
SG1.1	SG1	1
SG1.2	SG8	1
EU1.1	E-HU1	1

6.0 ATTACHMENTS

6.1 Attachment 1, Unit 1 Emergency Action Level Technical Bases

6.2 Attachment 2, Unit 1 Fission Product Barrier Matrix and Basis

6.3 Attachment 3, Unit 2 Emergency Action Level Technical Bases

6.4 Attachment 4, Unit 2 Fission Product Barrier Matrix and Basis

6.5 Attachment 5, Safe Operation & Shutdown Areas Tables R-2 & H-2 Bases

ATTACHMENT 1:

Unit 1 EAL Technical Bases

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ATTACHMENT 1:

Unit 1 EAL Technical Bases

ATTACHMENT 1:

Unit 1 EAL Technical Bases

Category R – Abnormal Rad Release / Rad Effluent

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

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

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

Events of this category pertain to the following subcategories:

1. Radiological Effluent

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

2. Irradiated Fuel Event

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

3. Area Radiation Levels

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

ATTACHMENT 1:

Unit 1 EAL Technical Bases

Category: R – Abnormal Rad Levels / Rad Effluent **RU1.1**

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

EITHER of the following gaseous effluent monitors > the reading shown for **≥ 60 min.**:

- SLCRS Vent (RM-1VS-110 LRNG) **7.58E+3 μCi/s**
- Ventilation Vent (RM-1VS-109 LRNG) **5.28E+3 μCi/s**

(Notes 1, 2, 3)

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

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

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

Mode Applicability:

All

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

RU1.1

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 effluent pathways.

The specified gaseous release values represent two times the ODCM release rate limits (ref. 1, 2).

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

Basis Reference(s):

1. 1/2-ODC-2.02, ODCM Gaseous Effluents
3. ERS-HHM-87-014 , Unit 1/Unit 2 ODCM Gaseous Effluent Monitor Setpoints
4. NEI 99-01 Rev. 6 AU1

Category: R – Abnormal Rad Levels / Rad Effluent **RU1.2**

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

EITHER of the following liquid effluent monitors **> 2 x high-high alarm setpoint** for **≥ 60 min.**:

- Liquid Waste (RM-1LW-104)
- Laundry & Contaminated Shower Drains (RM-1LW-116)

(Notes 1, 2, 3)

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

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

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

Mode Applicability:

All

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.

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Unit 1 EAL Technical Bases

RU1.2

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 liquid effluent pathways.

This EAL also 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).

The specified liquid release values represent two times the ODCM release rate limits. The liquid monitor high-high alarm setpoints are established to ensure the ODCM release limits are not exceeded (ref. 1, 2).

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

Basis Reference(s):

1. 1/2-ODC-2.01, ODCM Liquid Effluents
2. ERS-ATL-93-021 Process Alarm Setpoints for Liquid Effluent Monitors
3. NEI 99-01 Rev. 6 AU1

ATTACHMENT 1:

Unit 1 EAL Technical Bases

Category: R – Abnormal Rad Levels / Rad Effluent **RU1.3**

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:

RU1.3	**Unusual Event**

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

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

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

Mode Applicability:

All

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.

ATTACHMENT 1:

Unit 1 EAL Technical Bases

RU1.3

Basis Reference(s):

1. 1/2-ODC-2.01, ODCM Liquid Effluents
2. 1/2-ODC-2.02, ODCM Gaseous Effluents
3. 1/2-ODC-3.03, Controls for RETS and REMP programs
4. NEI 99-01 Rev. 6 AU1

ATTACHMENT 1:

Unit 1 EAL Technical Bases

Category: R – Abnormal Rad Levels / Rad Effluent **RA1.1**

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**

EITHER of the following gaseous effluent monitors > the reading shown for **≥ 15 min.**:

- SLCRS Vent (RM-1VS-110 HRNG) **1.56E+5 μCi/s**
- Ventilation Vent (RM-1VS-109 HRNG) **1.18E+5 μCi/s**

(Notes 1, 2, 3, 4)

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

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

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

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

Mode Applicability:

All

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

RA1.1

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

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

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

Basis Reference(s):

1. ERS-MPD-93-007 BVPS-U1 Gaseous Radioactivity Monitor Emergency Action Levels
2. ERS-HHM-87-014 , Unit 1/Unit 2 ODCM Gaseous Effluent Monitor Setpoints
3. NEI 99-01 Rev. 6 AA1

ATTACHMENT 1:

Unit 1 EAL Technical Bases

Category: R – Abnormal Rad Levels / Rad Effluent **RA1.2**

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

Gaseous release dose assessment using actual meteorology indicates doses **> 10 mrem** TEDE or **50 mrem** thyroid CDE at or beyond the site boundary (Note 4)

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

Mode Applicability:

All

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.

Basis Reference(s):

1. ERS-MPD-93-007 BVPS-U1 Gaseous Radioactivity Monitor Emergency Action Levels
2. NEI 99-01 Rev. 6 AA1

Category: R – Abnormal Rad Levels / Rad Effluent **RA1.3**

Subcategory: 1 – Radiological Effluent

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

EAL:

RA1.3	**Alert**
Analysis of a liquid effluent sample indicates a concentration or release rate that would result in doses **> 10 mrem** TEDE or **50 mrem** thyroid CDE at or beyond the site boundary for **60 min.** of exposure (Notes 1, 2)	

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

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

Mode Applicability:

All

Basis:

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

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

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

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

Basis Reference(s):

1. ERS-LMR-14-001, Liquid Monitor Emergency Action Level (EAL) Set Points
2. NEI 99-01 Rev. 6 AA1

ATTACHMENT 1:

Unit 1 EAL Technical Bases

Category: R – Abnormal Rad Levels / Rad Effluent **RA1.4**

Subcategory: 1 – Radiological Effluent

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

EAL:

RA1.4 Alert

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

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

(Notes 1, 2)

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

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

Mode Applicability:

All

Basis:

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

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

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

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

Basis Reference(s):

1. NEI 99-01 Rev. 6 AA1

ATTACHMENT 1:

Unit 1 EAL Technical Bases

Category: R – Abnormal Rad Levels / Rad Effluent **RS1.1**

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

EITHER of the following gaseous effluent monitors > the reading shown for **≥ 15 min.**:

- SLCRS Vent (RM-1VS-110 HRNG) **1.56E+6 μCi/s**
- Ventilation Vent (RM-1VS-109 HRNG) **1.18E+6 μCi/s**

(Notes 1, 2, 3, 4)

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

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

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

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

Mode Applicability:

All

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.

ATTACHMENT 1:

Unit 1 EAL Technical Bases

RS1.1

The gaseous effluent release values correspond to calculated doses of 10% of the EPA Protective Action Guidelines (TEDE or CDE Thyroid) (ref. 1).

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

Basis Reference(s):

1. ERS-MPD-93-007 BVPS-U1 Gaseous Radioactivity Monitor Emergency Action Levels
2. NEI 99-01 Rev. 6 AS1

ATTACHMENT 1:

Unit 1 EAL Technical Bases

Category: R – Abnormal Rad Levels / Rad Effluent **RS1.2**

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
Gaseous release dose assessment using actual meteorology indicates doses **> 100 mrem** TEDE or **500 mrem** thyroid CDE at or beyond the site boundary (Note 4)

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

Mode Applicability:

All

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.

Basis Reference(s):

1. NEI 99-01 Rev. 6 AS1

ATTACHMENT 1:

Unit 1 EAL Technical Bases

Category: R – Abnormal Rad Levels / Rad Effluent **RS1.3**

Subcategory: 1 – Radiological Effluent

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

EAL:

RS1.3 **Site Area Emergency**

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

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

(Notes 1, 2)

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

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

Mode Applicability:

All

Basis:

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

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

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

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

Basis Reference(s):

1. ERS-MPD-93-007 BVPS-U1 Gaseous Radioactivity Monitor Emergency Action Levels
2. NEI 99-01 Rev. 6 AS1

Category: R – Abnormal Rad Levels / Rad Effluent **RG1.1**

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

EITHER of the following gaseous effluent monitors > the reading shown for **≥ 15 min.**:

- SLCRS Vent (RM-1VS-110 HRNG) **1.56E+7 μCi/s**
- Ventilation Vent (RM-1VS-109 HRNG) **1.18E+7 μCi/s**

(Notes 1, 2, 3, 4)

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

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

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

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

Mode Applicability:

All

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.

ATTACHMENT 1:

Unit 1 EAL Technical Bases

RG1.1

The gaseous effluent release values correspond to calculated doses of 100% of the EPA Protective Action Guidelines (TEDE or CDE Thyroid) (ref. 1).

Basis Reference(s):

1. ERS-MPD-93-007 BVPS-U1 Gaseous Radioactivity Monitor Emergency Action Levels
2. NEI 99-01 Rev. 6 AG1

RG1.2

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

Gaseous release dose assessment using actual meteorology indicates doses
> 1,000 mrem TEDE or **5,000 mrem** thyroid CDE at or beyond the site boundary (Note 4)

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

Mode Applicability:

All

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.

Basis Reference(s):

1. ERS-MPD-93-007 BVPS-U1 Gaseous Radioactivity Monitor Emergency Action Levels
2. NEI 99-01 Rev. 6 AG1

Category: R – Abnormal Rad Levels / Rad Effluent **RG1.3**

Subcategory: 1 – Radiological Effluent

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

EAL:

RG1.3 General Emergency

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

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

(Notes 1, 2)

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

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

Mode Applicability:

All

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.

Basis Reference(s):

1. NEI 99-01 Rev. 6 AG1

RU2.1

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 on **ANY** of the following:

- Spent Fuel Pool Level (LI-1FC-200A/B)
- Spent Fuel Pool Level alarm (A6-3)
- Temporary RCS Refueling Level (LI-1RC-481C) (MODE 6 & Defueled Only)
- Temporary RCS Refueling Level Loop A (MODE 6 & Defueled Only)
- Local standpipe (tygon hose) (MODE 6 & Defueled Only)

AND

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

- RM-1RM-203 Manipulator Crane Area Monitor (MODE 6 & Defueled Only)
- RM-1RM-207 Fuel Pool Bridge Area Monitor

Mode Applicability:

All

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.

RU2.1

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.

Indication of decreasing level includes ANY of the following: (ref. 1):

- Spent Fuel Pool Level (LI-1FC-200A/B)
- Spent Fuel Pool Level alarm (A6-3)
- Temporary RCS Refueling Level (LI-1RC-481C)
- Temporary RCS Refueling Level Loop A
- Local standpipe (tygon hose)

Allowing level to decrease could result in spent fuel being uncovered, reducing spent fuel decay heat removal and creating an extremely hazardous radiation environment. During refueling, this maintains sufficient water level in the fuel transfer canal, refueling cavity, and SFP to retain iodine fission product activity in the water in the event of a fuel handling accident.

The fuel transfer canal is only of concern in assessing this EAL when irradiated fuel transfer is in progress, in which case the spent fuel pool transfer canal gate is open and connected to the fuel transfer canal.

The listed area radiation monitors are those which would likely see an increase in area radiation due to a loss of REFUELING PATHWAY inventory.

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

Basis Reference(s):

1. 1OM-53C.4.1.20.1 Spent Fuel Pool Cooling Trouble
2. BVPS-1&2 Technical Specification 3.7.15 Fuel Storage Pool Water Level
3. BVPS-1&2 Technical Specification 3.9.6 Refueling Cavity Water Level
4. NEI 99-01 Rev. 6 AU2

RA2.1

Category: R – Abnormal Rad Levels / Rad Effluent

Subcategory: 2 – Irradiated Fuel Event

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

EAL:

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

Mode Applicability:

All

Basis:

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

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

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

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

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

RA2.1

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.

Basis Reference(s):

1. 1OM-53C.4.1.20.1 Spent Fuel Pool Cooling Trouble
2. NEI 99-01 Rev. 6 AA2

ATTACHMENT 1:

Unit 1 EAL Technical Bases

Category: R – Abnormal Rad Levels / Rad Effluent **RA2.2**

Subcategory: 2 – Irradiated Fuel Event

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

EAL:

RA2.2 Alert

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

- RM-1VS-109 LRNG Ventilation Vent (High alarm)
- RM-1VS-110 LRNG SLCRS Vent (High alarm)
- RM-1RM-203 Manipulator Crane Area Monitor (High-High alarm)
- RM-1RM-207 Fuel Pool Bridge Area Monitor (High-High alarm)

Mode Applicability:

All

Basis:

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

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

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

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).

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

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

RA2.2

Basis Reference(s):

1. 1OM-53C.4.1.49.1 Irradiated Fuel Damage
2. 1OM-53C.4.1.20.1 Spent Fuel Pool Cooling Trouble
3. NEI 99-01 Rev. 6 AA2

ATTACHMENT 1:

Unit 1 EAL Technical Bases

Category: R – Abnormal Rad Levels / Rad Effluent **RA2.3**

Subcategory: 2 – Irradiated Fuel Event

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

EAL:

RA2.3	**Alert**
Spent fuel pool level (LI-1FC-200A/B) reading **≤ 10 ft.** (Level 2)	

Mode Applicability:

All

Basis:

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

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

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

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

Level 2 is the level that is adequate to provide substantial radiation shielding for a person standing on the spent fuel pool operating deck. It represents the range of water level where any necessary operations in the vicinity of the spent fuel pool can be completed without significant dose consequences from direct gamma radiation from the stored spent fuel. BVPS designated as Level 2 the water level ~10 feet above the top of the fuel racks (El 752') (ref. 2).

Spent Fuel Pool (SFP) draindown to elevation 750 ft-10 inches, as described in Technical Specification 4.3.2, from SFP cooling system piping break outside the SFP walls would result in an indicated level of approximately 8.9 ft.

Escalation of the emergency classification level would be via IC RS1 RS2.

RA2.3

Basis Reference(s):

1. NRC EA-12-51 Issuance of Order to Modify Licenses with Regard to Reliable Spent Fuel Pool Instrumentation
2. ECP No. 13-0561-000, Reference Documents for ECP-13-0561 – Installation of Spent Fuel Pool Level Instrumentation for Beyond Design Basis External Events
3 NEI 99-01 Rev. 6 AA2

ATTACHMENT 1:

Unit 1 EAL Technical Bases

Category: R – Abnormal Rad Levels / Rad Effluent **RS2.1**

Subcategory: 2 – Irradiated Fuel Event

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

EAL:

RS2.1	**Site Area Emergency**
Spent fuel pool level (LI-1FC-200A/B) reading **$\leq$ 0.5 ft.** (Level 3)	

Mode Applicability:

All

Basis:

This IC 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.

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

BVPS designated as Level 3 the water level greater than 6 inches (0.5 ft.) above the top of the fuel storage racks plus the accuracy of the SFP level instrument channel (El. 742' – 6.5") (ref. 2).

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

Basis Reference(s):

1. NRC EA-12-51 Issuance of Order to Modify Licenses with Regard to Reliable Spent Fuel Pool Instrumentation
2. ECP No. 13-0561-000, Reference Documents for ECP-13-0561 – Installation of Spent Fuel Pool Level Instrumentation for Beyond Design Basis External Events
3. NEI 99-01 Rev. 6 AS2

Category: R – Abnormal Rad Levels / Rad Effluent **RG2.1**

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 (LI-1FC-200A/B) **cannot** be restored to at least **0.5 ft.** (Level 3) for **≥ 60 min.** (Note 1)	

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

Mode Applicability:

All

Basis:

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

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

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

BVPS designated as Level 3 the water level greater than 6 inches (0.5 ft.) above the top of the fuel storage racks plus the accuracy of the SFP level instrument channel (El. 742' – 6.5") (ref. 2).

Basis Reference(s):

1. NRC EA-12-51 Issuance of Order to Modify Licenses with Regard to Reliable Spent Fuel Pool Instrumentation
2. ECP No. 13-0561-000, Reference Documents for ECP-13-0561 – Installation of Spent Fuel Pool Level Instrumentation for Beyond Design Basis External Events
3. NEI 99-01 Rev. 6 AG2

ATTACHMENT 1:

Unit 1 EAL Technical Bases

Category: R – Abnormal Rad Levels / Rad Effluent **RA3.1**

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

- Control Room (RM-1RM-218A/B)
- Central Alarm Station (by survey)

Mode Applicability:

All

Basis:

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

RM-1RM-218A/B are the installed Control Room area radiation monitors and may be used to assess this EAL threshold. However, no permanently installed area radiation monitoring is installed in the CAS and therefore this threshold must be assessed via local radiation survey (ref. 1).

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

Basis Reference(s):

1. NEI 99-01 Rev. 6 AA3

Category: R – Abnormal Rad Levels / Rad Effluent **RA3.2**

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 1R-1** rooms or areas (Notes 5, 12)

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 12: 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).

Table 1R-1 Safe Operation & Shutdown Rooms/Areas

Room/Area	Mode Applicability
Safeguards 735’ East and West Cable Vault (2 separate areas)	4
Safeguards 722’ Penetrations D	4
Auxiliary Building 735’ CCR Hx Area	4
Service Building 713’ AE Emergency Switchgear	4

Mode Applicability:

4 - Hot Shutdown

Basis:

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

For RA3.2, an Alert declaration is warranted if entry into the affected room/area is, or may be, procedurally required during the plant operating mode in effect at the time of the elevated radiation levels. The emergency classification is not contingent upon whether entry is actually necessary at the time of the increased radiation levels. Access should be considered as impeded if extraordinary measures are necessary to facilitate entry of personnel into the

RA3.2

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.
- 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 listed area specifies the plant mode(s) during which entry would be required for each room or area (ref. 1).

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

RA3.2 mode applicability has been limited to the applicable modes identified in Table 1R-1 Safe Operation and Safe Shutdown Rooms/Areas. If due to plant operating procedure or a plant configuration changes, the applicable plant modes specified in Table 1R-1 are changed, a corresponding change to Attachment 5 ‘Safe Operation and Shutdown Areas Tables RA3.2 and HA5.1 Bases’ and to EAL RA3.2 mode applicability is required.

Basis Reference(s):

1. EPLAN, Section 4, Attachment 5 Safe Operation & Shutdown Areas RA3.2 & HA5.1 Bases
2. NEI 99-01 Rev. 6 AA3

ATTACHMENT 1:

Unit 1 EAL Technical Bases

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 facility (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 storage of spent nuclear fuel.

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

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

Category: ISFSI **EU1.1**

Subcategory: Confinement Boundary

Initiating Condition: Damage to a loaded cask CONFINEMENT BOUNDARY

EAL:

EU1.1 Unusual Event

Damage to a loaded cask CONFINEMENT BOUNDARY as indicated by an on-contact radiation reading > **ANY** of the following:

- **1,050 mrem/hr** at the Horizontal Storage Module (HSM) bird screen
- **4 mrem/hr** outside HSM door
- **8 mrem/hr** on end shield wall exterior

Mode Applicability:

All

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.

The dry-cask storage system is the NUHOMS Horizontal Modular Storage System. (ref. 1).

The value shown represents 2 times the limits specified in the ISFSI Certificate of Compliance Technical Specification section 5.4.2 for radiation external to a HSM loaded with a Model 37PTH DSC (ref. 1).

ATTACHMENT 1:

Unit 1 EAL Technical Bases

Basis Reference(s): **EU1.1**

1. Technical Specifications for the Standardized NUHOMS Horizontal Modular Storage System, Section 5.4 HSM or HSM-H Dose Rate Evaluation Program
2. NEI 99-01 Rev. 6 E-HU1

ATTACHMENT 1:

Unit 1 EAL Technical Bases

Category C – Cold Shutdown / Refueling System Malfunction

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

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

The events of this category pertain to the following subcategories:

1. RCS Level

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

2. Loss of Emergency AC Power

   Loss of essential plant electrical power can compromise plant SAFETY SYSTEM operability including decay heat removal and emergency core cooling systems, which may be necessary to ensure fission product barrier integrity. This category includes loss of onsite and offsite power sources for 4KV 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 buses.

ATTACHMENT 1:

Unit 1 EAL Technical Bases

5. Loss of Communications

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

6. Hazardous Event Affecting SAFETY SYSTEMS

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

ATTACHMENT 1:

Unit 1 EAL Technical Bases

Category: C – Cold Shutdown / Refueling System Malfunction **CU1.1**

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

Mode Applicability:

5 - Cold Shutdown, 6 - Refueling

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 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.

With the plant in Cold Shutdown, RCS water level is normally maintained above the pressurizer low level setpoint of 14%. 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 (ref. 1, 2).

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

ATTACHMENT 1:

Unit 1 EAL Technical Bases

CU1.1

Basis Reference(s):

1. 1OM-53C.4.1.10.1 Loss of Residual Heat Removal Capability
2 1OM-52.4.R.1.F Station Shutdown from 100% to Mode 5
3. Technical Specification Section 3.9.6 Refueling Cavity Water Level
4. NEI 99-01 Rev. 6 CU1

Category: C – Cold Shutdown / Refueling System Malfunction **CU1.2**

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 1C-6 Sump/Tank level due to a loss of RCS inventory
- Visual observation of UNISOLABLE RCS leakage

Table 1C-6 Sump/Tank
• Containment Sumps • Incore Sump • Refueling Water Storage Tank (RWST) • Primary Drains Tank • Pressurizer Relief Tank (PRT) • CCR Surge Tank

Mode Applicability:

5 - Cold Shutdown, 6 – Refueling

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 RCS level have been lost. In this condition, operators may determine that an inventory loss is occurring by observing changes in sump and/or tank levels. Sump and/or tank level changes must be evaluated against other potential sources of water flow to ensure they are indicative of leakage from the RCS.

CU1.2

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

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

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

Basis Reference(s):

1. 1OM-53C.4.1.10.1 Loss of Residual Heat Removal Capability
2. NEI 99-01 Rev. 6 CU1

ATTACHMENT 1:

Unit 1 EAL Technical Bases

Category: C – Cold Shutdown / Refueling System Malfunction **CA1.1**

Subcategory: 1 – RCS Level

Initiating Condition: Loss of RCS inventory

EAL:

CA1.1	**Alert**
Loss of RCS inventory as indicated by reactor vessel level **≤ 20 in.** (LI-1RC-481C)	

Mode Applicability:

5 - Cold Shutdown, 6 – Refueling

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 20 in. indicates that operator actions have not been successful in restoring and maintaining RCSwater level. The heat-up rate of the coolant will increase as the available water inventory is reduced. A continuing decrease in water level will lead to core uncovery.

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

Reactor vessel level of ~14 in. is the minimum level for RHR pump operation in the decay heat removal mode @ an RHR flowrate of 1,000 gpm. (ref. 1). However, Refueling Outage Temporary Level Instrument LI-1RC-481C (typically available in Mode 6) **cannot** measure RCS level below 732 feet 3 15/16 inch elevation (reactor pressure vessel nozzle centerline elevations) which corresponds to the lowest increment of 14 inches on the instrument. The EAL value has been established at 20 inches to ensure instrument indication with significant ambient temperature increase in the CNMT, such as could accompany loss of residual heat removal and boiling of RCS inventory with the RCS vented to atmosphere (ref. 2, 3).

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

CA1.1

Basis Reference(s):

1. 1OM-53C.4.1.10.2 Loss of RHR While Operating at Reduced Inventory/Midloop Conditions Attachment 2 Required RCS Water Level for Reduced Inventory/Midloop
2. 1OM-53C.4.1.10.1 Loss of Residual Heat Removal Capability
3. BV Calculation SP-1RC-30, Instrument Uncertainty for Refueling Level Indicator LI-1RC-481C
4. NEI 99-01 Rev. 6 CA1

Category: C – Cold Shutdown / Refueling System Malfunction **CA1.2**

Subcategory: 1 – RCS Level

Initiating Condition: Loss of RCS inventory

EAL:

CA1.2 Alert

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

AND EITHER

- UNPLANNED increase in **ANY** Table 1C-6 Sump/Tank level due to a loss of RCS inventory
- Visual observation of UNISOLABLE RCS leakage

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

Table 1C-6 Sump/Tank
• Containment Sumps • Incore Sump • Refueling Water Storage Tank (RWST) • Primary Drains Tank • Pressurizer Relief Tank (PRT) • CCR Surge Tank

Mode Applicability:

5 - Cold Shutdown, 6 – Refueling

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

CA1.2

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.

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

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

In this EAL, all RCS water level indication would be unavailable for greater than 15 minutes, and the RCS inventory loss must be detected by indirect leakage indications. Level increases must be evaluated against other potential sources of leakage such as cooling water sources inside the containment to ensure they are indicative of RCS leakage. If the make-up rate to the RCS unexplainably rises above the pre-established rate, a loss of RCS inventory may be occurring even if the source of the leakage **cannot** be immediately identified. Visual observation of leakage from systems connected to the RCS that **cannot** be isolated could also be indicative of a loss of RCS inventory (ref. 1).

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

Basis Reference(s):

1. 1OM-53C.4.1.10.1 Loss of Residual Heat Removal Capability
2. NEI 99-01 Rev. 6 CA1

ATTACHMENT 1:

Unit 1 EAL Technical Bases

Category: C – Cold Shutdown / Refueling System Malfunction **CS1.1**

Subcategory: 1 – RCS Level

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

EAL:

CS1.1 Site Area Emergency
CONTAINMENT CLOSURE **not** established,
AND
RCS level **< 64%** RVLIS Full Range (**6"** below bottom of hotleg)

Mode Applicability:

5 – Cold Shutdown, 6 – Refueling

Basis:

This IC addresses a significant and prolonged loss of reactor vessel/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 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 reactor vessel levels of EALs CS1.1 and CS2.2 reflect the fact that with CONTAINMENT CLOSURE established, there is a lower probability of a fission product release to the environment.

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

When RVLIS Full Range water level decreases to 64% (ref. 1), water level is six inches below the elevation of the bottom of the RCS hot leg penetration. When RCS water level drops significantly below the elevation of the bottom of the RCS hot leg penetration, all sources of RCS injection have failed or are incapable of making up for the inventory loss.

In Refueling mode, RCS water level indication from RVLIS is likely unavailable but alternate means of level indication are normally installed (including visual observation) to assure that the ability to monitor water level will not be interrupted. If no RVLIS alternate means available, refer to CS1.3.

CS1.1

The status of CONTAINMENT CLOSURE is tracked if plant conditions change that could raise the risk of a fission product release as a result of a loss of decay heat removal (ref. 2, 3).

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

Basis Reference(s):

1. 1OM-6.5.B.7 RVLIS Full Range Level VS. Reactor Vessel Height
2. NOP-OP-1005 Shutdown Defense in Depth
3. 1/2-ADM-0712 Shutdown Defense in Depth Assessment
4. NEI 99-01 Rev. 6 CS1

ATTACHMENT 1:

Unit 1 EAL Technical Bases

Category: C – Cold Shutdown / Refueling System Malfunction **CS1.2**

Subcategory: 1 – RCS Level

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

EAL:

CS1.2	**Site Area Emergency**
CONTAINMENT CLOSURE established	
AND	
RCS level **< 56%** RVLIS Full Range (top of active fuel)	

Mode Applicability:

5 – Cold Shutdown, 6 – Refueling

Basis:

This IC addresses a significant and prolonged loss of reactor vessel 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 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 reactor vessel levels of EALs CS1.1 and CS1.2 reflect the fact that with CONTAINMENT CLOSURE established, there is a lower probability of a fission product release to the environment.

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

When Reactor Vessel water level drops below 56% RVLIS Full Range (ref. 1), core uncovery is about to occur.

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

CS1.2

barrier and Potential Loss of the Fuel Clad barrier. If no RVLIS alternate means available, refer to CS1.3.

The status of CONTAINMENT CLOSURE is tracked if plant conditions change that could raise the risk of a fission product release as a result of a loss of decay heat removal (ref. 2, 3).

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

Basis Reference(s):

1. 1OM-6.5.B.7 RVLIS Full Range Level VS. Reactor Vessel Height
2. NOP-OP-1005 Shutdown Defense in Depth
3. 1/2-ADM-0712 Shutdown Defense in Depth Assessment
4. NEI 99-01 Rev. 6 CS1

ATTACHMENT 1:

Unit 1 EAL Technical Bases

Category: C – Cold Shutdown / Refueling System Malfunction **CS1.3**

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

- UNPLANNED increase in **ANY** Table 1C-6 Sump/Tank level of sufficient magnitude to indicate core uncovery
- Erratic source range monitor indication
- Containment Radiation Monitor (RM-1RM-219A or B) **> 15 R/hr**

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

Table 1C-6 Sump/Tank
• Containment Sumps • Incore Sump • Refueling Water Storage Tank (RWST) • Primary Drains Tank • Pressurizer Relief Tank (PRT) • CCR Surge Tank

Mode Applicability:

5 – Cold Shutdown, 6 – Refueling

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.

CS1.3

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

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

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

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

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

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

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. Operating procedures provide instructions for calculating primary system leak rate by manual or computer-based water inventory balances. Level increases must be evaluated against other potential sources of leakage such as cooling water sources inside the containment to ensure they are indicative of RCS leakage. If the make-up rate to the RCS unexplainably rises above the pre-established rate, a loss of RCS inventory may be occurring even if the source of the leakage **cannot** be immediately identified. Visual observation of leakage from systems connected to the RCS that **cannot** be isolated could also be indicative of a loss of RCS inventory (ref. 1, 2).

The RCS inventory loss may be detected by the Containment Radiation Monitors or erratic source range monitor indication.

CS1.3

As water level in the reactor vessel lowers, the dose rate above the core will rise. The dose rate due to this core shine should result in Containment Radiation Monitor (CRM) indication > 15 R/hr. Containment radiation levels are indicated on containment radiation monitors RM-1RM-219A and 219B. These monitors are not located within line of sight of the reactor vessel. The containment radiation monitor alert alarm is set at 4.58E+2 R/hr and high alarm is set at 1.4E+4 R/hr. The alarm setpoints are considered operationally significant, but above what would be expected for a loss of vessel level while in the refuel mode. Therefore, the CRM threshold values have been established at 15 R/hr (~10x the low scale reading of 1.5 R/hr) to provide a reasonable and conservative indication of abnormal conditions associated with elevated radiation levels in containment due to a loss of water level with irradiated fuel in the vessel.

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. 3).

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

Basis Reference(s):

1. 1OM-53C.4.1.10.1 Loss of Residual Heat Removal Capability
2. Nuclear Safety Analysis Center (NSAC), 1980, “Analysis of Three Mile Island - Unit 2 Accident,” NSAC-1
3. NEI 99-01 Rev. 6 CS1

ATTACHMENT 1:

Unit 1 EAL Technical Bases

Category: C – Cold Shutdown / Refueling System Malfunction **CG1.1**

Subcategory: 1 – RCS Level

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

EAL:

CG1.1 General Emergency

RCS level **< 56%** RVLIS Full Range (top of active fuel) for ≥ **30 min.** (Note 1)

AND

ANY Containment Challenge indication, **Table 1C-1**

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

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

Table 1C-1 Containment Challenge Indications
• CONTAINMENT CLOSURE **not** established (Note 6)
• Containment hydrogen concentration > 4%
• UNPLANNED rise in Containment pressure

Mode Applicability:

5 – Cold Shutdown, 6 – Refueling

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.

CG1.1

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

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

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

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

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

When Reactor Vessel water level drops below 56% RVLIS Full Range (ref. 1), core uncovery is about to occur.

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

CG1.1

Three conditions are associated with a challenge to Containment integrity:

1. CONTAINMENT CLOSURE not established - The status of CONTAINMENT CLOSURE is tracked if plant conditions change that could raise the risk of a fission product release as a result of a loss of decay heat removal (ref. 2, 3). If containment closure is re-established prior to exceeding the 30 minute core uncovery time limit then escalation to GE would not occur.
2. Containment hydrogen > 4% - The 4% hydrogen concentration threshold is generally considered the lower limit for hydrogen deflagrations. Hydrogen monitors, although available at all times, are not in service during normal operations. They are started per 1OM-46.4.G (ref. 5).
3. UNPLANNED rise in containment pressure - An UNPLANNED pressure rise in containment while in cold Shutdown or Refueling modes can threaten CONTAINMENT CLOSURE capability and thus Containment potentially **cannot** be relied upon as a barrier to fission product release.

This 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*.

Basis Reference(s):

1. 1OM-6.5.B.7 RVLIS Full Range Level VS. Reactor Vessel Height
2. NOP-OP-1005 Shutdown Defense in Depth
3. 1/2CMP-47-Contingency Hatch Closure-1M, Contingency Hatch Closure
4. 1/2-ADM-0712 Shutdown Defense in Depth Assessment
5. 1OM-46.4.G Placing Wide Range Containent Hydrogen Monitoring System in Operation
6. NEI 99-01 Rev. 6 CG1

ATTACHMENT 1:

Unit 1 EAL Technical Bases

Category: C – Cold Shutdown / Refueling System Malfunction **CG1.2**

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 uncovery is indicated by **ANY** of the following:
• UNPLANNED increase in **ANY** Table 1C-6 Sump/Tank level of sufficient magnitude to indicate core uncovery
• Erratic source range monitor indication
• Containment Radiation Monitor (RM-1RM-219A or B) **> 15 R/hr**
AND
ANY Containment Challenge indication, **Table 1C-1**

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

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

Table 1C-1 Containment Challenge Indications
• CONTAINMENT CLOSURE **not** established (Note 6)
• Containment hydrogen concentration > 4%
• UNPLANNED rise in Containment pressure

CG1.2

Table 1C-6 Sump/Tank
• Containment Sumps • Incore Sump • Refueling Water Storage Tank (RWST) • Primary Drains Tank • Pressurizer Relief Tank (PRT) • CCR Surge Tank

Mode Applicability:

5 - Cold Shutdown, 6 – Refueling

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

CG1.2

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

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

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

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

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 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 (ref. 1).

The RCS inventory loss may be detected by the Containment Radiation Monitors or erratic source range monitor indication.

As water level in the reactor vessel lowers, the dose rate above the core will rise. The dose rate due to this core shine should result in Containment Radiation Monitor (CRM) indication > 15 R/hr. Containment radiation levels are indicated on containment radiation monitors RM-1RM-219A and 219B. These monitors are not located within line of sight of the reactor vessel. The containment radiation monitor alert alarm is set at 4.58E+2 R/hr and high alarm is set at 1.4E+4 R/hr. The alarm setpoints are considered operationally significant, but above what would be expected for a loss of vessel level while in the refuel mode. Therefore, the CRM threshold values have been established at 15 R/hr (~10x the low scale reading of 1.5 R/hr) to provide a reasonable and conservative indication of abnormal conditions associated with elevated radiation levels in containment due to a loss of water level with irradiated fuel in the vessel.

CG1.2

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. 2).

Three conditions are associated with a challenge to Containment integrity:

1. CONTAINMENT CLOSURE not established - The status of CONTAINMENT CLOSURE is tracked if plant conditions change that could raise the risk of a fission product release as a result of a loss of decay heat removal (ref. 3, 4). If containment closure is re-established prior to exceeding the 30 minute core uncovery time limit then escalation to GE would not occur.
2. Containment hydrogen > 4% - The 4% hydrogen concentration threshold is generally considered the lower limit for hydrogen deflagrations. Hydrogen monitors, although available at all times, are not in service during normal operations. They are started per 1OM-46.4.G (ref. 6).
3. UNPLANNED rise in Containment pressure - An UNPLANNED pressure rise in containment while in cold Shutdown or Refueling modes can threaten CONTAINMENT CLOSURE capability and thus Containment potentially **cannot** be relied upon as a barrier to fission product release.

This 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*.

Basis Reference(s):

1. 1OM-53C.4.1.10.1, Loss of Residual Heat Removal Capability
2. Nuclear Safety Analysis Center (NSAC), 1980, “Analysis of Three Mile Island - Unit 2 Accident,” NSAC-1
3. 1/2CMP-47-Contingency Hatch Closure-1M, Contingency Hatch Closure
4. NOP-OP-1005 Shutdown Defense in Depth
5. 1/2-ADM-0712 Shutdown Defense in Depth Assessment
6. 1OM-46.4.G Placing Wide Range Containent Hydrogen Monitoring System in Operation
7. NEI 99-01 Rev. 6 CG1

ATTACHMENT 1:

Unit 1 EAL Technical Bases

Category: C – Cold Shutdown / Refueling System Malfunction **CU2.1**

Subcategory: 2 – Loss of Emergency AC Power

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

EAL:

CU2.1 Unusual Event

AC power capability, **Table 1C-2**, to 4 KV emergency buses 1AE and 1DF reduced to a single power source for **≥ 15 min.** (Note 1)

AND

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

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

Table 1C-2 AC Power Sources
Offsite: • SSST 1A • SSST 1B • USST 1C (while on backfeed) • USST 1D (while on backfeed) **Onsite:** • 1DG1 • 1DG2 • Unit 2 SBO X-Tie (if already aligned)

Mode Applicability:

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

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

CU2.1

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 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 fed from the unaffected unit (SBO crosstie).
- 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 condition indicated by this EAL is the degradation of the offsite and onsite power sources such that ANY additional single failure would result in a loss of all AC power to the emergency buses.

Table 1C-2 provides a list of offsite and onsite AC power sources to the 4KV emergency buses (ref. 1, 2, 3). Credit can be taken for the Unit 2 SBO crosstie only if already aligned due to the time required to establish (> 15min.).

The subsequent loss of the remaining single power source would escalate the event to an Alert in accordance with IC CA2. This cold condition EAL is equivalent to the hot condition EAL SA1.1.

Basis Reference(s):

1. BV1 UFSAR Section 8.3 System Interconnections
2. BV1 UFSAR Figure 8.1-1 Electrical One Line Diagram BVPS Unit No. 1
3. 1OM-53C.4.1.36.2 Loss of 4KV Emergency Bus
4. NEI 99-01 Rev. 6 CU2

ATTACHMENT 1:

Unit 1 EAL Technical Bases

Category: C – Cold Shutdown / Refueling System Malfunction **CA2.1**

Subcategory: 2 – Loss of Emergency AC Power

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

EAL:

CA2.1	**Alert**
Loss of **ALL** offsite and **ALL** onsite AC power to 4 KV emergency buses 1AE and 1DF for **≥ 15 min.** (Note 1)	

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

Mode Applicability:

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

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

Basis Reference(s):

1. BV1 UFSAR Section 8.3 System Interconnections
2. BV1 UFSAR Figure 8.1-1 Electrical One Line Diagram BVPS Unit No. 1
3. 1OM-53C.4.1.36.2 Loss of 4KV Emergency Bus
4. 1OM-53C.4.1.36.1 Loss of All AC Power when Shutdown
5. NEI 99-01 Rev. 6 CA2

ATTACHMENT 1:

Unit 1 EAL Technical Bases

Category: C – Cold Shutdown / Refueling System Malfunction **CU3.1**

Subcategory: 3 – RCS Temperature

Initiating Condition: UNPLANNED increase in RCS temperature

EAL:

CU3.1	**Unusual Event**
UNPLANNED increase in RCS temperature to **> 200°F** (Note 9)	

Note 9: Begin monitoring hot condition EALs concurrently for any new event or condition not related to the loss of decay heat removal.

Mode Applicability:

5 - Cold Shutdown, 6 - Refueling

Basis:

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

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

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

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

The following instrumentation is capable of providing indication of an RCS temperature rise that approaches the Technical Specification Cold Shutdown temperature limit of (200° F) (ref. 1, 2, 3):

- CET's (incore thermocouples)
- RCS Wide Range Hot Leg Instruments
- RCS Wide Range Cold Leg Instruments
- RHR System Inlet Temperature

The note is a reminder that any temperature increase above 200ºF is an operating mode change from cold to hot conditions. Since each EAL is associated with operating mode

CU3.1

applicability, the set of EALs that must be monitored must now include EALs associated with hot condition operating modes.

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.

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

Basis Reference(s):

1. Technical Specifications Table 1.1-1
2. 1OM-53C.4.1.10.1 Loss of Residual Heat Removal Capability
3. 1OM-53C.4.1.10.2 Loss of RHR While Operating at Reduced InventoryMidloop Conditions Attachment 2 Required RCS Water Level for Reduced Inventory/Midloop
4. NEI 99-01 Rev. 6 CU3

Category: C – Cold Shutdown / Refueling System Malfunction **CU3.2**

Subcategory: 3 – RCS Temperature

Initiating Condition: UNPLANNED increase in RCS temperature

EAL:

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

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

Mode Applicability:

5 - Cold Shutdown, 6- Refueling

Basis:

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

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

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

The following instrumentation is capable of providing indication of an RCS temperature rise that approaches the Technical Specification Cold Shutdown temperature limit of (200° F) (ref. 1, 2, 3):

- CET's (incore thermocouples)
- RCS Wide Range Hot Leg Instruments
- RCS Wide Range Cold Leg Instruments
- RHR System Inlet Temperature

CU3.2

The following instrumentation would be available to provide RCS level:

- Temporary RCS Refueling Level (LI-1RC-481C)
- Temporary RCS Refueling Level Loop A
- Local standpipe (tygon hose)

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

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

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

Basis Reference(s):

1. Technical Specifications Table 1.1-1
2. 1OM-53C.4.1.10.1 Loss of Residual Heat Removal Capability
3. 1OM-53C.4.1.10.2 Loss of RHR While Operating at Reduced InventoryMidloop Conditions Attachment 2 Required RCS Water Level for Reduced Inventory/Midloop
4. NEI 99-01 Rev. 6 CU3

ATTACHMENT 1:

Unit 1 EAL Technical Bases

Category: C – Cold Shutdown / Refueling System Malfunction **CA3.1**

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 1C-3** duration (Notes 1, 9)

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

Note 9: Begin monitoring hot condition EALs concurrently for any new event or condition not related to the loss of decay heat removal.

<table>
<tr><th colspan="3">Table 1C-3: RCS Heat-up Duration Thresholds</th></tr>
<tr><th>RCS Status</th><th>CONTAINMENT CLOSURE Status</th><th>Heat-up Duration</th></tr>
<tr><td>Intact (but not Reduced Inventory)</td><td>N/A</td><td>60 min.*</td></tr>
<tr><td rowspan="2">Not intact
OR
Reduced Inventory</td><td>Established</td><td>20 min.*</td></tr>
<tr><td>Not established</td><td>0 min.</td></tr>
<tr><td colspan="3">* If an RCS heat removal system is in operation within this time frame and RCS temperature is being reduced, the EAL is not applicable.</td></tr>
</table>

Mode Applicability:

5 - Cold Shutdown, 6 – Refueling

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.

CA3.1

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 following instrumentation is capable of providing indication of an RCS temperature rise that approaches the Technical Specification Cold Shutdown temperature limit of (200° F) (ref. 1, 2, 3):

- CET's (incore thermocouples)
- RCS Wide Range Hot Leg Instruments
- RCS Wide Range Cold Leg Instruments
- RHR System Inlet Temperature

The status of CONTAINMENT CLOSURE is tracked if plant conditions change that could raise the risk of a fission product release as a result of a loss of decay heat removal (ref. 4, 5).

The note is a reminder that any temperature increase above 200ºF is an operating mode change from cold to hot conditions. Since each EAL is associated with operating mode applicability, the set of EALs that must be monitored must now include EALs associated with hot condition operating modes.

In the absence of reliable RCS temperature indication caused by the loss of decay heat removal capability, classification should be based on the RCS pressure increase criteria when the RCS is intact in Mode 5 or based on time to boil data when in Mode 6 or the RCS is not intact in Mode 5 (ref. 3).

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

CA3.1

Basis Reference(s):

1. Technical Specifications Table 1.1-1
2. 1OM-53C.4.1.10.1 Loss of Residual Heat Removal Capability
3. 1OM-53C.4.1.10.2 Loss of RHR While Operating at Reduced InventoryMidloop Conditions Attachment 2 Required RCS Water Level for Reduced Inventory/Midloop
4. NOP-OP-1005 Shutdown Defense in Depth
5. 1/2-ADM-0712 Shutdown Defense in Depth Assessment
6. NEI 99-01 Rev. 6 CA3

ATTACHMENT 1:

Unit 1 EAL Technical Bases

Category: C – Cold Shutdown / Refueling System Malfunction **CA3.2**

Subcategory: 3 – RCS Temperature

Initiating Condition: Inability to maintain plant in cold shutdown

EAL:

CA3.2 Alert

RCS temperature **cannot** be monitored

AND

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

Mode Applicability:

5 - Cold Shutdown, 6 – Refueling

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

A 10 psig RCS pressure increase can be monitored on RCS Wide Range Pressure Instruments (ref. 2).

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

Basis Reference(s):

1. 1OM-53C.4.1.10.2 Loss of RHR While Operating at Reduced InventoryMidloop Conditions Attachment 2 Required RCS Water Level for Reduced Inventory/Midloop
2. NEI 99-01 Rev. 6 CA3

ATTACHMENT 1:

Unit 1 EAL Technical Bases

Category: C – Cold Shutdown / Refueling System Malfunction **CU4.1**

Subcategory: 4 – Loss of Vital DC Power

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

EAL:

CU4.1 Unusual Event

Bus voltage indications on Technical Specification **required** 125 VDC buses < the following for **≥ 15 min.** (Notes 1, 17)

- **111 VDC** on Bus 1-1 or 1-2
- **110 VDC** on Bus 1-3 or 1-4

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

Note 17: Indications in the control room should be used to determine when the EAL threshold is approached and 1VM-BAT-1,2,3,4 should be used to validate the voltage for EAL declaration.

Mode Applicability:

5 - Cold Shutdown, 6 - Refueling

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.

The safety-related 125 VDC Power Distribution System is composed of the following (ref. 1, 2):

- two 1700 amp-hour [BAT-1-1 & 1-2] + two 2400 amp-hour [BAT-1-3 & 1-4] batteries
- four dual unit 100 amp battery chargers
- four 125 VDC DC Switchboards [DC-SWBD1-1, 1-2, 1-3 & 1-4]
- four 125 VDC distribution panels

CU4.1

The system also supports a 120 VAC Vital Bus System (that powers vital plant instrumentation), which is powered from 125 VDC / 120 VAC inverters (or by rectified 480 VAC power being inverted, when AC power is available).

The 125 VDC and 120 VAC Vital Bus Systems are designed to provide redundant and reliable power to components and systems that are essential to plant safety, including the Reactor Protective System (RPS) and the Engineered Safety Feature Actuation System (ESFAS) (ref. 3).

The station batteries supply essential and nonessential 125 VDC loads and distribution panels during a loss of the battery charger supply. The batteries are sized to supply the station DC and AC vital bus loads for a period of 2 hours without AC power (ref. 2).

The nominal 60 cell station batteries [BAT-1-1 & 1-2] have a minimum design end of battery cycle voltage of 110.4 VDC, which is equivalent to an average of 1.84 volts per cell (ref. 2, 4). The 110.4 value is rounded to 111 VDC to eliminate the decimal point, since the instrument **cannot** read this level of accuracy.

The nominal 59 cell station batteries [BAT-1-3 & 1-4] have a minimum design end of battery cycle voltage of 110.0 VDC, which is equivalent to an average of 1.864 volts per cell (ref. 2, 4). The 110.0 value is set at 110 VDC to eliminate the decimal point, since the instrument cannot read this level of accuracy.

The indications in the control room should be used to determine when the EAL threshold is approached and 1VM-BAT-1,2,3,4 should be used to validate the voltage for EAL declaration.

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

Basis Reference(s):

1. Technical Specification Bases 3.8.5 DC Sources - Shutdown
2. BV1 UFSAR Section 8.5.3 125 V D-C Power System
3. Technical Specification Bases 3.8.8 Inverters - Shutdown
4. 1DBD-39 Design Basis Document 125 VDC Power System
5. 1OM-39.4.AAI, 125VDC BUS 1 VOLTAGE LOW
6. 1OM-39.4.AAL, 125VDC BUS 2 VOLTAGE LOW
7. 1OM-39.4.AAO, 125VDC BUS 3 VOLTAGE LOW
8. 1OM-39.4.AAR, 125VDC BUS 4 VOLTAGE LOW
9. NEI 99-01 Rev. 6 CU4

ATTACHMENT 1:

Unit 1 EAL Technical Bases

Category: C – Cold Shutdown / Refueling System Malfunction **CU5.1**

Subcategory: 5 – Loss of Communications

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

EAL:

CU5.1 Unusual Event

Loss of **ALL Table 1C-4** onsite communication methods

Table 1C-4 Communication Methods			
System	**Onsite**	**ORO**	**NRC**
Station Page Party Telephone System (Gaitronics)	X		
BVPS Industrial Radios	X	X	
Plant Telephone (PAX)	X	X	X
Commercial Telephones (hardwired & wireless)	X	X	X
Emergency Telephone System (ETS)			X

Mode Applicability:

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

Basis:

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

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

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

Onsite/offsite communications include one or more of the systems listed in Table 1C-4 (ref. 1).

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

CU5.1

Basis Reference(s):

1. BVPS Emergency Plan Section 7.6 Communications
2. NEI 99-01 Rev. 6 CU5

ATTACHMENT 1:

Unit 1 EAL Technical Bases

Category: C – Cold Shutdown / Refueling System Malfunction **CU5.2**

Subcategory: 5 – Loss of Communications

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

EAL:

CU5.2 Unusual Event

Loss of **ALL Table 1C-4** Offsite Response Organization (ORO) communication methods

Table 1C-4 Communication Methods			
System	**Onsite**	**ORO**	**NRC**
Station Page Party Telephone System (Gaitronics)	X		
BVPS Industrial Radios	X	X	
Plant Telephone (PAX)	X	X	X
Commercial Telephones (hardwired & wireless)	X	X	X
Emergency Telephone System (ETS)			X

Mode Applicability:

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

Basis:

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

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

This EAL addresses a total loss of the communications methods used to notify all OROs of an emergency declaration. The OROs referred to here are the EOCs for the States of Pennsylvania, Ohio, West Virginia and counties of Beaver, Columbiana and Hancock.

CU5.2

Onsite/offsite communications include one or more of the systems listed in Table 1C-4 (ref. 1).

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

Basis Reference(s):

1. BVPS Emergency Plan Section 7.6 Communications
2. NEI 99-01 Rev. 6 CU5

ATTACHMENT 1:

Unit 1 EAL Technical Bases

Category: C – Cold Shutdown / Refueling System Malfunction **CU5.3**

Subcategory: 5 – Loss of Communications

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

EAL:

CU5.3 Unusual Event

Loss of **ALL Table 1C-4** NRC communication methods

Table 1C-4 Communication Methods			
System	**Onsite**	**ORO**	**NRC**
Station Page Party Telephone System (Gaitronics)	X		
BVPS Industrial Radios	X	X	
Plant Telephone (PAX)	X	X	X
Commercial Telephones (hardwired & wireless)	X	X	X
Emergency Telephone System (ETS)			X

Mode Applicability:

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

Basis:

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

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

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

Onsite/offsite communications include one or more of the systems listed in Table 1C-4 (ref. 1).

This EAL is the cold condition equivalent of the hot condition EAL SU7.3.

CU5.3

Basis Reference(s):

1. BVPS Emergency Plan Section 7.6 Communications
2. NEI 99-01 Rev. 6 CU5

Category: C – Cold Shutdown / Refueling System Malfunction **CA6.1**

Subcategory: 6 – Hazardous Event Affecting Safety Systems

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

EAL:

CA6.1 Alert

The occurrence of **ANY Table 1C-5** hazardous event

AND:

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

AND EITHER:

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

(Notes 15, 16)

Note 15: If the affected SAFETY SYSTEM train was already inoperable or out of service before the hazardous event occurred, then this emergency classification is not warranted.

Note 16: If the hazardous event only resulted in VISIBLE DAMAGE, with no indications of degraded performance to at least one train of a SAFETY SYSTEM, then this emergency classification is not warranted.

Table 1C-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

CA6.1

Basis:

This IC addresses a hazardous event that causes damage to SAFETY SYSTEMS needed for the current operating mode. In order to provide the appropriate context for consideration of an ALERT classification, the hazardous event must have caused indications of degraded SAFETY SYSTEM performance in one train, and there must be either indications of performance issues with the second SAFETY SYSTEM train or VISIBLE DAMAGE to the second train such that the potential exists for this second SAFETY SYSTEM train to have performance issues. In other words, in order for this EAL to be classified, the hazardous event must occur, at least one SAFETY SYSTEM train must have indications of degraded performance, and the second SAFETY SYSTEM train must have indications of degraded performance or VISIBLE DAMAGE such that the potential exists for performance issues. Note that this second SAFETY SYSTEM train is from the same SAFETY SYSTEM that has indications of degraded performance for the first AND EITHER statement of this EAL; commercial nuclear power plants are designed to be able to support single system issues without compromising public health and safety from radiological events.

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

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

- The Operating Basis Earthquake is 0.06g. It is the conservatively determined earthquake and associated ground motion that might reasonably or probably be expected to occur at the nuclear plant site. Control Room alarm indication of an earthquake greater than OBE is indicated on the seismic monitoring system cabinet 1ER-CCC-1. 1/2OM-53C.4A.75.3 Acts of Nature - Seismic provides the guidance for determining if the OBE earthquake threshold is exceeded and any required response actions (ref. 1). The signficance of seismic events are discussed under EAL HU2.1.
- Internal flooding may be caused by events such as component failures, equipment misalignment, or outage activity mishaps (ref. 2).
- External flooding may be due to river level (ref. 3, 4).

CA6.1

- Seismic Category I structures are analyzed to withstand a sustained, design wind velocity of at least 80 mph. (ref. 5, 6).
- Areas containing functions and systems required for safe shutdown of the plant are identified by fire area (ref. 7, 8, 9).

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

Basis Reference(s):

1. 1/2OM-53C.4A.75.3 Acts of Nature Seismic Event
2. DMC-2169 BVPS-1 PAB Flood
3. 1/2OM-53C.4A.75.2 Acts of Nature - Flood
4. 1/2OM-53C.4A.75.4 Acts of Nature – Dam Failure
5. 1/2OM-53C.4A.75.1 Acts of Nature – Severe Weather
6. BV1 UFSAR Section 2.7.2 Tornado Model
7. BV1 UFSAR Section 2.7.1.1 Seismic Category I Structures
8. BV1 UFSAR Table B.1-1 Structures and Systems Requiring Design for Seismic Loading
9. BV1 UFSAR Table B.3-1 NSSS Fluid Systems Component Seismic Category List
10. NEI 99-01 Rev. 6 CA6

ATTACHMENT 1:

Unit 1 EAL Technical 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 Technological Hazard

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

4. Fire

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

5. Hazardous Gas

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

6. Control Room Evacuation

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

7. Emergency Director Judgment

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

ATTACHMENT 1:

Unit 1 EAL Technical Bases

Category: H – Hazards and Other Conditions Affecting Plant Safety **HU1.1**

Subcategory: 1 – Security

Initiating Condition: Confirmed SECURITY CONDITION or threat

EAL:

HU1.1	**Unusual Event**
A SECURITY CONDITION that does **not** involve a HOSTILE ACTION as reported by the Security Shift Supervisor	

Mode Applicability:

All

Basis:

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

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

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

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

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

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

Basis Reference(s):

1. BVPS Physical Security Plan/Contingency Plan (Safeguards)
2. NEI 99-01 Rev. 6 HU1

ATTACHMENT 1:

Unit 1 EAL Technical Bases

Category: H – Hazards and Other Conditions Affecting Plant Safety **HU1.2**

Subcategory: 1 – Security

Initiating Condition: Confirmed SECURITY CONDITION or threat

EAL:

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

Mode Applicability:

All

Basis:

This IC addresses events that pose a threat to plant personnel or SAFETY SYSTEM equipment, and thus represent a potential degradation in the level of plant safety. Security events which do not meet one of these EALs are adequately addressed by the requirements of 10 CFR § 73.71 or 10 CFR § 50.72. Security events assessed as HOSTILE ACTIONS are classifiable under ICs HA1 and HS1.Timely and accurate communications between Security Shift Supervision and the Control Room is essential for proper classification of a security-related event. Classification of these events will initiate appropriate threat-related notifications to plant personnel and Offsite Response Organizations.

Security plans and terminology are based on the guidance provided by NEI 03-12, *Template for the Security Plan, Training and Qualification Plan, Safeguards Contingency Plan*

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

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

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

Basis Reference(s):

1. BVPS Physical Security Plan/Contingency Plan (Safeguards)
2. NEI 99-01 Rev. 6 HU1

ATTACHMENT 1:

Unit 1 EAL Technical Bases

Category: H – Hazards and Other Conditions Affecting Plant Safety **HU1.3**

Subcategory: 1 – Security

Initiating Condition: Confirmed SECURITY CONDITION or threat

EAL:

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

Mode Applicability:

All

Basis:

This IC addresses events that pose a threat to plant personnel or SAFETY SYSTEM equipment, and thus represent a potential degradation in the level of plant safety. Security events which do not meet one of these EALs are adequately addressed by the requirements of 10 CFR § 73.71 or 10 CFR § 50.72. Security events assessed as HOSTILE ACTIONS are classifiable under ICs HA1 and HS1.Timely and accurate communications between Security Shift Supervision and the Control Room is essential for proper classification of a security-related event. Classification of these events will initiate appropriate threat-related notifications to plant personnel and Offsite Response Organizations.

Security plans and terminology are based on the guidance provided by NEI 03-12, *Template for the Security Plan, Training and Qualification Plan, Safeguards Contingency*

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

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

Basis Reference(s):

1. BVPS Physical Security Plan/Contingency Plan (Safeguards)
2. NEI 99-01 Rev. 6 HU1

ATTACHMENT 1:

Unit 1 EAL Technical Bases

Category: H – Hazards and Other Conditions Affecting Plant Safety **HA1.1**

Subcategory: 1 – Security

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

EAL:

HA1.1	**Alert**
A HOSTILE ACTION is occurring or has occurred within the OWNER CONTROLLED AREA as reported by the Security Shift Supervisor	

Mode Applicability:

All

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 Supervisor and the Control Room is essential for proper classification of a security-related event.

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

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

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

This EAL is applicable for any HOSTILE ACTION occurring, or that has occurred, in the OWNER CONTROLLED AREA.

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 BVPS Physical Security Plan/Contingency Plan (ref. 1).

HA1.1

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

Basis Reference(s):

1. BVPS Physical Security Plan/Contingency Plan (Safeguards)
2. NEI 99-01 Rev. 6 HA1

Category: H – Hazards and Other Conditions Affecting Plant Safety **HA1.2**

Subcategory: 1 – Security

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

EAL:

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

Mode Applicability:

All

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 Supervisor or and the Control Room is essential for proper classification of a security-related event.

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

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

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

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

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

ATTACHMENT 1:

Unit 1 EAL Technical Bases

HA1.2

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 BVPS Physical Security Plan/Contingency Plan (ref. 1).

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

Basis Reference(s):

1. BVPS Physical Security Plan/Contingency Plan (Safeguards)
2. NEI 99-01 Rev. 6 HA1

ATTACHMENT 1:

Unit 1 EAL Technical Bases

Category: H – Hazards and Other Conditions Affecting Plant Safety **HS1.1**

Subcategory: 1 – Security

Initiating Condition: HOSTILE ACTION within the PROTECTED AREA

EAL:

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

Mode Applicability:

All

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 the Security Shift Supervisor and the Control Room is essential for proper classification of a security-related event.

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

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

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

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

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

ATTACHMENT 1:

Unit 1 EAL Technical Bases

HS1.1

Basis Reference(s):

1. BVPS Physical Security Plan/Contingency Plan (Safeguards)
2. NEI 99-01 Rev. 6 HS1

Category: H – Hazards and Other Conditions Affecting Plant Safety **HU2.1**

Subcategory: 2 – Seismic Event

Initiating Condition: Seismic event greater than OBE level

EAL:

HU2.1	**Unusual Event**
Seismic event **> OBE (> 0.06g)** as indicated by lit lamp on 1ER-CCC-1 Seismic Instrumentation Central Control Cabinet	

Mode Applicability:

All

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.

Seismic events of this magnitude require plant shutdown and evaluation to determine if any damage to plant SSCs has occurred. The post seismic condition of the plant is determined by plant walkdowns and monitoring of plant perimeters to determine if damage has occurred to plant safety systems.

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 0.06g). The Shift Manager or Emergency Director may seek external verification if deemed appropriate (e.g., a call to the USGS, check internet news sources, etc.); however, the verification action must not preclude a timely emergency declaration.

The Operating Basis Earthquake is 0.06g. It is the conservatively determined earthquake and associated ground motion that might reasonably or probably be expected to occur at the nuclear plant site (ref. 1).

1/2OM-53C.4A.75.3 Acts of Nature - Seismic provides the guidance for determining if the OBE earthquake threshold is exceeded and any required response actions. (ref. 2)

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

ATTACHMENT 1:

Unit 1 EAL Technical Bases

HU2.1

Basis Reference(s):

1. BV1 UFSAR Section 2.5.3 Seismic Design
2. 1/2OM-53C.4A.75.3 Acts of Nature – Seismic Event
3. NEI 99-01 Rev. 6 HU2

Category: H – Hazards and Other Conditions Affecting Plant Safety **HU3.1**

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

Basis:

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

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

Response actions associated with a tornado onsite is provided in 1/2OM-53C.4A.75.1 Acts of Nature – Severe Weather (ref. 1).

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

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

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

Basis Reference(s):

1. 1/2OM-53C.4A.75.1 Acts of Nature – Severe Weather
2. NEI 99-01 Rev. 6 HU3

Category: H – Hazards and Other Conditions Affecting Plant Safety **HU3.2**

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 (Note 13)	

Note 13: Flooding refers to 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.

Mode Applicability:

All

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.

Depending upon the plant mode at the time of the event, refer to EAL CA6.1 or SA9.1 for internal flooding affecting one or more SAFETY SYSTEM trains . Escalation of the emergency classification level would be based on ICs in Recoginition Categories R, F, S or C.

Basis Reference(s):

1. BV1 Calculation DMC-2169, PAB Flood Level Resulting from REJ-18 Failure
2. NEI 99-01 Rev. 6 HU3

Category: H – Hazards and Other Conditions Affecting Plant Safety **HU3.3**

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) (Notes 12 and 14)	

Note 12: 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).

Note 14: As used here, the term “offsite” is meant to be areas external to the BVPS PROTECTED AREA.

Mode Applicability:

All

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

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

Basis Reference(s):

1. NEI 99-01 Rev. 6 HU3

ATTACHMENT 1:

Unit 1 EAL Technical Bases

Category: H – Hazards and Other Conditions Affecting Plant Safety **HU3.4**

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

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.

Basis Reference(s):

1. NEI 99-01 Rev. 6 HU3

ATTACHMENT 1:

Unit 1 EAL Technical Bases

Category: H – Hazards and Other Conditions Affecting Plant Safety **HU4.1**

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 (Note 11)
- Field verification of a single fire alarm (Note 11)

AND

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

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

Note 11: Incipient Fire Detection alarms are **not** considered control room fire alarms for this EAL.

Table 1H-1 Safe Shutdown Fire Areas
• Cable Tunnel (CV-3) • CONTROL ROOM • Containment Building • Demin. Water Storage Tank (1WT-TK-10) • Diesel Generator Building • Fuel Building • Intake Structure Pump Cubicles • Safeguards (including AFW, Main Steam and Cable Vault Areas) • Primary Auxiliary Building (except elev. 768') • RWST (1QS-TK-1) • Service Building (below elev. 735')

Mode Applicability:

All

ATTACHMENT 1:

Unit 1 EAL Technical Bases

Basis: **HU4.1**

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

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

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

Basis-Related Requirements from Appendix R

Appendix R to 10 CFR 50, states in part:

> Criterion 3 of Appendix A to this part specifies that "Structures, systems, and components important to safety shall be designed and located to minimize, consistent with other safety requirements, the probability and effect of fires and explosions."
>
> When considering the effects of fire, those systems associated with achieving and maintaining safe shutdown conditions assume major importance to safety because damage to them can lead to core damage resulting from loss of coolant through boil-off.
>
> Because fire may affect safe shutdown systems and because the loss of function of systems used to mitigate the consequences of design basis accidents under post-fire conditions does not per se impact public safety, the need to limit fire damage to systems required to achieve and maintain safe shutdown conditions is greater than the need to limit fire damage to those systems required to mitigate the consequences of design basis accidents.

In addition, Appendix R to 10 CFR 50, requires, among other considerations, the use of 1-hour fire barriers for the enclosure of cable and equipment and associated non-safety circuits of one redundant train (G.2.c).

HU4.1

Table 1H-1 applies to buildings and areas housing equipment needed for safe shutdown (SAFETY SYSTEMS) (ref. 1). The list includes the structures containing the equipment for safe shutdown, certain structures may contain equipment not needed if the plant is already in a shutdown mode.

Incipient Fire Detection alarms are **not** considered control room fire alarms for this EAL. The purpose of Incipient Fire Detection is to detect conditions days/weeks before any FIRE develops.

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

Basis Reference(s):

1. BV1 UFSAR Appendix A Table A.1-1 Catergory I Structures, Systems and Components
2. NEI 99-01 Rev. 6 HU4

ATTACHMENT 1:

Unit 1 EAL Technical Bases

Category: H – Hazards and Other Conditions Affecting Plant Safety **HU4.2**

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) (Note 11)
AND
The fire alarm is indicating a FIRE within **ANY Table 1H-1** area (Note 11)
AND
The existence of a FIRE is **not** verified within **30 min.** of alarm receipt (Note 1, 11)

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

Note 11: Incipient Fire Detection alarms are **not** considered control room fire alarms for this EAL.

Table 1H-1 Safe Shutdown Fire Areas
• Cable Tunnel (CV-3) • CONTROL ROOM • Containment Building • Demin. Water Storage Tank (1WT-TK-10) • Diesel Generator Building • Fuel Building • Intake Structure Pump Cubicles • Safeguards (including AFW, Main Steam and Cable Vault Areas) • Primary Auxiliary Building (except elev. 768') • RWST (1QS-TK-1) • Service Building (below elev. 735')

Mode Applicability:

All

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.

HU4.2

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

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

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

Basis-Related Requirements from Appendix R

Appendix R to 10 CFR 50, states in part:

> Criterion 3 of Appendix A to this part specifies that "Structures, systems, and components important to safety shall be designed and located to minimize, consistent with other safety requirements, the probability and effect of fires and explosions."
>
> When considering the effects of fire, those systems associated with achieving and maintaining safe shutdown conditions assume major importance to safety because damage to them can lead to core damage resulting from loss of coolant through boil-off.
>
> Because fire may affect safe shutdown systems and because the loss of function of systems used to mitigate the consequences of design basis accidents under post-fire conditions does not per se impact public safety, the need to limit fire damage to systems required to achieve and maintain safe shutdown conditions is greater than the need to limit fire damage to those systems required to mitigate the consequences of design basis accidents.

In addition, Appendix R to 10 CFR 50, requires, among other considerations, the use of 1-hour fire barriers for the enclosure of cable and equipment and associated non-safety circuits of one redundant train (G.2.c). As used in HU4.2, the 30-minutes to verify a single alarm is well within this worst-case 1-hour time period.

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.

HU4.2

Table 1H-1 applies to buildings and areas housing equipment needed for safe shutdown (SAFETY SYSTEMS) (ref. 1). The list includes the structures containing the equipment for safe shutdown, certain structures may contain equipment not needed if the plant is already in a shutdown mode.

Incipient Fire Detection alarms are **not** considered control room fire alarms for this EAL. The purpose of Incipient Fire Detection is to detect conditions days/weeks before any FIRE develops.

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

Basis Reference(s):

1. BV1 UFSAR Appendix A Table A.1-1 Catergory I Structures, Systems and Components
2. NEI 99-01 Rev. 6 HU4

ATTACHMENT 1:

Unit 1 EAL Technical Bases

Category: H – Hazards and Other Conditions Affecting Plant Safety **HU4.3**

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, 11)	

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

Note 11: Incipient Fire Detection alarms are **not** considered control room fire alarms for this EAL.

Mode Applicability:

All

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.

Incipient Fire Detection alarms are **not** considered control room fire alarms for this EAL. The purpose of Incipient Fire Detection is to detect conditions days/weeks before any FIRE develops.

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

Basis Reference(s):

1. NEI 99-01 Rev. 6 HU4

ATTACHMENT 1:

Unit 1 EAL Technical Bases

Category: H – Hazards and Other Conditions Affecting Plant Safety **HU4.4**

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

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.

Basis Reference(s):

1. NEI 99-01 Rev. 6 HU4

ATTACHMENT 1:

Unit 1 EAL Technical Bases

Category: H – Hazards and Other Conditions Affecting Plant Safety **HA5.1**

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 1H-2** rooms or areas

AND

Entry into the room or area is prohibited or impeded (Notes 5, 12)

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 12: 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).

Table 1H-2 Safe Operation & Shutdown Rooms/Areas

Room/Area	Mode Applicability
Control Room	All
Safeguards 735' East and West Cable Vault (2 separate areas)	4
Safeguards 722' Penetrations D	4
Auxiliary Building 735' CCR Hx Area	4
Service Building 713' AE Emergency Switchgear	4

Mode Applicability:

Refer to Table 1H-2 for Mode Applicability

Basis:

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

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

HA5.1

Evaluation of the IC and EAL do not require atmospheric sampling; it only requires the Emergency Director judgment that the gas concentration in the affected room/area is sufficient to preclude or significantly impede procedurally required access. This judgment may be based on a variety of factors including an existing job hazard analysis, report of ill effects on personnel, advice from a subject matter expert or operating experience with the same or similar hazards. Access should be considered as impeded if extraordinary measures are necessary to facilitate entry of personnel into the affected room/area (e.g., requiring use of protective equipment, such as SCBAs, that is not routinely employed).

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

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

An asphyxiant is a gas capable of reducing the level of oxygen in the body to dangerous levels. Most commonly, asphyxiants work by merely displacing air in an enclosed environment. This reduces the concentration of oxygen below the normal level of around 19%, which can lead to breathing difficulties, unconsciousness or even death.

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

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).

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

Basis Reference(s): **HA5.1**

1. EPLAN, Section 4, Attachment 5 Safe Operation & Shutdown Areas RA3.2 & HA5.1 Bases
2. NEI 99-01 Rev. 6 HA5

ATTACHMENT 1:

Unit 1 EAL Technical Bases

Category: H – Hazards and Other Conditions Affecting Plant Safety **HA6.1**

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 Emergency Shutdown Panel (SDP) or Back-up Indicating Panel (BIP)	

Mode Applicability:

All

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.

AOP 1.33.1A specifies conditions under which CONTROL ROOM evacuation may be necessary. This EAL is only applicable when the decision has been made to evacuate the CONTROL ROOM, not when conditions are being evaluated per 1OM-53C.4.1.33.1A.
(Ref. 1, 2).

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

Basis Reference(s):

1. 1OM-53C.4.1.33.1A Control Room Inaccessibility
2. 1OM-56C.4.B Alternate Safe Shutdown from Outside Control Room
3. NEI 99-01 Rev. 6 HA6

ATTACHMENT 1:

Unit 1 EAL Technical Bases

Category: H – Hazards and Other Conditions Affecting Plant Safety **HS6.1**

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 Emergency Shutdown Panel (SDP) or Back-up Indicating Panel (BIP)

AND

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

- Reactivity control (modes 1, 2, and 3 only)
- RCS Inventory (inventory control to maintain core cooling)
- RCS heat removal

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

Mode Applicability:

1 - Power Operation, 2 - Startup, 3 - Hot Standby, 4 - Hot Shutdown, 5 – Cold Shutdown, 6 - Refueling

Basis:

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

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

The Shift Manager determines if the Control Room is inoperable and requires evacuation. Control Room inhabitability may be caused by FIRE, dense smoke, noxious fumes, bomb threat in or adjacent to the Control Room, or other life threatening conditions (Ref. 1, 2).

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

HS6.1

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

Basis Reference(s):

1. 1OM-53C.4.1.33.1A Control Room Inaccessibility
2. 1OM-56C.4.B Alternate Safe Shutdown from Outside Control Room
3. NEI 99-01 Rev. 6 HS6

ATTACHMENT 1:

Unit 1 EAL Technical Bases

Category: H – Hazards and Other Conditions Affecting Plant Safety **HU7.1**

Subcategory: 7 – Emergency Director Judgment

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

EAL:

HU7.1	**Unusual Event**
Other conditions exist which in the judgment of the Emergency Director indicate that events are in progress or have occurred which indicate a potential degradation of the level of safety of the plant or indicate a security threat to facility protection has been initiated. No releases of radioactive material requiring offsite response or monitoring are expected unless further degradation of SAFETY SYSTEMS occurs.	

Mode Applicability:

All

Basis:

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

Basis Reference(s):

1. BVPS Emergency Preparedness Plan Section 5.2 BVPS Emergency Organization
2. NEI 99-01 Rev. 6 HU7

ATTACHMENT 1:

Unit 1 EAL Technical Bases

Category: H – Hazards and Other Conditions Affecting Plant Safety **HA7.1**

Subcategory: 7 – Emergency Director Judgment

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

EAL:

HA7.1 Alert

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

Mode Applicability:

All

Basis:

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

Basis Reference(s):

1. BVPS Emergency Preparedness Plan Section 5.2 BVPS Emergency Organization
2. NEI 99-01 Rev. 6 HA7

ATTACHMENT 1:

Unit 1 EAL Technical Bases

Category: H – Hazards and Other Conditions Affecting Plant Safety **HS7.1**

Subcategory: 7 – Emergency Director Judgment

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

EAL:

HS7.1	**Site Area Emergency**
Other conditions exist which in the judgment of the Emergency Director indicate that events are in progress or have occurred which involve actual or likely major failures of plant functions needed for protection of the public or HOSTILE ACTION that results in intentional damage or malicious acts, (1) toward site personnel or equipment that could lead to the likely failure of or, (2) that prevent effective access to equipment needed for the protection of the public. Any releases are not expected to result in exposure levels which exceed EPA Protective Action Guideline exposure levels beyond the site boundary	

Mode Applicability:

All

Basis:

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

Basis Reference(s):

1. BVPS Emergency Preparedness Plan Section 5.2 BVPS Emergency Organization
2. NEI 99-01 Rev. 6 HS7

ATTACHMENT 1:

Unit 1 EAL Technical Bases

Category:	H – Hazards and Other Conditions Affecting Plant Safety	**HG7.1**
Subcategory:	7 – Emergency Director Judgment	
Initiating Condition:	Other conditions exist which in the judgment of the Emergency Director warrant declaration of a General Emergency	

EAL:

HG7.1 **General Emergency**

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

Mode Applicability:

All

Basis:

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

Basis Reference(s):

1. BVPS Emergency Preparedness Plan Section 5.2 BVPS Emergency Organization
2. NEI 99-01 Rev. 6 HG7

ATTACHMENT 1:

Unit 1 EAL Technical Bases

Category S – System Malfunction

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

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

The events of this category pertain to the following subcategories:

1. Loss of Emergency AC Power

Loss of emergency 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 4KV 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 essential plant 125 VDC power sources.

3. Loss of Control Room Indications

Certain events that degrade plant operator's ability to assess plant conditions within the plant warrant emergency classification. Losses of indicators are in this subcategory.

4. RCS Activity

During normal operation, reactor coolant fission product activity is very low. Small concentrations of fission products in the coolant are primarily from the fission of tramp uranium in the fuel clad or minor perforations in the clad itself. Any significant increase from these base-line levels (2% - 5% clad failures) is indicative of fuel failures and is covered under the Fission Product Barrier Degradation category. However, lesser amounts of clad damage may result in coolant activity exceeding Technical Specification limits. These fission products will be circulated with the reactor coolant and can be detected by coolant sampling.

5. RCS Leakage

The reactor vessel provides a volume for the coolant that covers the reactor core. The reactor pressure vessel and associated pressure piping (reactor coolant system) together provide a barrier to limit the release of radioactive material should the reactor fuel clad integrity fail. Excessive RCS leakage greater than Technical Specification limits indicates potential pipe cracks that may propagate to an extent threatening fuel clad, RCS and containment integrity.

ATTACHMENT 1:

Unit 1 EAL Technical Bases

6. RPS Failure

This subcategory includes events related to failure of the Reactor Protection System (RPS) to initiate and complete reactor trips. In the plant licensing basis, postulated failures of the RPS to complete a reactor trip comprise a specific set of analyzed events referred to as Anticipated Transient Without Scram (ATWS) events. For EAL classification, however, ATWS is intended to mean ANY trip failure event that does not achieve reactor shutdown. If RPS actuation fails to assure reactor shutdown, positive control of reactivity is at risk and could cause a threat to fuel clad, RCS and containment integrity.

7. Loss of Communications

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

8. Containment Failure

Failure of containment isolation capability (under conditions in which the containment is not currently challenged) warrants emergency classification. Failure of containment pressure control capability also warrants emergency classification.

9. Hazardous Event Affecting SAFETY SYSTEMS

Various natural and technological events that result in degraded plant SAFETY SYSTEM performance or significant visible damage warrant emergency classification under this subcategory.

ATTACHMENT 1:

Unit 1 EAL Technical Bases

Category: S – System Malfunction **SU1.1**

Subcategory: 1 – Loss of Emergency AC Power

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

EAL:

SU1.1 Unusual Event

Loss of **ALL** offsite AC power capability, **Table 1S-1**, to 4 KV emergency buses 1AE and 1DF for **≥ 15 min.** (Note 1)

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

Table 1S-1 AC Power Sources
Offsite: • SSST 1A • SSST 1B • USST 1C (while on backfeed) • USST 1D (while on backfeed) **Onsite:** • 1DG1 • 1DG2 • Unit 2 SBO X-Tie (if already aligned)

Mode Applicability:

1 - Power Operation, 2 - Startup, 3 - Hot Standby, 4 – Hot Shutdown

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.

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

Table 1S-1 provides a list of offsite and onsite AC power sources to the 4KV emergency buses (ref. 1, 2, 3). Credit can be taken for the Unit 2 SBO crosstie only if already aligned due to the time required to establish (> 15min.).

SU1.1

Escalation of the emergency classification level would be via IC SA1.

Basis Reference(s):

1. BV1 UFSAR Section 8.3 System Interconnections
2. BV1 UFSAR Figure 8.1-1 Electrical One Line Diagram BVPS Unit No. 1
3. 1OM-53C.4.1.36.2 Loss of 4KV Emergency Bus
4. NEI 99-01 Rev. 6 SU1

ATTACHMENT 1:

Unit 1 EAL Technical Bases

Category: S – System Malfunction **SA1.1**

Subcategory: 1 – Loss of Emergency AC Power

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

EAL:

SA1.1 Alert

AC power capability, **Table 1S-1**, to 4 KV emergency buses 1AE and 1DF reduced to a single power source for **≥ 15 min.** (Note 1)

AND

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

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

Table 1S-1 AC Power Sources
Offsite: • SSST 1A • SSST 1B • USST 1C (while on backfeed) • USST 1D (while on backfeed) **Onsite:** • 1DG1 • 1DG2 • Unit 2 SBO X-Tie (if already aligned)

Mode Applicability:

1 - Power Operation, 2 - Startup, 3 - Hot Standby, 4 - Hot Shutdown

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.

SA1.1

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 fed from the unaffected unit (SBO crosstie).
- 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.

Table 1S-1 provides a list of offsite and onsite AC power sources to the 4KV emergency buses (ref. 1, 2, 3). Credit can be taken for the Unit 2 SBO crosstie only if already aligned due to the time required to establish (> 15min.).

If the capability of a second source of emergency bus power is not restored within 15 minutes, an Alert is declared under this EAL

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

Escalation of the emergency classification level would be via IC SS1. This hot condition EAL is equivalent to the cold condition EAL CU2.1.

Basis Reference(s):

1. BV1 UFSAR Section 8.3 System Interconnections
2. BV1 UFSAR Figure 8.1-1 Electrical One Line Diagram BVPS Unit No. 1
3. 1OM-53C.4.1.36.2 Loss of 4KV Emergency Bus
4. NEI 99-01 Rev. 6 SA1

ATTACHMENT 1:

Unit 1 EAL Technical Bases

Category: S – System Malfunction **SS1.1**

Subcategory: 1 – Loss of Emergency AC Power

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

EAL:

SS1.1	**Site Area Emergency**
Loss of **ALL** offsite and **ALL** onsite AC power to 4 KV emergency buses 1AE and 1DF for **≥ 15 min.** (Note 1)	

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

Mode Applicability:

1 - Power Operation, 2 - Startup, 3 - Hot Standby, 4 - Hot Shutdown

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.

Basis Reference(s):

1. BV1 UFSAR Section 8.3 System Interconnections
2. BV1 UFSAR Figure 8.1-1 Electrical One Line Diagram BVPS Unit No. 1
3. 1OM-53C.4.1.36.2 Loss of 4KV Emergency Bus
4. 1OM53A.1.ECA-0.0 Loss of All Emergency 4KV AC Power
5. NEI 99-01 Rev. 6 SS1

ATTACHMENT 1:

Unit 1 EAL Technical Bases

Category: S –System Malfunction **SG1.1**

Subcategory: 1 – Loss of Emergency AC Power

Initiating Condition: Prolonged loss of **all** offsite and **all** onsite AC power to emergency buses

EAL:

SG1.1 General Emergency
Loss of **ALL** offsite and **ALL** onsite AC power to 4 KV emergency buses 1AE and 1DF
AND EITHER:
• Restoration of at least one emergency bus in **< 4 hours** is **not** likely (Note 1)
• Core Cooling RED Path conditions met

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

Mode Applicability:

1 - Power Operation, 2 - Startup, 3 - Hot Standby, 4 - Hot Shutdown

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.

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.

SG1.1

This EAL is indicated by the extended loss of all offsite and onsite AC power capability to 4 KV emergency buses AE and DF either for greater then the BVPS Station Blackout (SBO) coping analysis time (4 hrs.) (ref. 5) 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. 6).

Indication of continuing core cooling degradation must be based on fission product barrier monitoring with particular emphasis on Emergency Director judgment as it relates to IMMINENT loss 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. Critical Safety Function Status Tree (CSFST) Core Cooling-RED path indicates significant core exit superheating and core uncovery (ref. 6).

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.

Basis Reference(s):

1. BV1 UFSAR Section 8.3 System Interconnections
2. BV1 UFSAR Figure 8.1-1 Electrical One Line Diagram BVPS Unit No. 1
3. 1OM-53C.4.1.36.2 Loss of 4KV Emergency Bus
4. 1OM53A.1.ECA-0.0 Loss of All Emergency 4KV AC Power
5. BV1 Calculation DEC-0248, Coping Duration for Station Black Out
6. 1OM-53A.1.F-0.2 Core Cooling Status Tree
7. NEI 99-01 Rev. 6 SG1

ATTACHMENT 1:

Unit 1 EAL Technical Bases

SG1.2

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 to 4 KV emergency buses 1AE and 1DF for **≥ 15 min.**

AND

Bus voltage indications on **ALL** Technical Specification 125 VDC buses **<** the following for ≥ 15 min.:

- **111 VDC** on Bus 1-1 or 1-2
- **110 VDC** on Bus 1-3 or 1-4

(Notes 1, 17)

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

Note 17: Indications in the control room should be used to determine when the EAL threshold is approached and 1VM-BAT-1,2,3,4 should be used to validate the voltage for EAL declaration.

Mode Applicability:

1 - Power Operation, 2 - Startup, 3 - Hot Standby, 4 - Hot Shutdown

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.

This EAL is indicated by the loss of all offsite and onsite emergency AC power capability to 4KV safeguard buses AE and DF 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.

ATTACHMENT 1:

Unit 1 EAL Technical Bases

SG1.2

The system supports a 120 VAC Vital Bus System (that powers vital plant instrumentation), which is powered from 125 VDC / 120 VAC inverters (or by rectified 480 VAC power being inverted, when AC power is available).

The 125 VDC and 120 VAC Vital Bus Systems are designed to provide redundant and reliable power to components and systems that are essential to plant safety, including the Reactor Protective System (RPS) and the Engineered Safety Feature Actuation System (ESFAS) (ref. 5).

The station batteries supply essential and nonessential 125 VDC loads and distribution panels during a loss of the battery charger supply. The batteries are sized to supply the station DC and AC vital bus loads for a period of 2 hours without AC power (ref. 6).

The nominal 60 cell station batteries [BAT-1-1 & 1-2] have a minimum design end of battery cycle voltage of 110.4 VDC, which is equivalent to an average of 1.84 volts per cell (ref. 5, 8). The 110.4 value is rounded to 111 VDC to eliminate the decimal point, since the instrument cannot read this level of accuracy.

The nominal 59 cell station batteries [BAT-1-3 & 1-4] have a minimum design end of battery cycle voltage of 110.0 VDC, which is equivalent to an average of 1.864 volts per cell (ref. 5, 7). The 110.0 value is set at 110 VDC to eliminate the decimal point, since the instrument cannot read this level of accuracy.

The indications in the control room should be used to determine when the EAL threshold is approached and 1VM-BAT-1,2,3,4 should be used to validate the voltage for EAL declaration.

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.

SG1.2

Basis Reference(s):

1. BV1 UFSAR Section 8.3 System Interconnections
2. BV1 UFSAR Figure 8.1-1 Electrical One Line Diagram BVPS Unit No. 1
3. 1OM-53C.4.1.36.2 Loss of 4KV Emergency Bus
4. 1OM53A.1.ECA-0.0 Loss of All Emergency 4KV AC Power
5. Technical Specification Bases 3.8.5 DC Sources - Shutdown
6. BV1 UFSAR Section 8.5.3 125 V D-C Power System
7. Technical Specification Bases 3.8.8 Inverters - Shutdown
8. 1DBD-39 Design Basis Document 125 VDC Power System
9. 1OM-39.4.AAI, 125VDC BUS 1 VOLTAGE LOW
10. 1OM-39.4.AAL, 125VDC BUS 2 VOLTAGE LOW
11. 1OM-39.4.AAO, 125VDC BUS 3 VOLTAGE LOW
12. 1OM-39.4.AAR, 125VDC BUS 4 VOLTAGE LOW
13. NEI 99-01 Rev. 6 SG8

ATTACHMENT 1:

Unit 1 EAL Technical Bases

Category: S – System Malfunction **SS2.1**

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

Bus voltage indications on **ALL** Technical Specification 125 VDC buses < the following for **≥ 15 min.** (Notes 1, 17):

- **111 VDC** on Bus 1-1 or 1-2
- **110 VDC** on Bus 1-3 or 1-4

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

Note 17: Indications in the control room should be used to determine when the EAL threshold is approached and 1VM-BAT-1,2,3,4 should be used to validate the voltage for EAL declaration.

Mode Applicability:

1 - Power Operation, 2 - Startup, 3 - Hot Standby, 4 - Hot Shutdown

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.

The system supports a 120 VAC Vital Bus System (that powers vital plant instrumentation), which is powered from 125 VDC / 120 VAC inverters (or by rectified 480 VAC power being inverted, when AC power is available).

The 125 VDC and 120 VAC Vital Bus Systems are designed to provide redundant and reliable power to components and systems that are essential to plant safety, including the Reactor Protective System (RPS) and the Engineered Safety Feature Actuation System (ESFAS) (ref. 3).

The station batteries supply essential and nonessential 125 VDC loads and distribution panels during a loss of the battery charger supply. The batteries are sized to supply the station DC and AC vital bus loads for a period of 2 hours without AC power (ref. 2).

The nominal 60 cell station batteries [BAT-1-1 & 1-2] have a minimum design end of battery cycle voltage of 110.4 VDC, which is equivalent to an average of 1.84 volts per cell (ref. 1, 4). The 110.4 value is rounded to 111 VDC to eliminate the decimal point, since the analog instrument cannot be read this level of accuracy (ref 2).

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SS2.1

The nominal 59 cell station batteries [BAT-1-3 & 1-4] have a minimum design end of battery cycle voltage of 110.0 VDC, which is equivalent to an average of 1.864 volts per cell (ref. 1, 3). The 110.0 value is set at 110 VDC to eliminate the decimal point, since single unit precision is the best that can be read on an analog meter face with graduations every 2 VDC.

The indications in the control room should be used to determine when the EAL threshold is approached and 1VM-BAT-1,2,3,4 should be used to validate the voltage for EAL declaration.

Escalation of the emergency classification level would be via ICs RG1, FG1 or SG1.

Basis Reference(s):

1. Technical Specification Bases 3.8.4 DC Sources
2. BV1 UFSAR Section 8.5.3 125 V D-C Power System
3. Technical Specification Bases 3.8.7 Inverter
4. 1DBD-39 Design Basis Document 125 VDC Power System
5. 1OM-39.4.AAI, 125VDC BUS 1 VOLTAGE LOW
6. 1OM-39.4.AAL, 125VDC BUS 2 VOLTAGE LOW
7. 1OM-39.4.AAO, 125VDC BUS 3 VOLTAGE LOW
8. 1OM-39.4.AAR, 125VDC BUS 4 VOLTAGE LOW
9. NEI 99-01 Rev. 6 SS8

ATTACHMENT 1:

Unit 1 EAL Technical Bases

Category: S – System Malfunction **SU3.1**

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 1S-2** parameters from within the Control Room for **≥ 15 min.** (Note 1)

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

Table 1S-2 Safety System Parameters
• Reactor power • RCS level • RCS pressure • Core Exit T/C temperature • Level in at least one SG • Auxiliary or emergency feed flow in at least one SG

Mode Applicability:

1 - Power Operation, 2 - Startup, 3 - Hot Standby, 4 - Hot Shutdown

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,

SU3.1

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.

SAFETY SYSTEM parameters listed in Table 1S-2 are monitored in the Control Room through a combination of hard control panel indicators as well as computer based information systems. The Plant Computer, which displays Safety Parameter Display System (SPDS) required information, serves as a redundant compensatory indicator which may be utilized in lieu of normal Control Room indicators (ref. 1, 2).

Escalation of the emergency classification level would be via IC SA3.

Basis Reference(s):

1. BV1 UFSAR Section 7.5 Safety Related Display Information
2. 1DBD-05C Inadequate Core Cooling Monitoring System
3. NEI 99-01 Rev. 6 SU2

ATTACHMENT 1:

Unit 1 EAL Technical Bases

Category: S – System Malfunction **SA3.1**

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 1S-2** parameters from within the Control Room for **≥ 15 min.** (Note 1)

AND

ANY significant transient is in progress, **Table 1S-3**

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

Table 1S-2 Safety System Parameters
• Reactor power • RCS level • RCS pressure • Core Exit T/C temperature • Level in at least one SG • Auxiliary or emergency feed flow in at least one SG

Table 1S-3 Significant Transients
• Reactor trip • Automatic turbine runback ≥ 25% thermal power • Electrical load rejection > 25% full electrical load • Safety Injection actuation

Mode Applicability:

1 - Power Operation, 2 - Startup, 3 - Hot Standby, 4 - Hot Shutdown

Basis: **SA3.1**

This IC addresses the difficulty associated with monitoring rapidly changing plant conditions during a transient without the ability to obtain SAFETY SYSTEM parameters from within the Control Room. During this condition, the margin to a potential fission product barrier challenge is reduced. It thus represents a potential substantial degradation in the level of safety of the plant.

As used in this EAL, an "inability to monitor" means that values for one or more of the listed parameters cannot be determined from within the Control Room. This situation would require a loss of all of the Control Room sources for the given parameter(s). For example, the reactor power level cannot be determined from any analog, digital and recorder source within the Control Room.

An event involving a loss of plant indications, annunciators and/or display systems is evaluated in accordance with 10 CFR 50.72 (and associated guidance in NUREG-1022) to determine if an NRC event report is required. The event would be reported if it significantly impaired the capability to perform emergency assessments. In particular, emergency assessments necessary to implement abnormal operating procedures, emergency operating procedures, and emergency plan implementing procedures addressing emergency classification, accident assessment, or protective action decision-making.

This EAL is focused on a selected subset of plant parameters associated with the key safety functions of reactivity control, core cooling and RCS heat removal. The loss of the ability to determine one or more of these parameters from within the Control Room is considered to be more significant than simply a reportable condition. In addition, if all indication sources for one or more of the listed parameters are lost, then the ability to determine the values of other SAFETY SYSTEM parameters may be impacted as well. For example, if the value for reactor vessel level cannot be determined from the indications and recorders on a main control board, the SPDS or the plant computer, the availability of other parameter values may be compromised as well.

SAFETY SYSTEM parameters listed in Table 1S-2 are monitored in the Control Room through a combination of hard control panel indicators as well as computer based information systems. The Plant Computer, which displays Safety Parameter Display System (SPDS) required information, serves as a redundant compensatory indicator which may be utilized in lieu of normal Control Room indicators (ref. 1, 2).

Significant transients are listed in Table 1S-3 and include response to automatic or manually initiated functions such as reactor trips, runbacks involving greater than or equal to 25% thermal power change, electrical load rejections of greater than 25% full electrical load or ECCS (SI) injection actuations.

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.

Basis Reference(s): **SA3.1**

1. 1BV UFSAR Section 7.5 Safety Related Display Information
2. 1DBD-05C Inadequate Core Cooling Monitoring System
3. NEI 99-01 Rev. 6 SA2

ATTACHMENT 1:

Unit 1 EAL Technical Bases

Category: S – System Malfunction **SU4.1**

Subcategory: 4 – RCS Activity

Initiating Condition: Reactor coolant activity greater than Technical Specification allowable limits

EAL:

SU4.1	**Unusual Event**
Letdown Monitor (RM-1CH-101A or B) **> 6.0E+04 cpm** (Note 10)	

Note 10: Mode 3 applicable **only** when RCS temperature is ≥ 500ºF

Mode Applicability:

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

Basis:

This EAL addresses reactor coolant letdown line radiation levels sensed by RM-1CH-101A or B in excess of Technical Specification allowable limits (ref. 1). This condition is a precursor to a more significant event and represents a potential degradation of the level of safety of the plant.

This reading is not applicable if letdown is isolated since the monitor isolates with letdown. As such, this reading would be useful only in those events in which safety injection and containment isolation do not actuate.

The RM-1CH-101 A/B calculated EAL value of 58,000 cpm (based on 21 μCi/gm dose equivelant I-131) has been rounded to 60,000 cpm based on accuracy of the analog instrument display capability. 60,000 cpm is the closest visually distinguishable reading to the derived EAL value. Instrument markings that bound the calculated EAL value are 40,000 and 60,000 cpm (ref. 2, 3).

Escalation of the emergency classification level would be via ICs FA1 or the Recognition Category R ICs.

Basis Reference(s):

1. Technical Specifications Section 3.4.16 RCS Specific Activity
2. ERS-JTL-99-005, Unit 1 Letdown Radiation Monitor (RM-CH-101) Alarm Setpoint, Rev 3
3. 1OM-53C.4.1.6.6 High Reactor Coolant System Activity
4. NEI 99-01 SU3

ATTACHMENT 1:

Unit 1 EAL Technical Bases

Category: S – System Malfunction **SU4.2**

Subcategory: 4 – RCS Activity

Initiating Condition: Reactor coolant activity greater than Technical Specification allowable limits

EAL:

SU4.2	**Unusual Event**
Reactor coolant activity **> 21 μCi/gm** dose equivelant I-131 (Note 10)	

Note 10: Mode 3 applicable only when RCS temperature is ≥ 500ºF

Mode Applicability:

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

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.

This EAL addresses reactor coolant samples exceeding Technical Specification LCOs 3.4.16.A and 3.4.16.B which are applicable in Modes 1, 2, and 3 with T_{avg} ≥ 500ºF (ref. 1, 2).

Escalation of the emergency classification level would be via ICs FA1 or the Recognition Category R ICs.

Basis Reference(s):

1. Technical Specifications Section 3.4.16
2. Technical Specifications Section B3.4.16
3. NEI 99-01 Rev. 6 SU3

Category: S – System Malfunction **SU5.1**

Subcategory: 5 – RCS Leakage

Initiating Condition: RCS leakage for 15 minutes or longer

EAL:

SU5.1	**Unusual Event**
RCS unidentified or pressure boundary leakage **> 10 gpm** for **≥ 15 min.** (Note 1)	

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

Mode Applicability:

1 - Power Operation, 2 - Startup, 3 - Hot Standby, 4 - Hot Shutdown

Basis:

This IC addresses RCS leakage which may be a precursor to a more significant event. In this case, RCS leakage has been detected and operators, following applicable procedures, have been unable to promptly isolate the leak. This condition is considered to be a potential degradation of the level of safety of the plant.

This EAL is focused on a loss of mass from the RCS due to “unidentified leakage", "pressure boundary leakage" or “identified leakage” (as these leakage types are defined in the plant Technical Specifications). This EAL thus applies 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 this EAL was selected because it is usually observable with normal Control Room indications. Lesser values typically require time-consuming calculations to determine (e.g., a mass balance calculation). This EAL 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).

Unidentified leakage and identified leakage are determined by performance of the RCS water inventory balance. Pressure boundary leakage would first appear as unidentified leakage and can only be positively identified by inspection (ref. 1, 2).

Technical Specifications (ref. 1) defines RCS leakage.

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

SU5.1

the Component Cooling Water, or systems that directly see RCS pressure outside containment such as Chemical & Volume Control System and Primary Sampling (ref. 3, 4).

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.

Basis Reference(s):

1. Technical Specifications Section 1.1 Definitions
2. Technical Specifications 3.4.13 RCS Operational Leakge
3. 1OM-53C.4.1.6.7 Excessive Primary Plant Leakage
4. 1OM-53A.1.ECA-1.2 LOCA Outside Containment
5. 1OM-53C.4.1.6.4 Steam Generator Tube Leakage
6. NEI 99-01 Rev. 6 SU4

Category: S – System Malfunction **SU5.2**

Subcategory: 5 – RCS Leakage

Initiating Condition: RCS leakage for 15 minutes or longer

EAL:

SU5.2	**Unusual Event**
RCS identified leakage **> 25 gpm** for **≥ 15 min.**	
(Note 1)	

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

Mode Applicability:

1 - Power Operation, 2 - Startup, 3 - Hot Standby, 4 - Hot Shutdown

Basis:

This IC addresses RCS leakage which may be a precursor to a more significant event. In this case, RCS leakage has been detected and operators, following applicable procedures, have been unable to promptly isolate the leak. This condition is considered to be a potential degradation of the level of safety of the plant.

This EAL is focused on a loss of mass from the RCS due to “unidentified leakage”, “pressure boundary leakage” or "identified leakage” (as these leakage types are defined in the plant Technical Specifications). This EAL thus applies 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 EAL was 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 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.

Unidentified leakage and identified leakage are determined by performance of the RCS water inventory balance. Pressure boundary leakage would first appear as unidentified leakage and can only be positively identified by inspection (ref. 1, 2).

Technical Specifications (ref. 1) defines RCS leakage.

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

SU5.2

the Component Cooling Water, or systems that directly see RCS pressure outside containment such as Chemical & Volume Control System and Primary Sampling (ref. 3, 4).

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

Basis Reference(s):

1. Technical Specifications Section 1.1 Definitions
2. Technical Specifications 3.4.13 RCS Operational Leakge
3. 1OM-53C.4.1.6.7 Excessive Primary Plant Leakage
4. 1OM-53A.1.ECA-1.2 LOCA Outside Containment
5. 1OM-53C.4.1.6.4 Steam Generator Tube Leakage
6. NEI 99-01 Rev. 6 SU4

ATTACHMENT 1:

Unit 1 EAL Technical Bases

Category: S – System Malfunction **SU5.3**

Subcategory: 5 – RCS Leakage

Initiating Condition: RCS leakage for 15 minutes or longer

EAL:

SU5.3	**Unusual Event**
UNISOLABLE leakage from the RCS to a location outside containment **> 25 gpm** for **≥ 15 min**. (Note 1)	

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

Mode Applicability:

1 - Power Operation, 2 - Startup, 3 - Hot Standby, 4 - Hot Shutdown

Basis:

This IC addresses RCS leakage which may be a precursor to a more significant event. In this case, RCS leakage has been detected and operators, following applicable procedures, have been unable to promptly isolate the leak. This condition is considered to be a potential degradation of the level of safety of the plant.

This EAL addresses a RCS mass loss caused by an UNISOLABLE leak through an interfacing system. This EAL thus applies to leakage to a location outside of containment.

The leak rate values for each EAL were selected because they are usually observable with normal Control Room indications. Lesser values typically require time-consuming calculations to determine (e.g., a mass balance calculation).

The release of mass from the RCS due to the as-designed/expected operation of a relief valve does not warrant an emergency classification. For PWRs, an emergency classification would be required if a mass loss is caused by a relief valve that is not functioning as designed/expected (e.g., a relief valve sticks open and the line flow cannot be isolated).

The 15-minute threshold duration allows sufficient time for prompt operator actions to isolate the leakage, if possible.

Unidentified leakage and identified leakage are determined by performance of the RCS water inventory balance. Pressure boundary leakage would first appear as unidentified leakage and can only be positively identified by inspection (ref. 1, 2).

Technical Specifications (ref. 1) defines RCS leakage.

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

SU5.3

the Component Cooling Water, or systems that directly see RCS pressure outside containment such as Chemical & Volume Control System and Primary Sampling (ref. 3, 4).

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

Basis Reference(s):

1. Technical Specifications Section 1.1 Definitions
2. Technical Specifications 3.4.13 RCS Operational Leakge
3. 1OM-53C.4.1.6.7 Excessive Primary Plant Leakage
4. 1OM-53A.1.ECA-1.2 LOCA Outside Containment
5. 1OM-53C.4.1.6.4 Steam Generator Tube Leakage
6. NEI 99-01 Rev. 6 SU4

ATTACHMENT 1:

Unit 1 EAL Technical Bases

Category: S – System Malfunction **SU6.1**

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 after **ANY** RPS setpoint is exceeded

AND

A subsequent automatic trip or manual trip action taken at the Control Room Benchboards (reactor trip breaker switch or pushbutton or tripping the turbine) is successful in shutting down the reactor (Note 8)

Note 8: A manual trip action is any operator action, or set of actions, which causes the control rods to be rapidly inserted into the core, and **does not** include manually driving in control rods or implementation of boron injection strategies.

Mode Applicability:

1 - Power Operation

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 Control Room Benchboards 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 Control Room Benchboards 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 Control Room Benchboards 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.

SU6.1

A manual action at the Control Room Benchboards 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 Control Room Benchboards”.

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 Control Room Benchboards 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.

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. (ref. 1, 2).

ATTACHMENT 1:

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SU6.1

For the purposes of emergency classification, successful manual trip actions are those which can be quickly performed from the Control Room Benchboards; reactor trip breaker switch or pushbutton or tripping the turbine. Reactor shutdown achieved by use of other trip actions specified in FR-S.1 Response to Nuclear Power Generation/ATWS (such as manually inserting control rods or emergency boration) do not constitute a successful manual trip (ref. 2).

Following any automatic RPS trip signal, E-0 (ref. 1) and FR-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 (ref. 2).

In the event that the operator identifies a reactor trip is IMMINENT and initiates a successful manual reactor trip before the automatic RPS trip setpoint is reached, no declaration is required. The successful manual trip of the reactor before it reaches its automatic trip setpoint or reactor trip signals caused by instrumentation channel failures do not lead to a potential fission product barrier loss. However, if subsequent manual reactor trip actions fail to reduce reactor power 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.

Basis Reference(s):

1. 1OM-53A.1.E-0 Reactor Trip or Safety Injection
2. 1OM-53A.1.FR-S.1 Response to Nuclear Power Generation - ATWS
3 NEI 99-01 Rev. 6 SU5

Category: S – System Malfunction **SU6.2**

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 after **ANY** manual trip action was initiated

AND

A subsequent automatic trip or manual trip action taken at the Control Room Benchboards (reactor trip breaker switch or pushbutton or tripping the turbine) is successful in shutting down the reactor (Note 8)

Note 8: A manual trip action is any operator action, or set of actions, which causes the control rods to be rapidly inserted into the core, and **does not** include manually driving in control rods or implementation of boron injection strategies.

Mode Applicability:

1 - Power Operation

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 Control Room Benchboards 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 Control Room Benchboards 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 Control Room Benchboards 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 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.

SU6.2

A manual action at the Control Room Benchboards 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 Control Room Benchboards ".

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 Control Room Benchboards 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.

This EAL addresses a failure of a manually initiated trip in the absence of having exceeded an automatic RTS trip setpoint and a subsequent automatic or manual trip is successful in shutting down the reactor. (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. (ref. 1, 2).

SU6.2

For the purposes of emergency classification, successful manual trip actions are those which can be quickly performed from the Control Room Benchboards; reactor trip breaker switch or pushbutton or tripping the turbine. Reactor shutdown achieved by use of other trip actions specified in FR-S.1 Response to Nuclear Power Generation/ATWS (such as manually inserting control rods or emergency boration) do not constitute a successful manual trip (ref. 2).

Following the failure of any manual trip signal, E-0 (ref. 1) and FR-S.1 (ref. 2) prescribe insertion of redundant manual trip signals to back up the RPS trip function and ensure reactor shutdown is achieved. Even if a subsequent automatic trip signal or the first subsequent manual trip signal inserts all control rods to the full-in position immediately after the initial failure of the manual trip, the lowest level of classification that must be declared is an Unusual Event (ref. 2).

Basis Reference(s):

1. 1OM-53A.1.E-0 Reactor Trip or Safety Injection
2. 1OM-53A.1.FR-S.1 Response to Nuclear Power Generation - ATWS
3. NEI 99-01 Rev. 6 SU5

Category: S – System Malfunction **SA6.1**

Subcategory: 6 – RPS Failure

Initiating Condition: Automatic or manual trip fails to shut down the reactor and subsequent manual actions taken at the Control Room Benchboards are not successful in shutting down the reactor

EAL:

SA6.1 Alert

An automatic or manual trip fails to shut down the reactor

AND

Manual trip actions taken at the Control Room Benchboards (reactor trip breaker switch or pushbutton or tripping the turbine) are **not** successful in shutting down the reactor (Note 8)

Note 8: A manual trip action is any operator action, or set of actions, which causes the control rods to be rapidly inserted into the core, and **does not** include manually driving in control rods or implementation of boron injection strategies.

Mode Applicability:

1 - Power Operation

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 Control Room Benchboards 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 Control Room Benchboards since this event entails a significant failure of the RPS.

A manual action at the Control Room Benchboards 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 Control Room Benchboards (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 Control Room Benchboards”.

SA6.1

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.

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 (ref. 1).

For the purposes of emergency classification, successful manual trip actions are those which can be quickly performed from the Control Room Benchboards; reactor trip breaker switch or pushbutton or tripping the turbine. Reactor shutdown achieved by use of other trip actions specified in FR-S.1 Response to Nuclear Power Generation/ATWS (such as manually inserting control rods or emergency boration) do not constitute a successful manual trip (ref. 2).

Basis Reference(s):

1. 1OM-53A.1.E-0 Reactor Trip or Safety Injection
2. 1OM-53A.1.FR-S.1 Response to Nuclear Power Generation - ATWS
3. NEI 99-01 Rev. 6 SA5

ATTACHMENT 1:

Unit 1 EAL Technical Bases

Category: S – System Malfunction **SS6.1**

Subcategory: 6 – RPS Failure

Initiating Condition: Inability to shut down the reactor causing a challenge to core cooling or RCS heat removal

EAL:

SS6.1 Site Area Emergency

An automatic or manual trip fails to shut down the reactor

AND

All actions to shut down the reactor are **not** successful

AND EITHER:

- Core Cooling RED Path conditions met
- Heat Sink RED Path conditions met

Mode Applicability:

1 - Power Operation

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.

SS6.1

This EAL addresses the following:

- Any automatic reactor trip signal followed by a manual trip that fails to shut down the reactor to an extent the reactor is producing energy in excess of the heat load for which the SAFETY SYSTEMS were designed (EAL SA6.1), and
- Indications that either core cooling is extremely challenged or heat removal is extremely challenged.

The combination of failure of both front line and backup protection systems to function in response to a plant transient, along with the continued production of heat, poses a direct threat to the Fuel Clad and RCS barriers.

Reactor shutdown achieved by use of FR-S.1 Response to Nuclear Power Generation/ATWS (such as manually insert control rods or emergency boration) are also credited as a successful manual trip provided reactor power can be reduced below 5% before indications of an extreme challenge to either core cooling or heat removal exist (ref. 1, 2).

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 core exit T/Cs are reading greater than or equal to 1200ºF or a loss of adequate subcooling with elevated core exit T/Cs and low RVLIS level (ref. 3).

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 based on inadequate steam generator level and feedwater flow (ref. 4).

Basis Reference(s):

1. 1OM-53A.1.E-0 Reactor Trip or Safety Injection
2. 1OM-53A.1.FR-S.1 Response to Nuclear Power Generation - ATWS
3. 1OM-53A.1.F-0.2 Core Cooling Status Tree
4. 1OM-53A.1.F-0.3 Heat Sink Status Tree
5. NEI 99-01 Rev. 6 SS5

ATTACHMENT 1:

Unit 1 EAL Technical Bases

Category: S – System Malfunction **SU7.1**

Subcategory: 7 – Loss of Communications

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

EAL:

SU7.1 Unusual Event

Loss of **ALL Table 1S-4** onsite communication methods

Table 1S-4 Communication Methods			
System	**Onsite**	**ORO**	**NRC**
Station Page Party Telephone System (Gaitronics)	X		
BVPS Industrial Radios	X	X	
Plant Telephone (PAX)	X	X	X
Commercial Telephones (hardwired & wireless)	X	X	X
Emergency Telephone System (ETS)			X

Mode Applicability:

1 - Power Operation, 2 - Startup, 3 - Hot Standby, 4 - Hot Shutdown

Basis:

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

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

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

SU7.1

This EAL is the hot condition equivalent of the cold condition EAL CU5.1.

Basis Reference(s):

1. BVPS Emergency Plan Section 7.6 Communications
2. NEI 99-01 Rev. 6 SU6

Category: S – System Malfunction **SU7.2**

Subcategory: 7 – Loss of Communications

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

EAL:

SU7.2 Unusual Event

Loss of **ALL Table 1S-4** offsite response organizations (ORO) communication methods

Table 1S-4 Communication Methods			
System	**Onsite**	**ORO**	**NRC**
Station Page Party Telephone System (Gaitronics)	X		
BVPS Industrial Radios	X	X	
Plant Telephone (PAX)	X	X	X
Commercial Telephones (hardwired & wireless)	X	X	X
Emergency Telephone System (ETS)			X

Mode Applicability:

1 - Power Operation, 2 - Startup, 3 - Hot Standby, 4 - Hot Shutdown

Basis:

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

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

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Unit 1 EAL Technical Bases

SU7.2

This EAL addresses a total loss of the communications methods used to notify all OROs of an emergency declaration. The OROs referred to here are the EOCs for the States of Pennsylvania, Ohio, West Virginia and counties of Beaver, Columbiana and Hancock..

This EAL is the hot condition equivalent of the cold condition EAL CU5.2.

Basis Reference(s):

1. BVPS Emergency Plan Section 7.6 Communications
2. NEI 99-01 Rev. 6 SU6

ATTACHMENT 1:

Unit 1 EAL Technical Bases

Category: S – System Malfunction **SU7.3**

Subcategory: 7 – Loss of Communications

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

EAL:

SU7.3 **Unusual Event**
Loss of **ALL Table 1S-4** NRC communication methods

Table 1S-4 Communication Methods			
System	**Onsite**	**ORO**	**NRC**
Station Page Party Telephone System (Gaitronics)	X		
BVPS Industrial Radios	X	X	
Plant Telephone (PAX)	X	X	X
Commercial Telephones (hardwired & wireless)	X	X	X
Emergency Telephone System (ETS)			X

Mode Applicability:

1 - Power Operation, 2 - Startup, 3 - Hot Standby, 4 - Hot Shutdown

Basis:

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

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

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

This EAL is the hot condition equivalent of the cold condition EAL CU5.3.

SU7.3

Basis Reference(s):

1. BVPS Emergency Plan Section 7.6 Communications
2. NEI 99-01 Rev. 6 SU6

ATTACHMENT 1:

Unit 1 EAL Technical Bases

Category: S – System Malfunction **SU8.1**

Subcategory: 8 – Containment Failure

Initiating Condition: Failure to isolate containment or loss of containment pressure control.

EAL:

SU8.1	**Unusual Event**

ANY penetration is not isolated within **15 min.** of a VALID containment isolation signal

OR

Containment pressure **> 11 psig AND <** one full train of depressurization equipment operating per design for **≥ 15 min.**
(Note 1)

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

Mode Applicability:

1 - Power Operation, 2 - Startup, 3 - Hot Standby, 4 - Hot Shutdown

Basis:

This IC 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.

SU8.1

Each unit has a containment pressure quench spray system with two 100% capacity trains. These pumps take suction from the RWST and discharge to the spray header. The quench spray system starts on a CIB at the start of a LOCA accident.

The recirculation spray system has four 50% capacity subsystems that consist of a pump and a cooler. The recirculation spray pump takes suction from the containment sump and discharges through a cooler to the spray header. The recirculation spray system does not start during a LOCA until there is low level in the RWST to verify the sump has adequate water inventory. When the RWST level goes very low the quench spray pumps are secured.

A very short period of time could exist where the quench spray system and the recirculation spray system pumps could both be running. Normally it is either the quench spray or the recirculation spray running.

One train of QS System and one train of RS System comprise one full train of depressurization equipment as designed (ref. 1).

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.

Basis Reference(s):

1. BV1 UFSAR Section 6.4 Containment Depressurization System
2. NEI 99-01 Rev. 6 SU7

Category: S – System Malfunction **SA9.1**

Subcategory: 9 – Hazardous Event Affecting Safety Systems

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

EAL:

SA9.1 Alert

The occurrence of **ANY Table 1S-5** hazardous event

AND

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

AND EITHER:

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

(Notes 15, 16)

Note 15: If the affected SAFETY SYSTEM train was already inoperable or out of service before the hazardous event occurred, then this emergency classification is not warranted.

Note 16: If the hazardous event only resulted in VISIBLE DAMAGE, with no indications of degraded performance to at least one train of a SAFETY SYSTEM, then this emergency classification is not warranted.

Table 1S-5 Hazardous Events
• Seismic event (earthquake) • Internal or external flooding event • High winds or tornado strike • FIRE • EXPLOSION • Other events with similar hazard characteristics as determined by the Shift Manager

Mode Applicability:

1 - Power Operation, 2 - Startup, 3 - Hot Standby, 4 - Hot Shutdown

SA9.1

Basis:

This IC addresses a hazardous event that causes damage to SAFETY SYSTEMS needed for the current operating mode. In order to provide the appropriate context for consideration of an ALERT classification, the hazardous event must have caused indications of degraded SAFETY SYSTEM performance in one train, and there must be either indications of performance issues with the second SAFETY SYSTEM train or VISIBLE DAMAGE to the second train such that the potential exists for this second SAFETY SYSTEM train to have performance issues. In other words, in order for this EAL to be classified, the hazardous event must occur, at least one SAFETY SYSTEM train must have indications of degraded performance, and the second SAFETY SYSTEM train must have indications of degraded performance or VISIBLE DAMAGE such that the potential exists for performance issues. Note that this second SAFETY SYSTEM train is from the same SAFETY SYSTEM that has indications of degraded performance for the first AND EITHER statement of this EAL; commercial nuclear power plants are designed to be able to support single system issues without compromising public health and safety from radiological events.

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

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

- The Operating Basis Earthquake is 0.06g. It is the conservatively determined earthquake and associated ground motion that might reasonably or probably be expected to occur at the nuclear plant site. Control Room alarm indication of an earthquake greater than OBE is indicated on the seismic monitoring system cabinet 1ER-CCC-1. 1/2OM-53C.4A.75.3 Acts of Nature - Seismic provides the guidance for determining if the OBE earthquake threshold is exceeded and any required response actions (ref. 1). The signficance of seismic events are discussed under EAL HU2.1.
- Internal flooding may be caused by events such as component failures, equipment misalignment, or outage activity mishaps (ref. 2).
- External flooding may be due to river level (ref. 3, 4).

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SA9.1

- Seismic Category I structures are analyzed to withstand a sustained, design wind velocity of at least 80 mph. (ref. 5, 6).
- Areas containing functions and systems required for safe shutdown of the plant are identified by fire area (ref. 7, 8).

Escalation of the emergency classification level would be via IC FS1 or RS1.

Basis Reference(s):

1. 1/2OM-53C-4A.75.3 Acts of Nature Seismic Event
2. BV1 Calculation DMC-2169 BVPS-1 PAB Flood
3. 1/2OM-53C.4A.75.2 Acts of Nature - Flood
4. 1/2OM-53C.4A.75.4 Acts of Nature – Dam Failure
5. 1/2OM-53C.4A.75.1 Acts of Nature – Severe Weather
6. BV1 UFSAR Section 2.7.1.1 Seismic Category I Structures
7. BV1 UFSAR Section 2.7.2 Tornado Model
8. BV1 UFSAR Table B.1-1 Structures and Systems Requiring Design for Seismic Loading
9. BV1 UFSAR Table B.3-1 NSS Fluid Systems Component Seismic Category List
10. NEI 99-01 Rev. 6 SA9

ATTACHMENT 1:

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Category F – Fission Product Barrier Degradation

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

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

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

B. Reactor Coolant System (RCS): The RCS Barrier includes the RCS primary side and its connections up to and including the pressurizer safety and relief valves, and other connections up to and including the primary isolation valves.

C. Containment (CT): The Containment Barrier includes the containment building, connections up to and including the outermost containment isolation valves. This barrier also includes the main steam, feedwater, and blowdown line extensions outside the containment building up to and including the outermost secondary side isolation valve. Containment Barrier thresholds are used as criteria for escalation of the ECL from Alert to a Site Area Emergency or a General Emergency.

The EALs in this category require evaluation of the loss and potential loss thresholds listed in the fission product barrier matrix of Table 1F-1 (Attachment 2). “Loss” and “Potential Loss” signify the relative damage and threat of damage to the barrier. “Loss” means the barrier no longer assures containment of radioactive materials. “Potential Loss” means integrity of the barrier is threatened and could be lost if conditions continue to degrade. The number of barriers that are lost or potentially lost and the following criteria determine the appropriate emergency classification level:

Alert:
ANY loss or ANY potential loss of EITHER Fuel Clad or RCS

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

General Emergency:
Loss of ANY two barriers AND loss or potential loss of third barrier

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

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

- Unusual Event ICs associated with RCS and Fuel Clad Barriers are addressed under System Malfunction ICs.
- For accident conditions involving a radiological release, evaluation of the FISSION PRODUCT BARRIER THRESHOLDS will need to be performed in conjunction with dose assessments to ensure correct and timely escalation of the emergency classification. For example, an evaluation of the FISSION PRODUCT BARRIER THRESHOLDS may result in a Site Area Emergency classification while a dose assessment may indicate that an EAL for General Emergency IC RG1 has been exceeded.
- The FISSION PRODUCT BARRIER THRESHOLDS specified within a scheme reflect plant-specific BVPS design and operating characteristics.
- As used in this category, the term RCS leakage encompasses not just those types defined in Technical Specifications but also includes the loss of RCS mass to any location– inside the containment, an interfacing system, or outside of the containment. The release of liquid or steam mass from the RCS due to the as-designed/expected operation of a relief valve is not considered RCS leakage.
- At the Site Area Emergency level, EAL users should maintain cognizance of how far present conditions are from meeting a threshold that would require a General Emergency declaration. For example, if the Fuel Clad and RCS fission product barriers were both lost, then there should be frequent assessments of containment radioactive inventory and integrity. Alternatively, if both the Fuel Clad and RCS fission product barriers were potentially lost, the Emergency Director would have more assurance that there was no immediate need to escalate to a General Emergency.

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Category: Fission Product Barrier Degradation **FA1.1**

Subcategory: N/A

Initiating Condition: **Any** loss or **any** potential loss of either Fuel Clad or RCS

EAL:

FA1.1	**Alert**
ANY Loss or **ANY** Potential Loss of **EITHER** Fuel Clad or RCS (**Table 1F-1**)	

Mode Applicability:

1 - Power Operation, 2 - Startup, 3 - Hot Standby, 4 - Hot Shutdown

Basis:

Fuel Clad, RCS and Containment comprise the fission product barriers. Table 1F-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

Basis Reference(s):

1. NEI 99-01 Rev. 6 FA1

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Unit 1 EAL Technical Bases

Category: Fission Product Barrier Degradation **FS1.1**

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 1F-1**)	

Mode Applicability:

1 - Power Operation, 2 - Startup, 3 - Hot Standby, 4 - Hot Shutdown

Basis:

Fuel Clad, RCS and Containment comprise the fission product barriers. Table 1F-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 Director would have greater assurance that escalation to a General Emergency is less IMMINENT.

Basis Reference(s):

1. NEI 99-01 Rev. 6 FS1

ATTACHMENT 1:

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Category: Fission Product Barrier Degradation **FG1.1**

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 1F-1**)	

Mode Applicability:

1 - Power Operation, 2 - Startup, 3 - Hot Standby, 4 - Hot Shutdown

Basis:

Fuel Clad, RCS and Containment comprise the fission product barriers. Table 1F-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

Basis Reference(s):

1. NEI 99-01 Rev. 6 FG1

ATTACHMENT 2: Unit 1 Fission Product Barrier Loss/Potential Loss Matrix and Bases

Introduction

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

The first column of the table (to the left of the Fuel Clad Loss column) lists the categories (types) of FISSION PRODUCT BARRIER THRESHOLDS. The fission product barrier categories are:

A. RCS or SG Tube Leakage

B. Inadequate Heat removal

C. CT Radiation / RCS Activity

D. CT Integrity or Bypass

E. Emergency Director Judgment

Each category occupies a row in Table 1F-1 thus forming a matrix defined by the categories. The intersection of each row with each Loss/Potential Loss column forms a cell in which one or more FISSION PRODUCT BARRIER THRESHOLDS appear. If NEI 99-01 does not define a threshold for a barrier Loss/Potential Loss, the word “None” is entered in the cell.

Thresholds are assigned sequential numbers within each Loss and Potential Loss column beginning with number one. In this manner, a threshold can be identified by its category title and number. For example, the first Fuel Clad barrier Loss in Category A would be assigned “FC Loss A.1,” the third Containment barrier Potential Loss in Category C would be assigned “CT P-Loss C.3,” etc.

If a cell in Table 1F-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 1F-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 1F-1, locates the likely category and then reads across the fission product barrier Loss and Potential Loss thresholds in that category to determine if a threshold has been exceeded. If a threshold has not been exceeded, the EAL-user proceeds to the next likely category and continues review of the thresholds in the new category

If the EAL-user determines that any threshold has been exceeded, by definition, the barrier is lost or potentially lost – even if multiple thresholds in the same barrier column are exceeded, only that one barrier is lost or potentially lost. The EAL-user must examine each of the three fission product barriers to determine if other barrier thresholds in the category are lost or potentially lost. For example, if containment radiation is sufficiently high, a Loss of the Fuel Clad and RCS barriers and a Potential Loss of the Containment barrier can occur. Barrier Losses and Potential Losses are then applied to the algorithms given in EALs FG1.1, FS1.1, and FA1.1 to determine the appropriate emergency classification.

In the remainder of this Attachment, the Fuel Clad barrier threshold bases appear first, followed by the RCS barrier and finally the Containment barrier threshold bases. In each barrier, the bases are given according category Loss followed by category Potential Loss beginning with Category A, then B,…, E.

ATTACHMENT 2: Unit 1 Fission Product Barrier Loss/Potential Loss Matrix and Bases

Table 1F-1 Fission Product Barrier Threshold Matrix

	Fuel Clad (FC) Barrier		**Reactor Coolant System (RC) Barrier**		**Containment (CT) Barrier**	
Category	**Loss**	**Potential Loss**	**Loss**	**Potential Loss**	**Loss**	**Potential Loss**
A **RCS or SG Tube Leakage**	None	None	1. An automatic or manual ECCS (SI) actuation required by **EITHER:** • UNISOLABLE RCS leakage • SG tube RUPTURE	1. Operation of a standby charging pump is required by **EITHER:** • UNISOLABLE RCS leakage • SG tube leakage **OR** 2. RCS Integrity-RED Path conditions met	1. A leaking or RUPTURED SG is FAULTED outside of containment	None
B **Inadequate Heat Removal**	1. Core Cooling-RED Path conditions met	1. Core Cooling-ORANGE Path conditions met **OR** 2. Heat Sink-RED Path conditions met **AND** Heat sink is required	None	1. Heat Sink-RED Path conditions met **AND** Heat sink is required	None	1. Core Cooling-RED Path conditions met **AND** Restoration procedures **not** effective within **15 min.** (Note 1)
C **CT Radiation / RCS Activity**	1. Containment Radiation Monitor **> Table 1F-2, "FC Loss"** **OR** 2. Dose equivalent I-131 coolant activity **> 300 µCi/gm**	None	1. Containment Radiation Monitor **> Table 1F-2, "RC Loss"**	None	None	1. Containment Radiation Monitor **> Table 1F-2, "CT Potential Loss"**
D **CT Integrity or Bypass**	None	None	None	None	1. Containment isolation is required **AND EITHER:** • Containment integrity has been lost based on Emergency Director judgment • UNISOLABLE pathway from Containment to the environment exists **OR** 2. Indications of RCS leakage outside of Containment	1. Containment-RED Path conditions met **OR** 2. Containment hydrogen concentration **> 4%** **OR** 3. Containment pressure **> 11 psig AND <** one full train of depressurization equipment operating per design for **≥ 15 min.** (Note 1)
E **ED Judgment**	1. **ANY** condition in the opinion of the Emergency Director that indicates Loss of the Fuel Clad Barrier	1. **ANY** condition in the opinion of the Emergency Director that indicates Potential Loss of the Fuel Clad Barrier	1. **ANY** condition in the opinion of the Emergency Director that indicates Loss of the RCS Barrier	1. **ANY** condition in the opinion of the Emergency Director that indicates Potential Loss of the RCS Barrier	1. **ANY** condition in the opinion of the Emergency Director that indicates Loss of the Containment Barrier	1. **ANY** condition in the opinion of the Emergency Director that indicates Potential Loss of the Containment Barrier

Barrier: Fuel Clad **FC.A**

Category: A. RCS or SG Tube Leakage

Degradation Threat: Loss

Threshold:

None

Barrier: Fuel Clad **FC.A**

Category: A. RCS or SG Tube Leakage

Degradation Threat: Potential Loss

Threshold:

None

Barrier: Fuel Clad **FC.B**

Category: B. Inadequate Heat Removal

Degradation Threat: Loss

Threshold:

1. Core Cooling-RED Path conditions met

Basis:

This reading indicates temperatures within the core are sufficient to cause significant superheating of reactor coolant.

Critical Safety Function Status Tree (CSFST) Core Cooling-RED Path indicates significant core exit superheating and core uncovery. The CSFSTs are normally monitored using the Safety Parameter Display System (SPDS) display on the Plant Computer (ref. 1).

Basis Reference(s):

1. 1OM-53A.1.F-0.2 Core Cooling Status Tree
2. NEI 99-01 Rev. 6 Inadequate Heat Removal Fuel Clad Loss 2.A

Barrier: Fuel Clad **FC.B**

Category: B. Inadequate Heat Removal

Degradation Threat: Potential Loss

Threshold:

1. Core Cooling-ORANGE Path conditions met

Basis:

This reading indicates temperatures within the core are sufficient to allow the onset of heat-induced cladding damage.

Critical Safety Function Status Tree (CSFST) Core Cooling-ORANGE path indicates indicates subcooling has been lost and that some fuel clad damage may potentially occur. The CSFSTs are normally monitored using the Safety Parameter Display System (SPDS) display on the Plant Computer (ref. 1, 2).

Basis Reference(s):

1. 1OM-53A.1.F-0.2 Core Cooling Status Tree
2. NEI 99-01 Rev. 6 Inadequate Heat Removal Fuel Clad Loss 2.A

Barrier: Fuel Clad **FC.B**

Category: B. Inadequate Heat Removal

Degradation Threat: Potential Loss

Threshold:

2. Heat Sink-RED Path conditions met

   AND

   Heat sink is required

Basis:

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.

Meeting this threshold results in a Site Area Emergency because this threshold is identical to RCS Barrier Potential Loss threshold RC.A.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.

Critical Safety Function Status Tree (CSFST) Heat Sink-RED Path indicates the heat sink function is under extreme challenge and that some fuel clad damage may potentially occur (ref. 1).

The CSFSTs are normally monitored using the Safety Parameter Display System (SPDS) display on the Plant Computer (ref. 1).

The phrase “and heat sink required” precludes the need for classification for conditions in which RCS pressure is less than SG pressure or Heat Sink-RED path entry was created through operator action directed by an ERG. For example, FR-H.1 is entered from CSFST FR-H.1 Heat Sink-Red. Step 1 tells the operator to determine if heat sink is required by checking that RCS pressure is greater than any non-faulted SG pressure and RCS temperature is greater than 350ºF. If these conditions exist, Heat Sink is required. Otherwise, the operator is to either return to the procedure and step in effect and place RHR in service for heat removal. For large LOCA events inside the Containment, the SGs are moot because heat removal through the containment heat removal systems takes place. Therefore, Heat Sink Red should not be required and, should not be assessed for EAL classification because a LOCA event alone should not require higher than an Alert classification. (ref. 2).

FC.B

Basis Reference(s):

1. 1OM-53A.1.F-0.3 Heat Sink Status Tree
2. 1OM-53A.1.FR-H.1 Response to Loss of Secondary Heat Sink
3. NEI 99-01 Rev. 6 Inadequate Heat Removal Fuel Clad Loss 2.B

Barrier: Fuel Clad **FC.C**

Category: C. CT Radiation / RCS Activity

Degradation Threat: Loss

Threshold:

1. Containment Radiation Monitor **> Table 1F-2, "FC Loss"**

Table 1F-2 Containment Radiation – R/hr (RM-1RM-219A or B)			
Time After S/D (Hrs.)	RC Loss (R/hr)	FC Loss (R/hr)	CT Potential Loss (R/hr)
0-1	8	520	10,000
>1-2	8	360	7,100
>2-8	8	150	2,900
>8-16	8	93	1,800
>16-48	8	46	910

Basis:

The radiation monitor reading corresponds to an instantaneous release of all reactor coolant mass into the containment, assuming that reactor coolant activity equals 300 µCi/gm dose equivalent I-131. Reactor coolant activity above this level is greater than that expected for iodine spikes and corresponds to an approximate range of 2% to 5% fuel clad damage. Since this condition indicates that a significant amount of fuel clad damage has occurred, it represents a loss of the Fuel Clad Barrier.

The radiation monitor reading in this threshold is higher than that specified for RCS Barrier Loss threshold RC.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.

The gamma dose rate resulting from a postulated loss of coolant accident (LOCA) is monitored by the containment high range monitors, RM-1RM-219A & B and are located inside containment. The detector range is approximately 1 to 1E7 R/hr. Radiation Monitors RM-1RM-219A & B provide a diverse means of measuring the containment for high level gamma radiation (ref. 1).

FC.C

The Table 1F-2 values, column FC Loss represents, based on the referenced calculation, the expected containment high range radiation monitor (RM-1RM-219B) response based on a LOCA, for periods of 1, 2, 8 and 16 hours after shutdown with coolant activity of 300 Ci/gm DEI-131 or ~1% clad failure (ref. 1).

The value is derived as follows:

ERS-SMM-11-002 Attachment 2 CRM Readings vs. Time for 1% Clad Damage on RM-1RM-219B for 1, 2, 8 and 16 hours after shutdown (rounded) (ref. 1).

Basis Reference(s):

1. ERS-SMM-11-002, Containment Radiation Monitor Readings Following Clad Damage (FC2 Loss, FC7 Loss, RC2 Loss and CT2 Potential Loss)
2. NEI 99-01 Rev. 6 CMT Radiation / RCS Activity Fuel Clad Loss 3.A

Barrier: Fuel Clad **FC.C**

Category: C. CT Radiation / RCS Activity

Degradation Threat: Loss

Threshold:

2. Dose equivalent I-131 coolant activity **> 300 μCi/gm**

Basis:

This threshold indicates that RCS radioactivity concentration is greater than 300 μCi/gm dose equivalent I-131. Reactor coolant activity above this level is greater than that expected for iodine spikes and corresponds generically to an approximate range of 2% to 5% fuel clad damage (1% at BVPS) (ref. 1). Since this condition indicates that a significant amount of fuel clad damage has occurred, it represents a loss of the Fuel Clad Barrier.

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

Basis Reference(s):

1. ERS-SMM-11-002, Containment Radiation Monitor Readings Following Clad Damage (FC2 Loss, FC7 Loss, RC2 Loss and CT2 Potential Loss)
2. NEI 99-01 Rev. 6 CMT Radiation / RCS Activity Fuel Clad Loss 3.B

Barrier: Fuel Clad **FC.C**

Category: C. CT Radiation / RCS Activity

Degradation Threat: Potential Loss

Threshold:

None

Barrier: Fuel Clad **FC.D**

Category: D. CT Integrity or Bypass

Degradation Threat: Loss

Threshold:

None

Barrier: Fuel Clad **FC.D**

Category: D. CT Integrity or Bypass

Degradation Threat: Potential Loss

Threshold:

None

Barrier: Fuel Clad **FC.E**

Category: E. Emergency Director Judgment

Degradation Threat: Loss

Threshold:

1. **ANY** condition in the opinion of the Emergency Director that indicates Loss of the Fuel Clad Barrier

Basis:

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

Basis Reference(s):

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

Barrier: Fuel Clad **FC.E**

Category: E. Emergency Director Judgment

Degradation Threat: Potential Loss

Threshold:

1. **ANY** condition in the opinion of the Emergency Director that indicates Potential Loss of the Fuel Clad Barrier

Basis:

This threshold addresses any other factors that are to be used by the Emergency Director in determining whether the Fuel Clad Barrier is potentially lost. The Emergency Director should also consider whether or not to declare the barrier potentially lost in the event that barrier status cannot be monitored.

Basis Reference(s):

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

Barrier: Reactor Coolant System **RC.A**

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

Basis:

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 CT.A.1 will also be met.

ECCS (SI) actuation is caused by (ref. 1):

- Pressurizer low pressure
- Steamline low pressure
- Containment high pressure

Basis Reference(s):

1. 1OM-53A.1.E-0 Reactor Trip or Safety Injection
2. 1OM-53A.1.E-3 Steam Generator Tube Rupture
3. BVRM-OPS-0012 BV-1 EOP Setpoint Document
4. NEI 99-01 Rev. 6 RCS or SG Tube Leakage Reactor Coolant System Loss 1.A

Barrier: Reactor Coolant System **RC.A**

Category: A. RCS or SG Tube Leakage

Degradation Threat: Potential Loss

Threshold:

1. Operation of a standby charging pump is required by **EITHER:**
   - UNISOLABLE RCS leakage
   - SG tube leakage

Basis:

This threshold is based on an UNISOLABLE RCS leak that results in the inability to maintain pressurizer level within specified limits by operation of a normally used charging (makeup) pump, but an ECCS (SI) actuation has not occurred. The threshold is met when an operating procedure, or operating crew supervision, directs that a standby charging (makeup) pump be placed in service to restore and maintain pressurizer level.

This threshold is applicable to unidentified and pressure boundary leakage, as well as identified leakage. It is also applicable to UNISOLABLE RCS leakage through an interfacing system. The mass loss may be into any location – inside containment, to the secondary-side (i.e., steam generator tube leakage) or outside of containment.

If a leaking steam generator is also FAULTED outside of containment, the declaration escalates to a Site Area Emergency since the Containment Barrier Loss threshold CT.A.1 will also be met.

The Chemical and Volume Control System (CVCS) includes three single speed charging pumps that take suction from the volume control tank and return the cooled, purified reactor coolant to the RCS. The centrifugal charging pumps in the CVCS also serve as the high-head safety injection pumps in the Emergency Core Cooling System. The capacity of each centrifugal pump is ~129 gpm (including bypass flow). A second charging pump being required is indicative of a substantial RCS leak (ref. 1, 2).

Basis Reference(s):

1. BV1 UFSAR 9.1 Chemical and Volume Control System
2. BV1 UFSAR Table 9.1-2 CVCS Performance Requirements
3. NEI 99-01 Rev. 6 RCS or SG Tube Leakage Reactor Coolant System Potential Loss 1.A

Barrier: Reactor Coolant System **RC.A**

Category: A. RCS or SG Tube Leakage

Degradation Threat: Potential Loss

Threshold:

2. RCS Integrity-RED Path conditions met

Basis:

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).

Critical Safety Function Status Tree (CSFST) RCS Integrity-RED path indicates the RCS barrier is under significant challenge (ref. 1). The CSFSTs are normally monitored using the Safety Parameter Display System (SPDS) display on the Plant Computer.

Basis Reference(s):

1. 1OM-53A.1.F-0.4 Vessel Integrity Status Tree
2. NEI 99-01 Rev. 6 RCS or SG Tube Leakage Reactor Coolant System Potential Loss 1.B

Barrier: Reactor Coolant System **RC.B**

Category: B. Inadequate Heat Removal

Degradation Threat: Loss

Threshold:

None

Barrier: Reactor Coolant System **RC.B**

Category: B. Inadequate Heat Removal

Degradation Threat: Potential Loss

Threshold:

1. Heat Sink-RED path conditions met

   AND

   Heat sink is required

Basis:

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 FC.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.

In combination with FC Potential Loss FC.B.2, meeting this threshold results in a Site Area Emergency.

Critical Safety Function Status Tree (CSFST) Heat Sink-RED Path indicates the heat sink function is under extreme challenge and that some fuel clad damage may potentially occur (ref. 1).

The CSFSTs are normally monitored using the Safety Parameter Display System (SPDS) display on the Plant Computer (ref. 1).

RC.B

The phrase "and heat sink required" precludes the need for classification for conditions in which RCS pressure is less than SG pressure or Heat Sink-RED path entry was created through operator action directed by an ERG. For example, FRH-0.1 is entered from CSFST Heat Sink-Red. Step 1 tells the operator to determine if heat sink is required by checking that RCS pressure is greater than any non-faulted SG pressure and RCS temperature is greater than 350ºF. If these conditions exist, Heat Sink is required. Otherwise, the operator is to either return to the procedure and step in effect and place RHR in service for heat removal. For large LOCA events inside the Containment, the SGs are moot because heat removal through the containment heat removal systems takes place. Therefore, Heat Sink Red should not be required and, should not be assessed for EAL classification because a LOCA event alone should not require higher than an Alert classification. (ref. 2).

Basis Reference(s):

1. 1OM-53A.1.F-0.3 Heat Sink Status Tree
2. 1OM-53A.1.FR-H.1 Response to Loss of Secondary Heat Sink
3. NEI 99-01 Rev. 6 Inadequate Heat Removal RCS Loss 2.B

Barrier: Reactor Coolant System **RC.C**

Category: C. CT Radiation/ RCS Activity

Degradation Threat: Loss

Threshold:

1. Containment Radiation Monitor **> Table 1F-2, "RC Loss"**

Table 1F-2 Containment Radiation – R/hr (RM-1RM-219A or B)			
Time After S/D (Hrs.)	RC Loss (R/hr)	FC Loss (R/hr)	CT Potential Loss (R/hr)
0-1	8	520	10,000
>1-2	8	360	7,100
>2-8	8	150	2,900
>8-16	8	93	1,800
>16-48	8	46	910

Basis:

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 FC.C.1 since it indicates a loss of the RCS Barrier only.

The gamma dose rate resulting from a postulated loss of coolant accident (LOCA) is monitored by the containment high range monitors, RM-1RM-219A & B and are located inside containment. The detector range is approximately 1 to 1E7 R/hr. Radiation Monitors RM-1RM-219A & B provide a diverse means of measuring the containment for high level gamma radiation (ref. 1).

The Table 1F-2 values, column RC Loss represents, based on the referenced calculation, the expected containment high range radiation monitor (RM-1RM-219B) response based on a LOCA, with coolant activity corresponding to Technical Specification coolant activity of 21 μCi/gm DEI-131; 7.9 R/hr rounded to 8 R/hr (ref. 1).

RC.C

Basis Reference(s):

1. ERS-SMM-11-002, Containment Radiation Monitor Readings Following Clad Damage (FC2 Loss, FC7 Loss, RC2 Loss and CT2 Potential Loss)
2. NEI 99-01 Rev. 6 CMT Radiation / RCS Activity RCS Loss 3.A

Barrier: Reactor Coolant System **RC.C**

Category: C. CT Radiation/ RCS Activity

Degradation Threat: Potential Loss

Threshold:

None

Barrier: Reactor Coolant System **RC.D**

Category: D. CT Integrity or Bypass

Degradation Threat: Loss

Threshold:

None

Barrier: Reactor Coolant System **RC.D**

Category: D. CT Integrity or Bypass

Degradation Threat: Potential Loss

Threshold:

None

Barrier: Reactor Coolant System **RC.E**

Category: E. Emergency Director Judgment

Degradation Threat: Loss

Threshold:

1. **ANY** condition in the opinion of the Emergency Director that indicates Loss of the RCS Barrier

Basis:

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

Basis Reference(s):

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

Barrier: Reactor Coolant System **RC.E**

Category: E. Emergency Director Judgment

Degradation Threat: Potential Loss

Threshold:

1. **ANY** condition in the opinion of the Emergency Director that indicates Potential Loss of the RCS barrier

Basis:

This threshold addresses any other factors that may be used by the Emergency Director in determining whether the RCS Barrier is potentially lost. The Emergency Director should also consider whether or not to declare the barrier potentially lost in the event that barrier status cannot be monitored.

Basis Reference(s):

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

Barrier: Containment **CT.A**

Category: A. RCS or SG Tube Leakage

Degradation Threat: Loss

Threshold:

1. A leaking or RUPTURED SG is FAULTED outside of containment

Basis:

This threshold addresses a leaking or RUPTURED Steam Generator (SG) that is also FAULTED outside of containment. The condition of the SG, whether leaking or RUPTURED, is determined in accordance with the thresholds for RCS Barrier Potential Loss RC.A.1 and Loss RC.A.1, respectively. This condition represents a bypass of the containment barrier.

FAULTED is a defined term within the NEI 99-01 methodology; this determination is not necessarily dependent upon entry into, or diagnostic steps within, an EOP. For example, if the pressure in a steam generator is decreasing uncontrollably (part of the FAULTED definition) and the faulted steam generator isolation procedure is not entered because EOP user rules are dictating implementation of another procedure to address a higher priority condition, the steam generator is still considered FAULTED for emergency classification purposes.

The FAULTED criterion establishes an appropriate lower bound on the size of a steam release that may require an emergency classification. Steam releases of this size are readily observable with normal Control Room indications. The lower bound for this aspect of the containment barrier is analogous to the lower bound criteria specified in IC SU4 for the fuel clad barrier (i.e., RCS activity values) and IC SU5 for the RCS barrier (i.e., RCS leak rate values).

This threshold also applies to prolonged steam releases necessitated by operational considerations such as the forced steaming of a leaking or RUPTURED steam generator directly to atmosphere to cooldown the plant, or to drive an auxiliary (emergency) feed water pump. These types of conditions will result in a significant and sustained release of radioactive steam to the environment (and are thus similar to a FAULTED condition). The inability to isolate the steam flow without an adverse effect on plant cooldown meets the intent of a loss of containment.

Steam releases associated with the expected operation of a SG power operated relief valve or safety relief valve do not meet the intent of this threshold. Such releases may occur intermittently for a short period of time following a reactor trip as operators process through emergency operating procedures to bring the plant to a stable condition and prepare to initiate a plant cooldown. Steam releases associated with the unexpected operation of a valve (e.g., a stuck-open safety valve) do meet this threshold.

Following an SG tube leak or RUPTURE, there may be minor radiological releases through a secondary-side system component (e.g., air ejectors, gland seal exhausters, valve packing,

CT.A

etc.). These types of releases do not constitute a loss or potential loss of containment but should be evaluated using the Recognition Category R ICs.

The ECLs resulting from primary-to-secondary leakage, with or without a steam release from the FAULTED SG, are summarized below.

	Affected SG is FAULTED Outside of Containment?	
P-to-S Leak Rate	**Yes**	**No**
Less than or equal to 25 gpm	No classification	No classification
Greater than 25 gpm	Unusual Event per SU5.2	Unusual Event per SU5.2
Requires operation of a standby charging (makeup) pump (*RCS Barrier Potential Loss*)	Site Area Emergency per FS1.1	Alert per FA1.1
Requires an automatic or manual ECCS (SI) actuation (*RCS Barrier Loss*)	Site Area Emergency per FS1.1	Alert per FA1.1

Basis Reference(s):

1. 1OM-53A.1.E-3 Steam Generator Tube Rupture
2. 1OM-53A.1.ECA-3.1 SGTR with Loss of Reactor Coolant – Subcooled Recovery Desired
3. NEI 99-01 Rev. 6 RCS or SG Tube Leakage Containment Loss 1.A

Barrier: Containment **CT.A**

Category: A. RCS or SG Tube Leakage

Degradation Threat: Potential Loss

Threshold:

None

Barrier: Containment **CT.B**

Category: B. Inadequate Heat Removal

Degradation Threat: Loss

Threshold:

None

Barrier: Containment **CT.B**

Category: B. Inadequate Heat Removal

Degradation Threat: Potential Loss

Threshold:

1. Core Cooling-RED Path conditions met

   AND

   Restoration procedures **not** effective within **15 min.** (Note 1)

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

Basis:

This condition represents an IMMINENT core melt sequence which, if not corrected, could lead to vessel failure and an increased potential for containment failure. For this condition to occur, there must already have been a loss of the RCS Barrier and the Fuel Clad Barrier. If implementation of a procedure(s) to restore adequate core cooling is not effective (successful) within 15 minutes, it is assumed that the event trajectory will likely lead to core melting and a subsequent challenge of the Containment Barrier.

The restoration procedure is considered “effective” if core exit thermocouple readings are decreasing and/or if reactor vessel level is increasing. Whether or not the procedure(s) will be effective should be apparent within 15 minutes. The Emergency Director should escalate the emergency classification level as soon as it is determined that the procedure(s) will not be effective.

Severe accident analyses (e.g., NUREG-1150) have concluded that function restoration procedures can arrest core degradation in a significant fraction of core damage scenarios, and that the likelihood of containment failure is very small in these events. Given this, it is appropriate to provide 15 minutes beyond the required entry point to determine if procedural actions can reverse the core melt sequence.

Critical Safety Function Status Tree (CSFST) Core Cooling-RED path indicates significant core exit superheating and core uncovery. The CSFSTs are normally monitored using the Safety Parameter Display System (SPDS) display on the Plant Computer (ref. 1).

The function restoration procedures are those emergency operating procedures that address the recovery of the core cooling critical safety functions. The procedure is considered effective if the temperature is decreasing or if the vessel water level is increasing (ref. 2).

A direct correlation to status trees can be made if the effectiveness of the restoration procedures is also evaluated. If core exit thermocouple (TC) readings are greater than 1,200°F or other CSFST RED path conditions exist (ref. 1), Fuel Clad barrier is also lost.

CT.B

Basis Reference(s):

1. 1OM-53A.1.F-0.2 Core Cooling Status Trees
2. 1OM-53A.1.FR-C.1 Response to Inadequate Core Cooling
3. NEI 99-01 Rev. 6 Inadequate Heat Removal Containment Potential Loss 2.A

Barrier: Containment **CT.C**

Category: C. CT Radiation/RCS Activity

Degradation Threat: Loss

Threshold:

None

Barrier: Containment **CT.C**

Category: C. CT Radiation/RCS Activity

Degradation Threat: Potential Loss

Threshold:

1. Containment Radiation Monitor **> Table 1F-2, "CT Potential Loss"**

Table 1F-2 Containment Radiation – R/hr (RM-1RM-219A or B)			
Time After S/D (Hrs.)	RC Loss (R/hr)	FC Loss (R/hr)	CT Potential Loss (R/hr)
0-1	8	520	10,000
>1-2	8	360	7,100
>2-8	8	150	2,900
>8-16	8	93	1,800
>16-48	8	46	910

Basis:

The radiation monitor reading corresponds to an instantaneous release of all reactor coolant mass into the containment, assuming that 20% of the fuel cladding has failed. This level of fuel clad failure is well above that used to determine the analogous Fuel Clad Barrier Loss and RCS Barrier Loss thresholds.

NUREG-1228, *Source Estimations During Incident Response to Severe Nuclear Power Plant Accidents*, indicates the fuel clad failure must be greater than approximately 20% in order for there to be a major release of radioactivity requiring offsite protective actions. For this condition to exist, there must already have been a loss of the RCS Barrier and the Fuel Clad Barrier. It is therefore prudent to treat this condition as a potential loss of containment which would then escalate the ECL to a General Emergency.

The gamma dose rate resulting from a postulated loss of coolant accident (LOCA) is monitored by the containment high range monitors, RM-1RM-219A & B and are located inside containment. The detector range is approximately 1 to 1E7 R/hr. Radiation Monitors RM-1RM-219A & B provide a diverse means of measuring the containment for high level gamma radiation (ref. 1).

CT.C

The Table 1F-2 values, column CT Potential Loss represents, based on the referenced calculation, the expected containment high range radiation monitor (RM-1RM-219B) response based on a LOCA, for periods of 1, 2, 8 and 16 hours after shutdown with coolant activity corresponding to ~20% clad failure (ref. 1).

The value is derived as follows:

ERS-SMM-11-002 Attachment 2 CRM Readings vs. Time for 20% Clad Damage on RM-1RM-219B for 1, 2, 8 and 16 hours after shutdown (rounded) (ref. 1).

Basis Reference(s):

1. ERS-SMM-11-002, Containment Radiation Monitor Readings Following Clad Damage (FC2 Loss, FC7 Loss, RC2 Loss and CT2 Potential Loss)
2. NEI 99-01 Rev. 6 CMT Radiation / RCS Activity Containment Potential Loss 3.A

Barrier: Containment **CT.D**

Category: D. CT Integrity or Bypass

Degradation Threat: Loss

Threshold:

1. Containment isolation is required

   AND EITHER:

   - Containment integrity has been lost based on Emergency Director judgment
   - UNISOLABLE pathway from containment to the environment exists

Basis:

These thresholds address a situation where containment isolation is required and one of two conditions exists as discussed below. Users are reminded that there may be accident and release conditions that simultaneously meet both bulleted thresholds.

First Threshold – Containment integrity has been lost, i.e., the actual containment atmospheric leak rate likely exceeds that associated with allowable leakage (or sometimes referred to as design leakage). Following the release of RCS mass into containment, containment pressure will fluctuate based on a variety of factors; a loss of containment integrity condition may (or may not) be accompanied by a noticeable drop in containment pressure. Recognizing the inherent difficulties in determining a containment leak rate during accident conditions, it is expected that the Emergency Director will assess this threshold using judgment, and with due consideration given to current plant conditions, and available operational and radiological data (e.g., containment pressure, readings on radiation monitors outside containment, operating status of containment pressure control equipment, etc.).

Refer to the middle piping run of Figure 1. Two simplified examples are provided. One is leakage from a penetration and the other is leakage from an in-service system valve. Depending upon radiation monitor locations and sensitivities, the leakage could be detected by any of the four monitors depicted in the figure.

Another example would be a loss or potential loss of the RCS barrier, and the simultaneous occurrence of two FAULTED locations on a steam generator where one fault is located inside containment (e.g., on a steam or feedwater line) and the other outside of containment. In this case, the associated steam line provides a pathway for the containment atmosphere to escape to an area outside the containment.

CT.D

Following the leakage of RCS mass into containment and a rise in containment pressure, there may be minor radiological releases associated with allowable (design) containment leakage through various penetrations or system components. These releases do not constitute a loss or potential loss of containment but should be evaluated using the Recognition Category R ICs.

Second Threshold – Conditions are such that there is an UNISOLABLE pathway for the migration of radioactive material from the containment atmosphere to the environment. As used here, the term "environment" includes the atmosphere of a room or area, outside the containment, that may, in turn, communicate with the outside-the-plant atmosphere (e.g., through discharge of a ventilation system or atmospheric leakage). Depending upon a variety of factors, this condition may or may not be accompanied by a noticeable drop in containment pressure.

Refer to the top piping run of Figure 1. In this simplified example, the inboard and outboard isolation valves remained open after a containment isolation was required (i.e., containment isolation was not successful). There is now an UNISOLABLE pathway from the containment to the environment.

The existence of a filter is not considered in the threshold assessment. Filters do not remove fission product noble gases. In addition, a filter could become ineffective due to iodine and/or particulate loading beyond design limits (i.e., retention ability has been exceeded) or water saturation from steam/high humidity in the release stream.

Leakage between two interfacing liquid systems, by itself, does not meet this threshold. Refer to the bottom piping run of Figure 1. In this simplified example, leakage in an RCP seal cooler is allowing radioactive material to enter the Auxiliary Building. The radioactivity would be detected by the Process Monitor. If there is no leakage from the closed water cooling system to the Auxiliary Building, then no threshold has been met. If the pump developed a leak that allowed steam/water to enter the Auxiliary Building, then second threshold would be met. Depending upon radiation monitor locations and sensitivities, this leakage could be detected by any of the four monitors depicted in the figure and cause the first threshold to be met as well.

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

The status of the containment barrier during an event involving steam generator tube leakage is assessed using Loss Threshold CT.A.1.

Basis Reference(s):

1. NEI 99-01 Rev. 6 CMT Integrity or Bypass Containment Loss 4.A

Barrier: Containment **CT.D**

Category: D. CT Integrity or Bypass

Degradation Threat: Loss

Threshold:

2.	Indications of RCS leakage outside of containment

Basis:

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.

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.

1OM-53A.1.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):

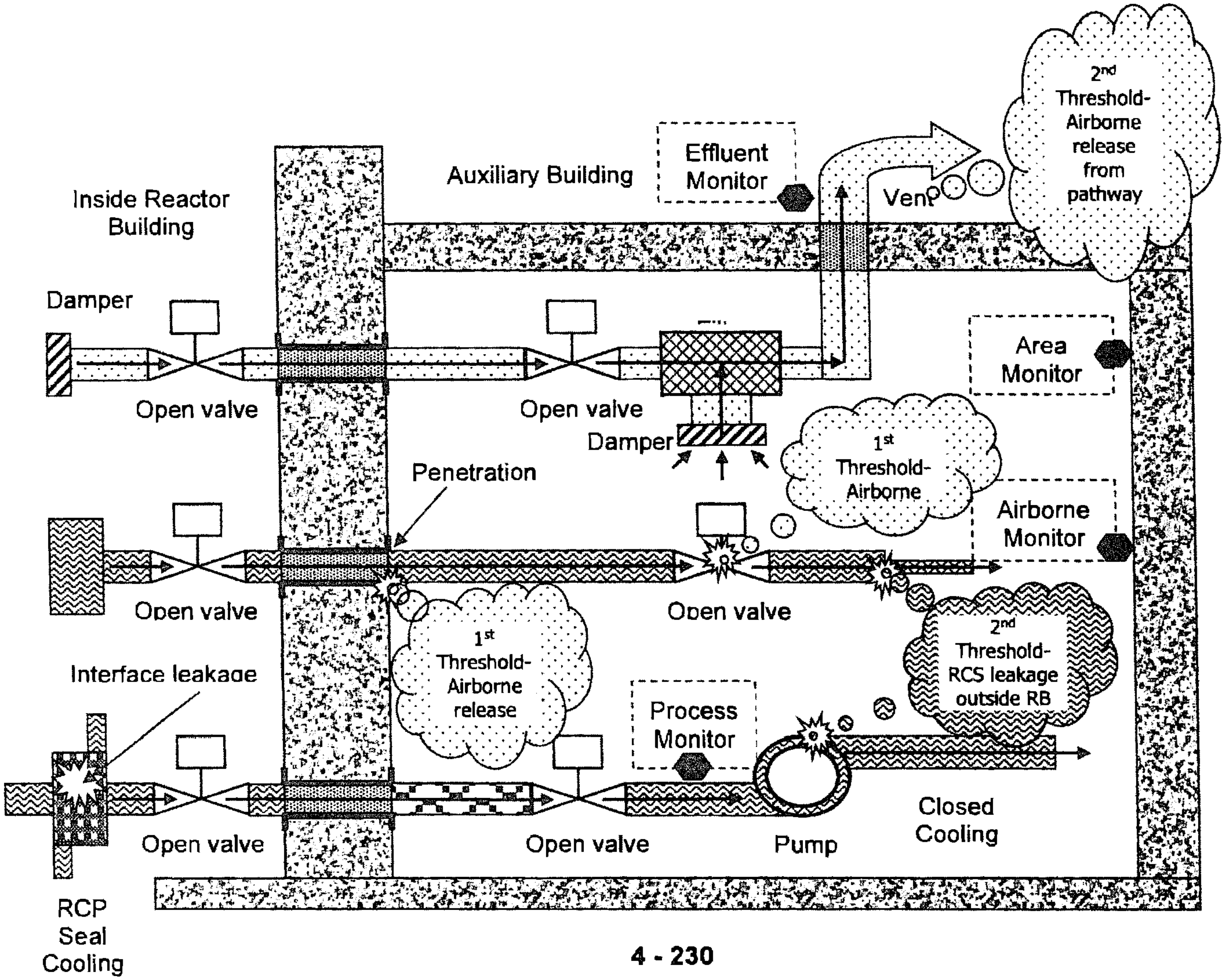
- Safety Injection
- Chemical & Volume Control
- RCP seals
- PZR/RCS Loop sample lines

Basis Reference(s):

1. 1OM-53A.1.ECA-1.2 LOCA Outside Containment
2. NEI 99-01 Rev. 6 CMT Integrity or Bypass Containment Loss

Figure 1: Containment Integrity or Bypass Examples

CT.D



ATTACHMENT 2:
Unit 1 Fission Product Barrier Loss/Potential Loss Matrix and Bases

Barrier: Containment **CT.D**

Category: D. CT Integrity or Bypass

Degradation Threat: Potential Loss

Threshold:

1. Containment-RED Path conditions met

Basis:

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.

Critical Safety Function Status Tree (CSFST) Containment-RED path is entered if containment pressure is greater than or equal to 45 psig and represents an extreme challenge to safety function. The CSFSTs are normally monitored using the the Safety Parameter Display System (SPDS) display on the Plant Computer (ref. 1).

45 psig is the containment design pressure and is the pressure used to define CSFST Containment Red Path conditions (ref. 1, 2).

Basis Reference(s):

1. 1OM-53A.1.F-0.5 Containment Status Tree
2. BV1 UFSAR Section 5.2.2 Design Basis and Loading Criteria
3. NEI 99-01 Rev. 6 CMT Integrity or Bypass Containment Potential Loss 4.A

ATTACHMENT 2:
Unit 1 Fission Product Barrier Loss/Potential Loss Matrix and Bases

Barrier: Containment **CT.D**

Category: D. CT Integrity or Bypass

Degradation Threat: Potential Loss

Threshold:

2. Containment hydrogen concentration **> 4%**

Basis:

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.

The containment hydrogen analyzer system consists of two redundant hydrogen monitors to provide protection against single failure and single loss of power. Containment samples are obtained through independent sample lines for each monitor. Indication is provided for each hydrogen analyzer, on the vertical board in the main control room, with an indicating range of 0-10 percent hydrogen. A recorder is provided to record the Train A hydrogen level. The hydrogen analyzer system is designed to provide a continuous positive indication of the containment hydrogen concentration within 30 minutes after the initiation of safety injection (ref. 1).

In the early stages of a core uncovery event, it is unlikely that hydrogen buildup due to a core uncovery could result in an explosive mixture of dissolved gasses in 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. A combustible mixture can be formed when hydrogen gas concentration in the Containment atmosphere is greater than 4% by volume. All hydrogen measurements are referenced to concentrations in dry air even though the actual Containment environment may contain significant steam concentrations.

To generate such levels of combustible gas, loss of the Fuel Clad and RCS barriers must have occurred. With the Potential Loss of the Containment barrier, the threshold hydrogen concentration, therefore, will likely warrant declaration of a General Emergency.

Basis Reference(s):

1. BV1 UFSAR Section 6.5 Post DBA Hydrogen Control System
2. NEI 99-01 Rev. 6 CMT Integrity or Bypass Containment Potential Loss 4.B

ATTACHMENT 2:
Unit 1 Fission Product Barrier Loss/Potential Loss Matrix and Bases

Barrier: Containment **CT.D**

Category: D. CT Integrity or Bypass

Degradation Threat: Potential Loss

Threshold:

3.	Containment pressure **> 11 psig AND** < one full train of depressurization equipment operating per design for **≥ 15 min.** (Note 1)

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

Basis:

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.

This threshold represents a Potential Loss of the Containment barrier because the Containment heat removal and depressurization equipment (but not including Containment venting strategies) is either lost or degraded.

Each unit has a containment pressure quench spray system with two 100% capacity trains. These pumps take suction from the RWST and discharge to the spray header. The quench spray system starts on a CIB at the start of a LOCA accident.

The recirculation spray system has four 50% capacity subsystems that consist of a pump and a cooler. The recirculation spray pump takes suction from the containment sump and discharges through a cooler to the spray header. The recirculation spray system does not start during a LOCA until there is low level in the RWST to verify the sump has adequate water inventory. When the RWST level goes very low the quench spray pumps are secured.

A very short period of time could exist where the quench spray system and the recirculation spray system pumps could both be running. Normally it is either the quench spray or the recirculation spray running.

One train of QS System and one train of RS System comprise one full train of depressurization equipment as designed (ref. 1).

CT.D

Basis Reference(s):

1. UFSAR Section 6.4 Containment Depressurization System
2. NEI 99-01 Rev. 6 CMT Integrity or Bypass Containment Potential Loss 4.C

ATTACHMENT 2:
Unit 1 Fission Product Barrier Loss/Potential Loss Matrix and Bases

Barrier: Containment **CT.E**

Category: E. Emergency Director Judgment

Degradation Threat: Loss

Threshold:

1. **ANY** condition in the opinion of the Emergency Director that indicates Loss of the Containment Barrier

Basis:

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

Basis Reference(s):

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

ATTACHMENT 2:
Unit 1 Fission Product Barrier Loss/Potential Loss Matrix and Bases

Barrier: Containment **CT.E**

Category: E. Emergency Director Judgment

Degradation Threat: Potential Loss

Threshold:

1. **ANY** condition in the opinion of the Emergency Director that indicates Potential Loss of the Containment Barrier

Basis:

This threshold addresses any other factors that may be used by the Emergency Director in determining whether the Containment Barrier is potentially lost. The Emergency Director should also consider whether or not to declare the barrier potentially lost in the event that barrier status cannot be monitored.

Basis Reference(s):

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

ATTACHMENT 3:

Unit 2 EAL Technical Bases

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Unit 2 EAL Technical Bases

Category R – Abnormal Rad Release / Rad Effluent

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

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

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

Events of this category pertain to the following subcategories:

1. Radiological Effluent

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

2. Irradiated Fuel Event

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

3. Area Radiation Levels

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

ATTACHMENT 3:

Unit 2 EAL Technical Bases

Category:	R – Abnormal Rad Levels / Rad Effluent	**RU1.1**
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**

EITHER of the following gaseous effluent monitors > the reading shown for **≥ 60 min.**:

- SLCRS Vent (2HVS-RQ109E-WRGM) **5.88E+3 μCi/s**
- Ventilation Vent (2HVS-RQ101B) **6.02E-4 μCi/cc**

(Notes 1, 2, 3)

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

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

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

Mode Applicability:

All

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.

RU1.1

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 effluent pathways.

The gaseous release values represent two times the ODCM release rate limits (ref. 1, 2).

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

Basis Reference(s):

1. 1/2-ODC-2.02, ODCM Gaseous Effluents
2. ERS-HHM-87-014, Unit 1/Unit 2 ODCM Gaseous Effluent Monitor Setpoints
3. NEI 99-01 Rev. 6 AU1

ATTACHMENT 3:

Unit 2 EAL Technical Bases

Category: R – Abnormal Rad Levels / Rad Effluent **RU1.2**

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**
Liquid Waste monitor 2SGC-RQ100 reading **> 2 x high alarm setpoint** for **≥ 60 min.** (Notes 1, 2, 3)	

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

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

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

Mode Applicability:

All

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.

RU1.2

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

This EAL also 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).

The liquid release values represent two times the ODCM release rate limits. The liquid monitor high-high alarm setpoints are established to ensure the ODCM release limits are not exceeded (ref. 1, 2).

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

Basis Reference(s):

1. 1/2-ODC-2.01, ODCM Liquid Effluents
2. ERS-ATL-93-021 Process Alarm Setpoints for Liquid Effluent Monitors
3. NEI 99-01 Rev. 6 AU1

ATTACHMENT 3:

Unit 2 EAL Technical Bases

Category: R – Abnormal Rad Levels / Rad Effluent **RU1.3**

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:

RU1.3	**Unusual Event**
Sample analysis for a gaseous or liquid release indicates a concentration or dose rate > **2 x ODCM limits** for **≥ 60 min.** (Notes 1, 2)	

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

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

Mode Applicability:

All

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.

Basis Reference(s): **RU1.3**

1. 1/2-ODC-2.02, ODCM Gaseous Effluents
2. 1/2-ODC-2.01, ODCM Liquid Effluents
3. 1/2-ODC-3.03, Controls for RETS and REMP Programs
3. NEI 99-01 Rev. 6 AU1

ATTACHMENT 3:

Unit 2 EAL Technical Bases

Category: R – Abnormal Rad Levels / Rad Effluent **RA1.1**

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

EITHER of the following gaseous effluent monitors > the reading shown for **≥ 15 min.**:

- SLCRS Vent (2HVS-RQ109E-WRGM) **1.95E+5 μCi/s**
- Ventilation Vent (2HVS-RQ101B) **1.67E-2 μCi/cc**

(Notes 1, 2, 3, 4)

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

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

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

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

Mode Applicability:

All

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.

ATTACHMENT 3:

Unit 2 EAL Technical Bases

RA1.1

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 gaseous effluent release values correspond to calculated doses of 1% (10% of the SAE thresholds) of the EPA Protective Action Guidelines (TEDE or CDE Thyroid) (ref. 1).

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

Basis Reference(s):

1. ERS-MPD-93-008 BVPS-U2 Gaseous Radioactivity Monitor Emergency Action Levels
2. NEI 99-01 Rev. 6 AA1

Category: R – Abnormal Rad Levels / Rad Effluent **RA1.2**

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**
Gaseous release dose assessment using actual meteorology indicates doses **> 10 mrem** TEDE or **50 mrem** thyroid CDE at or beyond the site boundary (Note 4)	

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

Mode Applicability:

All

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.

Basis Reference(s):

1. NEI 99-01 Rev. 6 AA1

ATTACHMENT 3:

Unit 2 EAL Technical Bases

Category: R – Abnormal Rad Levels / Rad Effluent **RA1.3**

Subcategory: 1 – Radiological Effluent

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

EAL:

RA1.3	**Alert**
Analysis of a liquid effluent sample indicates a concentration or release rate that would result in doses **> 10 mrem** TEDE or **50 mrem** thyroid CDE at or beyond the site boundary for **60 min.** of exposure (Notes 1, 2)	

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

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

Mode Applicability:

All

Basis:

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

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

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

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

Basis Reference(s):

1. ERS-LMR-14-001, Liquid Monitor Emergency Action Level (EAL) Set Points
2. NEI 99-01 Rev. 6 AA1

Category: R – Abnormal Rad Levels / Rad Effluent **RA1.4**

Subcategory: 1 – Radiological Effluent

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

EAL:

RA1.4 Alert

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

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

(Notes 1, 2)

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

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

Mode Applicability:

All

Basis:

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

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

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

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

Basis Reference(s):

1. NEI 99-01 Rev. 6 AA1

RS1.1

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

EITHER of the following gaseous effluent monitors > the reading shown for **≥ 15 min.**:

- SLCRS Vent (2HVS-RQ109E-WRGM) **1.95E+6 μCi/s**
- Ventilation Vent (2HVS-RQ101B) **1.67E-1 μCi/cc**

(Notes 1, 2, 3, 4)

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

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

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

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

Mode Applicability:

All

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.

RS1.1

The gaseous effluent release value corresponds to calculated doses of 10% of the EPA Protective Action Guidelines (TEDE or CDE Thyroid) (ref. 1).

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

Basis Reference(s):

1. ERS-MPD-93-008 BVPS-U2 Gaseous Radioactivity Monitor Emergency Action Levels
2. NEI 99-01 Rev. 6 AS1

Category: R – Abnormal Rad Levels / Rad Effluent **RS1.2**

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**
Gaseous release dose assessment using actual meteorology indicates doses **> 100 mrem** TEDE or **500 mrem** thyroid CDE at or beyond the site boundary (Note 4)	

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

Mode Applicability:

All

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.

Basis Reference(s):

1. NEI 99-01 Rev. 6 AS1

Category: R – Abnormal Rad Levels / Rad Effluent **RS1.3**

Subcategory: 1 – Radiological Effluent

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

EAL:

RS1.3 **Site Area Emergency**

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

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

(Notes 1, 2)

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

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

Mode Applicability:

All

Basis:

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

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

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

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

Basis Reference(s):

1. NEI 99-01 Rev. 6 AS1

ATTACHMENT 3:

Unit 2 EAL Technical Bases

Category: R – Abnormal Rad Levels / Rad Effluent **RG1.1**

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

SLCRS Vent (2HVS-RQ109E-WRGM) reading **> 1.95E+7 μCi/s** for **≥ 15 min.**
(Notes 1, 2, 3, 4)

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

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

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

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

Mode Applicability:

All

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.

ATTACHMENT 3:

Unit 2 EAL Technical Bases

RG1.1

The gaseous effluent release value corresponds to calculated doses of 100% of the EPA Protective Action Guidelines (TEDE or CDE Thyroid) (ref. 1).

Ventilation Vent (2HVS-RQ101B) monitor would be "off-scale" at this release level (maximum indication is 3.72E-01 μCi/cc) if the effluent flowpath was not isolated or aligned to the SLCRS vent. Since this value is only approximately 2x the SITE AREA EMERGENCY level vs. the 10x called for in the technical bases it is not used as a threshold value for the GENERAL EMERGENCY level.

Basis Reference(s):

1. ERS-MPD-93-008 BVPS-U2 Gaseous Radioactivity Monitor Emergency Action Levels
2. NEI 99-01 Rev. 6 AG1

ATTACHMENT 3:

Unit 2 EAL Technical Bases

Category: R – Abnormal Rad Levels / Rad Effluent **RG1.2**

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

Gaseous release dose assessment using actual meteorology indicates doses **> 1,000 mrem** TEDE or **5,000 mrem** thyroid CDE at or beyond the site boundary (Note 4)

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

Mode Applicability:

All

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.

Basis Reference(s):

1. NEI 99-01 Rev. 6 AG1

ATTACHMENT 3:

Unit 2 EAL Technical Bases

Category: R – Abnormal Rad Levels / Rad Effluent **RG1.3**

Subcategory: 1 – Radiological Effluent

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

EAL:

RG1.3 General Emergency

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

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

(Notes 1, 2)

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

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

Mode Applicability:

All

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.

Basis Reference(s):

1. NEI 99-01 Rev. 6 AG1

Category: R – Abnormal Rad Levels / Rad Effluent **RU2.1**

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 on **ANY** of the following:

- Spent Fuel Pool Level (2FNC-LT102A or B)
- Spent Fuel Pool Level Alarm (A6-1B)
- Spent Fuel Pool Level (2FNC-LI101A/B)
- PZR Cold Cal Level (2RCS-LT462) (MODE 5, 6 & Defueled Only)
- Temporary Level Instrument (2RCS-LT102) (MODE 6 & Defueled Only)
- Temporary Level Instrument (2RCS-LT105) (MODE 6 & Defueled Only)

AND

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

- 2RMR-RQ203 Manipulator Crane Area Monitor (MODE 6 & Defueled Only)
- 2RMF-RQ202 Fuel Pit Bridge Area Monitor

Mode Applicability:

All

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

RU2.1

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.

Indication of decreasing level includes **ANY** of the following: (ref. 1):

- Spent Fuel Pool Level (2FNC-LT102A/B)
- Spent Fuel Level Alarm (A6-1B)
- Spent Fuel Pool Level (2FNC-LI101A/B)
- PZR Cold Cal Level (2RCS-LT462)
- Temporary Level Instrument (2RCS-LT102)
- Temporary Level Instrument (2RCS-LT105)

Allowing level to decrease could result in spent fuel being uncovered, reducing spent fuel decay heat removal and creating an extremely hazardous radiation environment. During refueling, sufficient water level is required to be maintained in the fuel transfer canal, refueling cavity, and SFP to retain iodine fission product activity in the water in the event of a fuel handling accident.

The fuel transfer canal is only of concern in assessing this EAL when irradiated fuel transfer is in progress, in which case the spent fuel pool transfer canal gate is open and connected to the fuel transfer canal.

The listed area radiation monitors are those which would likely see an increase in area radiation due to a loss of REFUELING PATHWAY inventory.

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

Basis Reference(s):

1. 2OM-53C.4.2.20.1 Spent Fuel Pool Cooling Trouble
2. BVPS-1&2 Technical Specification 3.7.15 Fuel Storage Pool Water Level
3. BVPS-1&2 Technical Specification 3.9.6 Refueling Cavity Water Level
4. NEI 99-01 Rev. 6 AU2

ATTACHMENT 3:

Unit 2 EAL Technical Bases

Category: R – Abnormal Rad Levels / Rad Effluent **RA2.1**

Subcategory: 2 – Irradiated Fuel Event

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

EAL:

RA2.1 Alert

Uncovery of irradiated fuel in the REFUELING PATHWAY

Mode Applicability:

All

Basis:

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

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

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

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

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

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

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

Basis Reference(s):

1. 2OM-53C.4.2.20.1 Spent Fuel Pool Cooling Trouble
2. NEI 99-01 Rev. 6 AA2

Category: R – Abnormal Rad Levels / Rad Effluent **RA2.2**

Subcategory: 2 – Irradiated Fuel Event

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

EAL:

RA2.2 **Alert**

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

- 2HVS-RQ109E-WRGM SLCRS Vent (High alarm)
- 2HVS-RQ101B Ventilation Vent (High alarm)
- 2RMR-RQ203 Manipulator Crane Area Monitor (High alarm)
- 2RMF-RQ202 Fuel Bridge Area Monitor (High alarm)

Mode Applicability:

All

Basis:

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

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

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

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).

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

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

ATTACHMENT 3:

Unit 2 EAL Technical Bases

RA2.2

Basis Reference(s):

1. 2OM-53C.4.2.49.1 Irradiated Fuel Damage
2. 2OM-53C.4.2.20.1 Spent Fuel Pool Cooling Trouble
3. NEI 99-01 Rev. 6 AA2

ATTACHMENT 3:

Unit 2 EAL Technical Bases

Category: R – Abnormal Rad Levels / Rad Effluent **RA2.3**

Subcategory: 2 – Irradiated Fuel Event

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

EAL:

RA2.3	**Alert**
Spent fuel pool level (2FNC-LI101A/B) reading **≤ 10 ft.** (Level 2)	

Mode Applicability:

All

Basis:

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

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

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

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

Level 2 is the level that is adequate to provide substantial radiation shielding for a person standing on the spent fuel pool operating deck. It represents the range of water level where any necessary operations in the vicinity of the spent fuel pool can be completed without significant dose consequences from direct gamma radiation from the stored spent fuel. BVPS designated as Level 2 the water level ~10 feet above the top of the fuel racks (El 752'-6") (ref. 2).

Spent Fuel Pool (SFP) draindown to elevation 750 ft – 10 inches, as described in Technical Specification 4.3.2, from SFP cooling system piping break outside the SFP walls would result in an [2FNC-LI101A,B] indicated level of 8.3 ft. This SFP water level was evaluated by calculation 10080-UR(B)-512 as resulting in an operating deck dose rate of 280 mrem/hr after full core offload at 100 hours after shutdown. The NRC accepted the elevation change to 750 ft – 10 inches in BV2 Amendment 181.

RA2.3

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

Basis Reference(s):

1. NRC EA-12-51 Issuance of Order to Modify Licenses with Regard to Reliable Spent Fuel Pool Instrumentation
2. ECP No. 13-0562-000
3 NEI 99-01 Rev. 6 AA2

ATTACHMENT 3:

Unit 2 EAL Technical Bases

Category: R – Abnormal Rad Levels / Rad Effluent **RS2.1**

Subcategory: 2 – Irradiated Fuel Event

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

EAL:

RS2.1	**Site Area Emergency**
Spent fuel pool level (2FNC-LI101A/B) reading **≤ 0.5 ft.** (Level 3)	

Mode Applicability:

All

Basis:

This IC 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.

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

BVPS designated as Level 3 the water level greater than 6 inches (0.5 ft.) above the top of the fuel storage racks plus the accuracy of the SFP level instrument channel (El. 743' – 0.4") (ref. 2).

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

Basis Reference(s):

1. NRC EA-12-51 Issuance of Order to Modify Licenses with Regard to Reliable Spent Fuel Pool Instrumentation
2. ECP No. 13-0562-000
3. NEI 99-01 Rev. 6 AS2

ATTACHMENT 3:

Unit 2 EAL Technical Bases

Category: R – Abnormal Rad Levels / Rad Effluent **RG2.1**

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 (2FNC-LI101A/B) **cannot** be restored to at least **0.5 ft.** (Level 3) for **≥ 60 min.** (Note 1)	

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

Mode Applicability:

All

Basis:

This IC addresses a significant loss of spent fuel pool inventory control and makeup capability leading to a prolonged uncovery of spent fuel. This condition will lead to fuel damage and a radiological release to the environment.Post-Fukushima order EA-12-051 (ref. 1) required the installation of reliable SFP level indication capable of identifying normal level (Level 1), SFP level 10 ft. above the top of the fuel racks (Level 2) and SFP level at the top of the fuel racks (Level 3) (ref. 1).

BVPS designated as Level 3 the water level greater than 6 inches (0.5 ft.) above the top of the fuel storage racks plus the accuracy of the SFP level instrument channel (El. 743' – 0.4") (ref. 2).

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.

Basis Reference(s):

1. NRC EA-12-51 Issuance of Order to Modify Licenses with Regard to Reliable Spent Fuel Pool Instrumentation
2. ECP No. 13-0562-000
3. NEI 99-01 Rev. 6 AG2

Category: R – Abnormal Rad Levels / Rad Effluent **RA3.1**

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

- Control Room (2RMC*RQ201/202)
- Central Alarm Station (by survey)

Mode Applicability:

All

Basis:

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

2RMC*RQ201/202 are the installed Control Room area radiation monitors and may be used to assess this EAL threshold. However, no permanently installed area radiation monitoring is installed in the CAS and therefore this threshold must be assessed via local radiation survey (ref. 1).

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

Basis Reference(s):

1. 2-HPP-4.04.019, DRMS, Area Monitoring Subsystem
2. NEI 99-01 Rev. 6 AA3

Category: R – Abnormal Rad Levels / Rad Effluent **RA3.2**

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 Rod Control Building 735' (Notes 5, 12)

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 12: 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).

Mode Applicability:

4 - Hot Shutdown

Basis:

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

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

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

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

RA3.2

- 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 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 listed area 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 listed area specifies the plant mode(s) during which entry would be required for that area (ref. 1).

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

RA3.2 mode applicability has been limited to the applicable modes identified for the locations identified in RA3.2. If due to plant operating procedure or a plant configuration changes, the applicable plant modes specified in RA3.2 are changed, a corresponding change to Attachment 5 ‘Safe Operation and Shutdown Areas Tables RA3.2 and HA5.1 Bases’ and to EAL RA3.2 mode applicability is required.

Basis Reference(s):

1. EPLAN, Section 4, Attachment 5 Safe Operation & Shutdown Areas RA3.2 & HA5.1 Bases
2. NEI 99-01 Rev. 6 AA3

ATTACHMENT 3:

Unit 2 EAL Technical Bases

Category E – Independent Spent Fuel Storage Facility (ISFSI)

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

An Independent Spent Fuel Storage Facility(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.

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

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

ATTACHMENT 3:

Unit 2 EAL Technical Bases

Category: ISFSI **EU1.1**

Subcategory: Confinement Boundary

Initiating Condition: Damage to a loaded cask CONFINEMENT BOUNDARY

EAL:

EU1.1 Unusual Event

Damage to a loaded cask CONFINEMENT BOUNDARY as indicated by an on-contact radiation reading **> ANY** of the following:

- **1,050 mrem/hr** at the Horizonal Storage Module (HSM) bird screen
- **4 mrem/hr** outside HSM door
- **8 mrem/hr** on end shield wall exterior

Mode Applicability:

All

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.

The dry-cask storage system is the NUHOMS Horizontal Modular Storage System. (ref. 1).

The value shown represents 2 times the limits specified in the ISFSI Certificate of Compliance Technical Specification section 5.4.2 for radiation external to a HSM loaded with a Model 37PTH DSC (ref. 1).

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

Basis Reference(s): **EU1.1**

1. Technical Specifications for the Standardized NUHOMS Horizontal Modular Storage System, Section 5.4 HSM or HSM-H Dose Rate Evaluation Program
2. NEI 99-01 Rev. 6 E-HU1

ATTACHMENT 3:

Unit 2 EAL Technical Bases

Category C – Cold Shutdown / Refueling System Malfunction

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

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

The events of this category pertain to the following subcategories:

1. RCS Level

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

2. Loss of Emergency AC Power

Loss of essential plant electrical power can compromise plant SAFETY SYSTEM operability including decay heat removal and emergency core cooling systems, which may be necessary to ensure fission product barrier integrity. This category includes loss of onsite and offsite power sources for the 4 KV 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 buses.

ATTACHMENT 3:

Unit 2 EAL Technical Bases

5. Loss of Communications

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

6. Hazardous Event Affecting SAFETY SYSTEMS

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

ATTACHMENT 3:

Unit 2 EAL Technical Bases

CU1.1

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

Mode Applicability:

5 - Cold Shutdown, 6 - Refueling

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 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.

With the plant in Cold Shutdown, RCS water level is normally maintained above the pressurizer low level setpoint of 14%. 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 (ref. 1, 2).

With the plant in Refueling mode, RCS water level is normally maintained at or above the reactor vessel flange (Technical Specification LCO 3.9.6 requires at least 23 ft. of water above the top of the reactor vessel flange in the refueling cavity during refueling operations) (ref. 3).

CU1.1

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

Basis Reference(s):

1. 2OM-53C.4.2.10.1 Loss of Residual Heat Removal Capability
2. 2OM-52.4.R.2.F Station Shutdown from 100% to Mode 5
3. Technical Specification Section 3.9.6 Refueling Cavity Water Level
4. NEI 99-01 Rev. 6 CU1

ATTACHMENT 3:

Unit 2 EAL Technical Bases

Category: C – Cold Shutdown / Refueling System Malfunction **CU1.2**

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 2C-6 Sump/Tank level due to a loss of RCS inventory
- Visual observation of UNISOLABLE RCS leakage

Table 2C-6 Sump/Tank
• Containment Sumps • Incore Sump • Refueling Water Storage Tank (RWST) • Primary Drains Tank • Pressurizer Relief Tank (PRT) • CCP Surge Tank

Mode Applicability:

5 - Cold Shutdown, 6 – Refueling

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.

ATTACHMENT 3:

Unit 2 EAL Technical Bases

CU1.2

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

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

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

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

Basis Reference(s):

1. 2OM-53C.4.2.10.1 Loss of Residual Heat Removal Capability
2. NEI 99-01 Rev. 6 CU1

Category: C – Cold Shutdown / Refueling System Malfunction **CA1.1**

Subcategory: 1 – RCS Level

Initiating Condition: Loss of RCS inventory

EAL:

CA1.1	**Alert**
Loss of RCS inventory as indicated by reactor vessel level **≤ 14 in.** (2RCS-LI102)	

Mode Applicability:

5 - Cold Shutdown, 6 – Refueling

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 14 in. indicates that operator actions have not been successful in restoring and maintaining RCS water level. The heat-up rate of the coolant will increase as the available water inventory is reduced. A continuing decrease in water level will lead to core uncovery.

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

Reactor vessel level of ~14 in. is the minimum level for RHR pump operation in the decay heat removal mode @ an RHR flowrate of 1,000 gpm. (ref. 1).

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

Basis Reference(s):

1. 2OM-53C.4.2.10.2 Loss of RHS While Operating at Reduced InventoryMidloop Conditions Attachment 2 Required RCS Water Level for Reduced Inventory/Midloop
2. NEI 99-01 Rev. 6 CA1
3. 2OM-53C.4.2.10.1, Loss of Residual Heat Removal Capability

ATTACHMENT 3:

Unit 2 EAL Technical Bases

Category: C – Cold Shutdown / Refueling System Malfunction **CA1.2**

Subcategory: 1 – RCS Level

Initiating Condition: Loss of RCS inventory

EAL:

CA1.2 Alert

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

AND EITHER

- UNPLANNED increase in **ANY** Table 2C-6 Sump/Tank level due to a loss of RCS inventory
- Visual observation of UNISOLABLE RCS leakage

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

Table 2C-6 Sump/Tank
• Containment Sumps • Incore Sump • Refueling Water Storage Tank (RWST) • Primary Drains Tank • Pressurizer Relief Tank (PRT) • CCP Surge Tank

Mode Applicability:

5 - Cold Shutdown, 6 – Refueling

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.

ATTACHMENT 3:

Unit 2 EAL Technical Bases

CA1.2

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

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

In this EAL, all RCS water level indication would be unavailable for greater than 15 minutes, and the RCS inventory loss must be detected by indirect leakage indications. Surveillance procedures provide instructions for calculating primary system leak rate by manual or computer-based water inventory balances. Level increases must be evaluated against other potential sources of leakage such as cooling water sources inside the containment to ensure they are indicative of RCS leakage. If the make-up rate to the RCS unexplainably rises above the pre-established rate, a loss of RCS inventory may be occurring even if the source of the leakage cannot be immediately identified. Visual observation of leakage from systemsconnected to the RCS that cannot be isolated could also be indicative of a loss of RCS inventory (ref. 1).

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

Basis Reference(s):

1. 2OM-53C.4.2.10.1 Loss of Residual Heat Removal Capability
2. NEI 99-01 Rev. 6 CA1

ATTACHMENT 3:

Unit 2 EAL Technical Bases

Category: C – Cold Shutdown / Refueling System Malfunction **CS1.1**

Subcategory: 1 – RCS Level

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

EAL:

CS1.1 **Site Area Emergency**
CONTAINMENT CLOSURE **not** established
AND
RCS level **< 64%** RVLIS Full Range (6 in. below bottom of hotleg)

Mode Applicability:

5 – Cold Shutdown, 6 – Refueling

Basis:

This IC addresses a significant and prolonged loss of reactor vessel/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 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 reactor vessel levels of EALs CS1.1 and CS2.2 reflect the fact that with CONTAINMENT CLOSURE established, there is a lower probability of a fission product release to the environment.

When RVLIS Full Range water level decreases to 64% (ref. 1), water level is approximately 6 inches below the bottom of the RCS hot leg penetration. When RCS water level drops significantly below the elevation of the bottom of the RCS hot leg penetration, all sources of RCS injection have failed or are incapable of making up for the inventory loss.

In Refueling mode, RCS water level indication from RVLIS is likely unavailable but alternate means of level indication are normally installed (including visual observation) to assure that the ability to monitor water level will not be interrupted. If no RVLIS alternate means available, refer to CS1.3.

The status of CONTAINMENT CLOSURE is tracked if plant conditions change that could raise the risk of a fission product release as a result of a loss of decay heat removal (ref. 2, 3).

ATTACHMENT 3:

Unit 2 EAL Technical Bases

CS1.1

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.

Basis Reference(s):

1. 2OM-5D.5.A.37 Figure 5D-37 RVLIS Full Range Level VS. Reactor Vessel Height
2. NOP-OP-1005 Shutdown Defense in Depth
3. 1/2-ADM-0712 Shutdown Defense in Depth Assessment
4. NEI 99-01 Rev. 6 CS1

ATTACHMENT 3:

Unit 2 EAL Technical Bases

Category: C – Cold Shutdown / Refueling System Malfunction **CS1.2**

Subcategory: 1 – RCS Level

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

EAL:

CS1.2	**Site Area Emergency**

CONTAINMENT CLOSURE established

AND

RCS level **< 56%** RVLIS Full Range (top of active fuel)

Mode Applicability:

5 – Cold Shutdown, 6 – Refueling

Basis:

This IC addresses a significant and prolonged loss of reactor vessel 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 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 reactor vessel levels of EALs CS1.1 and CS1.2 reflect the fact that with CONTAINMENT CLOSURE established, there is a lower probability of a fission product release to the environment.

When Reactor Vessel water level drops below 56% RVLIS Full Range (ref. 1), core uncovery is about to occur.

Under the conditions specified by this EAL, continued lowering of RCS water level is indicative of a loss of inventory control. Inventory loss may be due to a vessel breach, RCS pressure boundary leakage or continued boiling in the reactor vessel. The magnitude of this loss of water indicates that makeup systems have not been effective and may not be capable of preventing further RCS or reactor vessel water level drop and potential core uncovery. The inability to restore and maintain level after reaching this setpoint infers a failure of the RCS barrier and Potential Loss of the Fuel Clad barrier. If no RVLIS alternate means available, refer to CS1.3.

The status of CONTAINMENT CLOSURE is tracked if plant conditions change that could raise the risk of a fission product release as a result of a loss of decay heat removal (ref. 2, 3).

CS1.2

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.

Basis Reference(s):

1. 2OM-5D.5.A.37 Figure 5D-37 RVLIS Full Range Level VS. Reactor Vessel Height
2. NOP-OP-1005 Shutdown Defense in Depth
3. 1/2-ADM-0712 Shutdown Defense in Depth Assessment
4. NEI 99-01 Rev. 6 CS1

ATTACHMENT 3:

Unit 2 EAL Technical Bases

Category: C – Cold Shutdown / Refueling System Malfunction **CS1.3**

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

- UNPLANNED increase in **ANY** Table 2C-6 Sump/Tank level of sufficient magnitude to indicate core uncovery
- Erratic source range monitor indication
- Containment Radiation Monitor (2RMR-RQ206/207) **> 15 R/hr**

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

Table 2C-6 Sump/Tank
• Containment Sumps • Incore Sump • Refueling Water Storage Tank (RWST) • Primary Drains Tank • Pressurizer Relief Tank (PRT) • CCP Surge Tank

Mode Applicability:

5 – Cold Shutdown, 6 – Refueling

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.

CS1.3

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

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

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

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

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. Operating procedures provide instructions for calculating primary system leak rate by manual or computer-based water inventory balances. Level increases must be evaluated against other potential sources of leakage such as cooling water sources inside the containment to ensure they are indicative of RCS leakage. If the make-up rate to the RCS unexplainably rises above the pre-established rate, a loss of RCS inventory may be occurring even if the source of the leakage cannot be immediately identified. (ref. 1).

The RCS inventory loss may be detected by the Containment Radiation Monitors or erratic source range monitor indication.

As water level in the reactor vessel lowers, the dose rate above the core will rise. The dose rate due to this core shine should result in Containment Radiation Monitor (CRM) indication > 15 R/hr. Containment radiation is indicated on containment radiation monitors (CRMs) 2RMR-RQ206 and 207. These monitors are not located within line of sight of the reactor vessel. The containment radiation monitor alert alarm is set at 6.18E+2 R/hr and high alarm is set at 2.0E+4 R/hr. The alarm setpoints are considered operationally significant, but above what would be expected for a loss of vessel level while in the refuel mode. The CRM threshold values have been established at 15 R/hr (~10x the low scale reading of 1.5 R/hr) to provide a reasonable and conservative indication of abnormal conditions associated with elevated radiation levels in containment due to a loss of water level with irradiated fuel in the vessel.

Post-TMI accident studies indicated that the installed PWR nuclear instrumentation will operate erratically when the core is uncovered and that this should be used as a tool for making such determinations (ref. 1).

CS1.3

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.

Basis Reference(s):

1. 2OM-53C.4.2.10.1 Loss of Residual Heat Removal Capability
2. Nuclear Safety Analysis Center (NSAC), 1980, "Analysis of Three Mile Island - Unit 2 Accident," NSAC-1
3. NEI 99-01 Rev. 6 CS1

ATTACHMENT 3:

Unit 2 EAL Technical Bases

Category: C – Cold Shutdown / Refueling System Malfunction **CG1.1**

Subcategory: 1 – RCS Level

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

EAL:

CG1.1 General Emergency
RCS level **< 56%** RVLIS Full Range (top of active fuel) for ≥ **30 min.** (Note 1)
AND
ANY Containment Challenge indication, **Table 2C-1**

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

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

Table 2C-1 Containment Challenge Indications
• CONTAINMENT CLOSURE **not** established (Note 6) • Containment hydrogen concentration > 4% • UNPLANNED rise in containment pressure

Mode Applicability:

5 – Cold Shutdown, 6 – Refueling

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.

ATTACHMENT 3:

Unit 2 EAL Technical Bases

CG1.1

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

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

When Reactor Vessel water level drops below 56% RVLIS Full Range (ref. 1), core uncovery is about to occur.

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

Three conditions are associated with a challenge to Containment integrity:

1. CONTAINMENT CLOSURE not established - The status of CONTAINMENT CLOSURE is tracked if plant conditions change that could raise the risk of a fission product release as a result of a loss of decay heat removal (ref. 2, 3, 4). If containment closure is re-established prior to exceeding the 30 minute core uncovery time limit then escalation to GE would not occur.
2. Containment hydrogen > 4% - The 4% hydrogen concentration threshold is generally considered the lower limit for hydrogen deflagrations. Hydrogen monitors, although available at all times, are not in service during normal operations. They are started per 2OM-46.4.F (ref. 5).
3. UNPLANNED rise in Containment pressure - An UNPLANNED pressure rise in containment while in cold Shutdown or Refueling modes can threaten CONTAINMENT CLOSURE capability and thus Containment potentially cannot be relied upon as a barrier to fission product release.

ATTACHMENT 3:

Unit 2 EAL Technical Bases

CG1.1

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."

Basis Reference(s):

1. 2OM-5D.5.A.37 Figure 5D-37 RVLIS Full Range Level VS. Reactor Vessel Height
2. NOP-OP-1005 Shutdown Defense in Depth
3. 1/2CMP-47-Contingency Hatch Closure-1M, Contingency Hatch Closure
4. 1/2-ADM-0712 Shutdown Defense in Depth Assessment
5. 2OM-46.4.F Containment Hydrogen Analyzer - Startup
6. NEI 99-01 Rev. 6 CG1

Category: C – Cold Shutdown / Refueling System Malfunction **CG1.2**

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

- UNPLANNED increase in **ANY** Table 2C-6 Sump/Tank level of sufficient magnitude to indicate core uncovery
- Erratic source range monitor indication
- Containment Radiation Monitor (2RMR-RQ206/207) **> 15 R/hr**

AND

ANY Containment Challenge indication, **Table 2C-1**

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

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

Table 2C-1 Containment Challenge Indications
• CONTAINMENT CLOSURE **not** established (Note 6) • Containment hydrogen concentration **> 4%** • UNPLANNED rise in containment pressure

CG1.2

Table 2C-6 Sump/Tank
• Containment Sumps • Incore Sump • Refueling Water Storage Tank (RWST) • Primary Drains Tank • Pressurizer Relief Tank (PRT) • CCP Surge Tank

Mode Applicability:

5 - Cold Shutdown, 6 – Refueling

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

CG1.2

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

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

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

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

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 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 (ref. 1).

The RCS inventory loss may be detected by the Containment Radiation Monitors or erratic source range monitor indication.

As water level in the reactor vessel lowers, the dose rate above the core will rise. The dose rate due to this core shine should result in Containment Radiation Monitor (CRM) indication > 15 R/hr. Containment radiation is indicated on containment radiation monitors (CRMs) 2RMRRQ206 and 207. These monitors are not located within line of sight of the reactor vessel. The containment radiation monitor alert alarm is set at 6.18E+2 R/hr and high alarm is set at 2.0E+4 R/hr. The alarm setpoints are considered operationally significant, but above what would be expected for a loss of vessel level while in the refuel mode. The CRM threshold values have been established at 15 R/hr (~10x the low scale reading of 1.5 R/hr) to provide a reasonable and conservative indication of abnormal conditions associated with elevated radiation levels in containment due to a loss of water level with irradiated fuel in the vessel.

ATTACHMENT 3:

Unit 2 EAL Technical Bases

CG1.2

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. 2).

Three conditions are associated with a challenge to Containment integrity:

1. CONTAINMENT CLOSURE not established - The status of CONTAINMENT CLOSURE is tracked if plant conditions change that could raise the risk of a fission product release as a result of a loss of decay heat removal (ref. 3, 4, 5). If containment closure is re-established prior to exceeding the 30 minute core uncovery time limit then escalation to GE would not occur.
2. Containment hydrogen > 4% - The 4% hydrogen concentration threshold is generally considered the lower limit for hydrogen deflagrations. Hydrogen monitors, although available at all times, are not in service during normal operations. They are started per 2OM-46.4.F (ref. 6).
3. UNPLANNED rise in Containment pressure - An UNPLANNED pressure rise in containment while in cold Shutdown or Refueling modes can threaten CONTAINMENT CLOSURE capability and thus Containment potentially cannot be relied upon as a barrier to fission product release.

This 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*.

Basis Reference(s):

1. 2OM-53C.4.2.10.1, Loss of Residual Heat Removal Capability
2. Nuclear Safety Analysis Center (NSAC), 1980, “Analysis of Three Mile Island - Unit 2 Accident,” NSAC-1
3. 1/2CMP-47-Contingency Hatch Closure-1M, Contingency Hatch Closure
4. NOP-OP-1005 Shutdown Defense in Depth
5. 1/2-ADM-0712 Shutdown Defense in Depth Assessment
6. 2OM-46.4.F Containment Hydrogen Analyzer - Startup
7. NEI 99-01 Rev. 6 CG1

ATTACHMENT 3:

Unit 2 EAL Technical Bases

Category: C – Cold Shutdown / Refueling System Malfunction **CU2.1**

Subcategory: 2 – Loss of Emergency AC Power

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

EAL:

CU2.1 Unusual Event

AC power capability, **Table 2C-2**, to 4 KV emergency buses 2AE and 2DF reduced to a single power source for **≥ 15 min.** (Note 1)

AND

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

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

Table 2C-2 AC Power Sources
Offsite: • SSST 2A • SSST 2B • USST 2C (while on backfeed) • USST 2D (while on backfeed) **Onsite:** • 2DG1 • 2DG2 • Unit 1 SBO X-Tie (if already aligned)

Mode Applicability:

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

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

CU2.1

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 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 fed from the unaffected unit (SBO crosstie).
- 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.

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

Table 2C-2 provides a list of offsite and onsite AC power sources to the 4KV emergency buses (ref. 1, 2, 3). Credit can be taken for the Unit 1 SBO crosstie only if already aligned due to the time required to establish (> 15min.).

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

The subsequent loss of the remaining single power source would escalate the event to an Alert in accordance with IC CA2. This cold condition EAL is equivalent to the hot condition EAL SA1.1.

Basis Reference(s):

1. BV2 UFSAR Section 8.3 Onsite Power Systems
2. BV2 UFSAR Figure 8.3-1 Main One Line Diagram BVPS Unit No. 2
3. 2OM-53C.4.2.36.2 Loss of 4KV Emergency Bus
4. NEI 99-01 Rev. 6 CU2

ATTACHMENT 3:

Unit 2 EAL Technical Bases

Category: C – Cold Shutdown / Refueling System Malfunction **CA2.1**

Subcategory: 2 – Loss of Emergency AC Power

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

EAL:

CA2.1 Alert
Loss of **ALL** offsite and **ALL** onsite AC power to 4 KV emergency buses 2AE and 2DF for **≥ 15 min.** (Note 1)

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

Mode Applicability:

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

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

Basis Reference(s):

1. BV2 UFSAR Section 8.3 Onsite Power Systems
2. BV2 UFSAR Figure 8.3-1 Main One Line Diagram
3. 2OM-53C.4.2.36.2 Loss of 4KV Emergency Bus
4. NEI 99-01 Rev. 6 CA2

ATTACHMENT 3:

Unit 2 EAL Technical Bases

Category: C – Cold Shutdown / Refueling System Malfunction **CU3.1**

Subcategory: 3 – RCS Temperature

Initiating Condition: UNPLANNED increase in RCS temperature

EAL:

CU3.1	**Unusual Event**
UNPLANNED increase in RCS temperature to **> 200°F** (Note 9)	

Note 9: Begin monitoring hot condition EALs concurrently for any new event or condition not related to the loss of decay heat removal.

Mode Applicability:

5 - Cold Shutdown, 6 - Refueling

Basis:

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

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

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

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

The following instrumentation is capable of providing indication of an RCS temperature rise that approaches the Technical Specification Cold Shutdown temperature limit of (200° F) (ref. 1, 2, 3):

- CET's (incore thermocouples)
- RCS Wide Range Hot Leg Instruments
- RCS Wide Range Cold Leg Instruments
- RHR System Inlet Temperature

CU3.1

The note is a reminder that any temperature increase above 200ºF is an operating mode change from cold to hot conditions. Since each EAL is associated with operating mode applicability, the set of EALs that must be monitored must now include EALs associated with hot condition operating modes.

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.

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

Basis Reference(s):

1. Technical Specifications Table 1.1-1
2. 2OM-53C.4.2.10.1 Loss of Residual Heat Removal Capability
3. 2OM-53C.4.2.10.2 Loss of RHR While Operating at Reduced InventoryMidloop Conditions Attachment 2 Required RCS Water Level for Reduced Inventory/Midloop
4. NEI 99-01 Rev. 6 CU3

Category: C – Cold Shutdown / Refueling System Malfunction **CU3.2**

Subcategory: 3 – RCS Temperature

Initiating Condition: UNPLANNED increase in RCS temperature

EAL:

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

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

Mode Applicability:

5 - Cold Shutdown, 6- Refueling

Basis:

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

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

The following instrumentation is capable of providing indication of an RCS temperature rise that approaches the Technical Specification Cold Shutdown temperature limit of (200° F) (ref. 1, 2, 3):

- CET's (incore thermocouples)
- RCS Wide Range Hot Leg Instruments
- RCS Wide Range Cold Leg Instruments
- RHR System Inlet Temperature

The following instrumentation would be available to provide RCS level:

- PZR Cold Cal Level (2RCS-LT462)
- Temporary Level Instrument (2RCS-LT102)
- Temporary Level Instrument (2RCS-LT105)

ATTACHMENT 3:

Unit 2 EAL Technical Bases

CU3.2

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

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

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.

Basis Reference(s):

1. Technical Specifications Table 1.1-1
2. 2OM-53C.4.2.10.1 Loss of Residual Heat Removal Capability
3. 2OM-53C.4.2.10.2 Loss of RHR While Operating at Reduced InventoryMidloop Conditions Attachment 2 Required RCS Water Level for Reduced Inventory/Midloop
4. NEI 99-01 Rev. 6 CU3

Category: C – Cold Shutdown / Refueling System Malfunction **CA3.1**

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 2C-3** duration (Notes 1, 9)

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

Note 9: Begin monitoring hot condition EALs concurrently for any new event or condition not related to the loss of decay heat removal.

<table>
<tr><th colspan="3">Table 2C-3: RCS Heat-up Duration Thresholds</th></tr>
<tr><th>RCS Status</th><th>CONTAINMENT CLOSURE Status</th><th>Heat-up Duration</th></tr>
<tr><td>Intact (but not Reduced Inventory)</td><td>N/A</td><td>60 min.*</td></tr>
<tr><td rowspan="2">Not intact
OR
Reduced Inventory</td><td>Established</td><td>20 min.*</td></tr>
<tr><td>Not established</td><td>0 min.</td></tr>
<tr><td colspan="3">* If an RCS heat removal system is in operation within this time frame and RCS temperature is being reduced, the EAL is not applicable.</td></tr>
</table>

Mode Applicability:

5 - Cold Shutdown, 6 – Refueling

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.

CA3.1

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 following instrumentation is capable of providing indication of an RCS temperature rise that approaches the Technical Specification Cold Shutdown temperature limit of (200° F) (ref. 1, 2, 3):

- CET's (incore thermocouples)
- RCS Wide Range Hot Leg Instruments
- RCS Wide Range Cold Leg Instruments
- RHR System Inlet Temperature

The status of CONTAINMENT CLOSURE is tracked if plant conditions change that could raise the risk of a fission product release as a result of a loss of decay heat removal (ref. 4, 5).

The note is a reminder that any temperature increase above 200ºF is an operating mode change from cold to hot conditions. Since each EAL is associated with operating mode applicability, the set of EALs that must be monitored must now include EALs associated with hot condition operating modes.

In the absence of reliable RCS temperature indication caused by the loss of decay heat removal capability, classification should be based on the RCS pressure increase criteria when the RCS is intact in Mode 5 or based on time to boil data when in Mode 6 or the RCS is not intact in Mode 5 (ref. 3).

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

Basis Reference(s): **CA3.1**

1. Technical Specifications Table 1.1-1
2. 2OM-53C.4.2.10.1 Loss of Residual Heat Removal Capability
3. 2OM-53C.4.2.10.2 Loss of RHR While Operating at Reduced InventoryMidloop Conditions Attachment 2 Required RCS Water Level for Reduced Inventory/Midloop
4. NOP-OP-1005 Shutdown Defense in Depth
5. 1/2-ADM-0712 Shutdown Defense in Depth Assessment
6. NEI 99-01 Rev. 6 CA3

ATTACHMENT 3:

Unit 2 EAL Technical Bases

Category: C – Cold Shutdown / Refueling System Malfunction **CA3.2**

Subcategory: 3 – RCS Temperature

Initiating Condition: Inability to maintain plant in cold shutdown

EAL:

CA3.2 Alert

RCS temperature **cannot** be monitored

AND

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

Mode Applicability:

5 - Cold Shutdown, 6 – Refueling

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.

This EAL provides a pressure-based indication of RCS heat-up in the absence of RCS tepmerature instrumentation.

A 10 psig RCS pressure increase can be monitored on RCS Narrow Range or RHR Pressure Instruments (ref. 2).

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

Basis Reference(s):

1. 2OM-53C.4.2.10.2 Loss of RHR While Operating at Reduced InventoryMidloop Conditions Attachment 2 Required RCS Water Level for Reduced Inventory/Midloop
2. NEI 99-01 Rev. 6 CA3

ATTACHMENT 3:

Unit 2 EAL Technical Bases

Category: C – Cold Shutdown / Refueling System Malfunction **CU4.1**

Subcategory: 4 – Loss of Vital DC Power

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

EAL:

CU4.1	**Unusual Event**
Bus voltage indications on Technical Specification **required** 125 VDC buses **< 111 VDC** for **≥ 15 min.** (Note 1)	

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

Mode Applicability:

5 - Cold Shutdown, 6 - Refueling

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.

This EAL is intended to be anticipatory in as much as the operating crew may not have necessary indication and control of equipment needed to respond to the loss. Fifteen minutes was selected as a threshold to exclude transient or momentary power losses.

The safety-related 125 VDC Power Distribution System is composed of the following (ref. 1, 2):

- two 1700 amp-hour [BAT-2-1 & 2-2] + two 1140 amp-hour [BAT-2-3 & 2-4] batteries
- four 100 amp battery chargers
- four 125 VDC DC Switchboards [DC-SWBD2-1, 2-2, 2-3 & 2-4]
- ten 125 VDC distribution panels (four each for [DC-SWBD2-1 & 2-2] and one each for [DC-SWBD2-3 & 2-4])

CU4.1

The system also supports a 120 VAC Vital Bus System (that powers vital plant instrumentation), which is powered from 125 VDC / 120 VAC inverters (or by rectified 480 VAC power being inverted, when AC power is available).

The 125 VDC and 120 VAC Vital Bus Systems are designed to provide redundant and reliable power to components and systems that are essential to plant safety, including the Reactor Protective System (RPS) and the Engineered Safety Feature Actuation System (ESFAS) (ref. 3).

The station batteries supply essential and nonessential 125 VDC loads and distribution panels during a loss of the battery charger supply. The batteries are sized to supply the station DC and AC vital bus loads for a period of 2 hours without AC power (ref. 2).

The nominal 60 cell station batteries are rated at 1700 amp-hour capacity [BAT-2-1 & 2-2] or 1140 amp-hour capacity [BAT-2-3 & 2-4] to an end voltage of 1.84 volts per cell, i.e., 110.4 VDC battery voltage. The 110.4 value is rounded to 111 VDC to eliminate the decimal point, since the instrument cannot read this level of accuracy (ref. 2).

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

Basis Reference(s):

1. Technical Specification Bases 3.8.5 DC Sources - Shutdown
2. BV2 UFSAR Section 8.3.2 DC Power Systems
3. Technical Specification Bases 3.8.8 Inverters - Shutdown
4. 2DBD-39 Design Basis Document for 125 VDC Power System
5. NEI 99-01 Rev. 6 CU4

Category: C – Cold Shutdown / Refueling System Malfunction **CU5.1**

Subcategory: 5 – Loss of Communications

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

EAL:

CU5.1 Unusual Event

Loss of **ALL Table 2C-4** onsite communication methods

Table 2C-4 Communication Methods			
System	**Onsite**	**ORO**	**NRC**
Station Page Party Telephone System (Gaitronics)	X		
BVPS Industrial Radios	X	X	
Plant Telephone (PAX)	X	X	X
Commercial Telephones (hardwired & wireless)	X	X	X
Emergency Telephone System (ETS)			X

Mode Applicability:

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

Basis:

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

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

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

Onsite communications include one or more of the systems listed in Table 2C-4 (ref. 1).

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

CU5.1

Basis Reference(s):

1. BVPS Emergency Plan Section 7.6 Communications
2. NEI 99-01 Rev. 6 CU5

Category: C – Cold Shutdown / Refueling System Malfunction **CU5.2**

Subcategory: 5 – Loss of Communications

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

EAL:

CU5.2 Unusual Event

Loss of **ALL Table 2C-4** Offsite Response Organizations (ORO) communication methods

Table 2C-4 Communication Methods			
System	**Onsite**	**ORO**	**NRC**
Station Page Party Telephone System (Gaitronics)	X		
BVPS Industrial Radios	X	X	
Plant Telephone (PAX)	X	X	X
Commercial Telephones (hardwired & wireless)	X	X	X
Emergency Telephone System (ETS)			X

Mode Applicability:

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

Basis:

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

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

CU5.2

EAL #2 addresses a total loss of the communications methods used to notify all OROs of an emergency delcatation. The OROs refrred to here are the EOCs for the States of Pennsylvania, Ohio, West Virginia and counties of Beaver, Columbiana and Hancock.

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

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

Basis Reference(s):

1. BVPS Emergency Plan Section 7.6 Communications
2. NEI 99-01 Rev. 6 CU5

ATTACHMENT 3:

Unit 2 EAL Technical Bases

Category: C – Cold Shutdown / Refueling System Malfunction **CU5.3**

Subcategory: 5 – Loss of Communications

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

EAL:

CU5.3 Unusual Event

Loss of **ALL Table 2C-4** NRC communication methods

Table 2C-4 Communication Methods			
System	**Onsite**	**ORO**	**NRC**
Station Page Party Telephone System (Gaitronics)	X		
BVPS Industrial Radios	X	X	
Plant Telephone (PAX)	X	X	X
Commercial Telephones (hardwired & wireless)	X	X	X
Emergency Telephone System (ETS)			X

Mode Applicability:

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

Basis:

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

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

CU5.3

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

NRC communications include one or more of the systems listed in Table 2C-4 (ref. 1).

This EAL is the cold condition equivalent of the hot condition EAL SU7.3.

Basis Reference(s):

1. BVPS Emergency Plan Section 7.6 Communications
2. NEI 99-01 Rev. 6 CU5

Category: C – Cold Shutdown / Refueling System Malfunction **CA6.1**

Subcategory: 6 – Hazardous Event Affecting Safety Systems

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

EAL:

CA6.1 Alert

The occurrence of **ANY Table 2C-5** hazardous event

AND

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

AND EITHER:

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

(Notes 15, 16)

Note 15: If the affected SAFETY SYSTEM train was already inoperable or out of service before the hazardous event occurred, then this emergency classification is not warranted.

Note 16: If the hazardous event only resulted in VISIBLE DAMAGE, with no indications of degraded performance to at least one train of a SAFETY SYSTEM, then this emergency classification is not warranted.

Table 2C-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

CA6.1

Mode Applicability:

5 - Cold Shutdown, 6 - Refueling

Basis:

This IC addresses a hazardous event that causes damage to SAFETY SYSTEMS needed for the current operating mode. In order to provide the appropriate context for consideration of an ALERT classification, the hazardous event must have caused indications of degraded SAFETY SYSTEM performance in one train, and there must be either indications of performance issues with the second SAFETY SYSTEM train or VISIBLE DAMAGE to the second train such that the potential exists for this second SAFETY SYSTEM train to have performance issues. In other words, in order for this EAL to be classified, the hazardous event must occur, at least one SAFETY SYSTEM train must have indications of degraded performance, and the second SAFETY SYSTEM train must have indications of degraded performance or VISIBLE DAMAGE such that the potential exists for performance issues. Note that this second SAFETY SYSTEM train is from the same SAFETY SYSTEM that has indications of degraded performance for the first AND EITHER statement of this EAL; commercial nuclear power plants are designed to be able to support single system issues without compromising public health and safety from radiological events.

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

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

- The Operating Basis Earthquake is 0.06g. It is the conservatively determined earthquake and associated ground motion that might reasonably or probably be expected to occur at the nuclear plant site. Control Room alarm indication of an earthquake greater than OBE is indicated on the seismic monitoring system cabinet 2ERS-CCC-1. 1/2OM-53C.4A.75.3 Acts of Nature - Seismic provides the guidance for determining if the OBE earthquake threshold is exceeded and any required response actions (ref. 1). The signficance of seismic events are discussed under EAL HU2.1.
- Internal flooding may be caused by events such as component failures, equipment misalignment, or outage activity mishaps.
- External flooding may be due to river level (ref. 2, 3).

CA6.1

- Seismic Category I structures are analyzed to withstand a sustained, design wind velocity of at least 80 mph. (ref. 4, 5).
- Areas containing functions and systems required for safe shutdown of the plant are identified by fire area (ref. 6, 7).

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

Basis Reference(s):

1. 1/2OM-53C.4A.75.3 Acts of Nature Seismic Event
2. 1/2OM-53C.4A.75.2 Acts of Nature - Flood
3. 1/2OM-53C.4A.75.4 Acts of Nature – Dam Failure
4. 1/2OM-53C.4A.75.1 Acts of Nature – Severe Weather
5. BV2 UFSAR Section 3.3.1.1 Wind Loadings
6. BV2 UFSAR Table 3.2-1 QA Category I and Seismic Catergory I Systems and Components
7. BV2 UFSAR Table 3.2-2 QA Classification of Structures
8. NEI 99-01 Rev. 6 CA6

ATTACHMENT 3:

Unit 2 EAL Technical Bases

Category H – Hazards and Other Conditions Affecting Plant Safety

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

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

1. Security

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

2. Seismic Event

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

3. Natural or Technology Hazard

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

4. Fire

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

5. Hazardous Gas

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

6. Control Room Evacuation

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

7. Emergency Director Judgment

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

ATTACHMENT 3:

Unit 2 EAL Technical Bases

Category: H – Hazards **HU1.1**

Subcategory: 1 – Security

Initiating Condition: Confirmed SECURITY CONDITION or threat

EAL:

HU1.1	**Unusual Event**
A SECURITY CONDITION that does **not** involve a HOSTILE ACTION as reported by the Security Shift Supervisor	

Mode Applicability:

All

Basis:

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

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

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

This EAL references the Shift Security Supervisor because these are the individuals trained to confirm that a security event is occurring or has occurred. Training on security event confirmation and classification is controlled due to the nature of Safeguards and10 CFR § 2.39 information.

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

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

Basis Reference(s):

1. BVPS Physical Security Plan/Contingency Plan (Safeguards)
2. NEI 99-01 Rev. 6 HU1

ATTACHMENT 3:

Unit 2 EAL Technical Bases

Category: H – Hazards **HU1.2**

Subcategory: 1 – Security

Initiating Condition: Confirmed SECURITY CONDITION or threat

EAL:

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

Mode Applicability:

All

Basis:

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

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

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

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

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

Basis Reference(s):

1. BVPS Physical Security Plan/Contingency Plan (Safeguards)
2. NEI 99-01 Rev. 6 HU1

ATTACHMENT 3:

Unit 2 EAL Technical Bases

Category: H – Hazards **HU1.3**

Subcategory: 1 – Security

Initiating Condition: Confirmed SECURITY CONDITION or threat

EAL:

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

Mode Applicability:

All

Basis:

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

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

Security plans and terminology are based on the guidance provided by NEI 03-12, Template for the Security Plan, Training and Qualification Plan, Safeguards Contingency.

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

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

Basis Reference(s):

1. BVPS Physical Security Plan/Contingency Plan (Safeguards)
2. NEI 99-01 Rev. 6 HU1

ATTACHMENT 3:

Unit 2 EAL Technical Bases

Category: H – Hazards **HA1.1**

Subcategory: 1 – Security

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

EAL:

HA1.1 **Alert**
A HOSTILE ACTION is occurring or has occurred within the OWNER CONTROLLED AREA as reported by the Security Shift Supervisor

Mode Applicability:

All

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 Supervisor and the Control Room is essential for proper classification of a security-related event.

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

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

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

This EAL is applicable for any HOSTILE ACTION occurring, or that has occurred, in the OWNER CONTROLLED AREA.

HA1.1

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 BVPS Physical Security Plan/Contingency Plan (ref. 1).

Escalation of the emergency classification level would be via HS1.

Basis Reference(s):

1. BVPS Physical Security Plan/Contingency Plan (Safeguards)
2. NEI 99-01 Rev. 6 HA1

ATTACHMENT 3:

Unit 2 EAL Technical Bases

Category: H – Hazards **HA1.2**

Subcategory: 1 – Security

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

EAL:

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

Mode Applicability:

All

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 Supervisor and the Control Room is essential for proper classification of a security-related event.

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

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

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

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

HA1.2

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

In some cases, it may not be readily apparent if an aircraft impact within the 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 BVPS Physical Security Plan/Contingency Plan (ref. 1).

Escalation of the emergency classification level would be via HS1.

Basis Reference(s):

1. BVPS Physical Security Plan/Contingency Plan (Safeguards)
2. NEI 99-01 Rev. 6 HA1

ATTACHMENT 3:

Unit 2 EAL Technical Bases

Category: H – Hazards **HS1.1**

Subcategory: 1 – Security

Initiating Condition: HOSTILE ACTION within the PROTECTED AREA

EAL:

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

Mode Applicability:

All

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 the Security Shift Supervisor and the Control Room is essential for proper classification of a security-related event.

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

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

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

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

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

HS1.1

Basis Reference(s):

1. BVPS Physical Security Plan/Contingency Plan (Safeguards)
2. NEI 99-01 Rev. 6 HS1

ATTACHMENT 3:

Unit 2 EAL Technical Bases

Category: H – Hazards and Other Conditions Affecting Plant Safety **HU2.1**

Subcategory: 2 – Seismic Event

Initiating Condition: Seismic event greater than OBE level

EAL:

HU2.1	**Unusual Event**
Seismic event **> OBE (> 0.06g)** as indicated by lit lamp on 2ERS-CCC-1 Seismic Instrumentation Central Control Cabinet	

Mode Applicability:

All

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.

Seismic events of this magnitude require plant shutdown and evaluation to determine if any damage to plant SSCs has occurred. The post seismic condition of the plant is determined by plant walkdowns and monitoring of plant perimeters to determine if damage has occurred to plant safety systems.

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 0.06g). The Shift Manager or Emergency Director may seek external verification if deemed appropriate (e.g., a call to the USGS, check internet news sources, etc.); however, the verification action must not preclude a timely emergency declaration.

The Operating Basis Earthquake is 0.06g. It is the conservatively determined earthquake and associated ground motion that might reasonably or probably be expected to occur at the nuclear plant site.

1/2OM-53C.4A.75.3 Acts of Nature - Seismic provides the guidance for determining if the OBE earthquake threshold is exceeded and any required response actions (ref. 2).

HU2.1

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.

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

Basis Reference(s):

1. BV2 UFSAR Section 2.5.2 Vibratory Ground Motion
2. 1/2OM-53C.4A.75.3 Acts of Nature – Seismic Event
3. NEI 99-01 Rev. 6 HU2

ATTACHMENT 3:

Unit 2 EAL Technical Bases

Category: H – Hazards and Other Conditions Affecting Plant Safety **HU3.1**

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

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.

Response actions associated with a tornado onsite is provided in 1/2OM-53C.4A.75.1 Acts of Nature – Severe Weather (ref. 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.

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.

Basis Reference(s):

1. 1/2OM-53C.4A.75.1 Acts of Nature – Severe Weather
2. NEI 99-01 Rev. 6 HU3

ATTACHMENT 3:

Unit 2 EAL Technical Bases

Category: H – Hazards and Other Conditions Affecting Plant Safety **HU3.2**

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 (Note 13)	

Note 13: Flooding refers to 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.

Mode Applicability:

All

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.

Depending upon the plant mode at the time of the event, refer to EAL CA6.1 or SA9.1 for internal flooding affecting one or more SAFETY SYSTEM trains.

Basis Reference(s):

1. NEI 99-01 Rev. 6 HU3

ATTACHMENT 3:

Unit 2 EAL Technical Bases

Category: H – Hazards and Other Conditions Affecting Plant Safety **HU3.3**

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) (Notes 12 and 14)

Note 12: 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).

Note 14: As used here, the term “offsite” is meant to be areas external to the BVPS PROTECTED AREA.

Mode Applicability:

All

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.

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

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

Basis Reference(s):

1. NEI 99-01 Rev. 6 HU3

ATTACHMENT 3:

Unit 2 EAL Technical Bases

Category: H – Hazards and Other Conditions Affecting Plant Safety **HU3.4**

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

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.

Basis Reference(s):

1. NEI 99-01 Rev. 6 HU3

ATTACHMENT 3:

Unit 2 EAL Technical Bases

Category: H – Hazards and Other Conditions Affecting Plant Safety **HU4.1**

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 (Note 11)
- Field verification of a single fire alarm (Note 11)

AND

The FIRE is located within **ANY Table 2H-1** area

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

Note 11: Incipient Fire Detection alarms are **not** considered control room fire alarms for this EAL.

Table 2H-1 Safe Shutdown Fire Areas
• Cable Vault and Rod Control Bldg
• Containment Building
• Control Building
• Demin. Water Storage (2FWE-TK210)
• Diesel Generator Building
• Fuel Handling Building
• Intake Structure Pump Cubicles
• Main Steam Valve Room
• Primary Aux. Building (except elev. 773')
• RWST (2QSS-TK21)
• Safeguards Building
• Service Building (except FW Reg Vlv Rm)

Mode Applicability:

All

Basis: **HU4.1**

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

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

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

Table 2H-1 applies to buildings and areas housing equipment needed for safe shutdown (SAFETY SYSTEMS) (ref. 1, 2). The list includes the structures containing the equipment for safe shutdown, certain structures may contain equipment not needed if the plant is already in a shutdown mode.

Incipient Fire Detection alarms are **not** considered control room fire alarms for this EAL. The purpose of Incipient Fire Detection is to detect conditions days/weeks before any FIRE develops.

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.

HU4.1

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

Basis Reference(s):

1. BV2 UFSAR Table 3.2-1 QA Catergory I Structures and Systems Category I Systems and Components
2. BV2 UFSAR Table 3.2-2 QA Classification of Structures
3. NEI 99-01 Rev. 6 HU4

Category: H – Hazards and Other Conditions Affecting Plant Safety **HU4.2**

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) (Note 11)

AND

The fire alarm is indicating a FIRE within **ANY Table 2H-1** area (Note 11)

AND

The existence of a FIRE is **not** verified within **30 min.** of alarm receipt (Notes 1, 11)

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

Note 11: Incipient Fire Detection alarms are **not** considered control room fire alarms for this EAL.

Table 2H-1 Safe Shutdown Fire Areas
• Cable Vault and Rod Control Bldg • Containment Building • Control Building • Demin. Water Storage (2FWE-TK210) • Diesel Generator Building • Fuel Handling Building • Intake Structure Pump Cubicles • Main Steam Valve Room • Primary Aux. Building (except elev. 773’) • RWST (2QSS-TK21) • Safeguards Building • Service Building (except FW Reg Vlv Rm)

Mode Applicability:

All

HU4.2

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.

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 2H-1 applies to buildings and areas housing equipment needed for safe shutdown (SAFETY SYSTEMS) (ref. 1, 2). The list includes the structures containing the equipment for safe shutdown, certain structures may contain equipment not needed if the plant is already in a shutdown mode.

Incipient Fire Detection alarms are **not** considered control room fire alarms for this EAL. The purpose of Incipient Fire Detection is to detect conditions days/weeks before any FIRE develops.

ATTACHMENT 3:

Unit 2 EAL Technical Bases

HU4.2

Basis-Related Requirements from Appendix R

Appendix R to 10 CFR 50, states in part:

> Criterion 3 of Appendix A to this part specifies that "Structures, systems, and components important to safety shall be designed and located to minimize, consistent with other safety requirements, the probability and effect of fires and explosions."
>
> When considering the effects of fire, those systems associated with achieving and maintaining safe shutdown conditions assume major importance to safety because damage to them can lead to core damage resulting from loss of coolant through boil-off.
>
> Because fire may affect safe shutdown systems and because the loss of function of systems used to mitigate the consequences of design basis accidents under post-fire conditions does not per se impact public safety, the need to limit fire damage to systems required to achieve and maintain safe shutdown conditions is greater than the need to limit fire damage to those systems required to mitigate the consequences of design basis accidents.

In addition, Appendix R to 10 CFR 50, requires, among other considerations, the use of 1-hour fire barriers for the enclosure of cable and equipment and associated non-safety circuits of one redundant train (G.2.c). As used in HU4.2, the 30-minutes to verify a single alarm is well within this worst-case 1-hour time period.

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

Basis Reference(s):

1. BV2 UFSAR Table 3.2-1 QA Catergory I Structures and Systems Category I Systems and Components
2. BV2 UFSAR Table 3.2-2 QA Classification of Structures
3. NEI 99-01 Rev. 6 HU4

ATTACHMENT 3:

Unit 2 EAL Technical Bases

Category: H – Hazards and Other Conditions Affecting Plant Safety **HU4.3**

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 (Notes 1, 11)

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

Note 11: Incipient Fire Detection alarms are **not** considered control room fire alarms for this EAL.

Mode Applicability:

All

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.

Incipient Fire Detection alarms are **not** considered control room fire alarms for this EAL. The purpose of Incipient Fire Detection is to detect conditions days/weeks before any FIRE develops.

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

Basis Reference(s):

1. NEI 99-01 Rev. 6 HU4

ATTACHMENT 3:

Unit 2 EAL Technical Bases

Category: H – Hazards and Other Conditions Affecting Plant Safety **HU4.4**

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

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.

Basis Reference(s):

1. NEI 99-01 Rev. 6 HU4

ATTACHMENT 3:

Unit 2 EAL Technical Bases

Category: H – Hazards and Other Conditions Affecting Plant Safety **HA5.1**

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 2H-2** rooms or areas

AND

Entry into the room or area is prohibited or impeded (Notes 5, 12)

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 12: 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).

Table 2H-2 Safe Operation & Shutdown Rooms/Areas

Room/Area	Mode Applicability
Control Room	All
Rod Control Building 735’	3, 4

Mode Applicability:

Refer to Table 2H-2 for Mode Applicability

Basis:

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

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

HA5.1

Evaluation of the IC and EAL do not require atmospheric sampling; it only requires the Emergency Director judgment that the gas concentration in the affected room/area is sufficient to preclude or significantly impede procedurally required access. This judgment may be based on a variety of factors including an existing job hazard analysis, report of ill effects on personnel, advice from a subject matter expert or operating experience with the same or similar hazards. Access should be considered as impeded if extraordinary measures are necessary to facilitate entry of personnel into the affected room/area (e.g., requiring use of protective equipment, such as SCBAs, that is not routinely employed).

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

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

An asphyxiant is a gas capable of reducing the level of oxygen in the body to dangerous levels. Most commonly, asphyxiants work by merely displacing air in an enclosed environment. This reduces the concentration of oxygen below the normal level of around 19%, which can lead to breathing difficulties, unconsciousness or even death.

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

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).

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

HA5.1

BVPS Basis Reference(s):

1. EPLAN, Section 4, Attachment 5 Safe Operation & Shutdown Areas RA3.2 & HA5.1 Bases
2. NEI 99-01 Rev. 6 HA5

ATTACHMENT 3:

Unit 2 EAL Technical Bases

Category: H – Hazards and Other Conditions Affecting Plant Safety **HA6.1**

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 Emergency Shutdown Panel (SDP) or Alternate Shutdown Panel (ASP)	

Mode Applicability:

All

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.

2OM-53C.4.2.33.1A and 2OM-56C.4.2.33.1A specify conditions under which CONTROL ROOM evacuation may be necessary. This EAL is only applicable when the decision has been made to evacuate the CONTROL ROOM, not when conditions are being evaluated per 2OM-53C.4.2.33.1A or 2OM-56C.4.B.
(Ref. 1, 2).

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

Basis Reference(s):

1. 2OM-53C.4.2.33.1A Control Room Inaccessibility
2. 2OM-56C.4.B Alternate Safe Shutdown from Outside Control Room
3. NEI 99-01 Rev. 6 HA6

Category: H – Hazards and Other Conditions Affecting Plant Safety **HS6.1**

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 Emergency Shutdown Panel (SDP) or Alternate Shutdown Panel (ASP)

AND

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

- Reactivity control (modes 1, 2, and 3 only)
- RCS Inventory (inventory control to maintain core cooling)
- RCS heat removal

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

Mode Applicability:

1 - Power Operation, 2 - Startup, 3 - Hot Standby, 4 - Hot Shutdown, 5 – Cold Shutdown, 6 – Refueling

Basis:

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

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

The Shift Manager determines if the Control Room is inoperable and requires evacuation. Control Room inhabitability may be caused by FIRE, dense smoke, noxious fumes, bomb threat in or adjacent to the Control Room, or other life threatening conditions (Ref. 1, 2).

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

HS6.1

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

Basis Reference(s):

1. 2OM-53C.4.2.33.1A Control Room Inaccessibility
2. 2OM-56C.4.B Alternate Safe Shutdown from Outside Control Room
3. NEI 99-01 Rev. 6 HS6

ATTACHMENT 3:

Unit 2 EAL Technical Bases

Category: H – Hazards and Other Conditions Affecting Plant Safety **HU7.1**

Subcategory: 7 – Emergency Director Judgment

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

EAL:

HU7.1 **Unusual Event**

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

Mode Applicability:

All

Basis:

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

Basis Reference(s):

1. NEI 99-01 Rev. 6 HU7

ATTACHMENT 3:

Unit 2 EAL Technical Bases

Category: H – Hazards and Other Conditions Affecting Plant Safety **HA7.1**

Subcategory: 7 – Emergency Director Judgment

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

EAL:

HA7.1 Alert
Other conditions exist which, in the judgment of the Emergency Director, indicate that events are in progress or have occurred which involve an actual or potential substantial degradation of the level of safety of the plant or a security event that involves probable life threatening risk to site personnel or damage to site equipment because of HOSTILE ACTION. Any releases are expected to be limited to small fractions of the EPA Protective Action Guideline exposure levels.

Mode Applicability:

All

Basis:

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

Basis Reference(s):

1. NEI 99-01 Rev. 6 HA7

ATTACHMENT 3:

Unit 2 EAL Technical Bases

Category: H – Hazards and Other Conditions Affecting Plant Safety **HS7.1**

Subcategory: 7 – Emergency Director Judgment

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

EAL:

HS7.1 Site Area Emergency

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

Mode Applicability:

All

Basis:

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

Basis Reference(s):

1. NEI 99-01 Rev. 6 HS7

Category: H – Hazards and Other Conditions Affecting Plant Safety **HG7.1**

Subcategory: 7 – Emergency Director Judgment

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

EAL:

HG7.1 **General Emergency**

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

Mode Applicability:

All

Basis:

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

Basis Reference(s):

1. NEI 99-01 Rev. 6 HG7

ATTACHMENT 3:

Unit 2 EAL Technical Bases

Category S – System Malfunction

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

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

The events of this category pertain to the following subcategories:

1. Loss of Emergency AC Power

Loss of emergency 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 4KV 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 essential plant 125 VDC power sources.

3. Loss of Control Room Indications

Certain events that degrade plant operator's ability to assess plant conditions within the plant warrant emergency classification. Losses of indicators are in this subcategory.

4. RCS Activity

During normal operation, reactor coolant fission product activity is very low. Small concentrations of fission products in the coolant are primarily from the fission of tramp uranium in the fuel clad or minor perforations in the clad itself. Any significant increase from these base-line levels (2% - 5% clad failures) is indicative of fuel failures and is covered under the Fission Product Barrier Degradation category. However, lesser amounts of clad damage may result in coolant activity exceeding Technical Specification limits. These fission products will be circulated with the reactor coolant and can be detected by coolant sampling.

5. RCS Leakage

The reactor vessel provides a volume for the coolant that covers the reactor core. The reactor pressure vessel and associated pressure piping (reactor coolant system) together provide a barrier to limit the release of radioactive material should the reactor fuel clad integrity fail. Excessive RCS leakage greater than Technical Specification limits indicates potential pipe cracks that may propagate to an extent threatening fuel clad, RCS and containment integrity.

ATTACHMENT 3:

Unit 2 EAL Technical Bases

6. RPS Failure

This subcategory includes events related to failure of the Reactor Protection System (RPS) to initiate and complete reactor trips. In the plant licensing basis, postulated failures of the RPS to complete a reactor trip comprise a specific set of analyzed events referred to as Anticipated Transient Without Scram (ATWS) events. For EAL classification, however, ATWS is intended to mean ANY trip failure event that does not achieve reactor shutdown. If RPS actuation fails to assure reactor shutdown, positive control of reactivity is at risk and could cause a threat to fuel clad, RCS and containment integrity.

7. Loss of Communications

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

8. Containment Failure

Failure of containment isolation capability (under conditions in which the containment is not currently challenged) warrants emergency classification. Failure of containment pressure control capability also warrants emergency classification.

9. Hazardous Event Affecting SAFETY SYSTEMS

Various natural and technological events that result in degraded plant SAFETY SYSTEM performance or significant visible

ATTACHMENT 3:

Unit 2 EAL Technical Bases

Category: S – System Malfunction **SU1.1**

Subcategory: 1 – Loss of Emergency AC Power

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

EAL:

SU1.1	**Unusual Event**
Loss of **ALL** offsite AC power capability, **Table 2S-1**, to 4 KV emergency buses 2AE and 2DF for **≥ 15 min.** (Note 1)	

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

Table 2S-1 AC Power Sources
Offsite: • SSST 2A • SSST 2B • USST 2C (while on backfeed) • USST 2D (while on backfeed) **Onsite:** • 2DG1 • 2DG2 • Unit 1 SBO X-Tie (if already aligned)

Mode Applicability:

1 - Power Operation, 2 - Startup, 3 - Hot Standby, 4 – Hot Shutdown

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.

Table 2S-1 provides a list of offsite and onsite AC power sources to the 4KV emergency buses (ref. 1, 2, 3). Credit can be taken for the Unit 1 SBO crosstie only if already aligned due to the time required to establish (> 15min.).

SU1.1

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.

Basis Reference(s):

1. BV2 UFSAR Section 8.3 Onsite Power Systems
2. BV2 UFSAR Figure 8.3-1 Main One Line Diagram
3. 2OM-53C.4.2.36.2 Loss of 4KV Emergency Bus
4. NEI 99-01 Rev. 6 SU1

ATTACHMENT 3:

Unit 2 EAL Technical Bases

Category: S – System Malfunction **SA1.1**

Subcategory: 1 – Loss of Emergency AC Power

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

EAL:

SA1.1 Alert

AC power capability, **Table 2S-1**, to 4 KV emergency buses 2AE and 2DF reduced to a single power source for **≥ 15 min.** (Note 1)

AND

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

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

Table 2S-1 AC Power Sources
Offsite:
• SSST 2A
• SSST 2B
• USST 2C (while on backfeed)
• USST 2D (while on backfeed)
Onsite:
• 2DG1
• 2DG2
• Unit 1 SBO X-Tie (if already aligned)

Mode Applicability:

1 - Power Operation, 2 - Startup, 3 - Hot Standby, 4 - Hot Shutdown

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.

SA1.1

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 fed from the unaffected unit (SBO crosstie).
- 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.

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

Table 2S-1 provides a list of offsite and onsite AC power sources to the 4KV emergency buses (ref. 1, 2, 3). Credit can be taken for the Unit 1 SBO crosstie only if already aligned due to the time required to establish (> 15min.).

Fifteen minutes was selected as a threshold to exclude transient or momentary losses of power. If the capability of a second source of emergency bus power is not restored within 15minutes, an Alert is declared under this EAL.

Escalation of the emergency classification level would be via IC SS1. This hot condition EAL is equivalent to the cold condition EAL CU2.1.

Basis Reference(s):

1. BV2 UFSAR Section 8.3 Onsite Power Systems
2. BV2 UFSAR Figure 8.3-1 Main One Line Diagram
3. 2OM-53C.4.2.36.2 Loss of 4KV Emergency Bus
4. NEI 99-01 Rev. 6 SA1

ATTACHMENT 3:

Unit 2 EAL Technical Bases

Category: S – System Malfunction **SS1.1**

Subcategory: 1 – Loss of Emergency AC Power

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

EAL:

SS1.1	**Site Area Emergency**
Loss of **ALL** offsite and **ALL** onsite AC power to 4 KV emergency buses 2AE and 2DF for **≥ 15 min.** (Note 1)	

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

Mode Applicability:

1 - Power Operation, 2 - Startup, 3 - Hot Standby, 4 - Hot Shutdown

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. The interval begins when both offsite and onsite AC power capability are lost.

Escalation of the emergency classification level would be via ICs RG1, FG1 or SG1.

Basis Reference(s):

1. BV2 UFSAR Section 8.3 Onsite Power Systems
2. BV2 UFSAR Figure 8.3-1 Main One Line Diagram
3. 2OM-53C.4.2.36.2 Loss of 4KV Emergency Bus
4. 2OM-53A.1.ECA-0.0 Loss of All AC Power
5. NEI 99-01 Rev. 6 SS1

ATTACHMENT 3:

Unit 2 EAL Technical Bases

Category: S –System Malfunction **SG1.1**

Subcategory: 1 – Loss of Emergency AC Power

Initiating Condition: Prolonged loss of **all** offsite and **all** onsite AC power to emergency buses

EAL:

SG1.1 General Emergency

Loss of **ALL** offsite and **ALL** onsite AC power to 4 KV emergency buses 2AE and 2DF

AND EITHER:

- Restoration of at least one emergency bus in **< 4 hours** is **not** likely (Note 1)
- Core Cooling RED Path conditions met

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

Mode Applicability:

1 - Power Operation, 2 - Startup, 3 - Hot Standby, 4 - Hot Shutdown

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.

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.

ATTACHMENT 3:

Unit 2 EAL Technical Bases

SG1.1

This EAL is indicated by the extended loss of all offsite and onsite AC power capability to 4 KV emergency buses AE and DF either for greater then the BVPS Station Blackout (SBO) coping analysis time (4 hrs.) (ref. 5) 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. 6).

Four hours is the station blackout coping time (ref 5).

Indication of continuing core cooling degradation must be based on fission product barrier monitoring with particular emphasis on Emergency Director judgment as it relates to IMMINENT loss 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. Critical Safety Function Status Tree (CSFST) Core Cooling-RED path indicates significant core exit superheating and core uncovery (ref. 6).

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.

Basis Reference(s):

1. BV2 UFSAR Section 8.3 Onsite Power Systems
2. BV2 UFSAR Figure 8.3-1 Main One Line Diagram
3. 2OM-53C.4.2.36.2 Loss of 4KV Emergency Bus
4. 2OM-53A.1.ECA-0.0 Loss of All AC Power
5. BV2 Calculation DEC-0246, Coping Duration for Station Black Out
6. 2OM-53A.1.F-0.2 Core Cooling Status Tree
7. NEI 99-01 Rev. 6 SG1

ATTACHMENT 3:

Unit 2 EAL Technical Bases

Category: S –System Malfunction **SG1.2**

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 to 4 KV emergency buses 2AE and 2DF for **≥ 15 min.**

AND

Bus voltage indications on **ALL** safety-related 125 VDC buses (2-1, 2-2, 2-3 and 2-4) **< 111 VDC** for **≥ 15 min.**

(Note 1)

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

Mode Applicability:

1 - Power Operation, 2 - Startup, 3 - Hot Standby, 4 - Hot Shutdown

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.

This EAL is indicated by the loss of all offsite and onsite emergency AC power capability to 4 KV safeguard buses 2AE and 2DF for greater than 15 minutes in combination with degraded vital DC power voltage. This EAL addresses operating experience from the March 2011 accident at Fukushima Daiichi.

The system also supports a 120 VAC Vital Bus System (that powers vital plant instrumentation), which is powered from 125 VDC / 120 VAC inverters (or by rectified 480 VAC power being inverted, when AC power is available).

The 125 VDC and 120 VAC Vital Bus Systems are designed to provide redundant and reliable power to components and systems that are essential to plant safety, including the Reactor Protective System (RPS) and the Engineered Safety Feature Actuation System (ESFAS) (ref. 5).

SG1.2

The station batteries supply essential and nonessential 125 VDC loads and distribution panels during a loss of the battery charger supply. The batteries are sized to supply the station DC and AC vital bus loads for a period of 2 hours without AC power (ref. 6).

The nominal 60 cell station batteries are rated at 1700 amp-hour capacity [BAT-2-1 & 2-2] or 1140 amp-hour capacity [BAT-2-3 & 2-4] to an end voltage of 1.84 volts per cell, i.e., 110.4 VDC battery voltage. The 110.4 value is rounded to 111 VDC to eliminate the decimal point, since the instrument cannot read this level of accuracy (ref. 5, 7).

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.

Basis Reference(s):

1. BV2 UFSAR Section 8.3 Onsite Power Systems
2. BV2 UFSAR Figure 8.3-1 Main One Line Diagram
3. 2OM-53C.4.2.36.2 Loss of 4KV Emergency Bus
4. 2OM-53A.1.ECA-0.0 Loss of All AC Power
5. Technical Specification Bases 3.8.5 DC Sources - Shutdown
6. BV2 UFSAR Section 8.3.2 DC Power Systems
7. Technical Specification Bases 3.8.8 Inverters - Shutdown
8. 2DBD-39 Design Basis Document for 125 VDC Power System
9. NEI 99-01 Rev. 6 SG8

ATTACHMENT 3:

Unit 2 EAL Technical Bases

Category: S – System Malfunction **SS2.1**

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

Bus voltage indications on **ALL** safety-related 125 VDC buses (2-1, 2-2, 2-3 and 2-4) **< 111 VDC** for **≥ 15 min.** (Note 1)

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

Mode Applicability:

1 - Power Operation, 2 - Startup, 3 - Hot Standby, 4 - Hot Shutdown

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.

The system supports a 120 VAC Vital Bus System (that powers vital plant instrumentation), which is powered from 125 VDC / 120 VAC inverters (or by rectified 480 VAC power being inverted, when AC power is available).

The 125 VDC and 120 VAC Vital Bus Systems are designed to provide redundant and reliable power to components and systems that are essential to plant safety, including the Reactor Protective System (RPS) and the Engineered Safety Feature Actuation System (ESFAS) (ref. 3).

The station batteries supply essential and nonessential 125 VDC loads and distribution panels during a loss of the battery charger supply. The batteries are sized to supply the station DC and AC vital bus loads for a period of 2 hours without AC power (ref. 2).

The nominal 60 cell station batteries are rated at 1700 amp-hour capacity [BAT-2-1 & 2-2] or 1140 amp-hour capacity [BAT-2-3 & 2-4] to an end voltage of 1.84 volts per cell, i.e., 110.4 VDC battery voltage. The 110.4 value is rounded to 111 VDC to eliminate the decimal point, since the instrument cannot read this level of accuracy (ref. 2).

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.

ATTACHMENT 3:

Unit 2 EAL Technical Bases

Basis Reference(s): **SS2.1**

1. Technical Specification Bases 3.8.4 DC Sources
2. BV2 UFSAR Section 8.3.2 DC Power Systems
3. Technical Specification Bases 3.8.7 Inverter
4. 2DBD-39 Design Basis Document for 125 VDC Power System
5. NEI 99-01 Rev. 6 SS8

Category:	S – System Malfunction	**SU3.1**
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 2S-2** parameters from within the Control Room for **≥ 15 min.** (Note 1)

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

Table 2S-2 Safety System Parameters
• Reactor power • RCS level • RCS pressure • Core Exit T/C temperature • Level in at least one SG • Auxiliary or emergency feed flow in at least one SG

Mode Applicability:

1 - Power Operation, 2 - Startup, 3 - Hot Standby, 4 - Hot Shutdown

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,

SU3.1

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.

SAFETY SYSTEM parameters listed in Table 2S-2 are monitored in the Control Room through a combination of hard control panel indicators as well as computer based information systems. The Plant Computer, which displays Safety Parameter Display System (SPDS) required information, serves as a redundant compensatory indicator which may be utilized in lieu of normal Control Room indicators (ref. 1, 2).

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.

Basis Reference(s):

1. BV2 UFSAR Section 7.5 Safety Related Display Information
2. 2DBD-05D Design Basis Document for Plant Safety Monitoring System
3. NEI 99-01 Rev. 6 SU2

ATTACHMENT 3:

Unit 2 EAL Technical Bases

Category: S – System Malfunction **SA3.1**

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 2S-2** parameters from within the Control Room for **≥ 15 min.** (Note 1)

AND

ANY significant transient is in progress, **Table 2S-3**

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

Table 2S-2 Safety System Parameters
• Reactor power • RCS level • RCS pressure • Core Exit T/C temperature • Level in at least one SG • Auxiliary or emergency feed flow in at least one SG

Table 2S-3 Significant Transients
• Reactor trip • Automatic turbine runback ≥ 25% thermal power • Electrical load rejection > 25% full electrical load • Safety Injection actuation

Mode Applicability:

1 - Power Operation, 2 - Startup, 3 - Hot Standby, 4 - Hot Shutdown

Basis: **SA3.1**

This IC addresses the difficulty associated with monitoring rapidly changing plant conditions during a transient without the ability to obtain SAFETY SYSTEM parameters from within the Control Room. During this condition, the margin to a potential fission product barrier challenge is reduced. It thus represents a potential substantial degradation in the level of safety of the plant.

As used in this EAL, an "inability to monitor" means that values for one or more of the listed parameters cannot be determined from within the Control Room. This situation would require a loss of all of the Control Room sources for the given parameter(s). For example, the reactor power level cannot be determined from any analog, digital and recorder source within the Control Room.

An event involving a loss of plant indications, annunciators and/or display systems is evaluated in accordance with 10 CFR 50.72 (and associated guidance in NUREG-1022) to determine if an NRC event report is required. The event would be reported if it significantly impaired the capability to perform emergency assessments. In particular, emergency assessments necessary to implement abnormal operating procedures, emergency operating procedures, and emergency plan implementing procedures addressing emergency classification, accident assessment, or protective action decision-making.

This EAL is focused on a selected subset of plant parameters associated with the key safety functions of reactivity control, core cooling and RCS heat removal. The loss of the ability to determine one or more of these parameters from within the Control Room is considered to be more significant than simply a reportable condition. In addition, if all indication sources for one or more of the listed parameters are lost, then the ability to determine the values of other SAFETY SYSTEM parameters may be impacted as well. For example, if the value for reactor vessel level cannot be determined from the indications and recorders on a main control board, the SPDS or the plant computer, the availability of other parameter values may be compromised as well.

SAFETY SYSTEM parameters listed in Table 2S-2 are monitored in the Control Room through a combination of hard control panel indicators as well as computer based information systems. The Plant Computer, which displays Safety Parameter Display System (SPDS) required information, serves as a redundant compensatory indicator which may be utilized in lieu of normal Control Room indicators (ref. 1, 2).

Significant transients are listed in Table S-3 and include response to automatic or manually initiated functions such as reactor trips, runbacks involving greater than or equal to 25% thermal power change, electrical load rejections of greater than 25% full electrical load or ECCS (SI) injection actuations.

SA3.1

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 AS1RS1

Basis Reference(s):

1. BV2 UFSAR Section 7.5 Safety Related Display Information
2. 2DBD-05D Design Basis Document for Plant Safety Monitoring System
3. NEI 99-01 Rev. 6 SA2

ATTACHMENT 3:

Unit 2 EAL Technical Bases

Category: S – System Malfunction **SU4.1**

Subcategory: 4 – RCS Activity

Initiating Condition: Reactor coolant activity greater than Technical Specification allowable limits

EAL:

SU4.1	**Unusual Event**
Letdown Monitor (2CHS-RQ101B) **> 2.98E+03 µCi/cc** (Note 10)	

Note 10: Mode 3 applicable **only** when RCS temperature is ≥ 500ºF

Mode Applicability:

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

Basis:

This EAL addresses reactor coolant letdown line radiation levels sensed by 2CHS-RQ101B in excess of Technical Specification allowable limits (ref. 1). This condition is a precursor to a more significant event and represents a potential degradation of the level of safety of the plant.

This reading is not applicable if letdown is isolated since the monitor isolates with letdown. As such, this reading would be useful only in those events in which safety injection and containment isolation do not actuate.

The 2CHS-RQ101B (high range) calculated EAL value based on 21 µCi/gm dose equivelant I-131 is 2,980 µCi/cc (ref. 2, 3). 2CHS-RQ101A (low range) monitor is off scale at this value.

Escalation of the emergency classification level would be via ICs FA1 or the Recognition Category R ICs.

Basis Reference(s):

1. Technical Specifications Section 3.4.16 RCS Specific Activity
2. ERS-SFL-88-027, Process Safety Limits, Alarm Setpoints and EAL Indicator Value for 2CHS-RQ101A/B
3. 2OM-53C.4.2.6.6 High Reactor Coolant System Activity
4. NEI 99-01 Rev. 6 SU3

Category: S – System Malfunction **SU4.2**

Subcategory: 4 – RCS Activity

Initiating Condition: Reactor coolant activity greater than Technical Specification allowable limits

EAL:

SU4.2	**Unusual Event**
Reactor coolant activity **> 21 µCi/gm** dose equivelant **I**-131 (Note 10)	

Note 10: Mode 3 applicable only when RCS temperature is ≥ 500ºF

Mode Applicability:

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

Basis:

This IC addresses a reactor coolant activity value that exceeds an allowable limit specified in Technical Specifications. This EAL addresses reactor coolant samples exceeding Technical Specification LCOs 3.4.16.A and 3.4.16.B which are applicable in Modes 1, 2, and 3 with T_{avg} ≥ 500ºF (ref. 1, 2). 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.

Basis Reference(s):

1. Technical Specifications Section 3.4.16
2. Technical Specifications Section B3.4.16
3. ERS-SFL-88-027, Process Safety Limits, Alarm Setpoints and EAL Indicator Value for 2CHS-RQ101A/B
4. NEI 99-01 Rev. 6 SU3

Category: S – System Malfunction **SU5.1**

Subcategory: 5 – RCS Leakage

Initiating Condition: RCS leakage for 15 minutes or longer

EAL:

SU5.1	**Unusual Event**
RCS unidentified or pressure boundary leakage **> 10 gpm** for **≥ 15 min.** (Note 1)	

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

Mode Applicability:

1 – Power Operation, 2 – Startup, 3 – Hot Standby, 4 – Hot Shutdown

Basis:

This IC addresses RCS leakage which may be a precursor to a more significant event. In this case, RCS leakage has been detected and operators, following applicable procedures, have been unable to promptly isolate the leak. This condition is considered to be a potential degradation of the level of safety of the plant.

This EAL is focused on a loss of mass from the RCS due to “unidentified leakage”, “pressure boundary leakage” or “identified leakage” (as these leakage types are defined in the plant Technical Specifications). This EAL thus applies 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 this EAL was selected because it is usually observable with normal Control Room indications. Lesser values typically require time-consuming calculations to determine (e.g., a mass balance calculation). This EAL 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).

SU5.1

Unidentified leakage and identified leakage are determined by performance of the RCS water inventory balance. Pressure boundary leakage would first appear as unidentified leakage and can only be positively identified by inspection (ref. 1, 2).

Technical Specifications (ref. 1) defines RCS leakage.

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.

Basis Reference(s):

1. Technical Specifications Section 1.1 Definitions
2. Technical Specifications 3.4.13 RCS Operational Leakge
3. 2OM-53C.4.2.6.7 Excessive Primary Plant Leakage
4. 2OM-53A.1.ECA-1.2 LOCA Outside Containment
5. NEI 99-01 Rev. 6 SU4

Category: S – System Malfunction **SU5.2**

Subcategory: 5 – RCS Leakage

Initiating Condition: RCS leakage for 15 minutes or longer

EAL:

SU5.2	**Unusual Event**
RCS identified leakage **> 25 gpm** for **≥ 15 min.** (Note 1)	

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

Mode Applicability:

1 – Power Operation, 2 – Startup, 3 – Hot Standby, 4 – Hot Shutdown

Basis:

This IC addresses RCS leakage which may be a precursor to a more significant event. In this case, RCS leakage has been detected and operators, following applicable procedures, have been unable to promptly isolate the leak. This condition is considered to be a potential degradation of the level of safety of the plant.

This EAL is focused on a loss of mass from the RCS due to “unidentified leakage”, “pressure boundary leakage” or “identified leakage” (as these leakage types are defined in the plant Technical Specifications). This EAL thus applies 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 this EAL was selected because it is usually observable with normal Control Room indications. Lesser values typically require time-consuming calculations to determine (e.g., a mass balance calculation).

The release of mass from the RCS due to the as-designed/expected operation of a relief valve does not warrant an emergency classification. An emergency classification would be required if a mass loss is caused by a relief valve that is not functioning as designed/expected (e.g., a relief valve sticks open and the line flow cannot be isolated).

Unidentified leakage and identified leakage are determined by performance of the RCS water inventory balance. Pressure boundary leakage would first appear as unidentified leakage and can only be positively identified by inspection (ref. 1, 2).

Technical Specifications (ref. 1) defines RCS leakage.

The 15-minute threshold duration allows sufficient time for prompt operator actions to isolate the leakage, if possible.

SU5.2

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

Basis Reference(s):

1. Technical Specifications Section 1.1 Definitions
2. Technical Specifications 3.4.13 RCS Operational Leakge
3. 2OM-53C.4.2.6.7 Excessive Primary Plant Leakage
4. 2OM-53A.1.ECA-1.2 LOCA Outside Containment
5. NEI 99-01 Rev. 6 SU4

ATTACHMENT 3:

Unit 2 EAL Technical Bases

Category: S – System Malfunction **SU5.3**

Subcategory: 5 – RCS Leakage

Initiating Condition: RCS leakage for 15 minutes or longer

EAL:

SU5.3 Unusual Event

UNISOLABLE leakage from the RCS to a location outside containment **> 25 gpm** for **≥ 15 min.**
(Note 1)

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

Mode Applicability:

1 – Power Operation, 2 – Startup, 3 – Hot Standby, 4 – Hot Shutdown

Basis:

This IC addresses RCS leakage which may be a precursor to a more significant event. In this case, RCS leakage has been detected and operators, following applicable procedures, have been unable to promptly isolate the leak. This condition is considered to be a potential degradation of the level of safety of the plant.

This EAL addresses a RCS mass loss caused by an UNISOLABLE leak through an interfacing system. This EAL thus applies to leakage to a location outside of containment.

The leak rate values for this EAL was selected because it is usually observable with normal Control Room indications. Lesser values typically require time-consuming calculations to determine (e.g., a mass balance calculation).

The release of mass from the RCS due to the as-designed/expected operation of a relief valve does not warrant an emergency classification. An emergency classification would be required if a mass loss is caused by a relief valve that is not functioning as designed/expected (e.g., a relief valve sticks open and the line flow cannot be isolated).

The 15-minute threshold duration allows sufficient time for prompt operator actions to isolate the leakage, if possible.

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 and Primary Sampling system (ref. 3, 4).

Technical Specifications (ref. 1) defines RCS leakage.

ATTACHMENT 3:

Unit 2 EAL Technical Bases

SU5.3

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

Basis Reference(s):

1. Technical Specifications Section 1.1 Definitions
2. Technical Specifications 3.4.13 RCS Operational Leakge
3. 2OM-53C.4.2.6.7 Excessive Primary Plant Leakage
4. 2OM-53A.1.ECA-1.2 LOCA Outside Containment
5. NEI 99-01 Rev. 6 SU4

Category: S – System Malfunction **SU6.1**

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 after **ANY** RPS setpoint is exceeded

AND

A subsequent automatic trip or manual trip action taken at the Control Room Benchboards (reactor trip and bypass switches or tripping the turbine) is successful in shutting down the reactor (Note 8)

Note 8: A manual trip action is any operator action, or set of actions, which causes the control rods to be rapidly inserted into the core, and **does not** include manually driving in control rods or implementation of boron injection strategies.

Mode Applicability:

1 – Power Operation

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 Control Room Benchboards 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 Control Room Benchboards 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 Control Room Benchboards 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.

SU6.1

A manual action at the Control Room Benchboards 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 Control Room Benchboards".

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 Control Room Benchboards 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.

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. (ref. 1, 2).

SU6.1

For the purposes of emergency classification, successful manual trip actions are those which can be quickly performed from the Control Room Benchboards; reactor trip and bypass switches or tripping the turbine. Reactor shutdown achieved by use of other trip actions specified in FR-S.1 Response to Nuclear Power Generation/ATWS (such as manually inserting control rods or emergency boration) do not constitute a successful manual trip (ref. 2).

Following any automatic RPS trip signal, E-0.0 (ref. 1) and FR-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 (ref. 2).

In the event that the operator identifies a reactor trip is IMMINENT and initiates a successful manual reactor trip before the automatic RPS trip setpoint is reached, no declaration is required. The successful manual trip of the reactor before it reaches its automatic trip setpoint or reactor trip signals caused by instrumentation channel failures do not lead to a potential fission product barrier loss. However, if subsequent manual reactor trip actions fail to reduce reactor power, 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.

Basis Reference(s):

1. 2OM-53A.1.E-0 Reactor Trip or Safety Injection
2. 2OM-53A.1.FR-S.1 Response to Nuclear Power Generation – ATWS
3 NEI 99-01 Rev. 6 SU5

Category: S – System Malfunction **SU6.2**

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 after **ANY** manual trip action was initiated

AND

A subsequent automatic trip or manual trip action taken at the Control Room Benchboards (reactor trip and bypass switches or tripping the turbine) is successful in shutting down the reactor (Note 8)

Note 8: A manual trip action is any operator action, or set of actions, which causes the control rods to be rapidly inserted into the core, and **does not** include manually driving in control rods or implementation of boron injection strategies.

Mode Applicability:

1 – Power Operation

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 Control Room Benchboards 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 Control Room Benchboards 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 Control Room Benchboards 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 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 3:

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SU6.2

A manual action at the Control Room Benchboards 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 Control Room Benchboards”.

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 Control Room Benchboards 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.

This EAL addresses a failure of a manually initiated trip in the absence of having exceeded an automatic RTS trip setpoint and a subsequent automatic or manual trip is successful in shutting down the reactor. (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. (ref. 1, 2).

SU6.2

For the purposes of emergency classification, successful manual trip actions are those which can be quickly performed from the Control Room Benchboards; reactor trip and bypass switches or tripping the turbine. Reactor shutdown achieved by use of other trip actions specified in FR-S.1 Response to Nuclear Power Generation/ATWS (such as manually inserting control rods or emergency boration) do not constitute a successful manual trip (ref. 2).

Following the failure of any manual trip signal, E-0.0 (ref. 1) and FR-S.1 (ref. 2) prescribe insertion of redundant manual trip signals to back up the RPS trip function and ensure reactor shutdown is achieved. Even if a subsequent automatic trip signal or the first subsequent manual trip signal inserts all control rods to the full-in position immediately after the initial failure of the manual trip, the lowest level of classification that must be declared is an Unusual Event (ref. 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 following a failure of an initial manual trip, the event escalates to an Alert under EAL SA6.1.

Basis Reference(s):

1. 2OM-53A.1.E-0 Reactor Trip or Safety Injection
2. 2OM-53A.1.FR-S.1 Response to Nuclear Power Generation – ATWS
3. NEI 99-01 Rev. 6 SU5

SA6.1

Category: S – System Malfunction

Subcategory: 6 – RPS Failure

Initiating Condition: Automatic or manual trip fails to shut down the reactor and subsequent manual actions taken at the Control Room Benchboards are not successful in shutting down the reactor

EAL:

SA6.1 Alert
An automatic or manual trip fails to shut down the reactor
AND
Manual trip actions taken at the Control Room Benchboards (reactor trip and bypass switches or tripping the turbine) are **not** successful in shutting down the reactor (Note 8)

Note 8: A manual trip action is any operator action, or set of actions, which causes the control rods to be rapidly inserted into the core, and **does not** include manually driving in control rods or implementation of boron injection strategies.

Mode Applicability:

1 – Power Operation

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 Control Room Benchboards 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 Control Room Benchboards since this event entails a significant failure of the RPS.

A manual action at the Control Room Benchboards 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 Control Room Benchboards (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 Control Room Benchboards”.

SA6.1

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.

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 (ref. 1).

For the purposes of emergency classification, successful manual trip actions are those which can be quickly performed from the Control Room Benchboards; ; reactor trip and bypass switches or tripping the turbine. Reactor shutdown achieved by use of other trip actions specified in FR-S.1 Response to Nuclear Power Generation/ATWS (such as manually inserting control rods or emergency boration) do not constitute a successful manual trip (ref. 2).

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.

Basis Reference(s):

1. 2OM-53A.1.E-0 Reactor Trip or Safety Injection
2. 2OM-53A.1.FR-S.1 Response to Nuclear Power Generation – ATWS
3. NEI 99-01 Rev. 6 SA5

Category: S – System Malfunction **SS6.1**

Subcategory: 6 – RPS Failure

Initiating Condition: Inability to shut down the reactor causing a challenge to core cooling or RCS heat removal

EAL:

SS6.1 Site Area Emergency

An automatic or manual trip fails to shut down the reactor

AND

ALL actions to shut down the reactor are **not** successful

AND EITHER:

- Core Cooling RED Path conditions met
- Heat Sink RED Path conditions met

Mode Applicability:

1 – Power Operation

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 shutdown 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.

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.

SS6.1

The combination of failure of both front line and backup protection systems to function in response to a plant transient, along with the continued production of heat, poses a direct threat to the Fuel Clad and RCS barriers.

Reactor shutdown achieved by use of FR-S.1 Response to Nuclear Power Generation/ATWS (such as manually insert control rods or emergency boration) are also credited as a successful manual trip provided reactor power can be reduced below 5% before indications of an extreme challenge to either core cooling or heat removal exist (ref. 1, 2).

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 core exit T/Cs are reading greater than or equal to 1200ºF or a loss of adequate subcooling with elevated core exit T/Cs and low RVLIS level (ref. 3).

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 based on inadequate steam generator level and feedwater flow (ref. 4).

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.

Basis Reference(s):

1. 2OM-53A.1.E-0 Reactor Trip or Safety Injection
2. 2OM-53A.1.FR-S.1 Response to Nuclear Power Generation – ATWS
3. 2OM-53A.1.F-0.2 Core Cooling Status Tree
4. 2OM-53A.1.F-0.3 Heat Sink Status Tree
5. NEI 99-01 Rev. 6 SS5

ATTACHMENT 3:

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Category: S – System Malfunction **SU7.1**

Subcategory: 7 – Loss of Communications

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

EAL:

SU7.1 Unusual Event

Loss of **ALL Table 2S-4** onsite communication methods

Table 2S-4 Communication Methods			
System	**Onsite**	**ORO**	**NRC**
Station Page Party Telephone System (Gaitronics)	X		
BVPS Industrial Radios	X	X	
Plant Telephone (PAX)	X	X	X
Commercial Telephones (hardwired & wireless)	X	X	X
Emergency Telephone System (ETS)			X

Mode Applicability:

1 – Power Operation, 2 – Startup, 3 – Hot Standby, 4 – Hot Shutdown

Basis:

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

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

SU7.1

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

Onsite communications include one or more of the systems listed in Table 2S-4 (ref. 1, 2).

This EAL is the hot condition equivalent of the cold condition EAL CU5.1.

Basis Reference(s):

1. BVPS Emergency Plan Section 7.6 Communications
2. NEI 99-01 Rev. 6 SU6

Category: S – System Malfunction **SU7.2**

Subcategory: 7 – Loss of Communications

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

EAL:

SU7.2 Unusual Event

Loss of **ALL Table 2S-4** offsite response organization (ORO) communication methods

Table 2S-4 Communication Methods			
System	**Onsite**	**ORO**	**NRC**
Station Page Party Telephone System (Gaitronics)	X		
BVPS Industrial Radios	X	X	
Plant Telephone (PAX)	X	X	X
Commercial Telephones (hardwired & wireless)	X	X	X
Emergency Telephone System (ETS)			X

Mode Applicability:

1 – Power Operation, 2 – Startup, 3 – Hot Standby, 4 – Hot Shutdown

Basis:

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

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

SU7.2

EAL #2 addresses a total loss of the communications methods used to notify all OROs of an emergency declaration. The offsite response organizations (OROs) referred to here are the EOCs for the States of Pennsylvania, Ohio, West Virginia and counties of Beaver, Columbiana and Hancock.

Offsite communications include one or more of the systems listed in Table 2S-4 (ref. 1, 2).

This EAL is the hot condition equivalent of the cold condition EAL CU5.2.

Basis Reference(s):

1. BVPS Emergency Plan Section 7.6 Communications
2. NEI 99-01 Rev. 6 SU6

Category: S – System Malfunction **SU7.3**

Subcategory: 7 – Loss of Communications

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

EAL:

SU7.3 **Unusual Event**
Loss of **ALL Table 2S-4** NRC communication methods

Table 2S-4 Communication Methods			
System	**Onsite**	**ORO**	**NRC**
Station Page Party Telephone System (Gaitronics)	X		
BVPS Industrial Radios	X	X	
Plant Telephone (PAX)	X	X	X
Commercial Telephones (hardwired & wireless)	X	X	X
Emergency Telephone System (ETS)			X

Mode Applicability:

1 – Power Operation, 2 – Startup, 3 – Hot Standby, 4 – Hot Shutdown

Basis:

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

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

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

This EAL is the hot condition equivalent of the cold condition EAL CU5.3.

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SU7.3

Basis Reference(s):

1. BVPS Emergency Plan Section 7.6 Communications
2. NEI 99-01 Rev. 6 SU6

Category: S – System Malfunction **SU8.1**

Subcategory: 8 – Containment Failure

Initiating Condition: Failure to isolate containment or loss of containment pressure control

EAL:

SU8.1 Unusual Event

ANY penetration is not isolated within **15 min.** of a VALID containment isolation signal

OR

Containment pressure **> 11 psig AND** < one full train of depressurization equipment operating per design for **≥ 15 min.**
(Note 1)

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

Mode Applicability:

1 – Power Operation, 2 – Startup, 3 – Hot Standby, 4 – Hot Shutdown

Basis:

This IC 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 or ice condenser fans) are either lost or performing in a degraded manner.

SU8.1

Each unit has a containment pressure quench spray system with two 100% capacity trains. These pumps take suction from the RWST and discharge to the spray header. The quench spray system starts on a CIB at the start of a LOCA accident.

The recirculation spray system has four 50% capacity subsystems that consist of a pump and a cooler. The recirculation spray pump takes suction from the containment sump and discharges through a cooler to the spray header. The recirculation spray system does not start during a LOCA until there is low level in the RWST to verify the sump has adequate water inventory. When the RWST level goes very low the quench spray pumps are secured.

A very short period of time could exist where the quench spray system and the recirculation spray system pumps could both be running. Normally it is either the quench spray or the recirculation spray running.

One train of QS System and one train of RS System comprise one full train of depressurization equipment as designed (ref. 1).

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.

Basis Reference(s):

1. BV2 UFSAR Section 6.2 Containment Systems
2. NEI 99-01 Rev. 6 SU7

Category: S – System Malfunction **SA9.1**

Subcategory: 9 – Hazardous Event Affecting Safety Systems

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

EAL:

SA9.1 Alert

The occurrence of **ANY Table 2S-5** hazardous event

AND

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

AND EITHER:

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

(Notes 15, 16)

Note 15: If the affected SAFETY SYSTEM train was already inoperable or out of service before the hazardous event occurred, then this emergency classification is warranted.

Note 16: If the hazardous event only resulted in VISIBLE DAMAGE, with no indications of degraded performance to at least one train of a SAFETY SYSTEM, then this emergency classification is not warranted.

Table 2S-5 Hazardous Events
• Seismic event (earthquake) • Internal or external flooding event • High winds or tornado strike • FIRE • EXPLOSION • Other events with similar hazard characteristics as determined by the Shift Manager

Mode Applicability:

1 – Power Operation, 2 – Startup, 3 – Hot Standby, 4 – Hot Shutdown

SA9.1

Basis:

This IC addresses a hazardous event that causes damage to SAFETY SYSTEMS needed for the current operating mode. In order to provide the appropriate context for consideration of an ALERT classification, the hazardous event must have caused indications of degraded SAFETY SYSTEM performance in one train, and there must be either indications of performance issues with the second SAFETY SYSTEM train or VISIBLE DAMAGE to the second train such that the potential exists for this second SAFETY SYSTEM train to have performance issues. In other words, in order for this EAL to be classified, the hazardous event must occur, at least one SAFETY SYSTEM train must have indications of degraded performance, and the second SAFETY SYSTEM train must have indications of degraded performance or VISIBLE DAMAGE such that the potential exists for performance issues. Note that this second SAFETY SYSTEM train is from the same SAFETY SYSTEM that has indications of degraded performance for the first AND EITHER statement of this EAL; commercial nuclear power plants are designed to be able to support single system issues without compromising public health and safety from radiological events.

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

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

- The Operating Basis Earthquake is 0.06g. It is the conservatively determined earthquake and associated ground motion that might reasonably or probably be expected to occur at the nuclear plant site. Control Room alarm indication of an earthquake greater than OBE is indicated on the seismic monitoring system cabinet 2ERS-CCC-1. 1/2OM-53C.4A.75.3 Acts of Nature - Seismic provides the guidance for determining if the OBE earthquake threshold is exceeded and any required response actions (ref. 1). The signficance of seismic events are discussed under EAL HU2.1.
- Internal flooding may be caused by events such as component failures, equipment misalignment, or outage activity mishaps.
- External flooding may be due to river level (ref. 2, 3).

SA9.1

- Seismic Category I structures are analyzed to withstand a sustained, design wind velocity of at least 80 mph. (ref. 4, 5).
- Areas containing functions and systems required for safe shutdown of the plant are identified by fire area (ref. 6, 7).

Escalation of the emergency classification level would be via IC FS1 or RS1.

Basis Reference(s):

1. 1/2OM-53-4A.75.3 Acts of Nature Seismic Event
2. 1/2OM-53C.4A.75.2 Acts of Nature – Flood
3. 1/2OM-53C.4A.75.4 Acts of Nature – Dam Failure
4. 1/2OM-53C.4A.75.1 Acts of Nature – Severe Weather
5. BV2 UFSAR Section 3.3.1.1 Design Wind Velocity
6. BV2 UFSAR Table 3.2-1 QA Category I and Seismic Catergory I Systems and Components
7. BV2 UFSAR Table 3.2-2 Classification of Structures
8. BV2 Calculation N-265, Flooding Analysis Outside Containment
9. NEI 99-01 Rev. 6 SA9

ATTACHMENT 3:

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Category F – Fission Product Barrier Degradation

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

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

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

B. Reactor Coolant System (RCS): The RCS Barrier includes the RCS primary side and its connections up to and including the pressurizer safety and relief valves, and other connections up to and including the primary isolation valves.

C. Containment (CT): The Containment Barrier includes the containment building, connections up to and including the outermost containment isolation valves. This barrier also includes the main steam, feedwater, and blowdown line extensions outside the containment building up to and including the outermost secondary side isolation valve. Containment Barrier thresholds are used as criteria for escalation of the ECL from Alert to a Site Area Emergency or a General Emergency.

The EALs in this category require evaluation of the loss and potential loss thresholds listed in the fission product barrier matrix of Table F-1 (Attachment 2). "Loss" and "Potential Loss" signify the relative damage and threat of damage to the barrier. "Loss" means the barrier no longer assures containment of radioactive materials. "Potential Loss" means integrity of the barrier is threatened and could be lost if conditions continue to degrade. The number of barriers that are lost or potentially lost and the following criteria determine the appropriate emergency classification level:

Alert:
ANY loss or ANY potential loss of EITHER Fuel Clad or RCS

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

General Emergency:
Loss of ANY two barriers AND loss or potential loss of third barrier

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

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

- Unusual Event ICs associated with RCS and Fuel Clad Barriers are addressed under System Malfunction ICs.
- For accident conditions involving a radiological release, evaluation of the FISSION PRODUCT BARRIER THRESHOLDS will need to be performed in conjunction with dose assessments to ensure correct and timely escalation of the emergency classification. For example, an evaluation of the FISSION PRODUCT BARRIER THRESHOLDS may result in a Site Area Emergency classification while a dose assessment may indicate that an EAL for General Emergency IC RG1 has been exceeded.
- The FISSION PRODUCT BARRIER THRESHOLDS specified within a scheme reflect plant-specific BVPS design and operating characteristics.
- As used in this category, the term RCS leakage encompasses not just those types defined in Technical Specifications but also includes the loss of RCS mass to any location– inside the containment, an interfacing system, or outside of the containment. The release of liquid or steam mass from the RCS due to the as-designed/expected operation of a relief valve is not considered RCS leakage.
- At the Site Area Emergency level, EAL users should maintain cognizance of how far present conditions are from meeting a threshold that would require a General Emergency declaration. For example, if the Fuel Clad and RCS fission product barriers were both lost, then there should be frequent assessments of containment radioactive inventory and integrity. Alternatively, if both the Fuel Clad and RCS fission product barriers were potentially lost, the Emergency Director would have more assurance that there was no immediate need to escalate to a General Emergency.

Category: Fission Product Barrier Degradation **FA1.1**

Subcategory: N/A

Initiating Condition: **Any** Loss or **any** Potential Loss of either Fuel Clad or RCS

EAL:

FA1.1 Alert
ANY Loss or **ANY** Potential Loss of **EITHER** Fuel Clad or RCS (**Table 2F-1**)

Mode Applicability:

1 – Power Operation, 2 – Startup, 3 – Hot Standby, 4 – Hot Shutdown

Basis:

Fuel Clad, RCS and Containment comprise the fission product barriers. Table 2F-1 (Attachment 4) 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

Basis Reference(s):

2. NEI 99-01 Rev. 6 FA1

ATTACHMENT 3:

Unit 2 EAL Technical Bases

Category: Fission Product Barrier Degradation **FS1.1**

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 2F-1**)	

Mode Applicability:

1 – Power Operation, 2 – Startup, 3 – Hot Standby, 4 – Hot Shutdown

Basis:

Fuel Clad, RCS and Containment comprise the fission product barriers. Table 2F-1 (Attachment 4) lists the FISSION PRODUCT BARRIER THRESHOLDS, bases and references.

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

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

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

Basis Reference(s):

3. NEI 99-01 Rev. 6 FS1

Category: Fission Product Barrier Degradation **FG1.1**

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 2F-1**)

Mode Applicability:

1 – Power Operation, 2 – Startup, 3 – Hot Standby, 4 – Hot Shutdown

Basis:

Fuel Clad, RCS and Containment comprise the fission product barriers. Table 2F-1 (Attachment 4) 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

Basis Reference(s):

4. NEI 99-01 Rev. 6 FG1

ATTACHMENT 4:

Unit 2 Fission Product Barrier Loss/Potential Loss Matrix and Bases

Introduction

Table 2F-1 lists the threshold conditions that define the Loss and Potential Loss of the three fission product barriers (Fuel Clad, Reactor Coolant System, and Containment). The table is structured so that each of the three barriers occupies adjacent columns. Each fission product barrier column is further divided into two columns; one for Loss thresholds and one for Potential Loss thresholds.

The first column of the table (to the left of the Fuel Clad Loss column) lists the categories (types) of FISSION PRODUCT BARRIER THRESHOLDS. The fission product barrier categories are:

A. RCS or SG Tube Leakage

B. Inadequate Heat removal

C. CT Radiation / RCS Activity

D. CT Integrity or Bypass

E. Emergency Director Judgment

Each category occupies a row in Table 2F-1 thus forming a matrix defined by the categories. The intersection of each row with each Loss/Potential Loss column forms a cell in which one or more FISSION PRODUCT BARRIER THRESHOLDS appear. If NEI 99-01 does not define a threshold for a barrier Loss/Potential Loss, the word “None” is entered in the cell.

Thresholds are assigned sequential numbers within each Loss and Potential Loss column beginning with number one. In this manner, a threshold can be identified by its category title and number. For example, the first Fuel Clad barrier Loss in Category A would be assigned “FC Loss A.1,” the third Containment barrier Potential Loss in Category C would be assigned “CT P-Loss C.3,” etc.

If a cell in Table 2F-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 2F-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 2F-1, locates the likely category and then reads across the fission product barrier Loss and Potential Loss thresholds in that category to determine if a threshold has been exceeded. If a threshold has not been exceeded, the EAL-user proceeds to the next likely category and continues review of the thresholds in the new category

If the EAL-user determines that any threshold has been exceeded, by definition, the barrier is lost or potentially lost – even if multiple thresholds in the same barrier column are exceeded, only that one barrier is lost or potentially lost. The EAL-user must examine each of the three

ATTACHMENT 4:

Unit 2 Fission Product Barrier Loss/Potential Loss Matrix and Bases

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 4:
Unit 2 Fission Product Barrier Loss/Potential Loss Matrix and Bases

Table 2F-1 Fission Product Barrier Threshold Matrix						
	Fuel Clad (FC) Barrier		**Reactor Coolant System (RC) Barrier**		**Containment (CT) Barrier**	
Category	**Loss**	**Potential Loss**	**Loss**	**Potential Loss**	**Loss**	**Potential Loss**
A **RCS or SG Tube Leakage**	None	None	1. An automatic or manual ECCS (SI) actuation required by **EITHER:** • UNISOLABLE RCS leakage • SG tube RUPTURE	1. Operation of a standby charging pump is required by **EITHER:** • UNISOLABLE RCS leakage • SG tube leakage **OR** 2. RCS Integrity-RED Path conditions met	1. A leaking or RUPTURED SG is FAULTED outside of containment	None
B **Inadequate Heat Removal**	1. Core Cooling-RED Path conditions met	1. Core Cooling-ORANGE Path conditions met **OR** 2. Heat Sink-RED Path conditions met **AND** Heat sink is required	None	1. Heat Sink-RED Path conditions met **AND** Heat sink is required	None	1. Core Cooling-RED Path conditions met **AND** Restoration procedures **not** effective within **15 min.** (Note 1)
C **CT Radiation / RCS Activity**	1. Containment Radiation Monitor **> Table 2F-2, "FC Loss"** **OR** 2. Dose equivalent I-131 coolant activity **> 300 μCi/gm**	None	**1.** Containment Radiation Monitor **> Table 2F-2, "RC Loss"**	None	None	1. Containment Radiation Monitor **> Table 2F-2, "CT Potential Loss"**
D **CT Integrity or Bypass**	None	None	None	None	1. Containment isolation is required **AND EITHER:** • Containment integrity has been lost based on Emergency Director judgment • UNISOLABLE pathway from Containment to the environment exists **OR** 2. Indications of RCS leakage outside of Containment	1. Containment-RED Path conditions met **OR** 2. Containment hydrogen concentration **> 4%** **OR** 3. Containment pressure **> 11 psig AND <** one full train of depressurization equipment operating per design for **≥ 15 min.** (Note 1)
E **ED Judgment**	1. **ANY** condition in the opinion of the Emergency Director that indicates Loss of the Fuel Clad Barrier	1. **ANY** condition in the opinion of the Emergency Director that indicates Potential Loss of the Fuel Clad Barrier	1. **ANY** condition in the opinion of the Emergency Director that indicates Loss of the RCS Barrier	1. **ANY** condition in the opinion of the Emergency Director that indicates Potential Loss of the RCS Barrier	1. **ANY** condition in the opinion of the Emergency Director that indicates Loss of the Containment Barrier	1. **ANY** condition in the opinion of the Emergency Director that indicates Potential Loss of the Containment Barrier

ATTACHMENT 4:
Unit 2 Fission Product Barrier Loss/Potential Loss Matrix and Bases

Barrier: Fuel Clad **FC.A**

Category: A. RCS or SG Tube Leakage

Degradation Threat: Loss

Threshold:

None

ATTACHMENT 4:
Unit 2 Fission Product Barrier Loss/Potential Loss Matrix and Bases

Barrier: Fuel Clad **FC.A**

Category: A. RCS or SG Tube Leakage

Degradation Threat: Potential Loss

Threshold:

None

ATTACHMENT 4:
Unit 2 Fission Product Barrier Loss/Potential Loss Matrix and Bases

Barrier: Fuel Clad **FC.B**

Category: B. Inadequate Heat Removal

Degradation Threat: Loss

Threshold:

1. Core Cooling-RED Path conditions met

Basis:

This reading indicates temperatures within the core are sufficient to cause significant superheating of reactor coolant.

Critical Safety Function Status Tree (CSFST) Core Cooling-RED Path indicates significant core exit superheating and core uncovery. The CSFSTs are normally monitored using the Safety Parameter Display System (SPDS) display on the Plant Computer (ref. 1).

Basis Reference(s):

1. 2OM-53A.1.F-0.2 Core Cooling Status Tree
2. NEI 99-01 Rev. 6 Inadequate Heat Removal Fuel Clad Loss 2.A

ATTACHMENT 4:
Unit 2 Fission Product Barrier Loss/Potential Loss Matrix and Bases

Barrier: Fuel Clad **FC.B**

Category: B. Inadequate Heat Removal

Degradation Threat: Potential Loss

Threshold:

1. Core Cooling-ORANGE Path conditions met

Basis:

This reading indicates temperatures within the core are sufficient to allow the onset of heat-induced cladding damage.

Critical Safety Function Status Tree (CSFST) Core Cooling-ORANGE path indicates indicates subcooling has been lost and that some fuel clad damage may potentially occur. The CSFSTs are normally monitored using the Safety Parameter Display System (SPDS) display on the Plant Computer (ref. 1, 2).

Basis Reference(s):

1. 2OM-53A.1.F-0.2 Core Cooling Status Tree
2. NEI 99-01 Rev. 6 Inadequate Heat Removal Fuel Clad Loss 2.A

ATTACHMENT 4:
Unit 2 Fission Product Barrier Loss/Potential Loss Matrix and Bases

Barrier: Fuel Clad **FC.B**

Category: B. Inadequate Heat Removal

Degradation Threat: Potential Loss

Threshold:

2. Heat Sink-RED Path conditions met

AND

Heat sink is required

Basis:

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.

Meeting this threshold results in a Site Area Emergency because this threshold is identical to RCS Barrier Potential Loss threshold RC.B.1; 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.

Critical Safety Function Status Tree (CSFST) Heat Sink-RED Path indicates the heat sink function is under extreme challenge and that some fuel clad damage may potentially occur (ref. 1).

The phrase “and heat sink required” precludes the need for classification for conditions in which RCS pressure is less than SG pressure or Heat Sink-RED path entry was created through operator action directed by an ERG. For example, FRH-0.1 is entered from CSFST Heat Sink-Red. Step 1 tells the operator to determine if heat sink is required by checking that RCS pressure is greater than any non-faulted SG pressure and RCS temperature is greater than 350ºF. If these conditions exist, Heat Sink is required. Otherwise, the operator is to either return to the procedure and step in effect and place RHR in service for heat removal. For large LOCA events inside the Containment, the SGs are moot because heat removal through the containment heat removal systems takes place. Therefore, Heat Sink Red should not be required and, should not be assessed for EAL classification because a LOCA event alone should not require higher than an Alert classification. (ref. 2).

The CSFSTs are normally monitored using the Safety Parameter Display System (SPDS) display on the Plant Computer (ref. 1).

FC.B

Basis Reference(s):

1. 2OM-53A.1.F-0.3 Heat Sink Status Tree
2. 2OM-53A.1.FR-H.1 Response to Loss of Secondary Heat Sink
3. NEI 99-01 Rev. 6 Inadequate Heat Removal Fuel Clad Loss 2.B

ATTACHMENT 4:
Unit 2 Fission Product Barrier Loss/Potential Loss Matrix and Bases

Barrier: Fuel Clad **FC.C**

Category: C. CT Radiation / RCS Activity

Degradation Threat: Loss

Threshold:

1. Containment Radiation Monitor **> Table 2F-2, “FC Loss”**

Table 2F-2 Containment Radiation – R/hr (2RMR-RQ206/207)			
Time After S/D (Hrs.)	RC Loss (R/hr)	FC Loss (R/hr)	CT Potential Loss (R/hr)
0-1	11	700	14,000
>1-2	11	490	9,600
>2-8	11	200	3,900
>8-16	11	120	2,400
>16-48	11	63	1200

Basis:

The radiation monitor reading corresponds to an instantaneous release of all reactor coolant mass into the containment, assuming that reactor coolant activity equals 300 μCi/gm dose equivalent I-131. Reactor coolant activity above this level is greater than that expected for iodine spikes and corresponds to an approximate range of 2% to 5% fuel clad damage. Since this condition indicates that a significant amount of fuel clad damage has occurred, it represents a loss of the Fuel Clad Barrier.

The radiation monitor reading in this threshold is higher than that specified for RCS Barrier Loss threshold RC.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.

The gamma dose rate resulting from a postulated loss of coolant accident (LOCA) is monitored by the containment high range monitors, 2RMR-RQ206 and 207 and are located inside containment. The detector range is approximately 1 to 1E8 R/hr. Radiation Monitors 2RMR-RQ206/207 provide a diverse means of measuring the containment for high level gamma radiation (ref. 1).

FC.C

The Table 2F-2 values, column FC Loss represents, based on the referenced calculation, the expected containment high range radiation monitor (2RMR-RQ206 and 207) response based on a LOCA, for periods of 1, 2, 8 and 16 hours after shutdown with coolant activity of 300 Ci/gm DEI-131 or ~1% clad failure (ref. 1).

The value is derived as follows:

ERS-SMM-11-002 Attachment 2 CRM Readings vs. Time for 1% Clad Damage on 2RMR-RQ206 and 207 for 1, 2, 8 and 16 hours after shutdown (rounded) (ref. 1).

Basis Reference(s):

1. ERS-SMM-11-002, Containment Radiation Monitor Readings Following Clad Damage (FC2 Loss, FC7 Loss, RC2 Loss and CT2 Potential Loss)
2. NEI 99-01 Rev. 6 CMT Radiation / RCS Activity Fuel Clad Loss 3.A

ATTACHMENT 4:
Unit 2 Fission Product Barrier Loss/Potential Loss Matrix and Bases

Barrier: Fuel Clad **FC.C**

Category: C. CT Radiation / RCS Activity

Degradation Threat: Loss

Threshold:

2. Dose equivalent I-131 coolant activity **> 300 µCi/gm**

Basis:

This threshold indicates that RCS radioactivity concentration is greater than 300 µCi/gm dose equivalent I-131. Reactor coolant activity above this level is greater than that expected for iodine spikes and corresponds generically to an approximate range of 2% to 5% fuel clad damage (1% at BVPS) (ref. 1). Since this condition indicates that a significant amount of fuel clad damage has occurred, it represents a loss of the Fuel Clad Barrier.

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

Basis Reference(s):

1. ERS-SMM-11-002, Containment Radiation Monitor Readings Following Clad Damage (FC2 Loss, FC7 Loss, RC2 Loss and CT2 Potential Loss)
2. NEI 99-01 Rev. 6 CMT Radiation / RCS Activity Fuel Clad Loss 3.B

ATTACHMENT 4:
Unit 2 Fission Product Barrier Loss/Potential Loss Matrix and Bases

Barrier: Fuel Clad **FC.C**

Category: C. CT Radiation / RCS Activity

Degradation Threat: Potential Loss

Threshold:

None

ATTACHMENT 4:
Unit 2 Fission Product Barrier Loss/Potential Loss Matrix and Bases

Barrier: Fuel Clad **FC.D**

Category: D. CT Integrity or Bypass

Degradation Threat: Loss

Threshold:

None

ATTACHMENT 4:
Unit 2 Fission Product Barrier Loss/Potential Loss Matrix and Bases

Barrier: Fuel Clad **FC.D**

Category: D. CT Integrity or Bypass

Degradation Threat: Potential Loss

Threshold:

None

ATTACHMENT 4:
Unit 2 Fission Product Barrier Loss/Potential Loss Matrix and Bases

Barrier: Fuel Clad **FC.E**

Category: E. Emergency Director Judgment

Degradation Threat: Loss

Threshold:

1. **ANY** condition in the opinion of the Emergency Director that indicates Loss of the Fuel Clad Barrier

Bases

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

Basis Reference(s):

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

ATTACHMENT 4:
Unit 2 Fission Product Barrier Loss/Potential Loss Matrix and Bases

Barrier: Fuel Clad **FC.E**

Category: E. Emergency Director Judgment

Degradation Threat: Potential Loss

Threshold:

1. **ANY** condition in the opinion of the Emergency Director that indicates Potential Loss of the Fuel Clad Barrier

Bases

This threshold addresses any other factors that are to be used by the Emergency Director in determining whether the Fuel Clad Barrier is potentially lost. The Emergency Director should also consider whether or not to declare the barrier potentially lost in the event that barrier status cannot be monitored.

Basis Reference(s):

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

ATTACHMENT 4:
Unit 2 Fission Product Barrier Loss/Potential Loss Matrix and Bases

Barrier: Reactor Coolant System **RC.A**

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

Basis:

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 CT.A.1 will also be met.

ECCS (SI) actuation is caused by (ref. 1):

- Pressurizer low pressure
- Steamline low pressure
- Containment high pressure

Basis Reference(s):

1. 2OM-53A.1.E-0 Reactor Trip or Safety Injection
2. 2OM-53A.1.E-3 Steam Generator Tube Rupture
3. BVRM-OPS-0013 BV-2 EOP Setpoint Document
4. NEI 99-01 Rev. 6 RCS or SG Tube Leakage Reactor Coolant System Loss 1.A

ATTACHMENT 4:
Unit 2 Fission Product Barrier Loss/Potential Loss Matrix and Bases

Barrier: Reactor Coolant System **RC.A**

Category: A. RCS or SG Tube Leakage

Degradation Threat: Potential Loss

Threshold:

1. Operation of a standby charging pump is required by **EITHER:**
   - UNISOLABLE RCS leakage
   - SG tube leakage

Basis:

This threshold is based on an UNISOLABLE RCS leak that results in the inability to maintain pressurizer level within specified limits by operation of a normally used charging (makeup) pump, but an ECCS (SI) actuation has not occurred. The threshold is met when an operating procedure, or operating crew supervision, directs that a standby charging (makeup) pump be placed in service to restore and maintain pressurizer level.

This threshold is applicable to unidentified and pressure boundary leakage, as well as identified leakage. It is also applicable to UNISOLABLE RCS leakage through an interfacing system. The mass loss may be into any location – inside containment, to the secondary-side (i.e., steam generator tube leakage) or outside of containment.

If a leaking steam generator is also FAULTED outside of containment, the declaration escalates to a Site Area Emergency since the Containment Barrier Loss threshold CT.A.1 will also be met.

The Chemical and Volume Control System (CVCS) includes three single speed charging pumps that take suction from the volume control tank and return the cooled, purified reactor coolant to the RCS. The centrifugal charging pumps in the CVCS also serve as the high-head safety injection pumps in the Emergency Core Cooling System. The capacity of each centrifugal pump is ~150 gpm. A second charging pump being required is indicative of a substantial RCS leak (ref. 1, 2).

Basis Reference(s):

1. BV2 UFSAR 9.3.4 Chemical and Volume Control System
2. BV2 UFSAR Table 9.3-8 CVCS Principle Components and Design Parameters
3. NEI 99-01 Rev. 6 RCS or SG Tube Leakage Reactor Coolant System Potential Loss 1.A

ATTACHMENT 4:
Unit 2 Fission Product Barrier Loss/Potential Loss Matrix and Bases

Barrier: Reactor Coolant System **RC.A**

Category: A. RCS or SG Tube Leakage

Degradation Threat: Potential Loss

Threshold:

2. RCS Integrity-RED Path conditions met

Basis:

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).

Critical Safety Function Status Tree (CSFST) RCS Integrity-RED path indicates the RCS barrier is under significant challenge (ref. 1). The CSFSTs are normally monitored using the Safety Parameter Display System (SPDS) display on the Plant Computer.

Basis Reference(s):

1. 2OM-53A.1.F-0.4 Vessel Integrity Status Tree
2. NEI 99-01 Rev. 6 RCS or SG Tube Leakage Reactor Coolant System Potential Loss 1.B

ATTACHMENT 4:
Unit 2 Fission Product Barrier Loss/Potential Loss Matrix and Bases

Barrier: Reactor Coolant System **RC.B**

Category: B. Inadequate Heat Removal

Degradation Threat: Loss

Threshold:

None

ATTACHMENT 4:
Unit 2 Fission Product Barrier Loss/Potential Loss Matrix and Bases

Barrier: Reactor Coolant System **RC.B**

Category: B. Inadequate Heat Removal

Degradation Threat: Potential Loss

Threshold:

1. Heat Sink-RED path conditions met

   AND

   Heat sink is required

Basis:

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 FC.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.

Critical Safety Function Status Tree (CSFST) Heat Sink-RED Path indicates the heat sink function is under extreme challenge and that some fuel clad damage may potentially occur (ref. 1).

The CSFSTs are normally monitored using the Safety Parameter Display System (SPDS) display on the Plant Computer (ref. 1).

The phrase “and heat sink required” precludes the need for classification for conditions in which RCS pressure is less than SG pressure or Heat Sink-RED path entry was created through operator action directed by an ERG. For example, FRH-0.1 is entered from CSFST Heat Sink-Red. Step 1 tells the operator to determine if heat sink is required by checking that RCS pressure is greater than any non-faulted SG pressure and RCS temperature is greater than 350ºF. If these conditions exist, Heat Sink is required. Otherwise, the operator is to either return to the procedure and step in effect and place RHR in service for heat removal. For large LOCA events inside the Containment, the SGs are moot because heat removal through the containment heat removal systems takes place. Therefore, Heat Sink Red should not be required and, should not be assessed for EAL classification because a LOCA event alone should not require higher than an Alert classification. (ref. 2).

RC.B

Basis Reference(s):

1. 2OM-53A.1.F-0.3 Heat Sink Status Tree
2. 2OM-53A.1.FR-H.1 Response to Loss of Secondary Heat Sink
3. NEI 99-01 Rev. 6 Inadequate Heat Removal RCS Loss 2.B

ATTACHMENT 4:
Unit 2 Fission Product Barrier Loss/Potential Loss Matrix and Bases

Barrier: Reactor Coolant System **RC.C**

Category: C. RCS Radiation/ RCS Activity

Degradation Threat: Loss

Threshold:

1. Containment Radiation Monitor **> Table 2F-2, "RC Loss"**

Table 2F-2 Containment Radiation – R/hr (2RMR-RQ206/207)			
Time After S/D (Hrs.)	RC Loss (R/hr)	FC Loss (R/hr)	CT Potential Loss (R/hr)
0-1	11	700	14,000
>1-2	11	490	9,600
>2-8	11	200	3,900
>8-16	11	120	2,400
>16-48	11	63	1200

Basis:

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 FC.C.1 since it indicates a loss of the RCS Barrier only.

The gamma dose rate resulting from a postulated loss of coolant accident (LOCA) is monitored by the containment high range monitors, 2RMR-RQ206 and 207 and are located inside containment. The detector range is approximately 1 to 1E8 R/hr. Radiation Monitors 2RMR-RQ206/207 provide a diverse means of measuring the containment for high level gamma radiation (ref. 1).

The Table 2F-2 values, column RC Loss represents, based on the referenced calculation, the expected containment high range radiation monitor (2RMR-RQ206 and 207) response based on a LOCA, with coolant activity corresponding to Technical Specification coolant activity of 21 µCi/gm DEI-131 (ref. 1).

RC.C

Basis Reference(s):

1. ERS-SMM-11-002, Containment Radiation Monitor Readings Following Clad Damage (FC2 Loss, FC7 Loss, RC2 Loss and CT2 Potential Loss)
2. NEI 99-01 Rev. 6 CMT Radiation / RCS Activity RCS Loss 3.A

ATTACHMENT 4:
Unit 2 Fission Product Barrier Loss/Potential Loss Matrix and Bases

Barrier: Reactor Coolant System **RC.C**

Category: C. CT Radiation/ RCS Activity

Degradation Threat: Potential Loss

Threshold:

None

ATTACHMENT 4:
Unit 2 Fission Product Barrier Loss/Potential Loss Matrix and Bases

Barrier: Reactor Coolant System **RC.D**

Category: D. CT Integrity or Bypass

Degradation Threat: Loss

Threshold:

None

ATTACHMENT 4:
Unit 2 Fission Product Barrier Loss/Potential Loss Matrix and Bases

Barrier: Reactor Coolant System **RC.D**

Category: D. CT Integrity or Bypass

Degradation Threat: Potential Loss

Threshold:

None

ATTACHMENT 4:
Unit 2 Fission Product Barrier Loss/Potential Loss Matrix and Bases

Barrier: Reactor Coolant System **RC.E**

Category: E. Emergency Director Judgment

Degradation Threat: Loss

Threshold:

1. **ANY** condition in the opinion of the Emergency Director that indicates Loss of the RCS Barrier

Basis:

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

Basis Reference(s):

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

ATTACHMENT 4:
Unit 2 Fission Product Barrier Loss/Potential Loss Matrix and Bases

Barrier: Reactor Coolant System **RC.E**

Category: E. Emergency Director Judgment

Degradation Threat: Potential Loss

Threshold:

1. **ANY** condition in the opinion of the Emergency Director that indicates Potential Loss of the RCS Barrier

Basis:

This threshold addresses any other factors that may be used by the Emergency Director in determining whether the RCS Barrier is potentially lost. The Emergency Director should also consider whether or not to declare the barrier potentially lost in the event that barrier status cannot be monitored.

Basis Reference(s):

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

ATTACHMENT 4:
Unit 2 Fission Product Barrier Loss/Potential Loss Matrix and Bases

Barrier: Containment **CT.A**

Category: A. RCS or SG Tube Leakage

Degradation Threat: Loss

Threshold:

1. A leaking or RUPTURED SG is FAULTED outside of containment

Basis:

This threshold addresses a leaking or RUPTURED Steam Generator (SG) that is also FAULTED outside of containment. The condition of the SG, whether leaking or RUPTURED, is determined in accordance with the thresholds for RCS Barrier Potential Loss RC.A.1 and Loss RC.A.1, respectively. This condition represents a bypass of the containment barrier.

FAULTED is a defined term within the NEI 99-01 methodology; this determination is not necessarily dependent upon entry into, or diagnostic steps within, an EOP. For example, if the pressure in a steam generator is decreasing uncontrollably (part of the FAULTED definition) and the FAULTED steam generator isolation procedure is not entered because EOP user rules are dictating implementation of another procedure to address a higher priority condition, the steam generator is still considered FAULTED for emergency classification purposes.

The FAULTED criterion establishes an appropriate lower bound on the size of a steam release that may require an emergency classification. Steam releases of this size are readily observable with normal Control Room indications. The lower bound for this aspect of the containment barrier is analogous to the lower bound criteria specified in IC SU4 for the fuel clad barrier (i.e., RCS activity values) and IC SU5 for the RCS barrier (i.e., RCS leak rate values).

This threshold also applies to prolonged steam releases necessitated by operational considerations such as the forced steaming of a leaking or RUPTURED steam generator directly to atmosphere to cooldown the plant, or to drive an auxiliary (emergency) feed water pump. These types of conditions will result in a significant and sustained release of radioactive steam to the environment (and are thus similar to a FAULTED condition). The inability to isolate the steam flow without an adverse effect on plant cooldown meets the intent of a loss of containment.

Steam releases associated with the expected operation of a SG power operated relief valve or safety relief valve do not meet the intent of this threshold. Such releases may occur intermittently for a short period of time following a reactor trip as operators process through emergency operating procedures to bring the plant to a stable condition and prepare to initiate a plant cooldown. Steam releases associated with the unexpected operation of a valve (e.g., a stuck-open safety valve) do meet this threshold.

ATTACHMENT 4:
Unit 2 Fission Product Barrier Loss/Potential Loss Matrix and Bases

CT.A

Following an SG tube leak or RUPTURE, there may be minor radiological releases through a secondary-side system component (e.g., air ejectors, gland seal exhausters, valve packing, etc.). These types of releases do not constitute a loss or potential loss of containment but should be evaluated using the Recognition Category R ICs.

The ECLs resulting from primary-to-secondary leakage, with or without a steam release from the FAULTED SG, are summarized below.

	Affected SG is FAULTED Outside of Containment?	
P-to-S Leak Rate	**Yes**	**No**
Less than or equal to 25 gpm	No classification	No classification
Greater than 25 gpm	Unusual Event per SU5.2	Unusual Event per SU5.2
Requires operation of a standby charging (makeup) pump (*RCS Barrier Potential Loss*)	Site Area Emergency per FS1.1	Alert per FA1.1
Requires an automatic or manual ECCS (SI) actuation (*RCS Barrier Loss*)	Site Area Emergency per FS1.1	Alert per FA1.1

Basis Reference(s):

1. 2OM-53A.1.E-3 Steam Generator Tube Rupture
2. 2OM-53A.1.ECA-3.1 SGTR with Loss of Reactor Coolant – Subcooled Recovery Desired
3. NEI 99-01 Rev. 6 RCS or SG Tube Leakage Containment Loss 1.A

ATTACHMENT 4:
Unit 2 Fission Product Barrier Loss/Potential Loss Matrix and Bases

Barrier: Containment **CT.A**

Category: A. RCS or SG Tube Leakage

Degradation Threat: Potential Loss

Threshold:

None

ATTACHMENT 4:
Unit 2 Fission Product Barrier Loss/Potential Loss Matrix and Bases

Barrier: Containment **CT.B**

Category: B. Inadequate Heat Removal

Degradation Threat: Loss

Threshold:

None

ATTACHMENT 4:
Unit 2 Fission Product Barrier Loss/Potential Loss Matrix and Bases

Barrier: Containment **CT.B**

Category: B. Inadequate Heat Removal

Degradation Threat: Potential Loss

Threshold:

1. Core Cooling-RED Path conditions met

   AND

   Restoration procedures **not** effective within **15 min.** (Note 1)

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

Basis:

This condition represents an IMMINENT core melt sequence which, if not corrected, could lead to vessel failure and an increased potential for containment failure. For this condition to occur, there must already have been a loss of the RCS Barrier and the Fuel Clad Barrier. If implementation of a procedure(s) to restore adequate core cooling is not effective (successful) within 15 minutes, it is assumed that the event trajectory will likely lead to core melting and a subsequent challenge of the Containment Barrier.

The restoration procedure is considered "effective" if core exit thermocouple readings are decreasing and/or if reactor vessel level is increasing. Whether or not the procedure(s) will be effective should be apparent within 15 minutes. The Emergency Director should escalate the emergency classification level as soon as it is determined that the procedure(s) will not be effective.

Severe accident analyses (e.g., NUREG-1150) have concluded that function restoration procedures can arrest core degradation in a significant fraction of core damage scenarios, and that the likelihood of containment failure is very small in these events. Given this, it is appropriate to provide 15 minutes beyond the required entry point to determine if procedural actions can reverse the core melt sequence.

Critical Safety Function Status Tree (CSFST) Core Cooling-RED path indicates significant core exit superheating and core uncovery. The CSFSTs are normally monitored using the Safety Parameter Display System (SPDS) display on the Plant Computer (ref. 1).

The function restoration procedures are those emergency operating procedures that address the recovery of the core cooling critical safety functions. The procedure is considered effective if the temperature is decreasing or if the vessel water level is increasing (ref. 2).

ATTACHMENT 4:
Unit 2 Fission Product Barrier Loss/Potential Loss Matrix and Bases

CT.B

A direct correlation to status trees can be made if the effectiveness of the restoration procedures is also evaluated. If core exit thermocouple (TC) readings are greater than 1,200°F or other CSFST RED path conditions exist (ref. 1), Fuel Clad barrier is also lost.

Basis Reference(s):

1. 2OM-53A.1.F-0.2 Core Cooling Status Trees
2. 2OM-53A.1.FR-C.1 Response to Inadequate Core Cooling
3. NEI 99-01 Rev. 6 Inadequate Heat Removal Containment Potential Loss 2.A

ATTACHMENT 4:
Unit 2 Fission Product Barrier Loss/Potential Loss Matrix and Bases

Barrier: Containment **CT.C**

Category: C. CT Radiation/RCS Activity

Degradation Threat: Loss

Threshold:

None

ATTACHMENT 4:
Unit 2 Fission Product Barrier Loss/Potential Loss Matrix and Bases

Barrier: Containment **CT.C**

Category: C. CT Radiation/RCS Activity

Degradation Threat: Potential Loss

Threshold:

1. Containment Radiation Monitor **> Table 2F-2, "CT Potential Loss"**

Table 2F-2 Containment Radiation – R/hr (2RMR-RQ206/207)			
Time After S/D (Hrs.)	RC Loss (R/hr)	FC Loss (R/hr)	CT Potential Loss (R/hr)
0-1	11	700	14,000
>1-2	11	490	9,600
>2-8	11	200	3,900
>8-16	11	120	2,400
>16-48	11	63	1200

Basis:

The radiation monitor reading corresponds to an instantaneous release of all reactor coolant mass into the containment, assuming that 20% of the fuel cladding has failed. This level of fuel clad failure is well above that used to determine the analogous Fuel Clad Barrier Loss and RCS Barrier Loss thresholds.

NUREG-1228, Source Estimations During Incident Response to Severe Nuclear Power Plant Accidents, indicates the fuel clad failure must be greater than approximately 20% in order for there to be a major release of radioactivity requiring offsite protective actions. For this condition to exist, there must already have been a loss of the RCS Barrier and the Fuel Clad Barrier. It is therefore prudent to treat this condition as a potential loss of containment which would then escalate the ECL to a General Emergency.

The gamma dose rate resulting from a postulated loss of coolant accident (LOCA) is monitored by the containment high range monitors, 2RMR-RQ206 and 207 and are located inside containment. The detector range is approximately 1 to 1E8 R/hr. Radiation Monitors 2RMR-RQ206/207 provide a diverse means of measuring the containment for high level gamma radiation (ref. 1).

ATTACHMENT 4:
Unit 2 Fission Product Barrier Loss/Potential Loss Matrix and Bases

CT.C

The Table 2F-2 values, column CT Potential Loss represents, based on the referenced calculation, the expected containment high range radiation monitor (2RMR-RQ206 and 207) response based on a LOCA, for periods of 1, 2, 8 and 16 hours after shutdown with coolant activity corresponding to ~20% clad failure (ref. 1).

The value is derived as follows:

ERS-SMM-11-002 Attachment 2 CRM Readings vs. Time for 20% Clad Damage on 2RMR-RQ206 for 1, 2, 8 and 16 hours after shutdown (rounded) (ref. 1).

Basis Reference(s):

1. ERS-SMM-11-002, Containment Radiation Monitor Readings Following Clad Damage (FC2 Loss, FC7 Loss, RC2 Loss and CT2 Potential Loss)
2. NEI 99-01 Rev. 6 CMT Radiation / RCS Activity Containment Potential Loss 3.A

ATTACHMENT 4:
Unit 2 Fission Product Barrier Loss/Potential Loss Matrix and Bases

Barrier: Containment **CT.D**

Category: D. CT Integrity or Bypass

Degradation Threat: Loss

Threshold:

1. Containment isolation is required

   AND EITHER:

   - Containment integrity has been lost based on Emergency Director judgment
   - UNISOLABLE pathway from containment to the environment exists

Basis:

These thresholds address a situation where containment isolation is required and one of two conditions exists as discussed below. Users are reminded that there may be accident and release conditions that simultaneously meet both bulleted thresholds.

First Threshold – Containment integrity has been lost, i.e., the actual containment atmospheric leak rate likely exceeds that associated with allowable leakage (or sometimes referred to as design leakage). Following the release of RCS mass into containment, containment pressure will fluctuate based on a variety of factors; a loss of containment integrity condition may (or may not) be accompanied by a noticeable drop in containment pressure. Recognizing the inherent difficulties in determining a containment leak rate during accident conditions, it is expected that the Emergency Director will assess this threshold using judgment, and with due consideration given to current plant conditions, and available operational and radiological data (e.g., containment pressure, readings on radiation monitors outside containment, operating status of containment pressure control equipment, etc.).

Refer to the middle piping run of Figure 1. Two simplified examples are provided. One is leakage from a penetration and the other is leakage from an in-service system valve. Depending upon radiation monitor locations and sensitivities, the leakage could be detected by any of the four monitors depicted in the figure.

Another example would be a loss or potential loss of the RCS barrier, and the simultaneous occurrence of two FAULTED locations on a steam generator where one fault is located inside containment (e.g., on a steam or feedwater line) and the other outside of containment. In this case, the associated steam line provides a pathway for the containment atmosphere to escape to an area outside the containment.

Following the leakage of RCS mass into containment and a rise in containment pressure, there may be minor radiological releases associated with allowable (design) containment leakage through various penetrations or system components. These releases do not constitute a loss or potential loss of containment but should be evaluated using the Recognition Category R ICs.

Second Threshold – Conditions are such that there is an UNISOLABLE pathway for the migration of radioactive material from the containment atmosphere to the environment. As used here, the term "environment" includes the atmosphere of a room or area, outside the containment, that may, in turn, communicate with the outside-the-plant atmosphere (e.g., through discharge of a ventilation system or atmospheric leakage). Depending upon a variety of factors, this condition may or may not be accompanied by a noticeable drop in containment pressure.

Refer to the top piping run of Figure 1. In this simplified example, the inboard and outboard isolation valves remained open after a containment isolation was required (i.e., containment isolation was not successful). There is now an UNISOLABLE pathway from the containment to the environment.

The existence of a filter is not considered in the threshold assessment. Filters do not remove fission product noble gases. In addition, a filter could become ineffective due to iodine and/or particulate loading beyond design limits (i.e., retention ability has been exceeded) or water saturation from steam/high humidity in the release stream.

Leakage between two interfacing liquid systems, by itself, does not meet this threshold.

Refer to the bottom piping run of Figure 1. In this simplified example, leakage in an RCP seal cooler is allowing radioactive material to enter the Auxiliary Building. The radioactivity would be detected by the Process Monitor. If there is no leakage from the closed water cooling system to the Auxiliary Building, then no threshold has been met. If the pump developed a leak that allowed steam/water to enter the Auxiliary Building, then second threshold would be met. Depending upon radiation monitor locations and sensitivities, this leakage could be detected by any of the four monitors depicted in the figure and cause the first threshold to be met as well.

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

The status of the containment barrier during an event involving steam generator tube leakage is assessed using Loss Threshold A.1.

Basis Reference(s):

1. NEI 99-01 Rev. 6 CMT Integrity or Bypass Containment Loss 4.A

ATTACHMENT 4:
Unit 2 Fission Product Barrier Loss/Potential Loss Matrix and Bases

Barrier: Containment **CT.D**

Category: D. CT Integrity or Bypass

Degradation Threat: Loss

Threshold:

2. Indications of RCS leakage outside of containment

Basis:

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.

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.

2OM-53A.1.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):

- Safety Injection
- Chemical & Volume Control
- RCP seals
- PZR/RCS Loop sample lines

CT.D

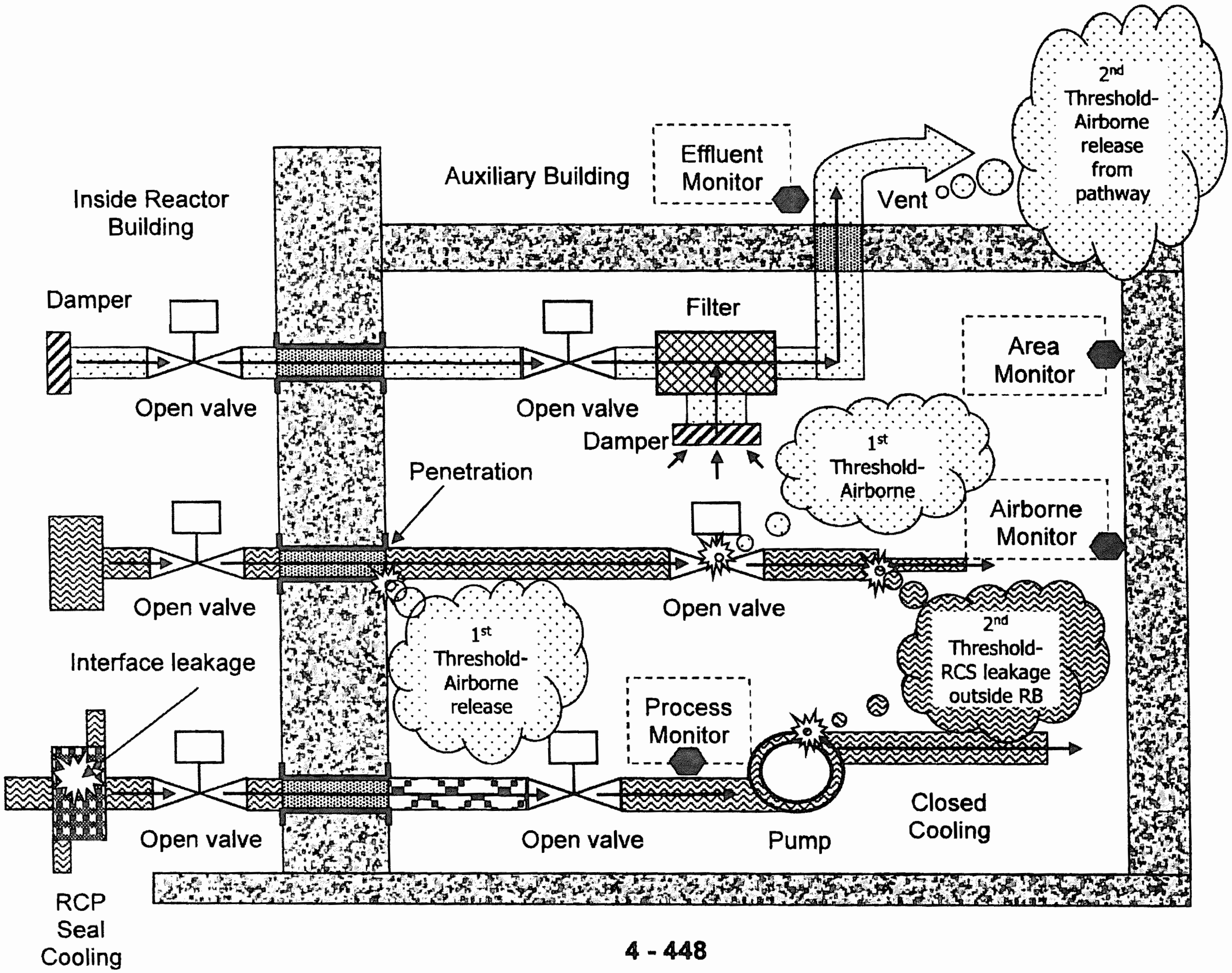
Basis Reference(s):

1. 2OM-53A.1.ECA-1.2 LOCA Outside Containment
2. NEI 99-01 Rev. 6 CMT Integrity or Bypass Containment Loss

ATTACHMENT 4: Unit 2 Fission Product Barrier Loss/Potential Loss Matrix and Bases

Figure 1: Containment Integrity or Bypass Examples

CT.D



ATTACHMENT 4:
Unit 2 Fission Product Barrier Loss/Potential Loss Matrix and Bases

Barrier: Containment **CT.D**

Category: D. CT Integrity or Bypass

Degradation Threat: Potential Loss

Threshold:

1. Containment-RED Path conditions met

Basis:

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.

Critical Safety Function Status Tree (CSFST) Containment-RED path is entered if containment pressure is greater than or equal to 45 psig and represents an extreme challenge to safety function. The CSFSTs are normally monitored using the the Safety Parameter Display System (SPDS) display on the Plant Computer (ref. 1).

45 psig is the containment design pressure and is the pressure used to define CSFST Containment Red Path conditions (ref. 1, 2).

Basis Reference(s):

1. 2OM-53A.1.F-0.5 Containment Status Tree
2. BV2 UFSAR Section 6.2.1 Design Basis
3. NEI 99-01 Rev. 6 CMT Integrity or Bypass Containment Potential Loss 4.A

ATTACHMENT 4:
Unit 2 Fission Product Barrier Loss/Potential Loss Matrix and Bases

Barrier: Containment **CT.D**

Category: D. CT Integrity or Bypass

Degradation Threat: Potential Loss

Threshold:

2.	Containment hydrogen concentration **> 4%**

Basis:

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.

The containment hydrogen analyzer system consists of two redundant hydrogen monitors to provide protection against single failure and single loss of power. Containment samples are obtained through independent sample lines for each monitor. Indication is provided for each hydrogen analyzer, on the vertical board in the main control room, with an indicating range of 0-10 percent hydrogen. A recorder is provided to record the Train A hydrogen level. The hydrogen analyzer system is designed to provide a continuous positive indication of the containment hydrogen concentration within 30 minutes after the initiation of safety injection (ref. 1).

In the early stages of a core uncovery event, it is unlikely that hydrogen buildup due to a core uncovery could result in an explosive mixture of dissolved gasses in 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. A combustible mixture can be formed when hydrogen gas concentration in the Containment atmosphere is greater than 4% by volume. All hydrogen measurements are referenced to concentrations in dry air even though the actual Containment environment may contain significant steam concentrations.

To generate such levels of combustible gas, loss of the Fuel Clad and RCS barriers must have occurred. With the Potential Loss of the Containment barrier, the threshold hydrogen concentration, therefore, will likely warrant declaration of a General Emergency.

Basis Reference(s):

1. BV2 UFSAR Section 6.2.5 Combustible Gas Control in Containment
2. NEI 99-01 Rev. 6 CMT Integrity or Bypass Containment Potential Loss 4.B

ATTACHMENT 4:
Unit 2 Fission Product Barrier Loss/Potential Loss Matrix and Bases

Barrier: Containment **CT.D**

Category: D. CT Integrity or Bypass

Degradation Threat: Potential Loss

Threshold:

3. Containment pressure **> 11 psig AND <** one full train of depressurization equipment operating per design for **≥ 15 min.** (Note 1)

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

Basis:

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.

Each unit has a containment pressure quench spray system with two 100% capacity trains. These pumps take suction from the RWST and discharge to the spray header. The quench spray system starts on a CIB at the start of a LOCA accident.

The recirculation spray system has four 50% capacity subsystems that consist of a pump and a cooler. The recirculation spray pump takes suction from the containment sump and discharges through a cooler to the spray header. The recirculation spray system does not start during a LOCA until there is low level in the RWST to verify the sump has adequate water inventory. When the RWST level goes very low the quench spray pumps are secured.

A very short period of time could exist where the quench spray system and the recirculation spray system pumps could both be running. Normally it is either the quench spray or the recirculation spray running.

One train of QS System and one train of RS System comprise one full train of depressurization equipment as designed (ref. 1).

Basis Reference(s):

1. BV2 UFSAR Section 6.2 Containment Systems
2. NEI 99-01 Rev. 6 CMT Integrity or Bypass Containment Potential Loss 4.C

ATTACHMENT 4:
Unit 2 Fission Product Barrier Loss/Potential Loss Matrix and Bases

Barrier: Containment **CT.E**

Category: E. Emergency Director Judgment

Degradation Threat: Loss

Threshold:

1. **ANY** condition in the opinion of the Emergency Director that indicates Loss of the Containment Barrier

Basis:

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

Basis Reference(s):

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

ATTACHMENT 4:
Unit 2 Fission Product Barrier Loss/Potential Loss Matrix and Bases

Barrier: Containment **CT.E**

Category: E. Emergency Director Judgment

Degradation Threat: Potential Loss

Threshold:

1. **ANY** condition in the opinion of the Emergency Director that indicates Potential Loss of the Containment Barrier

Basis:

This threshold addresses any other factors that may be used by the Emergency Director in determining whether the Containment Barrier is potentially lost. The Emergency Director should also consider whether or not to declare the barrier potentially lost in the event that barrier status cannot be monitored.

Basis Reference(s):

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

ATTACHMENT 5:
Safe Operation & Shutdown Areas RA3.2 & HA5.1 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 5:
Safe Operation & Shutdown Areas RA3.2 & HA5.1 Bases

BVPS Unit 1 Table 1R-1 and 1H-2 Bases

A review of station operating procedures identified the following mode dependent in-plant actions and associated areas that are required for normal plant operation, cooldown or shutdown:

Procedure	Area or Room	Requirement	Modes	Table? Y/N
N/A	U1 Control Room	Toxic gas release (1H-2 only)	All	Y
1OM-52.4.R.1.F	Aux Bldg 735 Sample panel	Shutdown/Cooldown T.S. blocking SI and Shutdown margin SR 3.1.1.1	3, 4, 5	N
1OM-52.4.R.1.F	Operator isolate PG water valve	Aux Bldg 722', Blender Room	3	N
1OM-52.4.R.1.F	Safeguards 735 East & West Cable Vault (2 separate areas)	Operator to de-energize the BIT isolation valves MCC's	4	N
1OM-52.4.R.1.F	Safeguards 735 East & West Cable Vault (2 separate)	Operator to de-energize thesafety injection accumulator isolation valves MCC's	4	N
10M-10.4.A	Safeguards 735 East & West Cable Vault (2 separate)	Shutdown/Cooldown to Mode 5 & RHR S/U	4	Y
10M-10.4.A	Safeguards 722' Penetrations D	Shutdown/Cooldown to Mode 5 & RHR S/U	4	Y
10M-10.4.A	Aux Bldg 735 CCR Hx Area	Shutdown/Cooldown to Mode 5 & RHR S/U	4	Y
10M-10.4.A	Service Bld 713 DF Emergency Switchgear	Shutdown/Cooldown to Mode 5 & RHR S/U	4	N
10M-10.4.A	Service Bld 713 AE Emergency Switchgear	Shutdown/Cooldown to Mode 5 & RHR S/U	4	Y

ATTACHMENT 5:
Safe Operation & Shutdown Areas RA3.2 & HA5.1 Bases

Table 1R-1 & 1H-2 Results

Table 1R-1/1H-2 Safe Operation & Shutdown Rooms/Areas	
Room/Area	**Mode Applicability**
Control Room *	All
Rod Control Bldg 735’	4
Safeguards 722’ Penetrations D	4
Auxiliary Building 735’ CCR Hx Area	4
Service Building 713’ AE Emergency Switchgear	4

* Applicable to Table 1H-2 only.

BVPS Unit 1 Plant Operating Procedures Reviewed

The following BVPS U1 procedures were reviewed,

- 1OM-52.4. R.1.F "Refueling Station Shutdown From 100% to MODE 5"
- 1OM-52.4.R.1.S "Secondary Plant Shutdown"
- 1OM-52.4.R.2.F "Refueling Station Shutdown - MODE 5 Activities"
- 10M-10.4.A "RHR System Startup and Operation"
- 10M-10.4.B "Residual Heat Removal System Running"
- 10M-15.4.G "Starting an Additional CCR Pump"

ATTACHMENT 5:
Safe Operation & Shutdown Areas RA3.2 & HA5.1 Bases

BVPS Unit 2 RA3.2 & Table 2H-2 Bases

A review of station operating procedures identified the following mode dependent in-plant actions and associated areas that are required for normal plant operation, cooldown or shutdown:

Procedure	Area or Room	Requirement	Modes	Table? Y/N
N/A	U2 Control Room	Toxic gas release (2H-2 only)	All	Y
2OM-52.4.R.1.F	Turbine Basement	Secure Heater Drain Pumps per 20M-23B.4.C	1	N
2OM-52.4.R.1.F	Turbine Basement	Secure Main Feed Pumps per 20M-24.4. F	1	N
2OM-52.4.R.1.F	Turbine Mezz 754'	Secure MSRs	1	N
2OM-52.4.R.1.S	Service Bldq 760'	Operator to align MainTransformer cooling	1	N
2OM-52.4.R.1.S	Turbine Bldq 752'	Operator to isolate CCS to Turbine Lube Oil Cooler & Exciter coolers; isolate MSRs; shut down Iso Phase Bus Duct Fans	3	N
2OM-52.4.R.1.S	Turbine Bldq 730'	SecureCondenser Tube Cleaning system; MUG H2 link removal; venting MUG; purging MUG	3	N
2OM-52.4.R.1.F	Aux Bldg 718' Sample panel	Chemist obtain RCS samples to verify shutdown margin for planned cooldown	3, 4, 5	N
2OM-52.4.R.1.F	Aux Bldg 710' Blender Room	Operator isolate PG water valve	3	N
2OM-52.4.R.1.F	Rod Control Bldg 735'	Place RHR in service per 20M-10.4.A (Attachment 1)	3, 4	Y
20M-10.4.A	Aux Bldg 710 CCP Hx Area	Operator placing additional reactor plant component cooling water heat exchanger in-service	3, 4	N
20M-52.4.R.1.F	PAB 755' and Rod Control 735	Operator to de-energize the High Head Safety Injection MOV's	4	N
20M-52.4.R.2.F	SFGDS 737', PAB 755', Rod Control 735"	MODE 5 Alignment Of ESF And ECCS Components	5	N

ATTACHMENT 5:
Safe Operation & Shutdown Areas RA3.2 & HA5.1 Bases

RA2.3 & Table 2H-2 Results

RA2.3/ Table 2H-2 Safe Operation & Shutdown Rooms/Areas	
Room/Area	**Mode Applicability**
Control Room *	All
Rod Control Building 735’	3, 4

* Applcable to Table 2H-2 only.

BVPS Unit 2 Plant Operating Procedures Reviewed

The following BVPS U2 procedures were reviewed,

- 20M-52.4. R.1.F "Refueling Station Shutdown From 100% to MODE 5"
- 20M-52.4.R.1.S "Secondary Plant Shutdown"
- 20M-52.4.R.2.F "Refueling Station Shutdown - MODE 5 Activities"
- 20M-10.4.A "RHR System Startup and Operation"
- 20M-10.4.B "Residual Heat Removal System Running"

Enclosure B

L-17-104

Beaver Valley Power Station, Unit Nos. 1 and 2, Emergency Preparedness Plan, Section 1, Definitions
(17 Pages Follow)

SECTION 1 [C61]

DEFINITIONS

Section 1

DEFINITIONS

Table of Contents

1. <u>DEFINITIONS</u>

The terms defined in this section are those which are used in special context in this document and/or are unique to the Beaver Valley Power Station (BVPS).

1.1. **ACCOUNTABILITY** -- Process to ascertain the whereabouts of all personnel within the plant PROTECTED AREA fence. Process is completed through the use of a computerized access security system.

1.2. **AFFECTING SAFE SHUTDOWN** -- Event in progress has adversely affected functions that are necessary to bring the plant to and maintain it in the applicable Hot or Cold Shutdown condition. Plant condition applicability is determined by Technical Specification LCOs in effect.

Example 1: Event causes damage that results in entry into an LCO that requires the plant to be placed in Hot Shutdown. Hot Shutdown is achievable, but Cold Shutdown is not. This event is not "AFFECTING SAFE SHUTDOWN."

Example 2: Event causes damage that results in entry into an LCO that requires the plant to be placed in Cold Shutdown. Hot Shutdown is achievable, but Cold Shutdown is not. This event is "AFFECTING SAFE SHUTDOWN."

1.3. **ALERT** -- See definition for EMERGENCY CLASSIFICATION LEVEL.

1.4. **ASSESSMENT ACTIONS** -- Those actions taken during or after an accident to obtain and process information that is necessary to make decisions to implement specific emergency measures.

1.5. **ASSESSMENT FACILITY** -- A facility for evaluation of information, including instrument data, to assess the severity and scope of an emergency condition. [Cxx]

1.6. **BEAVER VALLEY EMERGENCY RESPONSE SYSTEM** -- The BEAVER VALLEY EMERGENCY RESPONSE SYSTEM (BVERS) is a computer aided Voice Mail System to be utilized for ERO activation.

1.7. **BEAVER VALLEY SITE** -- The entire OWNER CONTROLLED AREA. Includes the BVPS Unit 1, BVPS Unit 2 and the EMERGENCY RESPONSE FACILITY. [Cxx]

1.8. **COMPENSATORY INDICATIONS** -- Computer points, In-Plant Computer - IPC (U1), Inadequate Core Cooling Monitor - ICCM (U1) , Sequence of Events Recorder - SER (U1), Plant Computer System - PCS (U2), Plant Safety Monitoring System - PSMS (U2) and PI Data (ProcessBook®).

1.9. **CONFINEMENT BOUNDARY** – The barrier(s) between spent fuel and the environment once the spent fuel is processed for dry storage. For BVPS the CONFINEMENT BOUNDARY is the Dry Shielded Canister (DSC) [Cxx]

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

1.11. **CONTROL ROOM** -- Area from which plant systems are operated and monitored.

1.12. **CORRECTIVE ACTIONS** -- Those emergency measures taken to terminate an emergency situation at or near the source of the problem.

1.13. **DOSE PROJECTION** -- A calculated estimate of the potential dose to individuals at a given location, normally OFFSITE; as determined from the quantity of radioactive material released and the appropriate meteorological transport and diffusion parameters.

1.14. **DRILL** -- A pre-planned training activity in which the participants are "walked" or "talked" through one or more procedures, or aspects of the Emergency Preparedness Plan.

1.15. **EMERGENCY ACTIONS** -- A collective term encompassing the Assessment, Corrective, and PROTECTIVE ACTIONS taken during the course of an emergency.

1.16. **EMERGENCY ACTION LEVEL (EAL) --** A pre-determined, site specific, observable threshold for a plant Initiating Condition that, when met or exceeded, places the plant in a given EMERGENCY CLASSIFICATION LEVEL. [Cxx]

1.17. **EMERGENCY CLASSIFICATION LEVEL (ECL)** -- One of a set of names or titles established by the NRC for grouping off-normal events or conditions according to (1) potential or actual effects or consequencses, and (2) resulting ONSITE and OFFSITE response actions. The EMERGENCY CLASSIFICATION LEVELS, in ascending order of severity, are: [Cxx]

- **UNUSUAL EVENT** -- Events are in progress or have occurred which indicate a potential degradation of the level of safety of the plant or indicate a security threat to facility protection has been initiated. No releases of radioactive material requiring OFFSITE response or monitoring are expected unless further degradation of SAFETY SYSTEMS occurs. [C46]

- **ALERT** -- Events are in progress or have occurred which involve an actual or potential substantial degradation of the level of safety of the plant or a security event that involves probable life threatening risk to site personnel or damage to site equipment because of HOSTILE ACTION. Any releases are expected to be limited to small fractions of the EPA Protective Action Guide exposure levels. [C46]

- **<u>SITE AREA EMERGENCY</u>** -- 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 Guide exposure levels beyond the site boundary. [C46]

- **<u>GENERAL EMERGENCY</u>** -- 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 Guide exposure levels OFFSITE for more than the immediate site area. [C46]

1.18. **EMERGENCY COORDINATORS** -- Designated BVPS staff members responsible for coordinating specific emergency organization functions. These coordinating positions are:

- (CONTROL ROOM) Operations Coordinator
- TSC Operations Coordinator
- EOF Operations Coordinator
- Communications and Records Coordinator
- Technical Support Coordinator
- OPERATIONS SUPPORT CENTER Coordinator
- Radiological Controls Coordinator
- Maintenance Coordinator
- Environmental Assessment and DOSE PROJECTION Coordinator
- Engineering Coordinator
- Security Coordinator
- Chemistry Coordinator
- Environmental Coordinator

- Computer Coordinator
- OPERATIONS SUPPORT CENTER Health Physics Coordinator[C15]
- Nuclear Communications/Onsite Coordinator

1.19. **EMERGENCY MANAGERS** -- Designated BVPS staff members responsible for coordinating specific emergency organization functions. These positions, primarily located in the EOF, are activated upon classification of a SITE AREA or GENERAL EMERGENCY and include:

- EMERGENCY/RECOVERY MANAGER
- Support Services Manager
- Nuclear Communications Manager
- Offsite Agency Liaison

1.20. **EMERGENCY DIRECTOR** -- The BVPS individual responsible for direction of ONSITE activities during any emergency at BVPS, and both ONSITE and OFFSITE activities during UNUSUAL EVENTS and ALERT Emergencies. The EMERGENCY DIRECTOR is the only individual authorized to declare an emergency condition, authorize emergency personnel radiation exposures greater than 10 CFR 20; and/or direct the issuance of KI.

1.21. **EMERGENCY IMPLEMENTING PROCEDURES** -- The detailed procedures which carry out the guidance of this Plan.

1.22. **EMERGENCY OPERATING PROCEDURES (EOP)** -- Those procedures utilized by the station operations staff in responding to CONTROL ROOM instrumentation alarms or indications (i.e., assessment and CORRECTIVE ACTIONS).

1.23. **EMERGENCY OPERATIONS CENTER (EOC)** -- Designated Federal, State, and County (i.e., Emergency or disaster services/management agencies) headquarters/facilities, especially designed and equipped for the purpose of exercising effective coordination and control for disaster operations carried out within their jurisdiction.

1.24. **EMERGENCY OPERATIONS FACILITY (EOF)** -- The facility designated for providing overall coordination of the utility's emergency response and coordination with offsite response agencies of the various jurisdictions for the protection of the general public. Space is provided for Federal, State, and local liaison officials. [C61]

1.25. **EMERGENCY PLANNING ZONE** -- There are two EMERGENCY PLANNING ZONES (EPZ). The first is an area approximately 10 miles in radius around BVPS, for which emergency planning consideration of the plume exposure pathway has been given in order to ensure that prompt and effective actions can and will be taken to protect the public in the event of an accident. The second is an area approximately 50 miles in radius around BVPS for which emergency planning consideration of the ingestion pathway has been given.

1.26. **EMERGENCY/RECOVERY MANAGER** -- Upon classification of a SITE AREA or GENERAL EMERGENCY, the EMERGENCY/RECOVERY MANAGER assumes responsibility and authority for overall direction and coordination of the BVPS emergency response, with primary responsibility for coordination of OFFSITE activities (monitoring, logistics, interagency liaison). When activated, the EMERGENCY/RECOVERY MANAGER is the only individual authorized to make recommendations of OFFSITE PROTECTIVE ACTIONS to OFFSITE response agencies.

1.27. **EMERGENCY RESPONSE FACILITY (ERF)** -- The near-site facility provided by BVPS. Incorporates the TECHNICAL SUPPORT CENTER, the Dosimetry Area, Counting Room and other facilities. [C68]

1.28. **ESSENTIAL PERSONNEL** -- Those personnel deemed necessary to the protection of the health and safety of the general public. The personnel from the following groups, and any others deemed necessary, are considered to be ESSENTIAL PERSONNEL:

- Operations
- Radiation Protection
- Chemistry
- Security
- Emergency Response Organization personnel (including Primary, Secondary, Call-out and On-Shift personnel[C44])

1.29. **EXERCISE** -- A realistic, pre-planned simulation of an accident, designed and coordinated in such a manner that the response of the emergency organization and other station personnel closely approximates the response to an actual incident. An EXERCISE may involve participation of OFFSITE organizations.

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

1.31. **EXTORTION** -- An attempt to cause an action at the station by threat of force.

1.32. **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 being completely depressurized. [Cxx]

1.33. **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. [Cxx]

1.34. **FISSION PRODUCT BARRIER THRESHOLD** -- A pre-determined, site-specific, observable threshold indicating the loss or potential loss of a fission product barrier. [Cxx]

1.35. **GENERAL EMERGENCY** -- See definition for EMERGENCY CLASSIFICATION LEVEL.

1.36. **GROUND RELEASE** -- Release of radioactive effluents from the facility via the Reactor Building and supplementary leak collection system vent (located on top of the Reactor Building), the ventilation vent (located on top of the Auxiliary Building), the PROCESS VENT (located on the Cooling Tower), or any other release pathway.

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

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

1.39. **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. [C46]

1.40. **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. [Cxx]

1.41. **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. [Cxx]

1.42. **INTIATING CONDITION (IC)** – An event or condition that aligns with the definition of one of the four emergency classification levels by virtue of the potential or actual effects of consequences. [Cxx].

1.43. **JOINT PUBLIC INFORMATION CENTER (JPIC)** -- The designated location from which news releases, press conferences, and other media interfacing can be provided.

1.44. **LARGE AIRCRAFT**– Any size or type of aircraft with the potential for causing significant damage to the plant (refer to the Security Plan for a more detailed definition).

1.45. **LOCAL AREA EVACUATION** -- Evacuation of personnel from localized affected areas within the station.

1.46. **NON-ESSENTIAL PERSONNEL** – Those personnel not determined to be ESSENTIAL PERSONNEL. [Cxx]

1.47. **NORMAL PLANT OPERATIONS** -- Activities at the plant site associated with routine testing, maintenance, or equipment operations, in accordance with normal operating or administrative procedures. Entry into abnormal or EMERGENCY OPERATING PROCEDURES, or deviation from normal security or radiological controls posture, is a departure from NORMAL PLANT OPERATIONS.

1.48. **OFFSITE** -- Any area outside of the BVPS property boundary surrounding the BEAVER VALLEY SITE.

1.49. **ONSITE** -- See Definition for BEAVER VALLEY SITE.

1.50. **OPERATIONS SUPPORT CENTER (OSC)** -- The designated location for assembly of on-duty and relief operations, health physics and maintenance support personnel.[C15]

1.51. **OWNER CONTROLLED AREA** – The property associated with the station and owned by the company. Access is normally limited to persons entering for official business. Cxx

1.52. **PRIMARY ASSEMBLY AREA** -- An area designated for the assembly of specific groups of individuals for ACCOUNTABILITY and/or in preparation for a plant evacuation within the PROTECTED AREA fence.

1.53. **PROCESS VENT** -- The effluent release path by which gaseous radioactive wastes are released following processing. The release point is located at the top of the cooling tower. In DOSE PROJECTION and accident analyses, this release pathway is considered a GROUND RELEASE.

1.54. **PROJECTILE** -- An object directed toward a NPP that could cause concern for its continued operability, reliability, or personnel safety.

1.55. **PROTECTED AREA** – Means an area encompassed by physical security barriers that is monitored by an intrusion detection system to which access is controlled. Access to the PROTECTED AREA requires proper security clearance and is controlled at the Site Security Alarm Stations. Cxx

1.56. **PROTECTIVE ACTIONS** -- Those emergency measures taken after an uncontrolled release of radioactive material, for the purpose of preventing or minimizing radiological exposures.

1.57. **PROTECTIVE ACTION GUIDES (PAG)** -- Projected radiological dose rate or dose commitment values to individuals in the general population that warrant protective action following a release of radioactive material.

1.58. **RADIOLOGICAL EMERGENCY RESPONSE PLAN (RERP)** -- Detailed incident response plans developed by the State of Pennsylvania and its agencies and County and Municipal Emergency Management agencies in coordination with the Pennsylvania Emergency Management Agency (PEMA) and the fixed nuclear facility.

1.59. **RECOVERY ACTIONS** -- Those actions taken after the emergency to restore the station as nearly as possible to its pre-emergency conditions.

1.60. **REFUELING PATHWAY** – The reactor refueling cavity, spent fuel pool and fuel transfer canal comprise the refueling pathway. Cxx

1.61. **REMOTE ASSEMBLY AREA** -- A designated area (or areas), outside the site, for the assembly of evacuated plant personnel during a SITE EVACUATION.

1.62. **RUPTURE(D)** -- The condition of a steam generator in which primary-to-secondary leakage is of sufficient magnitude to require a safety injection.[Cxx]

1.63. **SABOTAGE** -- Deliberate damage, mis-alignment, or mis-operation of plant equipment with the intent to render the equipment inoperable. Equipment found tampered with or damaged due to malicious mischief may not meet the definition of SABOTAGE until this determination is made by security supervision.

1.64. **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 10 CFR 50.2):

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. [Cxx]

1.65. **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.

1.66. **SITE ASSEMBLY** -- Process of gathering all personnel from areas within the PROTECTED AREA to PRIMARY ASSEMBLY AREAS.

1.67. **SITE AREA EMERGENCY** -- See definition for EMERGENCY CLASSIFICATION LEVEL.

1.68. **SITE EVACUATION** -- Evacuation of all NON-ESSENTIAL PERSONNEL within the BEAVER VALLEY SITE.

1.69. **STRIKE ACTION** -- A work stoppage within the PROTECTED AREA by a body of workers to enforce compliance with demands made on management. The STRIKE ACTION must threaten to interrupt NORMAL PLANT OPERATIONS.

1.70. **TECHNICAL SUPPORT CENTER (TSC)** -- A designated location where plant management coordination of emergency response is performed and where various Licensee, Federal, and vendor engineering disciplines can analyze the conditions within the reactor core during and after an accident to provide technical assessment of the accident and corrective action recommendations to the EMERGENCY DIRECTOR.

1.71. **UNAFFECTED AREA** -- Any area or location which is known to be not significantly affected by radiation levels or other hazardous conditions.

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

1.73. **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. [Cxx]

1.74. **UNUSUAL EVENT** -- See definition for EMERGENCY CLASSIFICATION LEVEL.

1.75. **VALID** -- An indication, report, or condition, is considered to be VALID when it is verified by (1) an instrument channel check, (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.

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

1.77. **VITAL AREA** -- Means any area that contains VITAL EQUIPMENT.

1.78. **VITAL EQUIPMENT** -- Means any equipment, system, device, or material, the failure, destruction, or release of which could directly or indirectly endanger the public health and safety by exposure to radiation. Equipment or systems which would be required to function to protect public health and safety following such failure, destruction, or release are also considered to be vital.

2. ABBREVIATIONS

AC .... Alternating Current

AFW .... Auxiliary Feed Water

AOP .... Abnormal Operating Procedure

ATWS .... Anticipated Transient Without Scram

B&W .... Babcock and Wilcox

BCEMA .... Beaver County Emergency Management Agency

BVERS .... BEAVER VALLEY EMERGENCY RESPONSE SYSTEM

BVPS .... Beaver Valley Power Station

BWST .... Borated Water Storage Tank

CCEMA .... Columbiana County Emergency Management Agency

CCW .... Component Cooling Water

CDE .... Committed Dose Equivalent

CE .... Combustion Engineering

CFR .... Code of Federal Regulations

CR .... CONTROL ROOM

CSF .... Critical Safety Function

CSFST .... Critical Safety Function Status Tree

CVCS .... Chemical and Volume Control System

DBA .... Design Basis Accident

DC .... Direct Current

DEP/BRP .... Dept of Environmental Protection/Bureau of Radiation Protection (Pennsylvania)

DHR .... Decay Heat Removal

DOE .... Department of Energy (US)

DOT .... Department of Transportation

EAL....................EMERGENCY ACTION LEVEL

ECCS....................Emergency Core Cooling System

ECL....................EMERGENCY CLASSIFICATION LEVEL

ED....................EMERGENCY DIRECTOR

EOC....................EMERGENCY OPERATIONS CENTER

EOF....................EMERGENCY OPERATIONS FACILITY

EOP....................EMERGENCY OPERATING PROCEDURE

EPA....................Environmental Protection Agency

EPG....................Emergency Procedure Guideline

EPIP....................Emergency Plan Implementing Procedure

EPRI....................Electric Power Research Institute

EPZ....................EMERGENCY PLANNING ZONE

ERDS....................Emergency Response Data System

ERF....................EMERGENCY RESPONSE FACILITY

ERG....................Emergency Response Guideline

E/RM....................EMERGENCY/RECOVERY MANAGER

ESF....................Engineered Safety Feature

ESW....................Emergency Service Water

FAA....................Federal Aviation Administration

FBI....................Federal Bureau of Investigation

FEMA....................Federal Emergency Management Agency

FENOC....................First Energy Nuclear Operating Company

FPB....................Fission Product Barrier

FRMAP....................Federal Radiation Monitoring and Assessment Plan

FSAR....................Final Safety Analysis Report

GE .... GENERAL EMERGENCY

HCOEM .... Hancock County Office of Emergency Management [C47]

IC.. Initiating Condition

INPO .... Institute for Nuclear Power Operations

IPC .... Inplant Process Computer

IPEEE............................ Individual Plant Examination of External Events (Generic Letter 88-20)

ISFSI .... INDEPENDENT SPENT FUEL STORAGE INSTALLATION

ITS.. Improved Technical Specifications

JPIC.. JOINT PUBLIC INFORMATION CENTER

Keff .... Effective Neutron Multiplication Factor [Cxx]

LEARN .... Law Enforcement Activity Radio Network

LER .... Licensee Event Report

LCO.. Limiting Condition for Operations

LOCA.. Loss of Coolant Accident

LRM .... Licensing Requirements Manual

LWR.. Light Water Reactor

MFW .... Main Feed Water

MIDAS... Meteorological Information and Dose Assessment System

mR.. milliRoentgen

MSIV... Main Steam Isolation Valve

MSL .... Main Steam Line

MSSV.. Main Steam Safety Valve

MW .... Megawatt

NAWAS... National Warning System

NEI .... Nuclear Energy Institute

NESP....................National Environmental Studies Project

NORAD....................North American Aerospace Defense Command

NPP....................Nuclear Power Plant

NRC....................Nuclear Regulatory Commission (US)

NSSS....................Nuclear Steam Supply System

NUMARC....................Nuclear Management and Resources Council

OBE....................Operating Basis Earthquake

OCA....................OWNER CONTROLLED AREA

ODCM/ODAM....................Offsite Dose Calculation (Assessment) Manual

OEMA....................Ohio Emergency Management Agency

ORC....................Offsite Review Committee

ORO....................Offsite Response Organization

OSC....................OPERATIONS SUPPORT CENTER, or Onsite Safety Committee

PA....................PROTECTED AREA

PEMA....................Pennsylvania Emergency Management Agency

POAH....................Point of Adding Heat

PORV....................Power Operated Relief Valve

PRA/PSA....................Probabilistic Risk Assessment / Probabilistic Safety Assessment

PSIG....................Pounds pe Square Inch Gauge

PWR....................Pressurized Water Reactor

R....................Roentgen

RCC....................Reactor Control Console

RCCA....................Rod Cluster Control Assembly

RCDT....................Reactor Coolant Drain Tank

RCP....................Reactor Coolant Pump

RCS .... Reactor Coolant System

REM .... Roentgen Equivalent Man

RPS .... Reactor Protection System

RPV .... Reactor Pressure Vessel

RVLIS .... Reactor Vessel Level Indicating System

SBO .... Station Blackout

SCBA .... Self-Contained Breathing Apparatus

SG .... Steam Generator

SI .... Safety Injection

SLCRS .... Supplemental Leak Collection and Release System

SPDS .... Safety Parameter Display System

SPING .... Special Particulate, Iodine, Noble Gas Monitoring System (Unit 1)

SRO .... Senior Reactor Operator

SSE .... Safe Shutdown Earthquake

TEDE .... Total Effective Dose Equivalent

TOAF .... Top of Active Fuel

TOP .... Temporary Operating Procedure

T/S .... Technical Specification

TID .... Technical Information Document

TSC .... TECHNICAL SUPPORT CENTER

UE .... UNUSUAL EVENT

WE .... Westinghouse Electric

WOG .... Westinghouse Owners Group

WRGM .... Wide Range Gas Monitor (Unit 2)

WVDHS/EM .... West Virginia Division of Homeland Security and Emergency Management [C47]