Ameren MISSOURI

Callaway Plant

July 16, 2015

ULNRC-06230

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

10 CFR 50.47
10 CFR 50.54(q)
10 CFR 50 Appendix E, IV .B.2
10 CFR 50.90

Ladies and Gentlemen:

DOCKET NUMBERS 50-483 AND 72-1045
CALLAWAY PLANT UNIT 1
UNION ELECTRIC CO.
RENEWED FACILITY OPERATING LICENSE NPF-30
SUPPLEMENT TO
RESPONSE TO REQUEST FOR ADDITIONAL INFORMATION (RAI)
RELATED TO LICENSE AMENDMENT REQUEST FOR
EMERGENCY ACTION LEVEL (EAL) UPGRADE
ADOPTING NRC-ENDORSED NEI 99-01, REVISION 6 (TAC NO. MF4945)

References: 1) ULNRC-06143, "License Amendment Request for Emergency Action Level (EAL) Upgrade Adopting NRC-Endorsed NEI 99-01, Revision 6," dated October 2, 2014 (ADAMS Accession Number ML14275A435)
2) ULNRC-06227, "Response to Request For Additional Information (RAI) Related to License Amendment Request for Emergency Action Level (EAL) Upgrade Adopting NRC-Endorsed NEI 99-01, Revision 6 (TAC No. MF4945)," dated July 6, 2015

In Reference 2, Ameren Missouri submitted a response to NRC staff requests for additional information (RAIs) that were needed to complete the staff's review of a license amendment request (submitted as Reference 1) to upgrade the Emergency Action Level scheme associated with the Radiological Emergency Response Plan (RERP) for Callaway Unit 1 by adopting Nuclear Energy Institute (NEI) 99-01, Revision 6, "Development of Emergency Action Levels for Non-Passive Reactors."

As a supplement to the Reference 2 RAI response, corrections and additions to the previously submitted material are provided in the Attachments to this letter. Attachment 1 provides a corrected RAI response matrix, wherein it is noted for the response to RAI 30 that EAL HS5.1 is applicable in Modes 1, 2, 3, 4, 5 and 6. Attachment 2 contains an amended summary of EAL changes that were not associated with RAI responses, including changes to the Bases for EALs CA1.1, CG1.1, SU1.1, SA1.1 and SS1.1 that had been omitted from the Reference 2 submittal. Attachment 3 provides a copy of the proposed RERP Technical Bases document that has been marked-up to reflect the changes made to the original submitted in Reference 1. Also, for information, Attachment 4 provides a copy of the proposed EAL wall charts that shows the corrected revision date (i.e., 7/2/15) in the chart footers.

The Callaway Onsite Review Committee has approved the proposed changes to the RERP and its Technical Bases. In addition, in accordance with 10 CFR 50.91, "Notice for public comment; State consultation," Section (b)(l), a copy of this letter is being provided to the designated Missouri State official.

This letter does not contain new commitments. For any questions concerning this letter, contact Gene Juricic at 573-676-4489 or Pat McKenna at 573-676-8504.

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

Sincerely,

Executed on: July 16, 2015

Scott A. M

Scott A. Maglio
Manager, Regulatory Affairs

JPK/nls

Attachments:

1) Corrected Response to Request for Additional Information (RAI) Emergency Action Level (EAL) Scheme Change
2) Corrected Summary of EAL Changes NOT Associated with RAI Responses
3) Revised Callaway NEI 99-01 Revision 6 EAL Technical Bases (Marked-up Copy)
4) Corrected Callaway NEI 99-01 Revision 6 EAL Wall Charts (Information Only)

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Attachment 1 to

ULNRC-06230

Corrected Response to Request for Additional Information (RAI) Emergency Action Level (EAL) Scheme Change

20 Pages

RAI-CP	SECTION/EAL	Question	CP Response
01	4.3	Section 4.3, "Instrumentation Used for EALs," to NEI 99-01, Revision 6, states "Scheme developers should ensure that specific values used as EAL setpoints are within the calibrated range of the referenced instrumentation." Please confirm that all setpoints and indications used in the CP EAL scheme are within the calibrated range(s) of the stated instrumentation and that the resolution of the instrumentation is appropriate for the setpoint/indication.	CP has confirmed that all setpoints and indications used in the CP EAL scheme are within the calibrated range(s) of the stated instrumentation and that the resolution of the instrumentation is appropriate for the setpoint/indication.
02	1.0	In regards to Section 1, "Purpose," of the proposed EAL Technical Basis: a. Section 4.6, "Basis Document," to NEI 99-01, Revision 6, states "A basis document is an integral part of an emergency classification scheme. The material in this document supports proper emergency classification decision-making by providing informing background and development information in a readily accessible format. It can be referred to in training situations and when making an actual emergency classification, if necessary." Please revise Section 1 of the proposed EAL Technical Basis to reflect the intent of the EAL Basis Document, as provided in NEI 99-01, Revision 6, and remove the proposed purpose discussion, "to facilitate a review of the Callaway EALs," or provide justification for failure to align with NRC endorsed guidance.	a. Deleted: "*It should be used to facilitate review of the Callaway EALs and provide historical documentation for future reference.*"

RAI-CP	SECTION/EAL	Question	CP Response
		b. Section 4.6, "Basis Document," to NEI 99-01, Revision 6, states "Because the information in a basis document can affect emergency classification decision-making…" Therefore, the NRC staff expects that changes to the basis document will be evaluated in accordance with the provisions of 10 CFR 50.54(q). Please incorporate information related to maintaining the technical basis document in accordance with 10 CFR 50.54(q) or provide justification for failure to align with NRC endorsed guidance.	b. The following has been added to Section 1.0 Introduction of the Technical Bases document: *"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)."*
		c. Section 4.7, "EAL/Threshold References to AOP [Abnormal Operating Procedure] and EOP [Emergency Operating Procedure] Setpoints/Criteria," to NEI 99-01, Revision 6, 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." The NRC staff expects that changes to AOPs and EOPs will be evaluated in accordance with the provisions of 10 CFR 50.54(q). Please incorporate information related to screening changes to AOPs or EOPs to determine if an evaluation pursuant to 10 CFR 50.54(q) is required or provide justification for failure to align with NRC endorsed guidance.	c. The following has been added to Section 1.0 Introduction of the Technical Bases document: *"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)."*

RAI-CP	SECTION/EAL	Question	CP Response
03	2.1	Sections 2.1, “Background,” and 4.0, “References,” of the proposed EAL Technical Basis reference an incorrect ADAMS Accession Number (ML110240324). Please verify that the proposed EAL Technical Basis is consistent with NRC endorsed guidance and appropriate ADAMS Accession number is referenced.	Revised the referenced ADAMS Accession No. to ML12326A805.
04	2.1	For Sections 2.1, “Background,” and 4.0, “References,” of the proposed EAL Technical Basis, please provide ADAMS Accession Number that references the endorsed version of NEI 99-01, Revision 6 (ML12326A805).	Revised the referenced ADAMS Accession No. to ML12326A805.
05	2.5	Section 2.5, “Technical Basis Information,” of the proposed EAL Technical Basis includes a Plant-Specific basis section, in addition to a Generic basis section. Considering that the EAL Technical Basis is provided to support proper emergency classification decision making, please explain why a Generic basis section is provided or revise accordingly.	Separate site-specific and generic bases were identified within the EAL bases to facilitate NRC review. These bases sections have now been combined into a single bases section for each EAL. (Throughout document)

RAI-CP	SECTION/EAL	Question	CP Response
06	2.6	Section 2.6, "Operating Mode Applicability," of the proposed EAL Technical Basis contains a brief discussion concerning EAL classification during mode changes. However, this discussion is not as clear as that provided in NRC endorsed guidance. Please justify the omission of significant portions of Section 5.4, "Consideration of Mode Changes During Classification," of NEI 99-01, Revision 6, or revise accordingly.	Revised the cited section to read consistent with Section 5.4 of the generic guidance: "*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.*"

RAI-CP	SECTION/EAL	Question	CP Response
07	3.1.1 3.1.2	In regards to Section 3 of the proposed EAL Technical Basis: Section 3.1.1, "Classification Timeliness," includes a reference to NSIR/DPR-ISG-01, "Interim Staff Guidance, Emergency Planning for Nuclear Power Plants," but does not include a discussion, as provided by NEI 99-01, Revision 6, Section 5.2, Classification Methodology," addressing "[w]hen 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'." Please justify excluding this information or revise accordingly. Section 3.1.2, "Valid Indications," does not include statement, "[t]he validation of indications should be completed in a manner that supports timely emergency declaration," as provided by NEI 99-01, Revision 6, Section 5.1, "General Considerations." Please justify excluding this information or revise accordingly.	Added the cited wording to Section 3.1.1: *"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."* Added the cited statement to Section 3.1.2: *"The validation of indications should be completed in a manner that supports timely emergency declaration,"*

RAI-CP	SECTION/EAL	Question	CP Response
08	5.0	Appendix B, "Definitions," to NEI 99-01, Revision 6, provides definitions for key terms necessary for overall understanding of the NEI 99-01 emergency classification scheme. For Section 5.1, "Definitions," please revise accordingly to add definitions for the following or justify excluding: • CONFINEMENT BOUNDARY, • EMERGENCY ACTION LEVEL, • EMERGENCY CLASSIFICATION LEVEL, • FISSION PRODUCT BARRIER THRESHOLD, and • INITIATING CONDITION. In addition, please consider removing one of the two provided definitions for INDEPENDENT SPENT FUEL STORAGE INSTALLATION to eliminate redundancy.	Added the following definitions to Section 5.1: • Confinement Boundary, • Emergency Action Level • Emergency Classification Level • Fission Product Barrier Threshold • Initiating Condition. Deleted the duplicate ISFSI definition from Section 5.1.
09	6.0	Section 6.0, "Callaway to NEI 99-01 Rev. 6 EAL Cross-Reference," contains the several apparent inconsistencies, as listed below. Please review the Callaway to NEI 99-01, Revision 6, EAL Cross-Reference for accuracy and make corrections as needed. a. Callaway emergency action level RA2.3 is not included in the Callaway to NEI 99-01, Revision 6, EAL Cross-Reference matrix. b. Callaway emergency action level CG1.1 corresponds to NEI 99-01 Rev. 6 EAL CG1 example 1. The Callaway to NEI 99-01, Revision 6, EAL Cross-Reference indicates example 2. c. Callaway emergency action level CG1.2 is not included in the Callaway to NEI 99-01,	a. RA2.3 added to EAL Cross-Reference. b. Revised EAL Cross-Reference to cite correct reference. c. CG1.2 added to EAL Cross-Reference.

		Revision 6, EAL Cross-Reference.	
		d. Callaway emergency action level SU 4.1 corresponds to NEI 99-01, Revision 6, EAL SU3 example 2. Callaway to NEI 99-01, Revision 6, EAL Cross-Reference indicates example 1.	d. Corrected cited reference.
		e. The Callaway to NEI 99-01, Revision 6, EAL Cross-Reference includes Callaway EAL SU4.2. EAL SU4.2 could not be located in the Callaway EAL basis document.	e. Deleted SU4.2 from EAL Cross-Reference. Generic SU 3 example 1 is not implemented at Callaway.
		f. Callaway emergency action level SU8.1 is not included in the Callaway to NEI 99-01, Revision 6, EAL Cross-Reference matrix.	f. Added SU8.1 to EAL Cross-Reference
		g. The Callaway to NEI 99-01, Revision 6, EAL Cross-Reference includes Callaway EAL SA8.1. EAL SA8.1 could not be located in the Callaway EAL basis document.	g. Corrected to read “SA9.1”
		h. Callaway emergency action level SA9.1 is not included in the Callaway to NEI 99-01, Revision 6, EAL Cross-Reference matrix.	h. See “g” above.
		i. Callaway EAL EU 1.1 shows as IU1.1 on the Callaway to NEI 99-01, Revision 6, EAL Cross-Reference. (Note: The “E” designation is correct.)	i. Corrected designation to read “E”

RAI-CP	SECTION/EAL	Question	CP Response
10	RU1.1	For EAL RU1.1, it is not clear how a determination can be made that a "2 X Hi – Hi alarm" condition exists. Please provide justification that a value of two times the alarms identified in Table R-1, "Effluent Monitor Classification Thresholds," can be accurately determined in a timely and accurate manner.	GT-RE-21B, GH-RE-10B & HB-RE18 alarms, Hi-Hi alarm setpoints, and monitor indications are displayed on the RM-11 in the Control Room. It is a simple matter of multiplying by the Hi-Hi alarm setpoint by 2 to get the EAL indicator value. If the RM-11 Hi-Hi alarm were to come in, the Control room Operator would monitor the parameter for EAL applicability.
11	RA1.2, RS1.2, RG1.2	For EALs RA1.2, RS1.2, and RG1.2, please explain why the proposed Note 3, which relates to "effluent flow past an effluent monitor," should be included for EALs that are based on dose assessments or revise accordingly.	Deleted Note 3 applicability to RA1.2, RS1.2 and RG1.2.
12	RA1.1, RS1.1, RG1.1	For EALS RA1.1, RS1.1, and RG1.1, there was a substantial change from the previous to the proposed Table R-1 values. The provided calculations did not contain information that could be used to justify this change. Please provide justification that supports the changes in the Table R-1 values from the previous values to the current values.	The NEI 99-01 Revision 5 Table R-1 calculation is documented in EPCI-08-01 (refer to Attachment 5). The software program MAGNEM was used. Selection of source terms for MAGNEM is limited compared to the current Unified Rascal Interface (URI). A default mix from the FSAR was used. Fifteen-minute release duration was used. All other parameters were duplicated for the Revision 6 Table R-1 values. The NEI 99-01 Revision 6 Table R-1 calculation is documented in EPCI-14-02. URI software is the current dose assessment software used at the Callaway Energy Center. URI replaced MAGNEM to meet the requirement to calculate multiple release points. The URI source term of Clad Damage was selected based on this occurrence being more likely than a core melt down. The 1-hour release duration is based on NEI 99-01, "Methodology for the Development of Emergency Action Levels for Non-Passive Reactors." (For additional details, please refer to Attachment 4.)
13	N/A	Omitted	N/A

RAI-CP	SECTION/EAL	Question	CP Response
14	Generic AA3/HA5	For NEI 99-01, Revision 6, EALs AA3 and HA5, CP is proposing two deviations. NEI Initiating Condition (IC) AA3 example 2, and HA5 example 1 will not be included because a review of CP normal operating and shutdown procedures by Operations Subject Matter Experts concluded that there are no areas external to the Main Control Room that require access to perform a normal plant shutdown and cooldown to Cold Shutdown conditions. a. Please verify that all required manipulations to shut down the plant and enter shutdown cooling can be performed from the Main Control Room or revise accordingly. b. Please verify that no local breaker operations are required or revise accordingly. c. Please provide evidence that an assessment of Control Room availability was performed to support these deviations.	A review of all procedures associated with down power from 100% to Mode 5 (cold shutdown) reviewing field actions that have to be taken. After review, there are no rooms which need to be accessed to shut the plant down to Mode 5. RHR shutdown cooling does not have to be placed in service because CP can cool down to Mode 5 using ASD, steam dumps and MSIV bypasses. Closing the breakers for SI Accumulators and RHR loop suctions valves would not be required. The end result was that a shutdown to cold shutdown can be accomplished from the Control Room alone, without additional actions being performed. The Control Room Ventilation System provides adequate protection from external hazardous gases.

RAI-CP	SECTION/EAL	Question	CP Response
15	RU2.1 RA2.1 RA2.2 RA2.3	For EAL RU2.1, site-specific refueling pathway level indications are not provided per guidance in NEI 99-01, Revision 6. Additionally, the NEI 99-01 Basis discussion does not include the NEI 99-01, Revision 6, EAL AA2 guidance that "This IC applies to irradiated fuel that is licensed for dry storage up to the point that the loaded storage cask is sealed. Once sealed, damage to a loaded cask causing loss of the CONFINEMENT BOUNDARY is classified in accordance with IC E-HU1." a. Please provide site-specific level indications for EAL RU2.1 that could be used to support timely and accurate assessments; include applicable mode availability for this level instrumentation. b. Please justify excluding the NEI 99-01, Revision 6, EAL AA2 guidance that relates to RA2.1, RA2.2, and RA2.3 applicability or revise accordingly. c. Please verify that RA2.1 should be an Alert and revise accordingly.	a. Added "(EC LI-0039A, EC LI-0039B, local observation of SFP level)" as site-specific SFP level indication to RU2.1. b. Added the cited applicability statement to the RA2.1 and RA2.2 bases: *"This EAL applies to irradiated fuel that is licensed for dry storage up to the point that the loaded storage cask is sealed. Once sealed, damage to a loaded cask causing loss of the CONFINEMENT BOUNDARY is classified in accordance with EU1.1."* c. Corrected typo to read "Alert."

RAI-CP	SECTION/EAL	Question	CP Response
16	RA2.2	For EAL RA2.2, the logic was changed from NEI 99-01, Revision 6, guidance that uses an increase in radiation monitor readings to determine that irradiated fuel has been damaged to a proposed logic that requires the operator to know that damage has occurred to irradiated fuel AND an there is an increase in radiation monitor indications. Please develop EAL RA2.2 per NEI 99-01, Revision 6, as endorsed or provide further justification for this deviation.	RA2.2 has been revised to read: "Damage to irradiated fuel resulting in a release of radioactivity from the fuel as indicated by **any** of the following: • Hi-Hi Alarm on Fuel Building exhaust monitors (GG-RE-27 or 28) • Manipulator crane radiation monitor (SD-RE-41) >100 mR/hr • Fuel Pool Bridge Crane OR Spent Fuel Pool Area radiation monitor (SD-RE-37 or 38) > 30 mR/hr"
17	EU1.1	For EAL EU1.1, please explain why symbols were used rather than spelling out "gamma" and "neutron" or revise accordingly.	Revised EU1.1 by replacing the gamma and neutron symbols with the terms "gamma" and "neutron." Reworded the EAL as follows: Damage to a loaded cask CONFINEMENT BOUNDARY as indicated by an on-contact radiation reading > **EITHER** of the following: • 60 mrem/hr (gamma + neutron) on the top of the closure lid of the overpack • 7,000 mrem/hr (gamma + neutron) on the side of the transfer cask This allows for this EAL to cover an accident while moving the spent fuel from the SFP to ISFSI. These numbers are 2x the TS limit. Refer to Certificate of Compliance No.1040, Appendix A, Section 5.3.4 (see Attachment 3). Technical bases were revised to support this change.

RAI-CP	SECTION/EAL	Question	CP Response
18	CA1.1	For EAL CA1.1, a BBLI-53 A/B level of 0 inches is provided as an indication that RCS level is lower than the bottom of the RCS hot leg. The Callaway Basis provides that BBLI-53A/B cannot sense level changes in the Reactor Vessel below the elevation of the RCS loop hot leg penetration. Please provide justification that supports using the minimum value of BBLI-53 A/B for EAL classification as this reading may not be readily differentiated from an instrument failure or revise accordingly.	BBLI-53 A/B indications are trended and displayed when providing RCS level indication. Comparison between BBLI-53A and BBLI-53B, along with their trending together, provides adequate indication of failure of a channel. (No document change)
19	N/A	Omitted	N/A
20	CS1.1 CS1.2	For EAL CS1.1 and CS1.2, the logic was changed from NEI 99-01, Revision 6, guidance without justification. As changed, the EAL appears vague and interpretive. Please develop EAL CS1.1 and CS1.2 per NEI 99-01, Revision 6, as endorsed, or provide further justification for deviation.	Both CS1.1 and CS1.2 wording is consistent with the generic guidance intent. The status of containment closure modifies the level threshold consistent with the generic guidance. (No document change)
21	CS1.3 CG1.2	For EALs CS1.3 and CG1.2, CP did not include “of sufficient magnitude to indicate core uncovery” to the unplanned increase in any sump/tank level to the IC wording. Additionally, CP added “Visual observation of UNISOLABLE RCS leakage” to the IC wording. As proposed, EALs CS1.3 and CG1.2 could result in unnecessary Site and General Emergency declarations. Please provide further justification or revise EAL CS1.3 and CG1.2 accordingly consistent with NEI 99-01, Revision 6, as endorsed.	Added “of sufficient magnitude to indicate core uncovery” to sump/tank level increases in CS1.3 and CG1.2. Deleted visual observation of unisolable RCS leakage from CS1.3 and CG1.2.

RAI-CP	SECTION/EAL	Question	CP Response
22	CG1.2	For EAL CG1.2 and the Containment Fission Product Barrier Potential Loss D.2, it is not clear how CA-3, “Hydrogen Flammability in Containment,” can be used to estimate containment atmosphere hydrogen concentration. Please explain how a procedure to determine hydrogen flammability in containment can be used to estimate containment atmosphere hydrogen concentration or remove the reference to CA-3 from Technical Basis.	Deleted reference to CA-3 in CG1.2.

RAI-CP	SECTION/EAL	Question	CP Response
23	CU2.1, CA2.1, SA1.1, SS1.1, SG1.1, SG1.2	For EALs CU2.1, CA2.1, SU1.1, SA1.1, SS1.1, SG1.1, and SG1.2, AC power sources are provided by Table C-3. Additionally, the Callaway Basis provides that "credit can be taken" for additional sources of power "if they are capable of carrying" an emergency bus. a. Omitted b. Please justify the addition of "Additional sources of offsite power are available from diesel generators such as the Alternate Emergency Power Supply (AEPS) or portable generation sources. Credit can be taken for these sources if they are capable of carrying an NB bus and are aligned within 15 minutes" to the Callaway Basis as this statement could potentially be applied to power supplies not listed on Table C-3 or revise according to NRC endorsed guidance.	a. N/A b. Per the developer notes: *"The EAL and/or Basis section may specify use of a non-safety-related power source provided that operation of this source is controlled in accordance with abnormal or emergency operating procedures, or beyond design basis accident response guidelines (e.g., FLEX support guidelines). Such power sources should generally meet the "Alternate ac source" definition provided in 10 CFR 50.2"* The Alternate Emergency Power Supply (AEPS) was built for, and has been demonstrated to be capable of carrying one of the two Safety Related Emergency Buses. Thus, it would be able to energize and maintain a Safety Related bus and provide the electric power needed to mitigate the consequences of an accident from this non-safety related power supply. In practice, without some initial setup it takes more than 15 minutes to line-up AEPS to a safety related bus. Thus the wording in Callaway's current EAL basis for SA1.1 stating "...if they are capable of carrying an NB bus and are aligned within 15 minutes". The second half of this question concerning "or portable generation sources" has been put in place based on the future actions we intend to perform to comply with the FLEX requirements. This capability has not been put in place, but is anticipated that is will meet the requirements needed to energize a Safety Related 4.160 kV bus. It is not anticipated that Callaway will be able to meet the 15 minute requirement with the portable generator, but if already lined up it would maintain the bus energized and allow for mitigating the accident. (No document change)

RAI-CP	SECTION/EAL	Question	CP Response
24	CU3.1 CA3.1	For EAL CU3.1 and CA3.1, please explain how the addition of "…due to the loss of decay heat removal capability…" to EAL CU3.1 and "…due to a loss of RCS cooling…" to EAL CA3.1 would not result in potential misclassification for an event other than a loss of decay heat removal that leads to an unplanned RCS temperature and/or RCS pressure rise. Please provide justification or revise accordingly consistent with endorsed guidance	Deleted "…due to the loss of decay heat removal capability…" from CU3.1. Deleted "…due to a loss of RCS cooling…" from CA3.1. Added the following omitted generic EAL wording: *"(This EAL does not apply during water-solid plant conditions.)"*
25	CU5.1 SU7.1	For EALs CU5.1 and SU7.1, the Sentry Notification System is provided as an offsite response organization (ORO) communication method for the electronic transmission of a notification form to the OROs. Please provide reference to specific section of the site emergency plan that that identifies the Sentry Notification System as a means of timely notification to OROs for a spectrum of potential event responses or revise accordingly.	Callaway Plant Radiological Emergency Response Plan (RERP), Section 7.2.4. *"The Sentry System provides a means of performing required ORO emergency notifications in a timely manner (within 15 minutes of classification). "* When each ORO acknowledges receipt of a Sentry notification, Sentry provides Callaway Plant with positive confirmation that ORO received the message.

RAI-CP	SECTION/EAL	Question	CP Response
26	CA3.1	For EAL CA3.1, Note 10: "Begin monitoring hot condition EALs concurrently," was added to the provided EAL Technical Basis. It is not clear to the staff how Note 10 would be applied during an UNPLANNED increase in RCS temperature event. Please provide justification for this difference or revise accordingly.	Revised Note 10 to read: "*Begin monitoring hot condition EALs concurrently for any new event or condition not related to the loss of decay heat removal.*" This note re-enforces the implementation guidance in NEI 99-01 that states: *"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.*** The note reminds the end-user that hot condition EALs become applicable for any new event or condition once Mode 4 is entered from Mode 5 during a loss of decay heat removal. (Note 10 revised throughout document and wallchart)
27	CA6.1 SA9.1	For EAL CA6.1 and SA9.1, the Callaway Basis discussion for seismic events refers to a discussion under EAL HU2.1. Please include the discussion on seismic events in the EAL CA6.1 and SA9.1 Callaway Basis or provide justification for not including the discussion as this could impact the timeliness of event assessment.	Added the applicable portion of the seismic event discussion in HU2.1 to the CA6.1 and SA9.1 bases.

RAI-CP	SECTION/EAL	Question	CP Response
28	HU2.1	For EAL HU2.1, the proposed EAL may not be consistent with from NEI 99-01, Revision 6, guidance, which provides that "site-specific indication that a seismic event met or exceeded OBE [operating basis earthquake] limits" should be based on the indications, alarms, and displays of site-specific monitoring equipment. The proposed EAL appears to base the declaration on implementation of an alarm response manual (OTO-SG-00001). Please provide justification for using OTO-SG-00001 for event classification rather than the appropriate seismic monitoring equipment as provided by NRC endorsed guidance or revise accordingly.	The HU2.1 wording was intended for the Unusual Event classification to be driven by receipt of seismic activity annunciator 98D (OBE) in the control room and not by actions performed in the seismic event AOP (OTO-SG-00001). Revised HU2.1 to read" *"Seismic event > OBE as indicated by Seismic Activity, Annunciator 98D."*
29	HU3.2	For EAL HU3.2, the proposed Callaway Basis identifies the Control Building, Battery Room, and ESF Switchgear Room as internal flooding areas of concern. Additionally, the Callaway Basis for HU3.2, which is applicable for all modes, references CA6.1, which is applicable in modes 5 and 6, for internal flooding affecting one or more safety trains. a. Please explain how the statement in EAL HU3.2 that limits flooding areas of concern will not potentially be used to limit a flooding related EAL declaration to only equipment in the Control Building, Battery Room, and the ESF Switchgear Room or revise accordingly. b. Please explain why EAL HU3.2 only references an EAL that is applicable in lower modes.	a. Deleted bases statement related to internal flooding areas. b. Added reference to SA9.1 in addition to CA6.1 in the HU3.2 bases.

RAI-CP	SECTION/EAL	Question	CP Response
30	HS5.1	For EAL HS5.1, please consider the following or provide an explanation how this EAL can be consistently applied: • Addition of operating mode specificity to the listed safety functions to preclude event classification when these safety functions are no longer needed in accordance with site technical specifications; and • Including a "Clock" start time in the Callaway Basis discussion.	1st bullet – Revised to restrict mode applicability of HS5.1 to Modes 1, 2, 3, 4, 5, and 6. None of the three listed safety functions are required when the reactor vessel is defueled. Restricted reactivity control safety function to Modes 1, 2 and 3 only since, by definition, the reactor has adequate shutdown margin while in Modes 4 and 5. 2nd bullet – Added the following to the bases regarding "clock" start time: *"For the purpose of this EAL the 15 minute clock starts when the last licensed operator leaves the Control Room."*
31	SU3 [SU4.1]	For NEI 99-01, Revision 6, EAL SU3.1, CP does not provide an EAL that uses site-specific radiation monitor(s). Please provide additional justification that a Callaway EAL cannot be developed consistent with endorsed guidance or revise accordingly.	Callaway does not have a site-specific radiation monitor correlation that supports identifying a monitor reading that correspond to TS coolant activity limits. The associated EAL is based on whether or not Technical Specification limits for RCS activity have been exceeded. Callaway's Technical Specifications include limits for Iodines governed by Dose Equivalent Iodine (DEI) and noble gases governed by Dose Equivalent Xenon (DEXe). DEI and DEXe both represent weighted sums of measured isotopic concentrations. Determination of DEI and DEXe is performed by laboratory analysis. Callaway's inline process radiation monitors do not have the capability to perform gamma spectrum measurements and determine concentrations of the specific isotopes that contribute to DEI or DEXe. Additionally, the process radiation monitors would not have the capability to perform weighted sums to compare measured count rates with applicable Technical Specification limits. (No Document change)

RAI-CP	SECTION/EAL	Question	CP Response
32	SU4.1	For EAL SU4.1, please explain why the proposed wording is different from the NEI 99-01, Revision 6, guidance which clearly states "sample analysis indicates that…," or revise accordingly.	Revised SU4.1 to read: *"Sample analysis indicates RCS activity > Technical Specification Section 3.4.16 limits"*
33	SU5.1	For EAL SU5.1, please explain how timely declaration can be performed without reliance on a potentially time consuming "manual" method of performing an RCS inventory balance or revise the Callaway Basis accordingly.	Callaway does not rely on a manual method of performing an RCS inventory balance. As stated in the bases: *"Manual or computer-based methods of performing an RCS inventory balance are normally used to determine RCS leakage.* ***The Personal Computer (PC) is preferred method of calculating RCS leak rate.*** *When the PC is used, plant status information and all calculations are generated by the OSPBB9 software program. When the PC software is not available, procedural guidance is available to perform the manual RCS inventory balance."* If the preferred method (computer) is not available, then the manual analysis method is performed, but only in the absence of computer based methods. Both computer and manual methods can be completed within a 15 minute time period. (No document change)
34	SU6.1	For EAL SU6.1, please provide a justification for including a "subsequent automatic trip" to the EAL condition or revise accordingly	The words "subsequent automatic trip" were added to both SU6.1 and SU6.2 to address the condition where an automatic trip signal other than the initial automatic trip failure successfully shuts down the reactor prior to any manual trip action being initiated. For example, if the reactor receives a valid reactor trip signal on high pressurizer pressure but fails to trip, but AMSAC automatically initiates and successfully trips the reactor before the manual trip signal was inserted, the EAL will still have been exceeded and an Unusual Event declared. (No document change)

RAI-CP	SECTION/EAL	Question	CP Response
35	SU6.1, SU6.2, SA6.1, SS6.1	For EALs SU6.1, SU6.2, SA6.1, and SS6.1, please provide further justification as to why greater than or equal to five percent reactor power was added or revise accordingly. (Note: Westinghouse EOPs do not solely rely on Reactor Power level to determine the status of reactor criticality.)	The method used to determine that the reactor is shutdown following a reactor trip, for the purposes of emergency classification, is consistent with the Callaway EOPs (E-0), i.e., indication of reactor power < 5%. This is also the power level that defines power operation in the Technical Specifications. As specified in the generic developers guidance: *"Developers may include site-specific EOP criteria indicative of a successful reactor shutdown in an EAL statement, the Basis or both (e.g., a reactor power level)."* Reactor power < 5% is therefore the site-specific indication of a successful reactor trip for emergency classification. (No document change)

Attachment 2 to

ULNRC-06230

Corrected Summary of EAL Changes NOT Associated with RAI Responses

1 Page

The table below summarizes changes that have been introduced to the EAL submittal documentation for reasons other than the responses to the NRC RAIs.

Section/EAL	Description
5.1 Definitions	Added the abbreviation (OCA) to the Owner Controlled Area definition.
5.1 Definitions	Added the abbreviation (PA) to the Protected Area definition.
5.2 Abbreviations/Acronyms	ODCM - Removed the “-” from Off-site for consistency throughout the document.
7.2	Corrected the Attachment 2 Title to Fission Product Barrier Loss / Potential Loss Matrix and Bases, by adding “Loss / Potential Loss”.
EU1.1	Corrected the section titled “VCSNS Basis Reference” to “Callaway Basis Reference”.
CA1.1	In the Basis section, corrected "6.5%" to "the specified level."
CG1.1	In the Basis and Callaway Basis Reference(s) sections, deleted references to CA-3.
SU1.1,SA1.1 and SS1.1	In the Basis section, deleted the sentence "The 15-minute interval was selected as a threshold to exclude transient or momentary power losses."
SU4.1	In the Basis, corrected “XE-133” to “Xe-133”.
SU6.2	In the Basis, ninth paragraph, second sentence, corrected the sentence by adding “trip” after manually.
SU7.1	In the Basis, corrected the reference from “Table C-5” to “Table S-4”.
SU7.1	In the Basis, added the sentence “This EAL is the hot condition equivalent of the cold condition EAL CU5.1.”, to match the similar sentence in EAL CU5.1.
Fuel Clad B.1. Potential Loss	In the Basis, deleted duplicate “indicates” in first sentence.
RCS C Potential Loss	In the Category line, corrected “**B**. CMT Radiation/RCS Activity” to “**C**. CMT Radiation/RCS Activity”.
RCS E.1 Potential Loss	In the Basis section, corrected “Emergency Director” to “Emergency Coordinator”.

Attachment 3 to

ULNRC-06230

Revised Callaway NEI 99-01 Revision 6 EAL Technical Bases (Marked-up Copy)

246 Pages

EIP-ZZ-00101 ADDENDUM 2

Revision xxx

Draft 07/02/15 (GLJ)

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 Callaway Energy Center (Callaway). Decision-makers responsible for implementation of EIP-ZZ-00101 Classification of Emergencies, may use this document as a technical reference in support of EAL interpretation. This information may assist the Emergency Coordinator in making classifications, particularly those involving judgment or multiple events. The basis information may also be useful in training and for explaining event classifications to offsite officials.

Deleted:). It should be used to facilitate review of the Callaway EALs and provide historical documentation for future reference.

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 Callaway Plant Radiological Emergency Response Plan (RERP).

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

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

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

Subsequently, Revision 6 of NEI 99-01 has been issued which incorporates resolutions to numerous implementation issues including the NRC EAL Frequently Asked Questions (FAQs). Using NEI 99-01 Revision 6, "Methodology for the Development of Emergency Action Levels for Non-Passive Reactors,” November 2012 (ADAMS Accession Number ML12326A805) (ref. 4.1.1), Callaway conducted an EAL implementation upgrade project that produced the EALs discussed herein.

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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. <u>Fuel Clad (FC):</u> The Fuel Clad Barrier consists of the cladding material that contains the fuel pellets.

B. <u>Reactor Coolant System (RCS):</u> 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. <u>Containment (CMT):</u> 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:

<u>Alert:</u>

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

<u>Site Area Emergency:</u>

Loss or potential loss of any two barriers

<u>General Emergency:</u>

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

2.4 EAL Organization

The Callaway EAL scheme includes the following features:

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

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

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

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

EAL Groups, Categories and Subcategories

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

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

2.5 Technical Bases Information

EAL technical bases are provided in Attachment 1 for each EAL according to EAL group (Any, Hot, Cold), EAL category (R, C, H, S, I and F) and EAL subcategory. A summary explanation

of each category and subcategory is given at the beginning of the technical bases discussions of the EALs included in the category. For each EAL, the following information is provided:

Category Letter & Title

Subcategory Number & Title

Initiating Condition (IC)

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

EAL Identifier (enclosed in rectangle)

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

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

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

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

Classification (enclosed in rectangle):

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

EAL (enclosed in rectangle)

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

Mode Applicability

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

Definitions:

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

Basis:

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

Callaway Basis Reference(s):

Site-specific source documentation from which the EAL is derived

2.6 Operating Mode Applicability (ref. 4.1.8)

1 Power Operation
$K_{eff} \geq 0.99$ and reactor thermal power > 5%

2 Startup
$K_{eff} \geq 0.99$ and reactor thermal power ≤ 5%

3 Hot Standby
$K_{eff} < 0.99$ and average coolant temperature ≥ 350ºF

4 Hot Shutdown
$K_{eff} < 0.99$ and average coolant temperature 350ºF > T_{avg} > 200 ºF and at least 53 of 54 reactor vessel head closure bolts fully tensioned

5 Cold Shutdown
$K_{eff} < 0.99$ and average coolant temperature ≤ 200ºF

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

D Defueled
All fuel assemblies have been removed from Containment and placed in the spent fuel pit and the SFP transfer canal gate valve is closed.

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.

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

3.0 GUIDANCE ON MAKING EMERGENCY CLASSIFICATIONS

3.1 General Considerations

When making an emergency classification, the Emergency Coordinator must consider all information having a bearing on the proper assessment of an Initiating Condition (IC). This includes the Emergency Action Level (EAL) plus the associated Operating Mode Applicability, Notes, and the informing basis information. In the Recognition Category F matrices, EALs are based on loss or potential loss of Fission Product Barrier Thresholds.

3.1.1 Classification Timeliness

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

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.”

3.1.2 Valid Indications

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

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

3.1.3 Imminent Conditions

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

3.1.4 Planned vs. Unplanned Events

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

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

3.1.5 Classification Based on Analysis

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

3.1.6 Emergency Coordinator Judgment

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

3.2 Classification Methodology

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

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

3.2.1 Classification of Multiple Events and Conditions

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

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

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

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

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

3.2.2 Consideration of Mode Changes During Classification

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

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

3.2.3 Classification of Imminent Conditions

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

3.2.4 Emergency Classification Level Upgrading and Downgrading

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

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

3.2.5 Classification of Short-Lived Events

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

3.2.6 Classification of Transient Conditions

Many of the ICs and/or EALs employ time-based criteria. These criteria will require that the IC/EAL conditions be present for a defined period of time before an emergency declaration is warranted. In cases where no time-based criterion is specified, it is recognized that some

transient conditions may cause an EAL to be met for a brief period of time (e.g., a few seconds to a few minutes). The following guidance should be applied to the classification of these conditions.

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

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

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

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

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

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

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

3.2.8 Retraction of an Emergency Declaration

Guidance on the retraction of an emergency declaration reported to the NRC is discussed in NUREG-1022 (ref. 4.1.3).

4.0 REFERENCES

4.1 Developmental

4.1.1 NEI 99-01 Revision 6, Methodology for the Development of Emergency Action Levels for Non-Passive Reactors, ADAMS Accession Number ML12326A805

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

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

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

4.1.5 10 § CFR 50.73 License Event Report System

4.1.6 Drawing 8600-X-88100 Property-Site Layout Owner Controlled Area and Surrounding Area

4.1.7 Callaway UFSAR Figure 1.2-44 Plant Area Layout

4.1.8 Technical Specifications Table 1.1-1 Modes

4.1.9 OSP-GT-00003 Containment Closure

4.1.10 Procedure Writers Manual Callaway Plant Procedure Writers Manual

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

4.1.12 Callaway Plant Radiological Emergency Response Plan Emergency Plan (RERP)

4.1.13 OTG-ZZ-00007 Refueling Preparation, Performance and Recovery

4.2 Implementing

4.2.1 EIP-ZZ-00101 Classification of Emergencies

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

4.2.3 Callaway EAL Matrix

5.0 DEFINITIONS, ACRONYMS & ABBREVIATIONS

5.1 Definitions (ref. 4.1.1 except as noted)

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

Alert

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

Confinement Boundary

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

Containment Closure

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

As applied to Callaway, Containment Closure is established when the requirements of OSP-GT-00003 Containment Closure are met (ref. 4.1.9).

Emergency Action Level

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

Emergency Classification Level

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

EPA PAGs

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

Explosion

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

Faulted

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

Fire

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

Fission Product Barrier Threshold

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

Flooding

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

General Emergency

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

Hostage

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

Hostile Action

An act toward Callaway 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 Callaway. Non-terrorism-based EALs should be used to address such activities (i.e., this may include violent acts between individuals in the owner controlled area).

Hostile Force

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

Imminent

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

Impede(d)

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

Independent Spent Fuel Storage Installation (ISFSI)

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

Initiating Condition

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

Maintain

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

Deleted: 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.¶

Owner Controlled Area (OCA)

The fenced area contiguous to the Protected Area, designated by AmerenUE (Callaway Plant) to be controlled for security purposes (ref 4.1.6).

Projectile

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

Protected Area (PA)

An area encompassed by physical barriers and to which access is controlled. The Protected Area refers to the designated security area around the process buildings and is depicted in Drawing 8600-X-88100 Property-Site Layout, Owner Controlled Area and Surrounding Area. (ref. 4.1.7).

RCS Intact

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

Reduced Inventory

Plant condition when fuel is in the reactor vessel and Reactor Coolant System level is lower than 3 feet below the Reactor Vessel flange (< 64.0 in.) (ref. 4.1.13).

Refueling Pathway

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

Ruptured

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

Restore

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

Safety System

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

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

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

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

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

Security Condition

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

Site Area Emergency

Events are in process or have occurred which involve actual or likely major failures of plant functions needed for protection of the public or hostile actions that result in intentional damage or malicious acts; (1) toward site personnel or equipment that could lead to the likely failure of or; (2) that prevent effective access to equipment needed for the protection of the public. Any releases are not expected to result in exposure levels which exceed EPA Protective Action Guidelines exposure levels beyond the site boundary.

Site Boundary

Exclusion Area Boundary is a synonymous term for Site Boundary. The Exclusion Area is defined as the area that encompasses the land surrounding the Plant to a radius of 1,200 meters (3,937 feet) from the midpoint of the Unit 1 Reactor Building and the canceled Unit 2 Reactor Building. Control of access to this is by virtue of ownership and in accordance with 10CFR100 (ref. 4.1.12).

Unisolable

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

Unplanned

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

Unusual Event

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

Valid

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

Visible Damage

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

5.2 Abbreviations/Acronyms

°F .... Degrees Fahrenheit
° .... Degrees
AC .... Alternating Current
ATWS .... Anticipated Transient Without Scram
Callaway .... Callaway Energy Center
CDE .... Committed Dose Equivalent
CFR .... Code of Federal Regulations
CMT .... Containment
CSFST .... Critical Safety Function Status Tree
DBA .... Design Basis Accident
DC .... Direct Current
EAL .... Emergency Action Level
ECCS .... Emergency Core Cooling System
ECL .... Emergency Classification Level
EOF .... Emergency Operations Facility
EOP .... Emergency Operating Procedure
EPA .... Environmental Protection Agency
ERG .... Emergency Response Guideline
EPIP .... Emergency Plan Implementing Procedure
ESF .... Engineered Safety Feature
ESW .... Emergency Service Water
FAA .... Federal Aviation Administration
FBI .... Federal Bureau of Investigation
FEMA .... Federal Emergency Management Agency
FSAR .... Final Safety Analysis Report
GE .... General Emergency
IC .... Initiating Condition
IPEEE .... Individual Plant Examination of External Events (Generic Letter 88-20)
K_{eff} .... Effective Neutron Multiplication Factor
LCO .... Limiting Condition of Operation
LER .... Licensee Event Report
LOCA .... Loss of Coolant Accident
LWR .... Light Water Reactor
MPC .... Maximum Permissible Concentration/Multi-Purpose Canister
mR, mRem, mrem, mREM .... milli-Roentgen Equivalent Man
MSL .... Main Steam Line
MW .... Megawatt
NEI .... Nuclear Energy Institute
NESP .... National Environmental Studies Project

NPP .... Nuclear Power Plant
NRC .... Nuclear Regulatory Commission
NSSS .... Nuclear Steam Supply System
NORAD .... North American Aerospace Defense Command
(NO)UE .... Notification of Unusual Event
OBE .... Operating Basis Earthquake
OCA .... Owner Controlled Area
ODCM .... Offsite Dose Calculation Manual
ORO .... Offsite Response Organization
OTO .... Off-Normal Operating Procedure
PA .... Protected Area
PAG .... Protective Action Guideline
PRA/PSA .... Probabilistic Risk Assessment / Probabilistic Safety Assessment
PWR .... Pressurized Water Reactor
PSIG .... Pounds per Square Inch Gauge
R .... Roentgen
RCC .... Reactor Control Console
RCS .... Reactor Coolant System
Rem, rem, REM .... Roentgen Equivalent Man
RETS .... Radiological Effluent Technical Specifications
RPS .... Reactor Protection System
R(P)V .... Reactor (Pressure) Vessel
RVLIS .... Reactor Vessel Level Indicating System
SAR .... Safety Analysis Report
SBO .... Station Blackout
SCBA .... Self-Contained Breathing Apparatus
SG .... Steam Generator
SI .... Safety Injection
ODCM .... Selected Licensee Commitment
SPDS .... Safety Parameter Display System
SRO .... Senior Reactor Operator
SSF .... Safe Shutdown Facility
TEDE .... Total Effective Dose Equivalent
TOAF .... Top of Active Fuel
TSC .... Technical Support Center
WOG .... Westinghouse Owners Group

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

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

Callaway	NEI 99-01 Rev. 6	
EAL	**IC**	**Example EAL**
RU1.1	AU1	1, 2
RU1.2	AU1	3
RU2.1	AU2	1
RA1.1	AA1	1
RA1.2	AA1	2
RA1.3	AA1	3
RA1.4	AA1	4
RA2.1	AA2	1
RA2.2	AA2	2
RA2.3	AA2	3
RA3.1	AA3	1
RS1.1	AS1	1
RS1.2	AS1	2
RS1.3	AS1	3
RS2.1	AS2	1
RG1.1	AG1	1
RG1.2	AG1	2
RG1.3	AG1	3
RG2.1	AG2	1
CU1.1	CU1	1
CU1.2	CU1	2

Callaway	NEI 99-01 Rev. 6	
EAL	**IC**	**Example EAL**
CU2.1	CU2	1
CU3.1	CU3	1
CU3.2	CU3	2
CU4.1	CU4	1
CU5.1	CU5	1, 2, 3
CA1.1	CA1	1
CA1.2	CA1	2
CA2.1	CA2	1
CA3.1	CA3	1, 2
CA6.1	CA6	1
CS1.1	CS1	1
CS1.2	CS1	2
CS1.3	CS1	3
CG1.1	CG1	1
CG1.2	CG1	2
FA1.1	FA1	1
FS1.1	FS1	1
FG1.1	FG1	1
HU1.1	HU1	1, 2 3
HU2.1	HU2	1
HU3.1	HU3	1
HU3.2	HU3	2
HU3.3	HU3	3
HU3.4	HU3	4
HU4.1	HU4	1

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Callaway	NEI 99-01 Rev. 6	
EAL	IC	Example EAL
HU4.2	HU4	2
HU4.3	HU4	3
HU4.4	HU4	4
HU6.1	HU7	1
HA1.1	HA1	1, 2
HA5.1	HA6	1
HA6.1	HA7	1
HS1.1	HS1	1
HS5.1	HS6	1
HS6.1	HS7	1
HG1.1	HG1	1
HG6.1	HG7	1
SU1.1	SU1	1
SU3.1	SU2	1
SU4.1	SU3	2
SU5.1	SU4	1, 2, 3
SU6.1	SU5	1
SU6.2	SU5	2
SU7.1	SU6	1, 2, 3
SU8.1	SU7	1, 2
SA1.1	SA1	1
SA3.1	SA2	1
SA6.1	SA5	1
SA9.1	SA9	1
SS1.1	SS1	1

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Deleted: SU4.2 ... [1]

Deleted: SA8

Callaway	NEI 99-01 Rev. 6	
EAL	IC	Example EAL
SS2.1	SS8	1
SS6.1	SS5	1
SG1.1	SG1	1
SG1.2	SG8	1
EU1.1	E-HU1	1

Deleted: IU1

7.0 ATTACHMENTS

7.1 Attachment 1, Emergency Action Level Technical Bases

7.2 Attachment 2, Fission Product Barrier Loss/Potential Loss Matrix and Basis

Category R – Abnormal Rad Release / Rad Effluent

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

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

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

Events of this category pertain to the following subcategories:

1. Radiological Effluent

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

2. Irradiated Fuel Event

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

3. Area Radiation Levels

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

Category: R – Abnormal Rad Levels / Rad Effluent

Subcategory: 1 – Radiological Effluent

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

EAL:

RU1.1 Unusual Event

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

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

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

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

Table R-1 Effluent Monitor Classification Thresholds

	Release Point	Monitor	GE	SAE	Alert	UE
Gaseous	Unit Vent	GT-RE-21B	6.59E+7 µCi/sec	6.59E+6 µCi/sec	6.59E+5 µCi/sec	2 X Hi-Hi alarm
Gaseous	ASD Monitors (A/B/C/D)	AB-RE-111/112/113/114	12 mR/hr	1.2 mR/hr	----	----
Gaseous	TD AFW Steam Discharge	FC-RE-385	163 mR/hr	16.3 mR/hr	1.6 mR/hr	----
Gaseous	Radwaste Bldg Vent	GH-RE-10B	----	----	----	2 X Hi-Hi alarm
Liquid	Liquid Radwaste Discharge	HB-RE-18	----	----	----	2 X Hi-Hi alarm

Mode Applicability:

All

Definition(s):

None

Basis:

The column "UE" gaseous and liquid release values in Table R-1 represent two times the appropriate ODCM release rate limits associated with the specified monitors (ref. 1, 2, 3).

The effluent monitor Hi-Hi alarm setpoints correspond to the Hi alarm (red) setpoint as displayed on RM-11.

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

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

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

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

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

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

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

Callaway Basis Reference(s):

1. APA-ZZ-01003 Callaway Plant Offsite Dose Calculation Manual Section 2.2.3
2. FSAR Section 16.11.1.3, Radioactive Effluent Monitoring Instrumentation LCO
3. EPCI 11402 EAL Table R-1 Calculations
4. NEI 99-01 AU1

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Deleted: NEI 99-01 Basis:¶

Category: R – Abnormal Rad Levels / Rad Effluent

Subcategory: 1 – Radiological Effluent

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

EAL:

RU1.2 Unusual Event

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

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

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

Mode Applicability:

All

Definition(s):

None

Basis:

Deleted: Callaway

None

Deleted: NEI 99-01 Basis:¶

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

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

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

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

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

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

Callaway Basis Reference(s):

1. APA-ZZ-01003 Callaway Plant Offsite Dose Calculation Manual Section 2.2.3
2. NEI 99-01 AU1

Category: R – Abnormal Rad Levels / Rad Effluent

Subcategory: 1 – Radiological Effluent

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

EAL:

RA1.1 Alert

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

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

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

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

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

Table R-1 Effluent Monitor Classification Thresholds

	Release Point	Monitor	GE	SAE	Alert	UE
Gaseous	Unit Vent	GT-RE-21B	6.59E+7 µCi/sec	6.59E+6 µCi/sec	6.59E+5 µCi/sec	2 X Hi-Hi alarm
Gaseous	ASD Monitors (A/B/C/D)	AB-RE-111/112/113/114	12 mR/hr	1.2 mR/hr	----	----
Gaseous	TD AFW Steam Discharge	FC-RE-385	163 mR/hr	16.3 mR/hr	1.6 mR/hr	----
Gaseous	Radwaste Bldg Vent	GH-RE-10B	----	----	----	2 X Hi-Hi alarm
Liquid	Liquid Radwaste Discharge	HB-RE-18	----	----	----	2 X Hi-Hi alarm

Mode Applicability:

All

Definition(s):

None

Basis:

Deleted: Callaway

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

- 10 mRem TEDE
- 50 mRem CDE Thyroid

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

Deleted: NEI 99-01 Basis:¶

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

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

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

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

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

Callaway Basis Reference(s):

1. EPCI 11402 EAL Table R-1 Calculations
2. NEI 99-01 AA1

Category: R – Abnormal Rad Levels / Rad Effluent

Subcategory: 1 – Radiological Effluent

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

EAL:

RA1.2 Alert

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

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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.

Deleted: 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

Definition(s):

SITE BOUNDARY - Exclusion Area Boundary is a synonymous term for Site Boundary. The Exclusion Area is defined as the area that encompasses the land surrounding the Plant to a radius of 1,200 meters (3,937 feet) from the midpoint of the Unit 1 Reactor Building and the canceled Unit 2 Reactor Building. Control of access to this is by virtue of ownership and in accordance with 10CFR100.

Basis:

Deleted: Callaway

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

Deleted: NEI 99-01 Basis:¶

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

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

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

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

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

Callaway Basis Reference(s):

1. EIP-ZZ-01211 Accident Dose Assessment
2. EPCI 11402 EAL Table R-1 Calculations
3. NEI 99-01 AA1

Category: R – Abnormal Rad Levels / Rad Effluent

Subcategory: 1 – Radiological Effluent

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

EAL:

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

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

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

Mode Applicability:

All

Definition(s):

SITE BOUNDARY - Exclusion Area Boundary is a synonymous term for Site Boundary. The Exclusion Area is defined as the area that encompasses the land surrounding the Plant to a radius of 1,200 meters (3,937 feet) from the midpoint of the Unit 1 Reactor Building and the canceled Unit 2 Reactor Building. Control of access to this is by virtue of ownership and in accordance with 10CFR100.

Basis:

Deleted: Callaway

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

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

Deleted: NEI 99-01 Basis:¶

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.

Callaway Basis Reference(s):

1. APA-ZZ-01003 Callaway Plant Offsite Dose Calculation Manual Section 2.2.3
2. NEI 99-01 AA1

Category: R – Abnormal Rad Levels / Rad Effluent

Subcategory: 1 – Radiological Effluent

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

EAL:

RA1.4 Alert

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

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

(Notes 1, 2)

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

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

Mode Applicability:

All

Definition(s):

SITE BOUNDARY - Exclusion Area Boundary is a synonymous term for Site Boundary. The Exclusion Area is defined as the area that encompasses the land surrounding the Plant to a radius of 1,200 meters (3,937 feet) from the midpoint of the Unit 1 Reactor Building and the canceled Unit 2 Reactor Building. Control of access to this is by virtue of ownership and in accordance with 10CFR100.

Basis:

Deleted: Callaway

EIP-ZZ-00211, Field Monitoring provides guidance for emergency or post-accident radiological environmental monitoring (ref. 1).

Deleted: NEI 99-01 Basis:¶

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

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

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

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

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

Callaway Basis Reference(s):

1. EIP-ZZ-00211, Field Monitoring
2. NEI 99-01 AA1

Category: R – Abnormal Rad Levels / Rad Effluent

Subcategory: 1 – Radiological Effluent

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

EAL:

RS1.1 Site Area Emergency

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

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

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

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

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

Table R-1 Effluent Monitor Classification Thresholds

	Release Point	Monitor	GE	SAE	Alert	UE
Gaseous	Unit Vent	GT-RE-21B	6.59E+7 µCi/sec	6.59E+6 µCi/sec	6.59E+5 µCi/sec	2 X Hi-Hi alarm
Gaseous	ASD Monitors (A/B/C/D)	AB-RE-111/112/113/114	12 mR/hr	1.2 mR/hr	----	----
Gaseous	TD AFW Steam Discharge	FC-RE-385	163 mR/hr	16.3 mR/hr	1.6 mR/hr	----
Gaseous	Radwaste Bldg Vent	GH-RE-10B	----	----	----	2 X Hi-Hi alarm
Liquid	Liquid Radwaste Discharge	HB-RE-18	----	----	----	2 X Hi-Hi alarm

Mode Applicability:

All

Definition(s):

None

Basis:

Deleted: Callaway

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

- 100 mRem TEDE
- 500 mRem CDE Thyroid

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

Deleted: NEI 99-01 Basis:¶

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

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

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

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

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

Callaway Basis Reference(s):

1. EPCI 11402 EAL Table R-1 Calculations
2. NEI 99-01 AS1

Category: R – Abnormal Rad Levels / Rad Effluent

Subcategory: 1 – Radiological Effluent

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

EAL:

RS1.2 Site Area Emergency

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

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

Deleted: s 3,

Deleted: 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

Definition(s):

SITE BOUNDARY - Exclusion Area Boundary is a synonymous term for Site Boundary. The Exclusion Area is defined as the area that encompasses the land surrounding the Plant to a radius of 1,200 meters (3,937 feet) from the midpoint of the Unit 1 Reactor Building and the canceled Unit 2 Reactor Building. Control of access to this is by virtue of ownership and in accordance with 10CFR100.

Basis:

Deleted: Callaway

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

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.

Deleted: NEI 99-01 Basis:¶

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.

Callaway Basis Reference(s):

1. EIP-ZZ-01211 Accident Dose Assessment
2. EPCI 11402 EAL Table R-1 Calculations
3. NEI 99-01 AS1

Category: R – Abnormal Rad Levels / Rad Effluent

Subcategory: 1 – Radiological Effluent

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

EAL:

RS1.3 Site Area Emergency

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

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

(Notes 1, 2)

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

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

Mode Applicability:

All

Definition(s):

SITE BOUNDARY - Exclusion Area Boundary is a synonymous term for Site Boundary. The Exclusion Area is defined as the area that encompasses the land surrounding the Plant to a radius of 1,200 meters (3,937 feet) from the midpoint of the Unit 1 Reactor Building and the canceled Unit 2 Reactor Building. Control of access to this is by virtue of ownership and in accordance with 10CFR100.

Basis:

Deleted: Callaway

EIP-ZZ-00211, Field Monitoring provides guidance for emergency or post-accident radiological environmental monitoring (ref. 1).

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

Deleted: NEI 99-01Basis:¶

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.

Callaway Basis Reference(s):

1. EIP-ZZ-00211, Field Monitoring
2. NEI 99-01 AS1

Category: R – Abnormal Rad Levels / Rad Effluent

Subcategory: 1 – Radiological Effluent

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

EAL:

RG1.1 General Emergency

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

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

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

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

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

Table R-1 Effluent Monitor Classification Thresholds

	Release Point	Monitor	GE	SAE	Alert	UE
Gaseous	Unit Vent	GT-RE-21B	6.59E+7 µCi/sec	6.59E+6 µCi/sec	6.59E+5 µCi/sec	2 X Hi-Hi alarm
Gaseous	ASD Monitors (A/B/C/D)	AB-RE-111/112/113/114	12 mR/hr	1.2 mR/hr	----	----
Gaseous	TD AFW Steam Discharge	FC-RE-385	163 mR/hr	16.3 mR/hr	1.6 mR/hr	----
Gaseous	Radwaste Bldg Vent	GH-RE-10B	----	----	----	2 X Hi-Hi alarm
Liquid	Liquid Radwaste Discharge	HB-RE-18	----	----	----	2 X Hi-Hi alarm

Mode Applicability:

All

Definition(s):

None

Basis:

Deleted: Callaway

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

- 1000 mRem TEDE
- 5000 mRem CDE Thyroid

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

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.

Deleted: NEI 99-01Basis:¶

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.

Callaway Basis Reference(s):

1. EPCI 11402 EAL Table R-1 Calculations
2. NEI 99-01 AG1

Category: R – Abnormal Rad Levels / Rad Effluent

Subcategory: 1 – Radiological Effluent

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

EAL:

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

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

Deleted: s 3,

Deleted: 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

Definition(s):

SITE BOUNDARY - Exclusion Area Boundary is a synonymous term for Site Boundary. The Exclusion Area is defined as the area that encompasses the land surrounding the Plant to a radius of 1,200 meters (3,937 feet) from the midpoint of the Unit 1 Reactor Building and the canceled Unit 2 Reactor Building. Control of access to this is by virtue of ownership and in accordance with 10CFR100.

Basis:

Deleted: Callaway

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

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.

Deleted: NEI 99-01 Basis:¶

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.

Callaway Basis Reference(s):

1. EIP-ZZ-01211 Accident Dose Assessment

2. EPCI 11402 EAL Table R-1 Calculations
3. NEI 99-01 AG1

Category: R – Abnormal Rad Levels / Rad Effluent

Subcategory: 1 – Radiological Effluent

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

EAL:

RG1.3 General Emergency

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

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

(Notes 1, 2)

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

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

Mode Applicability:

All

Definition(s):

SITE BOUNDARY - Exclusion Area Boundary is a synonymous term for Site Boundary. The Exclusion Area is defined as the area that encompasses the land surrounding the Plant to a radius of 1,200 meters (3,937 feet) from the midpoint of the Unit 1 Reactor Building and the canceled Unit 2 Reactor Building. Control of access to this is by virtue of ownership and in accordance with 10CFR100.

Basis:

Deleted: Callaway

EIP-ZZ-00211, Field Monitoring s provides guidance for emergency or post-accident radiological environmental monitoring (ref. 1).

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

Deleted: NEI 99-01 Basis:¶

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.

Callaway Basis Reference(s):

1. EIP-ZZ-00211, Field Monitoring
2. NEI 99-01 AG1

Category: R – Abnormal Rad Levels / Rad Effluent

Subcategory: 2 – Irradiated Fuel Event

Initiating Condition: Unplanned loss of water level above irradiated fuel

EAL:

RU2.1 Unusual Event

UNPLANNED water level drop in the REFUELING PATHWAY as indicated by low water level alarm or indication (EC LI-0039A, EC LI-0039B, local observation of SFP level)

AND

UNPLANNED rise in corresponding area radiation levels as indicated by **any** Table R-2 radiation monitors

Table R-2 Fuel Building & Containment Area Radiation Monitors
Fuel Building: • SD-RE-34, Cask Handle Area Radiation • SD-RE-35, New Fuel Storage Area Radiation • SD-RE-36, New Fuel Storage Area Radiation • SD-RE-37, Fuel Pool Bridge Crane Radiation • SD-RE-38, Spent Fuel Pool Area Radiation **Containment:** • SD-RE-40, Personnel Access Hatch Area • SD-RE-41, Manipulator Crane Radiation Monitor • SD-RE-42, Containment Building Radiation • GT-RE-59 Containment High Area Radiation Monitor • GT-RE-60 Containment High Area Radiation Monitor

Mode Applicability:

All

Definition(s):

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

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

Basis:

Deleted: Callaway

The low water level alarm in this EAL refers to the Spent Fuel Pool (SFP) low level alarm (Window Number 76D, SFP LEV HI LO) (ref. 1). During the fuel transfer phase of refueling operations, the fuel transfer canal is normally in communication with the spent fuel pool and the refueling pool in the Containment is in communication with the fuel transfer canal when the fuel transfer tube is open. A lowering in water level in the SFP, fuel transfer canal or refueling pool is therefore sensed by the SFP low level alarm. Neither the refueling pool nor the fuel transfer canal is equipped with a low level alarm (ref. 1). The SFP level is remotely monitored by level indicator EC LI-0039A. The level switch initiates high and low level annunciators

Technical Specification Section 3.7.15 (ref. 2) requires at least 23 ft of water above the Spent Fuel Pool storage racks. Technical Specification Section 3.9.7 (ref. 3) requires at least 23 ft of water above the Reactor Vessel flange in the refueling pool. During refueling, this maintains sufficient water level in the fuel transfer canal, refueling pool, and SFP to retain iodine fission product activity in the water in the event of a fuel handling accident.

The Table R-2 radiation monitors are those expected to see increase area radiation levels as a result of a loss of REFUELING PATHWAY inventory (ref. 1). Increasing radiation indications on these monitors in the absence of indications of decreasing REFUELING CAVITY level are not classifiable under this EAL.

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

Deleted: NEI 99-01 Basis:¶

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

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

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

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

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

Callaway Basis Reference(s):

1. OTO-EC-00001 Loss of Spent Fuel Pool/Refuel Pool Level
2. Technical Specification Section 3.7.15 Fuel Storage Pool Water Level
3. Technical Specification Section 3.9.7 Refueling Pool Water Level
4. NEI 99-01 AU2

Category: R – Abnormal Rad Levels / Rad Effluent

Subcategory: 2 – Irradiated Fuel Event

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

EAL:

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

Deleted: Unusual Event

Mode Applicability:

All

Definition(s):

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

Basis:

Deleted: Callaway

Deleted: None.¶
NEI 99-01 Basis:¶

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

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

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.

Callaway Basis Reference(s):

1. OTO-EC-00001 Loss of Spent Fuel Pool/Refuel Pool Level
2. NEI 99-01 AA2

ATTACHMENT 1
EAL Bases

Category: R – Abnormal Rad Levels / Rad Effluent

Subcategory: 2 – Irradiated Fuel Event

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

EAL:

RA2.2 Alert

Damage to irradiated fuel resulting in a release of radioactivity from the fuel as indicated by **any** of the following:

- Hi-Hi Alarm on Fuel Building exhaust monitors (GG-RE-27 or 28)
- Manipulator crane radiation monitor (SD-RE-41) >100 mR/hr
- Fuel Pool Bridge Crane OR Spent Fuel Pool Area radiation monitor (SD-RE-37 or 38) > 30 mR/hr

Deleted: Damage to irradiated fuel resulting in a release of radioactivity ¶ **AND**¶ **Any**

Deleted: radiation monitor indications

Mode Applicability:

All

Definition(s):

None

Basis:

Deleted: Callaway

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

The bases for the SFP ventilation radiation Hi-Hi alarm and the SFP and containment area radiation readings are a spent fuel handling accident (ref. 2, 3). In the Fuel Handling Building, a fuel assembly could be dropped in the fuel transfer canal or in the SFP. Should a fuel assembly be dropped in the fuel transfer canal or in the SFP and release radioactivity above a prescribed level, the fuel handling building ventilation monitors sound an alarm, alerting personnel to the problem (ref. 1, 2, 3, 4).

Deleted: NEI 99-01 Basis:¶

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

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

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

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

potential fuel damaging event (e.g., a fuel handling accident).

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

Callaway Basis Reference(s):

1. OTO-EC-00001 Loss of Spent Fuel Pool/Refuel Pool Level
2. OTO-KE-00001 Fuel Handling Accident
3. Calc. EPCI 98-01 Emergency Action Level Bases
4. Calc. HPCI 05-02 Gaseous and Liquid Radiation Monitor Setpoints
5. NEI 99-01 AA2

Category: R – Abnormal Rad Levels / Rad Effluent

Subcategory: 2 – Irradiated Fuel Event

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

EAL:

RA2.3	**Alert**
Lowering of spent fuel pool level to 2031 ft. 1.25 in. (Level 2)	

Mode Applicability:

All

Definition(s):

None

Basis:

Deleted: Callaway

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).

For Callaway Plant SFP Level 2 is plant elevation 2031 ft. 1.25 in. (~9 ft. 11 in. above the top of the spent fuel racks) as indicated on EC-LI-0059B in the Control Room or EC-LI-0059A in the Auxiliary Building MG Set Room. Backup indication is also available on EC-LI-0060A in the Auxiliary Building MG Set Room.

Deleted: NEI 99-01 Basis:¶

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

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

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

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

Callaway Basis Reference(s):

1. NRC EA-12-51 Issuance of Order to Modify Licenses with Regard to Reliable Spent Fuel Pool Instrumentation
2. SFPIS Mod Overview for EP
3 NEI 99-01 AA2

Category: R – Abnormal Rad Levels / Rad Effluent

Subcategory: 2 – Irradiated Fuel Event

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

EAL:

RS2.1	**Site Area Emergency**
Lowering of spent fuel pool level to 2022 ft. 1.25 in. (Level 3)	

Mode Applicability:

All

Definition(s):

None

Basis:

Deleted: Callaway

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).

For Callaway Plant SFP Level 3 has been set at a plant elevation 2022 ft. 1.25 in. (~11 in. above the top of the spent fuel racks) as indicated on EC-LI-0059B in the Control Room or EC-LI-0059A in the Auxiliary Building MG Set Room. Backup indication is also available on EC-LI-0060A in the Auxiliary Building MG Set Room.

Deleted: NEI 99-01 Basis:¶

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

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

Escalation of the emergency classification level would be via IC AG1 or RG2.

Callaway Basis Reference(s):

1. NRC EA-12-51 Issuance of Order to Modify Licenses with Regard to Reliable Spent Fuel Pool Instrumentation
2. SFPIS Mod Overview for EP
3. NEI 99-01 AS2

Category: R – Abnormal Rad Levels / Rad Effluent

Subcategory: 2 – Irradiated Fuel Event

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

EAL:

RG2.1	**General Emergency**
Spent fuel pool level cannot be restored to at least 2022 ft. 1.25 in. (Level 3) for ≥ 60 min. (Note 1)	

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

Mode Applicability:

All

Definition(s):

None

Basis:

Deleted: Callaway

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).

For Callaway Plant SFP Level 3 has been set at plant elevation 2022 ft. 1.25 in. (~11 in. above the top of the spent fuel racks) as indicated on EC-LI-0059B in the Control Room or EC-LI-0059A in the Auxiliary Building MG Set Room. Backup indication is also available on EC-LI-0060A in the Auxiliary Building MG Set Room.

Deleted: NEI 99-01 Basis:¶

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

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

Callaway Basis Reference(s):

1. NRC EA-12-51 Issuance of Order to Modify Licenses with Regard to Reliable Spent Fuel Pool Instrumentation
2. SFPIS Mod Overview for EP
3. NEI 99-01 AG2

Category: R – Abnormal Rad Levels / Rad Effluent

Subcategory: 3 – Area Radiation Levels

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

EAL:

RA3.1	**Alert**
Dose rates > 15 mR/hr in **EITHER** of the following areas:	
Control Room (SD-RE-33)	
OR	
Central Alarm Station (by survey)	

Mode Applicability:

All

Definition(s):

None

Basis:

Deleted: Callaway

Areas that meet this threshold include the Control Room and the Central Alarm Station (CAS). SD-RE-33 monitors the Control room for area radiation (ref. 1). The CAS is included in this EAL because of its' importance to permitting access to areas required to assure safe plant operations.

There is no permanently installed CAS area radiation monitors that may be used to assess this EAL threshold. Therefore this threshold must be assessed via local radiation survey for the CAS (ref. 1).

Deleted: NEI 99-01 Basis:¶

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

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

Callaway Basis Reference(s):

1. FSAR Section 12.3 Table 12.3-2 Area Radiation Monitors
2. NEI 99-01 AA3

Category E – Independent Spent Fuel Storage Installation (ISFSI)

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

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

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

Category: ISFSI

Subcategory: Confinement Boundary

Initiating Condition: Damage to a loaded cask CONFINEMENT BOUNDARY

EAL:

EU1.1 Unusual Event

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

- 60 mrem/hr (gamma + neutron) on the top of the closure lid of the overpack
- 7,000 mrem/hr (gamma + neutron) on the side of the transfer cask

Deleted: $\gamma + \eta$

Mode Applicability:

All

Definition(s):

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

Basis:

Deleted: Plant-Specific¶

Overpacks are the HI-STORM UMAX VVM casks which receive and contain the sealed MPCs for interim storage in the ISFSI. They provide gamma and neutron shielding, and provide for ventilated air flow to promote heat transfer from the MPC to the environs. The term overpack does not include the transfer cask (ref. 1).

The values shown represents 2 times the limits specified in the ISFSI Certificate of Compliance Technical Specification section 5.3.4 for radiation external to either a loaded MPC overpack or transfer cask (ref. 1).

Deleted: Generic¶

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.

Callaway Basis Reference(s):

Deleted: VCSNS

1. Certificate of Compliance No. 1040 Appendix A Technical Specifications for the HI-STORM UMAX Canister Storage System
2. NEI 99-01 E-HU1

Category C – Cold Shutdown / Refueling System Malfunction

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

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

The events of this category pertain to the following subcategories:

1. RCS Level

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

2. Loss of Emergency AC Power

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

3. RCS Temperature

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

4. Loss of Vital DC Power

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

5. Loss of Communications

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

6. Hazardous Event Affecting Safety Systems

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

Category: C – Cold Shutdown / Refueling System Malfunction

Subcategory: 1 – RCS Level

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

EAL:

CU1.1	**Unusual Event**
UNPLANNED loss of reactor coolant results in RCS water level less than a required lower limit for ≥ 15 min. (Note 1)	

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

Mode Applicability:

5 - Cold Shutdown, 6 - Refueling

Definition(s):

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

Basis:

Deleted: Callaway

With the plant in Cold Shutdown, RCS water level is normally maintained above the pressurizer low level setpoint of 17% (ref. 1). However, if RCS level is being controlled below the pressurizer low level setpoint, or if level is being maintained in a designated band in the reactor vessel it is the inability to maintain level above the low end of the designated control band due to a loss of inventory resulting from a leak in the RCS that is the concern.

With the plant in Refueling mode, RCS water level is normally maintained at or above the reactor vessel flange (Technical Specification LCO 3.9.7 requires at least 23 ft. of water above the top of the reactor vessel flange in the refueling cavity during refueling operations) (ref. 2).

The Plant Computer System Display called Refuel Level Indications (turn on code RLI) is available to assist in monitoring important parameters crucial to RCS draining operations (ref. 3).

Deleted: NEI 99-01 Basis:¶

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

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

This EAL recognizes that the minimum required RCS level can change several times during the course of a refueling outage as different plant configurations and system lineups are implemented. This EAL is met if the minimum level, specified for the current plant conditions,

cannot be maintained for 15 minutes or longer. The minimum level is typically specified in the applicable operating procedure but may be specified in another controlling document.

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

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

Callaway Basis Reference(s):

1. FR-I.2 Response to Low Pressurizer Level
2. OTN-BB-00002 Reactor Coolant System Draining
3. Technical Specification Section 3.9.7 Refueling Pool Water Level
4. NEI 99-01 CU1

Category: C – Cold Shutdown / Refueling System Malfunction

Subcategory: 1 – RCS Level

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

EAL:

CU1.2 Unusual Event

RCS water level cannot be monitored

AND EITHER

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

Table C-1 Sumps / Tanks
• Containment Sumps • Containment Normal Sumps • Containment Instrument Sump • PRT • RCDT • Auxiliary Building Sump

Mode Applicability:

5 - Cold Shutdown, 6 – Refueling

Definition(s):

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

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

Basis:

Deleted: Callaway

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).

The Plant Computer System Display called Refuel Level Indications (turn on code RLI) is available to assist in monitoring important parameters crucial to RCS draining operations (ref. 3).

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.

Deleted: NEI 99-01 Basis:¶

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 level have been lost. In this condition, operators may determine that an inventory loss is occurring by observing changes in sump and/or tank levels (Table C-1). Sump and/or tank level changes must be evaluated against other potential sources of water flow to ensure they are indicative of leakage from the RCS.

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

Callaway Basis Reference(s):

1. OTO-BB-00003-R014 Excess RCS Leakage
2. OSP-BB-00009 RCS Inventory Balance
3. OTN-BB-00002 Reactor Coolant System Draining
4. NEI 99-01 CU1

Category: C – Cold Shutdown / Refueling System Malfunction

Subcategory: 1 – RCS Level

Initiating Condition: Loss of RCS inventory

EAL:

CA1.1	**Alert**
Loss of RCS inventory as indicated by Reactor Vessel level < bottom of RCS hot leg ID (RVLIS Pumps Off < 73% or BBLI-53 A/B at 0 inches)	

Mode Applicability:

5 - Cold Shutdown, 6 – Refueling

Definition(s):

None

Basis:

Deleted: Callaway

When Reactor Vessel water level lowers to 2013.29 ft (ref. 1), the inside diameter (ID) of the bottom of the RCS hot leg penetration is uncovered. The elevation of the bottom of the RCS hot leg penetration can be monitored only by RVLIS. (Note that this threshold is the loop penetration at the Reactor Vessel not the low point of the loop.) Level monitoring instruments BB LI-53A/B and Computer Point BBL0053BB cannot sense level changes in the Reactor Vessel below the elevation of the RCS loop hot leg penetration. The Plant Computer System Display called Refuel Level Indications (turn on code RLI) is available to assist in monitoring important parameters crucial to RCS draining operations (ref. 3). The RVLIS Pumps Off threshold has been determined as follows (ref. 1, 2):

Elevation of bottom of Reactor Vessel (ft) A	1987.150
Elevation of bottom ID of RCS hot leg penetration (ft) B	2013.290
Hot leg penetration (above vessel bottom) C = B - A (ft)	26.140
Height of vessel D (ft)	41.245
RVLIS indication corresponding to the top of the core: H = 100 x C / D (%)	63.377
RVLIS overall channel accuracy: OCA = 7.48% + (0.0104 x H) + 0.81%	---
OCA at H (%)	8.949
Bottom ID of RCS loop, including channel uncertainties: H + OCA (%)	72.327
Rounded upward to nearest 1% (RVLIS range is 0 - 120% in 2% increments)	73

The threshold was chosen because level indication may be lost (RVLIS is normally inoperable in Refueling mode (ref. 2)) and loss of suction to decay heat removal systems has occurred. The inability to restore and maintain level after reaching this setpoint infers a failure of the RCS barrier.

Deleted: NEI 99-01 Basis:¶

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

Deleted: 6.5%

For this EAL, a lowering of RCS water level below the specified level indicates that operator actions have not been successful in restoring and maintaining RCS water level. The heat-up

rate of the coolant will increase as the available water inventory is reduced. A continuing decrease in water level will lead to core uncovery.

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

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

Callaway Basis Reference(s):

1. OOA-BB-00003 Refuel Level Indications
2. Calculation No. BB-177 (387.1 - CAL RVLIS Setpoints)
3. OTN-BB-00002 Reactor Coolant System Draining
4. NEI 99-01 CA1

Category: C – Cold Shutdown / Refueling System Malfunction

Subcategory: 1 – RCS Level

Initiating Condition: Loss of RCS inventory

EAL:

CA1.2 Alert

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

AND EITHER

- UNPLANNED increase in **any** Table C-1 Sump / Tank level
- Visual observation of UNISOLABLE RCS leakage

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

Table C-1 Sumps / Tanks
• Containment Sumps • Containment Normal Sumps • Containment Instrument Sump • PRT • RCDT • Auxiliary Building Sump

Mode Applicability:

5 - Cold Shutdown, 6 – Refueling

Definition(s):

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

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

Basis:

Deleted: Callaway

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 (Table C-1). 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 systems connected to the RCS that cannot be isolated could also be indicative of a loss of RCS inventory (ref. 1, 2).

The Plant Computer System Display called Refuel Level Indications (turn on code RLI) is available to assist in monitoring important parameters crucial to RCS draining operations (ref. 3).

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.

Deleted: NEI 99-01 Basis:¶

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

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

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

Callaway Basis Reference(s):

1. OTO-BB-00003-R014 Excess RCS Leakage
2. OSP-BB-00009 RCS Inventory Balance
3. OTN-BB-00002 Reactor Coolant System Draining
4. NEI 99-01 CA1

Category: C – Cold Shutdown / Refueling System Malfunction

Subcategory: 1 – RCS Level

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

EAL:

CS1.1	**Site Area Emergency**
With CONTAINMENT CLOSURE **not** established, RVLIS Pumps Off < 72%	

Mode Applicability:

5 – Cold Shutdown, 6 – Refueling

Definition(s):

CONTAINMENT CLOSURE - The procedurally defined conditions or actions taken to secure Primary or Secondary Containment and its associated structures, systems, and components as a functional barrier to fission product release under shutdown conditions.

As applied to Callaway, Containment Closure is established when the requirements of OSP-GT-00003 Containment Closure are met.

Basis:

Deleted: Callaway

When Reactor Vessel water level lowers to 2012.79 ft (ref. 1), water level is six inches below the elevation of the bottom of the RCS hot leg penetration. When Reactor Vessel water level drops significantly below the elevation of the bottom of the RCS hot leg penetration, all sources of RCS injection have failed or are incapable of making up for the inventory loss. Six inches below the elevation of the bottom of the RCS hot leg penetration can be monitored only by RVLIS. Level monitoring instruments BB LI-53A/B and Computer Point BBL0053BB cannot sense level changes in the Reactor Vessel below the elevation of the RCS loop hot leg penetration. The Plant Computer System Display called Refuel Level Indications (turn on code RLI) is available to assist in monitoring important parameters crucial to RCS draining operations (ref. 3). The RVLIS Pumps Off threshold has been determined as follows (ref. 1, 2):

Elevation of bottom of Reactor Vessel (ft) A	1987.150
Elevation of bottom ID of RCS hot leg penetration (ft) B	2013.290
Six inches below hot leg penetration (above vessel bottom) C = B - A - 0.5 (ft)	25.640
Height of vessel D (ft)	41.245
RVLIS indication corresponding to the top of the core: H = 100 x C / D (%)	62.165
RVLIS overall channel accuracy: OCA = 7.48% + (0.0104 x H) + 0.81%	---
OCA at H (%)	8.937
Bottom ID of RCS loop, including channel uncertainties: H + OCA (%)	71.102
Rounded upward to nearest 1% (RVLIS range is 0 - 120% in 2% increments)	72

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

inability to restore and maintain level after reaching this setpoint infers a failure of the RCS barrier and potential loss of the Fuel Clad barrier.

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

This IC addresses a significant and prolonged loss of reactor vessel/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.

Deleted: NEI 99-01 Basis:¶

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

Outage/shutdown contingency plans typically provide for re-establishing or verifying CONTAINMENT CLOSURE following a loss of heat removal or RCS inventory control functions. The difference in the specified RCS/reactor vessel levels of EALs CS1.1 and CS2.2 reflect the fact that with CONTAINMENT CLOSURE established, there is a lower probability of a fission product release to the environment.

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

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

Callaway Basis Reference(s):

1. OOA-BB-00003 Refuel Level Indications
2. Calculation No. BB-177 (387.1 - CAL RVLIS Setpoints)
3. OTN-BB-00002 Reactor Coolant System Draining
4. OSP-GT-00003 Containment Closure
5. NEI 99-01 CS1

Category: C – Cold Shutdown / Refueling System Malfunction

Subcategory: 1 – RCS Level

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

EAL:

CS1.2	**Site Area Emergency**
With CONTAINMENT CLOSURE established, RVLIS Pumps Off < 65% (Top of Fuel)	

Mode Applicability:

5 – Cold Shutdown, 6 – Refueling

Definition(s):

CONTAINMENT CLOSURE - The procedurally defined conditions or actions taken to secure Primary or Secondary Containment and its associated structures, systems, and components as a functional barrier to fission product release under shutdown conditions.

As applied to Callaway, Containment Closure is established when the requirements of OSP-GT-00003 Containment Closure are met.

Basis:

Deleted: Callaway

When Reactor Vessel water level drops below RVLIS Pumps Off indication of 65% (2010.29 ft), core uncovery is about to occur. The Plant Computer System Display called Refuel Level Indications (turn on code RLI) is available to assist in monitoring important parameters crucial to RCS draining operations (ref. 3). The RVLIS Pumps Off threshold has been determined as follows (ref. 1, 2):

Elevation of bottom of Reactor Vessel (ft) A	1987.150
Elevation of top of fuel (ft) B	2010.290
Height of top of core (above vessel bottom) C = B - A (ft)	23.140
Height of vessel D (ft)	41.245
RVLIS indication corresponding to the top of the core: H = 100 x C / D (%)	56.104
RVLIS overall channel accuracy: OCA = 7.48% + (0.0104 x H) + 0.81%	—
OCA at H (%)	8.873
Top of core, including channel uncertainties: H + OCA (%)	64.977
Rounded upward to nearest 1% (RVLIS range is 0 - 120% in 2% increments)	65

Under the conditions specified by this EAL, continued lowering of Reactor Vessel water level is indicative of a loss of inventory control. Inventory loss may be due to a vessel breach, RCS pressure boundary leakage or continued boiling in the Reactor Vessel. The magnitude of this

loss of water indicates that makeup systems have not been effective and may not be capable of preventing further RCS or Reactor Vessel water level drop and potential core uncovery. The inability to restore and maintain level after reaching this setpoint infers a failure of the RCS barrier and Potential Loss of the Fuel Clad barrier.

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

This IC addresses a significant and prolonged loss of reactor vessel/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.

Deleted: NEI 99-01 Basis:¶

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

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

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

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

Callaway Basis Reference(s):

1. OOA-BB-00003 Refuel Level Indications
2. Calculation No. BB-177 (387.1 - CAL RVLIS Setpoints)
3. OTN-BB-00002 Reactor Coolant System Draining
4. OSP-GT-00003 Containment Closure
5. NEI 99-01 CS1

Category: C – Cold Shutdown / Refueling System Malfunction

Subcategory: 1 – RCS Level

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

EAL:

CS1.3 Site Area Emergency

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

AND

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

- UNPLANNED increase in **any** Table C-1 sump/tank level of sufficient magnitude to indicate core uncovery
- Manipulator crane radiation monitor SD-RE-41 > 10,000 mR/hr
- Erratic Source Range Monitor indication

Deleted: <#>Visual observation of UNISOLABLE RCS leakage ¶

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

Table C-1 Sumps / Tanks
• Containment Sumps • Containment Normal Sumps • Containment Instrument Sump • PRT • RCDT • Auxiliary Building Sump

Mode Applicability:

5 – Cold Shutdown, 6 – Refueling

Definition(s):

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

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

Basis:

Deleted: Callaway

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

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

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 (Table C-1).

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 systems connected to the RCS that cannot be isolated could also be indicative of a loss of RCS inventory (ref. 1, 2).

The Plant Computer System Display called Refuel Level Indications (turn on code RLI) is available to assist in monitoring important parameters crucial to RCS draining operations (ref. 3).

The Reactor Vessel inventory loss may be detected by the manipulator crane radiation monitor 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 up-scaled manipulator crane radiation monitor (SD-RE-41) indication (ref. 4, 5, 6).

Post-TMI accident studies indicated that the installed PWR nuclear instrumentation will operate erratically when the core is uncovered and that this should be used as a tool for making such determinations (ref. 7, 8).

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.

Deleted: NEI 99-01 Basis:¶

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.

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

Callaway Basis Reference(s):

1. OTO-BB-00003-R014 Excess RCS Leakage
2. OSP-BB-00009 RCS Inventory Balance
3. OTN-BB-00002 Reactor Coolant System Draining
4. FSAR Section 12.3.3.4
5. FSAR Table 12.3-2
6. Calc. No. HPCI -0701 SD-RE-41 Response to Core Uncovery in Refueling Mode
7. Severe Accident Management Guidance Technical Basis Report, Volume 1: Candidate High-Level Actions and Their Effects, pgs 2-18, 2-19
8. Nuclear Safety Analysis Center (NSAC), 1980, “Analysis of Three Mile Island - Unit 2 Accident,” NSAC-1
9. NEI 99-01 CS1

Category: C – Cold Shutdown / Refueling System Malfunction

Subcategory: 1 – RCS Level

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

EAL:

CG1.1	**General Emergency**
RVLIS Pumps Off < 65% (Top of Fuel) for ≥ 30 min. (Note 1)	
AND	
Any Containment Challenge indication, Table C-2	

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

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

Table C-2 Containment Challenge Indications
• CONTAINMENT CLOSURE **not** established (Note 6) • Containment hydrogen concentration ≥ 4% • Unplanned rise in Containment pressure

Mode Applicability:

5 – Cold Shutdown, 6 – Refueling

Definition(s):

CONTAINMENT CLOSURE - The procedurally defined conditions or actions taken to secure Primary or Secondary Containment and its associated structures, systems, and components as a functional barrier to fission product release under shutdown conditions.

As applied to Callaway, Containment Closure is established when the requirements of OSP-GT-00003 Containment Closure are met.

Basis:

Deleted: Callaway

When Reactor Vessel water level drops below RVLIS Pumps Off indication of 65% (2010.29 ft), core uncovery is about to occur. The Plant Computer System Display called Refuel Level Indications (turn on code RLI) is available to assist in monitoring important parameters crucial to RCS draining operations (ref. 3). The RVLIS Pumps Off threshold has been determined as follows (ref. 1, 2):

Elevation of bottom of Reactor Vessel (ft) A	1987.150
Elevation of top of fuel (ft) B	2010.290
Height of top of core (above vessel bottom) C = B - A (ft)	23.140
Height of vessel D (ft)	41.245
RVLIS indication corresponding to the top of the core: H = 100 x C / D (%)	56.104
RVLIS overall channel accuracy: OCA = 7.48% + (0.0104 x H) + 0.81%	—
OCA at H (%)	8.873
Top of core, including channel uncertainties: H + OCA (%)	64.977
Rounded upward to nearest 1% (RVLIS range is 0 - 120% in 2% increments)	65

Three conditions are associated with a challenge to Containment integrity:

1. CONTAINMENT COSURE not established - The status of Containment closure is tracked if plant conditions change that could raise the risk of a fission product release as a result of a loss of decay heat removal (ref. 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. Callaway is equipped with a Hydrogen Control System (HCS) which serves to limit or reduce combustible gas concentrations in the Containment. The HCS is an engineered safety feature with redundant hydrogen recombiners, hydrogen mixing system, hydrogen monitoring subsystem, and a backup hydrogen purge subsystem. The HCS is designed to maintain the Containment hydrogen concentration below 4% by volume (ref. 5). Two Containment hydrogen monitors (GS AI-10 and GS AI-19) with a range of 0% to 10% provide indication on Control Room Panel RL020 and ERFIS (ref. 6, 7). The hydrogen monitors require a 2 hour warmup period when starting from the OFF position and 15 minutes when starting from STANDBY (ref. 8, 9).
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 (ref. 4).

Deleted: If an actual hydrogen concentration measurement is unavailable, CA-3 may be used to estimate the Containment atmosphere hydrogen concentration (ref. 10).

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

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

Deleted: NEI 99-01 Basis:¶

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.

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*.

Callaway Basis Reference(s):

1. OOA-BB-00003 Refuel Level Indications
2. Calculation No. BB-177 (387.1 - CAL RVLIS Setpoints)
3. OTN-BB-00002 Reactor Coolant System Draining

4. OSP-GT-00003 Containment Closure
5. FSAR Section 6.2.5
6. FSAR Table 7A-3 (Sheet 31)
7. Technical Specifications 3.3.3
8. OTN-GS-00001 Containment Hydrogen Control System
9. Calc No. 392.2 XX-95 Callaway Containment Parameters EOP Action Values, Setpoint ID T101 & T102
10. NEI 99-01 CS1

Deleted: 10. CA-3 Hydrogen Flammability in Containment¶
11

Category: C – Cold Shutdown / Refueling System Malfunction

Subcategory: 1 – RCS Level

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

EAL:

CG1.2 General Emergency

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

AND

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

- UNPLANNED increase in **any** Table C-1 sump/tank level of sufficient magnitude to indicate core uncovery
- Manipulator crane radiation monitor SD-RE-41 > 10,000 mR/hr
- Erratic Source Range Monitor indication

AND

Any Containment Challenge indication, Table C-2

Deleted: <#>Visual observation of UNISOLABLE RCS leakage ¶

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

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

Table C-1 Sumps / Tanks
• Containment Sumps • Containment Normal Sumps • Containment Instrument Sump • PRT • RCDT • Auxiliary Building Sump

Table C-2 Containment Challenge Indications
• CONTAINMENT CLOSURE **not** established (Note 6) • Containment hydrogen concentration ≥ 4% • Unplanned rise in Containment pressure

Mode Applicability:

5 - Cold Shutdown, 6 – Refueling

Definition(s):

CONTAINMENT CLOSURE - The procedurally defined conditions or actions taken to secure Primary or Secondary Containment and its associated structures, systems, and components as a functional barrier to fission product release under shutdown conditions.

As applied to Callaway, Containment Closure is established when the requirements of OSP-GT-00003 Containment Closure are met.

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

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

Basis:

Deleted: Callaway

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

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

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 (Table C-1). 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 systems connected to the RCS that cannot be isolated could also be indicative of a loss of RCS inventory (ref. 1, 2).

The Plant Computer System Display called Refuel Level Indications (turn on code RLI) is available to assist in monitoring important parameters crucial to RCS draining operations (ref. 3).

The Reactor Vessel inventory loss may be detected by the manipulator crane radiation monitor 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 up-scaled manipulator crane radiation monitor (SD-RE-41) indication (ref. 4, 5, 6).

Post-TMI accident studies indicated that the installed PWR nuclear instrumentation will operate erratically when the core is uncovered and that this should be used as a tool for making such determinations (ref. 7, 8).

Three conditions are associated with a challenge to Containment integrity:

1. CONTAINMENT COSURE not established - The status of Containment closure is tracked if plant conditions change that could raise the risk of a fission product release as a result of a loss of decay heat removal (ref. 15). 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. Callaway is equipped with a Hydrogen Control System (HCS) which serves to limit or reduce combustible gas concentrations in the Containment. The HCS is an engineered safety feature with redundant hydrogen recombiners, hydrogen mixing system, hydrogen monitoring subsystem, and a backup hydrogen purge subsystem. The HCS is designed to maintain the Containment hydrogen concentration below 4% by volume (ref. 9). Two Containment hydrogen monitors (GS AI-10 and GS AI-19) with a range of 0% to 10% provide indication on Control Room Panel RL020 and ERFIS (ref. 10, 11). The hydrogen monitors require a 2 hour warmup period when starting from the OFF position and 15 minutes when starting from STANDBY (ref. 12, 13
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 (ref. 15).

Deleted:). If an actual hydrogen concentration measurement is unavailable, CA-3 may be used to estimate the Containment atmosphere hydrogen concentration (ref. 14).

Deleted: NEI 99-01 Basis:¶

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

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

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

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

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

The 30-minute criterion is tied to a readily recognizable event start time (i.e., the total loss of ability to monitor level), and allows sufficient time to monitor, assess and correlate reactor and plant conditions to determine if core uncovery has actually occurred (i.e., to account for various

accident progression and instrumentation uncertainties). It also allows sufficient time for performance of actions to terminate leakage, recover inventory control/makeup equipment and/or restore level monitoring.

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

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

Callaway Basis Reference(s):

1. OTO-BB-00003-R014 Excess RCS Leakage
2. OSP-BB-00009 RCS Inventory Balance
3. OTN-BB-00002 Reactor Coolant System Draining
4. FSAR Section 12.3.3.4
5. FSAR Table 12.3-2
6. Calc. No. HPCI -0701 SD-RE-41 Response to Core Uncovery in Refueling Mode
7. Severe Accident Management Guidance Technical Basis Report, Volume 1: Candidate High-Level Actions and Their Effects, pgs 2-18, 2-19
8. Nuclear Safety Analysis Center (NSAC), 1980, "Analysis of Three Mile Island - Unit 2 Accident," NSAC-1
9. FSAR Section 6.2.5
10. FSAR Table 7A-3 (Sheet 31)
11. Technical Specifications 3.3.3
12. OTN-GS-00001 Containment Hydrogen Control System
13. Calc No. 392.2 XX-95 Callaway Containment Parameters EOP Action Values, Setpoint ID T101 & T102
14. OSP-GT-00003 Containment Closure
15. NEI 99-01 CG1

Deleted: 14. CA-3 Hydrogen Flammability in Containment¶
15

Deleted: 16

Category: C – Cold Shutdown / Refueling System Malfunction

Subcategory: 2 – Loss of Emergency AC Power

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

EAL:

CU2.1	**Unusual Event**

AC power capability, Table C-3, to emergency 4.16KV buses NB01 and NB02 reduced to a single power source for ≥ 15 min. (Note 1)

AND

Any additional single power source failure will result in loss of **all** AC power to SAFETY SYSTEMS

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

Table C-3 AC Power Sources
Offsite: • Safeguards XFMR A or B via ESF LTC XFMR XNB01 • Startup XFMR XMR01 via ESF LTC XFMR XNB02 • Main XFMR XMA01 backfed via UAT XFMR XMA02 (only if already aligned) **Onsite:** • EDG NE01 • EDG NE02

Mode Applicability:

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

Definition(s):

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

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

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

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

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

Basis:

Deleted: Callaway

For emergency classification purposes, "capability" means that an offsite AC power source(s) is available to the emergency buses, whether or not the buses are powered from it.

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.

4.16KV buses NB01 and NB02 are the emergency (essential) buses. NB01 supplies power to Load Group 1 (Red Train) safety related loads and NB02 supplies power to Load Group 2 (Yellow Train) safety related loads. Each bus has two sources of offsite power. One source is from 13.8 KV safeguards transformer A or B via ESF Load Tap Changing (LTC) transformer XNB01 and the other source is from the startup transformer XMR01 via ESF LTC transformer XNB02. Transformer XNB01 is the normal supply to bus NB01; XNB02 is the normal supply to bus NB02 (ref. 1, 2, 3).

In addition, NB01 and NB02 each have an emergency diesel generator which supply electrical power to the bus automatically in the event that the preferred source becomes unavailable (ref. 1).

Another method to obtain offsite power is by backfeeding the emergency buses through the main transformer XMA01 and unit auxiliary transformer XMR02. This is only done during cold shutdown unless nuclear safety considerations require it to be done during hot shutdown when no other power sources are available (ref. 4).

Additional sources of offsite power are available from diesel generators such as the Alternate Emergency Power Supply (AEPS) or portable generation sources. Credit can be taken for these sources if they are capable of carrying an NB bus and are aligned within 15 minutes.

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

Deleted: NEI 99-01 Basis:¶

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

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

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

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

emergency buses being back-fed from an offsite power source.

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

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

Callaway Basis Reference(s):

1. E-21001(Q) Main Single Line Diagram (Electrical Distribution Diagram)
2. FSAR Site Addenda Section 8.2.1.2
3. FSAR Section 8.3.1
4. OTS-MA-00001-R011 Main Step-Up Transformer Backfeed - IPTE
5. NEI 99-01 CU2

Category: C – Cold Shutdown / Refueling System Malfunction

Subcategory: 2 – Loss of Emergency AC Power

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

EAL:

CA2.1 Alert

Loss of **all** offsite and **all** onsite AC power capability, Table C-3, to emergency 4.16KV buses NB01 and NB02 for ≥ 15 min. (Note 1)

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

Table C-3 AC Power Sources
Offsite: • Safeguards XFMR A or B via ESF LTC XFMR XNB01 • Startup XFMR XMR01 via ESF LTC XFMR XNB02 • Main XFMR XMA01 backfed via UAT XFMR XMA02 (only if already aligned) **Onsite:** • EDG NE01 • EDG NE02

Mode Applicability:

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

Basis:

Deleted: Callaway

For emergency classification purposes, "capability" means that an offsite AC power source(s) is available to the emergency buses, whether or not the buses are powered from it.

The emergency 4.16KV AC System provides the power requirements for operation and safe shutdown of the plant. The essential switchgear are buses NB01 and NB02 (ref. 1).

4.16KV buses NB01 and NB02 are the emergency (essential) buses. NB01 supplies power to Load Group 1 (Red Train) safety related loads and NB02 supplies power to Load Group 2 (Yellow Train) safety related loads. Each bus has two sources of offsite power. One source is from 13.8 KV safeguards transformer A or B via ESF Load Tap Changing (LTC) transformer XNB01 and the other source is from the startup transformer XMR01 via ESF LTC transformer XNB02. Transformer XNB01 is the normal supply to bus NB01; XNB02 is the normal supply to bus NB02 (ref. 1, 2, 3).

In addition, NB01 and NB02 each have an emergency diesel generator which supply electrical power to the bus automatically in the event that the preferred source becomes unavailable (ref. 1).

Another method to obtain offsite power is by backfeeding the emergency buses through the main transformer XMA01 and unit auxiliary transformer XMR02. This is only done during cold shutdown unless nuclear safety considerations require it to be done during hot shutdown when no other power sources are available (ref. 4).

Additional sources of offsite power are available from diesel generators such as the Alternate Emergency Power Supply (AEPS) or portable generation sources. Credit can be taken for these sources if they are capable of carrying an NB bus and are aligned within 15 minutes.

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

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.

Deleted: NEI 99-01 Basis:¶

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.

Callaway Basis Reference(s):

1. E-21001(Q) Main Single Line Diagram (Electrical Distribution Diagram)
2. FSAR Site Addenda Section 8.2.1.2
3. FSAR Section 8.3.1
4. OTS-MA-00001-R011 Main Step-Up Transformer Backfeed - IPTE
5. NEI 99-01 CA2

Category: C – Cold Shutdown / Refueling System Malfunction

Subcategory: 3 – RCS Temperature

Initiating Condition: UNPLANNED increase in RCS temperature

EAL:

CU3.1	**Unusual Event**
UNPLANNED increase in RCS temperature to > 200°F (Note 10)	

Deleted: due to loss of decay heat removal capability

Note 10: 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

Definition(s):

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

Basis:

Deleted: Callaway

Several instruments are capable of providing indication of RCS temperature with respect to the Technical Specification cold shutdown temperature limit (200°F, ref. 1). These include core exit thermocouples (T/Cs) and Wide Range hot leg temperature indications. Plant computer screens are available for monitoring heatup and cooldown (e.g., MODE3, HEATU, COOLD, MODE4, ACCUM, and MODE5). The most limiting temperature indication should be used. For example, during heatup, the highest reading temperature indication should be used; during cooldown, the lowest (ref. 2, 3, 4).

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.

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

Deleted: NEI 99-01 Basis:¶

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

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

During an outage, the level in the reactor vessel will normally be maintained at or above the reactor vessel flange. Refueling evolutions that lower water level below the reactor vessel flange are carefully planned and controlled. A loss of forced decay heat removal at reduced

inventory may result in a rapid increase in reactor coolant temperature depending on the time after shutdown.

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

Callaway Basis Reference(s):

1. Callaway Technical Specifications Table 1.1-1
2. OTG-ZZ-00001 Plant Heatup Cold Shutdown to Hot Standby
3. OSP-BB-00007 RCS Heatup and Cooldown Limitations, Note before Section 6.1 and Attachment 2
4. FSAR Section 7.2.2.3.2
5. NEI 99-01 CU3

Category: C – Cold Shutdown / Refueling System Malfunction

Subcategory: 3 – RCS Temperature

Initiating Condition: UNPLANNED increase in RCS temperature

EAL:

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

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

Mode Applicability:

5 - Cold Shutdown, 6- Refueling

Definition(s):

None

Basis:

Deleted: Callaway

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

- RCS Loop level indications):
  - Indicators on RL018:
    BB LI-53A, RCS (LOOP 1) HOT LEG LEV
    BB LI-53B, RCS (LOOP 4) HOT LEG LEV
  - Computer points:
    BBL0053A, RCS LOOP 1 HOT LEG LEVEL
    BBL0053B, RCS LOOP 4 HOT LEG LEVEL
    BBL053BB, RCS LOOP LEVEL – CTMT VENTED
- RVLIS LI-1311, LI-1312, LI-1321, and LI-1322 (if in service) (ref. 3, 4)
- Visual observation (if vessel head is removed) (ref. 5)

The Plant Computer System Display called Refuel Level Indications (turn on code RLI) is available to assist in monitoring important parameters crucial to RCS draining operations (ref. 3).

Several instruments are capable of providing indication of RCS temperature with respect to the Technical Specification cold shutdown temperature limit (200°F, ref. 1). These include core exit thermocouples (T/Cs) and Wide Range hot leg temperature indications. Plant computer screens are available for monitoring heatup and cooldown (e.g., MODE3, HEATU, COOLD, MODE4, ACCUM, and MODE5). The most limiting temperature indication should be used. For

example, during heatup, the highest reading temperature indication should be used; during cooldown, the lowest (ref. 6, 7, 8).

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

Deleted: NEI 99-01 Basis:¶

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

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

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

Callaway Basis Reference(s):

1. Callaway Technical Specifications Table 1.1-1
2. OOA-BB-00003 Refuel Level Indications
3. OTN-BB-00002 Reactor Coolant System Draining
4. FSAR Section 18.2.13.2
5. OTS-KE-00018 Draining the Refueling Pool
6. OTG-ZZ-00001 Plant Heatup Cold Shutdown to Hot Standby
7. OSP-BB-00007 RCS Heatup and Cooldown Limitations, Note before Section 6.1 and Attachment 2
8. FSAR Section 7.2.2.3.2
9. NEI 99-01 CU3

Category: C – Cold Shutdown / Refueling System Malfunction

Subcategory: 3 – RCS Temperature

Initiating Condition: Inability to maintain plant in cold shutdown

EAL:

CA3.1 Alert

UNPLANNED increase in RCS temperature to > 200°F for > Table C-4 duration (Notes 1, 10)

OR

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

Deleted: due to a loss of RCS cooling

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

Note 10: Begin monitoring hot condition EALs concurrently for any new event or condition not related to the loss of decay heat removal.

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

Mode Applicability:

5 - Cold Shutdown, 6 – Refueling

Definition(s):

CONTAINMENT CLOSURE - The procedurally defined conditions or actions taken to secure Primary or Secondary Containment and its associated structures, systems, and components as a functional barrier to fission product release under shutdown conditions.

As applied to Callaway, Containment Closure is established when the requirements of OSP-GT-00003 Containment Closure are met.

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

REDUCED INVENTORY -. Plant condition when fuel is in the reactor vessel and Reactor Coolant System level is lower than 3 feet below the Reactor Vessel flange (< 64.0 in.).

Deleted: Callaway

Basis:

Several instruments are capable of providing indication of RCS temperature with respect to the Technical Specification cold shutdown temperature limit (200℉, ref. 1). These include core exit thermocouples (T/Cs) and Wide Range hot leg temperature indications. Plant computer screens are available for monitoring heatup and cooldown (e.g., MODE3, HEATU, COOLD, MODE4, ACCUM, and MODE5). The most limiting temperature indication should be used. For example, during heatup, the highest reading temperature indication should be used; during cooldown, the lowest (ref. 2, 3, 4).

RCS pressure instrument BB PI-403A is capable of measuring pressure to less than 10 psig (ref. 5).

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.

Deleted: NEI 99-01 Basis:¶

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

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

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

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

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

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

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

Callaway Basis Reference(s):

1. Callaway Technical Specifications Table 1.1-1

2. OTG-ZZ-00001 Plant Heatup Cold Shutdown to Hot Standby
3. OSP-BB-00007 RCS Heatup and Cooldown Limitations, Note before Section 6.1 and Attachment 2
4. FSAR Section 7.2.2.3.2
5. OTG-ZZ-00006 Plant Cooldown Hot Standby To Cold Shutdown
6. NEI 99-01 CA3

Category: C – Cold Shutdown / Refueling System Malfunction

Subcategory: 4 – Loss of Vital DC Power

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

EAL:

CU4.1	**Unusual Event**
< 107 VDC bus voltage indications on Technical Specification **required** 125 VDC buses for ≥ 15 min. (Note 1)	

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

Mode Applicability:

5 - Cold Shutdown, 6 - Refueling

Definition(s):

None

Basis:

Deleted: Callaway

The purpose of this EAL is to recognize a loss of DC power compromising the ability to monitor and control the removal of decay heat during cold shutdown or refueling operations. 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. The fifteen minute interval is intended to exclude transient or momentary power losses.

The vital DC buses are the following 125 VDC Class 1E buses (ref. 1):

Division 1:
- NK01
- NK03

Division 2:
- NK02
- NK04

There are four, 60 cell, lead-calcium storage batteries (NK11, NK12, NK13 and NK14) that supplement the output of the battery chargers. They supply DC power to the distribution buses when AC power to the chargers is lost or when transient loads exceed the 300 amp capacity of the battery chargers.

Due to the load distribution on each of the 125VDC buses, the four batteries for each bus do not have the same rating. All four of the 125VDC buses supply inverters for 120VAC NN bus power as well as control power for various safety related systems. NK01 and NK04 supply additional DC loads such as diesel field flashing, breaker control power, main control board power and emergency lighting. These loads are not supplied by the other two buses, NK02 and NK03. For this reason, batteries NK11 and NK14 require additional capacity. Each battery is designed to have sufficient stored energy to supply the required emergency loads for 240 minutes following a loss of AC power (station blackout) (ref. 2, 3, 4).

Minimum DC bus voltage is 107.0 VDC (ref. 4, 5). Bus voltage may be obtained from the following instruments (ref. 6):

- NK EI-1 (NK01)
- NK EI-2 (NK02)
- NK EI-3 (NK03)
- NK EI-4 (NK04)

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

This IC addresses a loss of vital DC power which compromises the ability to monitor and control operable SAFETY SYSTEMS when the plant is in the cold shutdown or refueling mode. In these modes, the core decay heat load has been significantly reduced, and coolant system temperatures and pressures are lower; these conditions increase 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.

Deleted: NEI 99-01 Basis¶

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

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

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

Callaway Basis Reference(s):

1. E-21010(Q) DC Single Line Diagram
2. FSAR Tables 8.3-1, -2, -3
3. FSAR Section 8.3.2
4. Calculation NK-10, NK System DC Voltage Drop
5. FSAR Table 8.3A-1 III.B
6. ECA-0.0 Loss of All AC Power
7. NEI 99-01 CU4

Category: C – Cold Shutdown / Refueling System Malfunction

Subcategory: 5 – Loss of Communications

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

EAL:

CU5.1 Unusual Event

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

OR

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

OR

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

Table C-5 Communication Methods

System	Onsite	ORO	NRC
Gaitronics	X		
Plant Radios	X		
Plant Emergency Dedicated Phones	X		
Plant Telephone System	X	X	X
ENS (Red Phone) Line		X	X
Back-Up Radio System		X	
Sentry Notification System		X	

Mode Applicability:

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

Definition(s):

None

Basis:

Deleted: Callaway

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

1. Gai-tronics system

   The Gai-tronics system provides six separate independent communication channels--one general page, one Control Room page and four party lines. Communication between parties within the plant can be easily and quickly established by using the general page channel. Communication between parties in the plant and the Control Room can be easily and quickly established using the Control Room page channel. The party line channel is normally used after the page call is completed. As many as four party lines may communicate simultaneously. The portion of the PA system connecting the fuel transfer area in the Containment, the spent fuel area and new fuel handling area in the fuel building, and the control room can be isolated from the remainder of the PA system from the control room. This permits extended use of the fuel handling communications system without disruption to the remainder of the system.

2. Plant Radios

   A six channel 800 MHZ trunked radio system for overall plant site area coverage reaches out as far as the intake structure. This two-way radio system provides communications for operating purposes with plant radio-equipped vehicles and plant hand-held portable radios. These systems are for use during normal operation or during a plant emergency. This radio system is available on the Control Room radio consoles, on the security radio consoles, on the EOF radio console, and the TSC radio console. This system is also in the field monitoring team vehicles and is used to communicate during emergencies.

3. Plant Emergency Dedicated Phones

   Three independent telephone systems are available for communications between the Emergency Response Facilities: the Technical Assessment Bridge Line, the Dose Assessment Bridge Line and the Emergency Management Bridge Line. Each system operates independently from the other systems and allows for conference calls between the members of that bridge line group

4. Plant telephone system

   The telephone system consists of digital automatic switchboard (DPBX) equipment and telephone stations. The DPBX is provided with redundant processors for reliability. The telephone stations are located throughout the power block, in the main control room, in the various buildings around the site, in the security building, and in the service building where the administrative offices are located. For emergency use, unlisted telephone numbers are provided for direct access to the outside local public telephone system.

5. ENS (Red Phone) line

   The NRC Emergency Notification System (ENS) is an FTS telephone used for official communications with NRC Headquarters. The NRC Headquarters has the capability to patch into the NRC Regional offices. The primary purpose of this phone is to provide a reliable method for the initial notification of the NRC and to maintain continuous communications with the NRC after initial notification. ENS telephones are located in the Control Room, TSC and EOF.

6. Back-Up Radio System (BURS)

   The Back-up Radio System is a communication link between the Callaway Plant and offsite emergency response agencies. The primary use of this system is the back up notification of offsite agencies and the coordination of offsite activities during a radio logic a1 emergency. The system uses 800 MHz radios. There are radio control base: units in the Plant Control Room, TSC and EOF, as well as each county EOC and the State EOC. The backup to this system is the commercial touchtone telephone system Notifications may also be initiated through the Callaway County/City of Fulton EOC via the Security radio.

7. Sentry Notification System

   A computerized notification system linked between the Callaway Plant, the State Emergency Management Agency and the four (4) EPZ risk counties. It allows the Communicator to fill out a notification form on screen and transmit the data simultaneously. Notifications on Sentry can be initiated from the Control Room, the Emergency Operations Facility (EOF), or the Technical Support Center (TSC).

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

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

Deleted: NEI 99-01 Basis:¶

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

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

The second EAL condition addresses a total loss of the communications methods used to notify all OROs of an emergency declaration. The OROs referred to here are the State, Callaway, Gasconade, Montgomery and Osage County EOCs

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

Callaway Basis Reference(s):

1. Callaway Plant Radiological Emergency Response Plan (RERP), Section 7.2
2. FSAR Section 9.5.2
3. NEI 99-01 CU5

Category: C – Cold Shutdown / Refueling System Malfunction

Subcategory: 6 – Hazardous Event Affecting Safety Systems

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

EAL:

CA6.1 Alert

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

AND EITHER:

- Event damage has caused indications of degraded performance in at least one train of a SAFETY SYSTEM needed for the current operating mode
- The event has caused VISIBLE DAMAGE to a SAFETY SYSTEM component or structure needed for the current operating mode

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

Mode Applicability:

5 - Cold Shutdown, 6 - Refueling

Definition(s):

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

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

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

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

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

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

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

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

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

Basis:

Deleted: Callaway

- Annunciator 98D, OBE will illuminate if the seismic instrument detects ground motion in excess of the OBE threshold. OTO-SG-00001, Seismic Event provides the guidance for determining if an OBE earthquake threshold is exceeded and any required response actions (ref. 1).
- Internal FLOODING may be caused by events such as component failures, equipment misalignment, or outage activity mishaps (ref. 2).

Deleted: <#>The significance of seismic events are discussed under EAL HU2.1 (ref. 1).¶

- External flooding may be due to high lake level. Callaway plant grade elevation is 840.0 ft MSL. (ref. 3).
- Seismic Category I structures are analyzed to withstand a sustained, design wind velocity of at least 100 mph. (ref. 4).
- Areas containing functions and systems required for safe shutdown of the plant are identified by fire area (ref. 5).
- An explosion that degrades the performance of a SAFETY SYSTEM train or visibly damages a SAFETY SYSTEM component or structure would be classified under this EAL.

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

Deleted: NEI 99-01 Basis:¶

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

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

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

Callaway Basis Reference(s):

1. OTO-SG-00001 Seismic Event
2. IPE Section 3.4.2.3 Results of the Vulnerability Screening
3. UFSAR Section 3.4 Water Level (Flood) Design Table 3.4-1 PMF, Groundwater, Reference, and Actual Plant Elevations
4. UFSAR Section 3.3.1.1 Design Wind Loadings
5. UFSAR Appendix 9.5B Fire Hazard Analysis and Site Combustion Restrictions
6. NEI 99-01 CA6

Category H – Hazards and Other Conditions Affecting Plant Safety

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

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

1. Security

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

2. Seismic Event

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

3. Natural or Technology Hazard

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

4. Fire

Fires can pose significant hazards to personnel and reactor safety. Appropriate for classification are fires within the site Protected Area or which may affect operability of equipment needed for safe shutdown

5. Control Room Evacuation

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

6. Emergency Coordinator Judgment

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

on operator/management experience and judgment is still necessary. The EALs of this category provide the Emergency Coordinator the latitude to classify emergency conditions consistent with the established classification criteria based upon Emergency Coordinator judgment.

Category: H – Hazards

Subcategory: 1 – Security

Initiating Condition: Confirmed SECURITY CONDITION or threat

EAL:

HU1.1 Unusual Event

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

OR

Notification of a credible security threat directed at the site

OR

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

Mode Applicability:

All

Definition(s):

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

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

Basis:

Deleted: Callaway

The security shift supervision is defined as the Security Shift Supervisor.

This EAL is based on the Callaway Plant Security Plan and DBT (ref. 1).

Deleted: NEI 99-01 Basis:

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

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

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

The first threshold 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.

The second threshold addresses the receipt of a credible security threat. The credibility of the threat is assessed in accordance with the Callaway Plant Security Plan and DBT.

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

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

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

Callaway Basis Reference(s):

1. Callaway Plant Security Plan and DBT (Safeguards)
2. EIP-ZZ-SK001, Response to Security Threat
3. SDP-CP-00003 Security Contingency Events
4. OTO-SK-00002 - Plant Security Event - Aircraft Threat
5. NEI 99-01 HU1

Category: H – Hazards

Subcategory: 1 – Security

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

EAL:

HA1.1 Alert

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

OR

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

Mode Applicability:

All

Definition(s):

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

OWNER CONTROLLED AREA - Area outside the PROTECTED AREA fence that immediately surrounds the plant. Access to this area is generally restricted to those entering on official business.

Basis:

Deleted: Callaway

The security shift supervision is defined as the Security Shift Supervisor.

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.

Deleted: NEI 99-01 Basis:¶

Timely and accurate communications between the Security Shift Supervisor and the Control Room is essential for proper classification of a security-related event (ref. 2, 3, 4).

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

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

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

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

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

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

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

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

Callaway Basis Reference(s):

1. Callaway Plant Security Plan and DBT (Safeguards)
2. EIP-ZZ-SK001, Response to Security Threat
3. SDP-CP-00003 Security Contingency Events
4. OTO-SK-00002 - Plant Security Event - Aircraft Threat
5. NEI 99-01 HA1

Category: H – Hazards

Subcategory: 1 – Security

Initiating Condition: HOSTILE ACTION within the PROTECTED AREA

EAL:

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

Mode Applicability:

All

Definition(s):

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

PROTECTED AREA - An area encompassed by physical barriers and to which access is controlled. The Protected Area refers to the designated security area around the process buildings and is depicted in Drawing 8600-X-88100 Property-Site Layout, Owner Controlled Area and Surrounding Area.

Basis:

Deleted: Callaway

The security shift supervision is defined as the Security Shift Supervisor.

These individuals are the designated on-site personnel qualified and trained to confirm that a security event is occurring or has occurred. Training on security event classification confirmation is closely controlled due to the strict secrecy controls placed on the Callaway Plant Security Plan and DBT (Safeguards) information. (ref. 1)

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.

Deleted: NEI 99-01 Basis:¶

Timely and accurate communications between Security Shift Supervision and the Control Room is essential for proper classification of a security-related event (ref. 2, 3, 4 5).

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 a HOSTILE ACTION directed at an ISFSI PROTECTED AREA located outside the plant PROTECTED AREA; such an attack should be assessed using IC HA1. It also does not apply to incidents that are accidental events, acts of civil disobedience, or otherwise are not a HOSTILE ACTION perpetrated by a HOSTILE FORCE. Examples include the crash of a small aircraft, shots from hunters, physical disputes between employees, etc. Reporting of these types of events is adequately addressed by other EALs, or the requirements of 10 CFR § 73.71 or 10 CFR § 50.72.

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

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

Callaway Basis Reference(s):

1. Callaway Plant Security Plan and DBT (Safeguards)
2. EIP-ZZ-SK001, Response to Security Threat
3. SDP-CP-00003 Security Contingency Events
4. OTO-SK-00001 - Plant Security Event – Hostile Intrusion
5. OTO-SK-00002 - Plant Security Event - Aircraft Threat
6. NEI 99-01 HS1

ATTACHMENT 1
EAL Bases

Category: H – Hazards

Subcategory: 1 – Security

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

EAL:

HG1.1 General Emergency

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

AND EITHER of the following has occurred:

- **Any** of the following safety functions cannot be controlled or maintained
  - Reactivity control
  - Core cooling
  - RCS heat removal

  OR
- Damage to spent fuel has occurred or is IMMINENT

Mode Applicability:

All

Definition(s):

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

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

PROTECTED AREA - An area encompassed by physical barriers and to which access is controlled. The Protected Area refers to the designated security area around the process buildings and is depicted in Drawing 8600-X-88100 Property-Site Layout, Owner Controlled Area and Surrounding Area.

Basis:

Deleted: Callaway

The security shift supervision is defined as the Security Shift Supervisor.

This IC addresses an event in which a HOSTILE FORCE has taken physical control of the facility to the extent that the plant staff can no longer operate equipment necessary to maintain key safety functions. It also addresses a HOSTILE ACTION leading to a loss of physical control that results in actual or IMMINENT damage to spent fuel due to 1) damage to a spent

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fuel pool cooling system (e.g., pumps, heat exchangers, controls, etc.) or, 2) loss of spent fuel pool integrity such that sufficient water level cannot be maintained.

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

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

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

Callaway Basis Reference(s):

1. Callaway Plant Security Plan and DBT (Safeguards)
2. EIP-ZZ-SK001, Response to Security Threat
3. SDP-CP-00003 Security Contingency Events
4. OTO-SK-00001 - Plant Security Event – Hostile Intrusion
5. OTO-SK-00002 - Plant Security Event - Aircraft Threat
6. NEI 99-01 HG1

Category: H – Hazards and Other Conditions Affecting Plant Safety

Subcategory: 2 – Seismic Event

Initiating Condition: Seismic event greater than OBE level

EAL:

HU2.1	**Unusual Event**
Seismic event > OBE as indicated by Seismic Activity, Annunciator 98D	

Deleted: per OTO-SG-00001,

Mode Applicability:

All

Definition(s):

None

Basis:

Deleted: Callaway

Ground motion acceleration of 0.12g horizontal or vertical is the Operating Basis Earthquake for Callaway (ref. 1).

Annunciator 98D, OBE will illuminate if the seismic instrument detects ground motion in excess of the OBE threshold (ref. 2).

When the seismic recorder indicates that the OBE has been exceeded, the reactor must be shut down and remain shut down until inspection of the facility shows that no damage has been incurred which would jeopardize safe operation of the facility or until such damage is repaired. Callaway was designed such that, for ground motion less than the OBE, those features of the plant necessary for continued operation without undue risk to the health and safety of the public will remain functional. Any ground motion in excess of this results in an uncertainty as to the extent of the damage which must be resolved before continued operation can be considered safe (ref. 1). Ground motion of this magnitude is unmistakably a "felt" earthquake.

OTO-SG-00001, Seismic Event provides the guidance for determining if the OBE earthquake threshold is exceeded and any required response actions. (ref. 2)

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. The NEIC can be contacted by calling **(303) 273-8500**. Select **option #1** and inform the analyst you wish to confirm recent seismic activity in the vicinity of Callaway. Alternatively, near real-time seismic activity can be accessed via the NEIC website:

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

Deleted: NEI 99-01 Basis:¶

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

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

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

Callaway Basis Reference(s):

1. FSAR Section 3.8.6.4.1 Design and Analysis Procedures
2. OTO-SG-00001, Seismic Event
3. NEI 99-01 HU2

Category: H – Hazards and Other Conditions Affecting Plant Safety

Subcategory: 3 – Natural or Technology Hazard

Initiating Condition: Hazardous event

EAL:

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

Mode Applicability:

All

Definition(s):

PROTECTED AREA - An area encompassed by physical barriers and to which access is controlled. The Protected Area refers to the designated security area around the process buildings and is depicted in Drawing 8600-X-88100 Property-Site Layout, Owner Controlled Area and Surrounding Area.

Basis:

Deleted: Callaway

Response actions associated with a tornado onsite is provided in OTO-ZZ-00012 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. A tornado is defined as a violently rotating column of air in contact with the ground and extending from the base of a thunderstorm.

Deleted: NEI 99-01 Basis:¶

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

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

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

Callaway Basis Reference(s):

1. OTO-ZZ-00012 Severe Weather
2. NEI 99-01 HU3

Category: H – Hazards and Other Conditions Affecting Plant Safety

Subcategory: 3 – Natural or Technology Hazard

Initiating Condition: Hazardous event

EAL:

HU3.2	**Unusual Event**
Internal room or area FLOODING of a magnitude sufficient to require manual or automatic electrical isolation of a SAFETY SYSTEM component needed for the current operating mode	

Mode Applicability:

All

Definition(s):

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

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

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

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

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

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

Basis:

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

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

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

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

Deleted: Callaway

Deleted: The internal flooding areas of concern are the Control Building, Battery Room and ESF Switchgear Room (ref.1).

Deleted: NEI 99-01 Basis:¶

Callaway Basis Reference(s):

1. IPE Section 3.4.2.3 Results of the Vulnerability Screening
2. NEI 99-01 HU3

Category: H – Hazards and Other Conditions Affecting Plant Safety

Subcategory: 3 – Natural or Technology Hazard

Initiating Condition: Hazardous event

EAL:

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

Mode Applicability:

All

Definition(s):

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

PROTECTED AREA - An area encompassed by physical barriers and to which access is controlled. The Protected Area refers to the designated security area around the process buildings and is depicted in Drawing 8600-X-88100 Property-Site Layout, Owner Controlled Area and Surrounding Area.

Basis:

Deleted: Callaway

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

Deleted: NEI 99-01 Basis:¶

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

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

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

Callaway Basis Reference(s):

1. NEI 99-01 HU3

Category: H – Hazards and Other Conditions Affecting Plant Safety

Subcategory: 3 – Natural or Technology Hazard

Initiating Condition: Hazardous event

EAL:

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

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

Mode Applicability:

All

Definition(s):

None

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.

Callaway Basis Reference(s):

1. NEI 99-01 HU3

Deleted: Callaway

Deleted: None
NEI 99-01 Basis:

Category: H – Hazards and Other Conditions Affecting Plant Safety

Subcategory: 4 – Fire

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

EAL:

HU4.1 Unusual Event

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

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

AND

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

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

Table H-1 Fire Areas
• Area 5 • Containment • Aux Feed Pump Rooms • Auxiliary Building • Diesel Generator Building • UHS Cooling Tower • ESW Pumphouse • Control Building/Communications Corridor • RWST • Fuel Building

Mode Applicability:

All

Definition(s):

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

Basis:

Deleted: Callaway

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

time limit or a classification must be made. If a fire is verified to be occurring by field report, the 15 minute time limit is from the original receipt of the fire detection alarm.

Table H-1 Fire Areas are based on FSAR Section 5.4A.2 System Required to Go From Hot Standby to Cold Shutdown. Table H-1 Fire Areas include those structures containing functions and systems required for safe shutdown of the plant (SAFETY SYSTEMS) (ref. 1).

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

Deleted: NEI 99-01 Basis:¶

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

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

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

Callaway Basis Reference(s):

1. FSAR Section 5.4A.2 System Required to Go From Hot Standby to Cold Shutdown
2. NEI 99-01 HU4

Category: H – Hazards and Other Conditions Affecting Plant Safety

Subcategory: 4 – Fire

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

EAL:

HU4.2 Unusual Event

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

AND

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

AND

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

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

Table H-1 Fire Areas
• Area 5 • Containment • Aux Feed Pump Rooms • Auxiliary Building • Diesel Generator Building • UHS Cooling Tower • ESW Pumphouse • Control Building/Communications Corridor • RWST • Fuel Building

Mode Applicability:

All

Definition(s):

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

Basis:

Deleted: Callaway

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

Table H-1 Fire Areas are based on FSAR Section 5.4A.2 System Required to Go From Hot Standby to Cold Shutdown. Table H-1 Fire Areas include those structures containing functions and systems required for safe shutdown of the plant (SAFETY SYSTEMS) (ref. 1).

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

Deleted: NEI 99-01 Basis:¶

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.

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

Callaway Basis Reference(s):

1. FSAR Section 5.4A.2 System Required to Go From Hot Standby to Cold Shutdown
2. NEI 99-01 HU4

Category: H – Hazards and Other Conditions Affecting Plant Safety

Subcategory: 4 – Fire

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

EAL:

HU4.3	**Unusual Event**
A FIRE within the plant PROTECTED AREA **not** extinguished within 60 min. of the initial report, alarm or indication (Note 1)	

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

Mode Applicability:

All

Definition(s):

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

PROTECTED AREA - An area encompassed by physical barriers and to which access is controlled. The Protected Area refers to the designated security area around the process buildings and is depicted in Drawing 8600-X-88100 Property-Site Layout, Owner Controlled Area and Surrounding Area.

Basis:

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

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

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

Callaway Basis Reference(s):

1. NEI 99-01 HU4

Deleted: Callaway

Deleted: None
NEI 99-01 Basis:

Category: H – Hazards and Other Conditions Affecting Plant Safety

Subcategory: 4 – Fire

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

EAL:

HU4.4	**Unusual Event**
A FIRE within the plant PROTECTED AREA that requires firefighting support by an offsite fire response agency to extinguish	

Mode Applicability:

All

Definition(s):

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

PROTECTED AREA - An area encompassed by physical barriers and to which access is controlled. The Protected Area refers to the designated security area around the process buildings and is depicted in Drawing 8600-X-88100 Property-Site Layout, Owner Controlled Area and Surrounding Area.

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.

Callaway Basis Reference(s):

1. NEI 99-01 HU4

Deleted: Callaway

Deleted: None¶
NEI 99-01 Basis:¶

Category: H – Hazards and Other Conditions Affecting Plant Safety

Subcategory: 5 – Control Room Evacuation

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

EAL:

HA5.1	**Alert**
An event has resulted in plant control being transferred from the Control Room to the Auxiliary Shutdown Panel (ASP)	

Mode Applicability:

All

Definition(s):

None

Basis:

Deleted: Callaway

The Shift Manager (SM) 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. OTO-ZZ-00001 Control Room Inaccessibility, provides the instructions for tripping the unit, and maintaining RCS inventory and Hot Shutdown conditions from outside the Control Room (Ref. 1).

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

Deleted: NEI 99-01 Basis:¶

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

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

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

Callaway Basis Reference(s):

1. OTO-ZZ-00001 Control Room Inaccessibility
2. NEI 99-01 HA6

Category: H – Hazards and Other Conditions Affecting Plant Safety

Subcategory: 5 – Control Room Evacuation

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

EAL:

HS5.1 Site Area Emergency

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

AND

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

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

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

Mode Applicability:

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

Definition(s):

None

Basis:

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

The Shift Manager (SM) 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. OTO-ZZ-00001 Control Room Inaccessibility, provides the instructions for tripping the unit, and maintaining RCS inventory and Hot Shutdown conditions from outside the Control Room (Ref. 1, 2).

The intent of this EAL is to capture events in which control of the plant cannot be reestablished in a timely manner. The fifteen minute time for transfer starts when the Control Room begins to be evacuated (not when OTO-ZZ-00001 is entered). The time interval is based on how quickly control must be reestablished without core uncovery and/or core damage. The determination of whether or not control is established from outside the Control Room is based on Emergency Coordinator judgment. The Emergency Coordinator is expected to make a reasonable, informed judgment that control of the plant from outside the Control Room cannot be established within the fifteen minute interval.

Once the Control Room is evacuated, the objective is to establish control of important plant equipment and maintain knowledge of important plant parameters in a timely manner. Primary

Deleted: All¶

Deleted: Callaway

emphasis should be placed on components and instruments that supply protection for and information about safety functions. Typically, these safety functions are reactivity control (ability to shutdown the reactor and maintain it shutdown), RCS inventory (ability to cool the core), and secondary heat removal (ability to maintain a heat sink).

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.

Deleted: NEI 99-01 Basis:¶

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

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

Callaway Basis Reference(s):

1. OTO-ZZ-00001 Control Room Inaccessibility
2. OTS-ZZ-00001 Cooldown from Outside the Control Room
3. NEI 99-01 HS6

Category: H – Hazards and Other Conditions Affecting Plant Safety

Subcategory: 7 – Emergency Coordinator Judgment

Initiating Condition: Other conditions existing that in the judgment of the Emergency Coordinator warrant declaration of a UE

EAL:

HU6.1	**Unusual Event**
Other conditions exist which in the judgment of the Emergency Coordinator indicate that events are in progress or have occurred which indicate a potential degradation of the level of safety of the plant or indicate a security threat to facility protection has been initiated. No releases of radioactive material requiring offsite response or monitoring are expected unless further degradation of SAFETY SYSTEMS occurs.	

Mode Applicability:

All

Definition(s):

None

Basis:

Deleted: Callaway

The Emergency Coordinator is the designated onsite individual having the responsibility and authority for implementing the Callaway Radiological Emergency Response Plan (ref. 1). The Operations Shift Manager (SM) initially acts in the capacity of the Emergency Coordinator and takes actions as outlined in the Emergency Plan implementing procedures (ref. 2). If required by the emergency classification or if deemed appropriate by the Emergency Coordinator, emergency response personnel are notified and instructed to report to their emergency response locations. In this manner, the individual usually in charge of activities in the Control Room is responsible for initiating the necessary emergency response, but Plant Management is expected to manage the emergency response as soon as available to do so in anticipation of the possible wide-ranging responsibilities associated with managing a major emergency.

Deleted: NEI 99-01 Basis:¶

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

Callaway Basis Reference(s):

1. Callaway Radiological Emergency Response Plan section 5.2.1 Emergency Coordinator
2. Callaway Radiological Emergency Response Plan section 5.1.1 Shift Manager
3. NEI 99-01 HU7

Category: H – Hazards and Other Conditions Affecting Plant Safety

Subcategory: 7 – Emergency Coordinator Judgment

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

EAL:

HA6.1	**Alert**
Other conditions exist which, in the judgment of the Emergency Coordinator, indicate that events are in progress or have occurred which involve an actual or potential substantial degradation of the level of safety of the plant or a security event that involves probable life threatening risk to site personnel or damage to site equipment because of HOSTILE ACTION. Any releases are expected to be limited to small fractions of the EPA Protective Action Guideline exposure levels.	

Mode Applicability:

All

Definition(s):

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

Basis:

Deleted: Callaway

The Emergency Coordinator is the designated onsite individual having the responsibility and authority for implementing the Callaway Radiological Emergency Response Plan (ref. 1). The Operations Shift Manager (SM) initially acts in the capacity of the Emergency Coordinator and takes actions as outlined in the Emergency Plan implementing procedures (ref. 2). If required by the emergency classification or if deemed appropriate by the Emergency Coordinator, emergency response personnel are notified and instructed to report to their emergency response locations. In this manner, the individual usually in charge of activities in the Control Room is responsible for initiating the necessary emergency response, but Plant Management is expected to manage the emergency response as soon as available to do so in anticipation of the possible wide-ranging responsibilities associated with managing a major emergency.

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

Deleted: NEI 99-01 Basis:¶

Callaway Basis Reference(s):

1. Callaway Radiological Emergency Response Plan section 5.2.1 Emergency Coordinator
2. Callaway Radiological Emergency Response Plan section 5.1.1 Shift Manager

3. NEI 99-01 HA7

Category: H – Hazards and Other Conditions Affecting Plant Safety

Subcategory: 7 – Emergency Coordinator Judgment

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

EAL:

HS6.1 Site Area Emergency

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

Mode Applicability:

All

Definition(s):

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

Basis:

Deleted: Callaway

The Emergency Coordinator is the designated onsite individual having the responsibility and authority for implementing the Callaway Radiological Emergency Response Plan (ref. 1). The Operations Shift Manager (SM) initially acts in the capacity of the Emergency Coordinator and takes actions as outlined in the Emergency Plan implementing procedures (ref. 2). If required by the emergency classification or if deemed appropriate by the Emergency Coordinator, emergency response personnel are notified and instructed to report to their emergency response locations. In this manner, the individual usually in charge of activities in the Control Room is responsible for initiating the necessary emergency response, but Plant Management is expected to manage the emergency response as soon as available to do so in anticipation of the possible wide-ranging responsibilities associated with managing a major emergency.

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

Deleted: NEI 99-01 Basis:¶

Callaway Basis Reference(s):

ATTACHMENT 1
EAL Bases

1. Callaway Radiological Emergency Response Plan section 5.2.1 Emergency Coordinator
2. Callaway Radiological Emergency Response Plan section 5.1.1 Shift Manager
3. NEI 99-01 HS7

Category: H – Hazards and Other Conditions Affecting Plant Safety

Subcategory: 7 – Emergency Coordinator Judgment

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

EAL:

HG6.1	**General Emergency**
Other conditions exist which in the judgment of the Emergency Coordinator indicate that events are in progress or have occurred which involve actual or IMMINENT substantial core degradation or melting with potential for loss of containment integrity or HOSTILE ACTION that results in an actual loss of physical control of the facility. Releases can be reasonably expected to exceed EPA Protective Action Guideline exposure levels offsite for more than the immediate site area	

Mode Applicability:

All

Definition(s):

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

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

Basis:

Deleted: Callaway

The Emergency Coordinator is the designated onsite individual having the responsibility and authority for implementing the Callaway Radiological Emergency Response Plan (ref. 1). The Operations Shift Manager (SM) initially acts in the capacity of the Emergency Coordinator and takes actions as outlined in the Emergency Plan implementing procedures (ref. 2). If required by the emergency classification or if deemed appropriate by the Emergency Coordinator, emergency response personnel are notified and instructed to report to their emergency response locations. In this manner, the individual usually in charge of activities in the Control Room is responsible for initiating the necessary emergency response, but Plant Management is expected to manage the emergency response as soon as available to do so in anticipation of the possible wide-ranging responsibilities associated with managing a major emergency.

Releases can reasonably be expected to exceed EPA PAG plume exposure levels outside the Site Boundary.

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

Deleted: NEI 99-01 Basis:¶

Callaway Basis Reference(s):

1. Callaway Radiological Emergency Response Plan section 5.2.1 Emergency Coordinator
2. Callaway Radiological Emergency Response Plan section 5.1.1 Shift Manager
3. NEI 99-01 HG7

Category S – System Malfunction

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

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

The events of this category pertain to the following subcategories:

1. Loss of Emergency AC Power

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

2. Loss of Vital DC Power

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

3. Loss of Control Room Indications

Certain events that degrade plant operator ability to effectively assess plant conditions within the plant warrant emergency classification. Losses of indicators are in this subcategory.

4. RCS Activity

During normal operation, reactor coolant fission product activity is very low. Small concentrations of fission products in the coolant are primarily from the fission of tramp uranium in the fuel clad or minor perforations in the clad itself. Any significant increase from these base-line levels (2% - 5% clad failures) is indicative of fuel failures and is covered under the Fission Product Barrier Degradation category. However, lesser amounts of clad damage may result in coolant activity exceeding Technical Specification limits. These fission products will be circulated with the reactor coolant and can be detected by coolant sampling.

5. RCS Leakage

The reactor vessel provides a volume for the coolant that covers the reactor core. The reactor pressure vessel and associated pressure piping (reactor coolant system) together provide a barrier to limit the release of radioactive material should the reactor fuel clad integrity fail. Excessive RCS leakage greater than Technical Specification limits indicates potential pipe cracks that may propagate to an extent threatening fuel clad, RCS and containment integrity.

6. RTS Failure

This subcategory includes events related to failure of the Reactor Trip System (RTS) to initiate and complete reactor trips. In the plant licensing basis, postulated failures of the RTS to complete a reactor trip comprise a specific set of analyzed events referred to as

Anticipated Transient Without Scram (ATWS) events. For EAL classification, however, ATWS is intended to mean any trip failure event that does not achieve reactor shutdown. If RTS actuation fails to assure reactor shutdown, positive control of reactivity is at risk and could cause a threat to fuel clad, RCS and containment integrity.

7. Loss of Communications

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

8. Containment Failure

Failure of containment isolation capability (under conditions in which the containment is not currently challenged) warrants emergency classification. Failure of containment pressure control capability also warrants emergency classification.

9. Hazardous Event Affecting Safety Systems

Various natural and technological events that result in degraded plant safety system performance or significant visible damage warrant emergency classification under this subcategory.

ATTACHMENT 1
EAL Bases

Category: S – System Malfunction

Subcategory: 1 – Loss of Emergency AC Power

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

EAL:

SU1.1	**Unusual Event**
Loss of **all** offsite AC power capability, Table S-1, to emergency 4.16KV buses NB01 and NB02 for ≥ 15 min. (Note 1)	

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

Table S-1 AC Power Sources
Offsite: • Safeguards XFMR A or B via ESF LTC XFMR XNB01 • Startup XFMR XMR01 via ESF LTC XFMR XNB02 • Main XFMR XMA01 backfed via UAT XFMR XMA02 (only if already aligned)
Onsite: • EDG NE01 • EDG NE02

Mode Applicability:

1 - Power Operation, 2 - Startup, 3 - Hot Standby, 4 – Hot Shutdown

Definition(s):

None

Basis:

Basis:

Deleted: Callaway

The 4.16KV AC System provides the power requirements for operation and safe shutdown of the plant. The essential switchgear are buses NB01 and NB02 (ref. 1).

NB01 supplies power to Load Group 1 (Red Train) safety related loads and NB02 supplies power to Load Group 2 (Yellow Train) safety related loads. Each bus has two sources of offsite power. One source is from 13.8 KV safeguards transformer A or B via ESF Load Tap Changing (LTC) transformer XNB01 and the other source is from the startup transformer XMR01 via ESF LTC transformer XNB02. Transformer XNB01 is the normal supply to bus NB01; XNB02 is the normal supply to bus NB02 (ref. 1, 2 3).

In addition, NB01 and NB02 each have an emergency diesel generator (onsite power supply) which supply electrical power to the bus automatically in the event that the preferred source becomes unavailable (ref. 1).

Another method to obtain offsite power is by backfeeding the emergency buses through the main transformer XMA01 and unit auxiliary transformer XMR02. However, this is only done during cold shutdown unless nuclear safety considerations require it to be done during hot shutdown when no other power sources are available (ref. 4).

Additional sources of offsite power are available from diesel generators such as the Alternate Emergency Power Supply (AEPS) or portable generation sources. Credit can be taken for these sources if they are capable of carrying an NB bus and are aligned within 15 minutes.

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.

Deleted: The 15-minute interval was selected as a threshold to exclude transient or momentary power losses.¶
NEI 99-01 Basis:¶

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

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

Escalation of the emergency classification level would be via IC SA1.

Callaway Basis Reference(s):

1. E-21001(Q) Main Single Line Diagram (Electrical Distribution Diagram)
2. FSAR Site Addenda Section 8.2.1.2
3. FSAR Section 8.3.1
4. OTS-MA-00001-R011 Main Step-Up Transformer Backfeed - IPTE
5. NEI 99-01 SU1

Category: S – System Malfunction

Subcategory: 1 – Loss of Emergency AC Power

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

EAL:

SA1.1 Alert

AC power capability, Table S-1, to emergency 4.16KV buses NB01 and NB02 reduced to a single power source for ≥ 15 min. (Note 1)

AND

Any additional single power source failure will result in loss of **all** AC power to SAFETY SYSTEMS

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

Table S-1 AC Power Sources
Offsite:
• Safeguards XFMR A or B via ESF LTC XFMR XNB01
• Startup XFMR XMR01 via ESF LTC XFMR XNB02
• Main XFMR XMA01 backfed via UAT XFMR XMA02 (only if already aligned)
Onsite:
• EDG NE01
• EDG NE02

Mode Applicability:

1 - Power Operation, 2 - Startup, 3 - Hot Standby, 4 - Hot Shutdown

Definition(s):

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

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

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

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

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

Basis:

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

The 4.16KV AC System provides the power requirements for operation and safe shutdown of the plant. The essential switchgear are buses NB01 and NB02 (ref. 1).

NB01 supplies power to Load Group 1 (Red Train) safety related loads and NB02 supplies power to Load Group 2 (Yellow Train) safety related loads. Each bus has two sources of offsite power. One source is from 13.8 KV safeguards transformer A or B via ESF Load Tap Changing (LTC) transformer XNB01 and the other source is from the startup transformer XMR01 via ESF LTC transformer XNB02. Transformer XNB01 is the normal supply to bus NB01; XNB02 is the normal supply to bus NB02 (ref. 1, 2 3).

In addition, NB01 and NB02 each have an emergency diesel generator (onsite power supply) which supply electrical power to the bus automatically in the event that the preferred source becomes unavailable (ref. 1).

Another method to obtain offsite power is by backfeeding the emergency buses through the main transformer XMA01 and unit auxiliary transformer XMR02. However, this is only done during cold shutdown unless nuclear safety considerations require it to be done during hot shutdown when no other power sources are available (ref. 4).

Additional sources of offsite power are available from diesel generators such as the Alternate Emergency Power Supply (AEPS) or portable generation sources. Credit can be taken for these sources if they are capable of carrying an NB bus and are aligned within 15 minutes.

If the capability of a second source of emergency bus power is not restored within 15 minutes, an Alert is declared under this EAL.

This IC describes a significant degradation of offsite and onsite AC power sources such that any additional single failure would result in a loss of all AC power to SAFETY SYSTEMS. In this condition, the sole AC power source may be powering one, or more than one, train of safety-related equipment. This IC provides an escalation path from IC SU1.

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

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

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

Escalation of the emergency classification level would be via IC SS1.

Callaway Basis Reference(s):

1. E-21001(Q) Main Single Line Diagram (Electrical Distribution Diagram)

Deleted: Callaway Basis:¶

Deleted: The 15-minute interval was selected as a threshold to exclude transient or momentary power losses.

Deleted: NEI 99-01 Basis:¶

2. FSAR Site Addenda Section 8.2.1.2
3. FSAR Section 8.3.1
4. OTS-MA-00001-R011 Main Step-Up Transformer Backfeed - IPTE
5. NEI 99-01 SA1

Category: S – System Malfunction

Subcategory: 1 – Loss of Emergency AC Power

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

EAL:

SS1.1	**Site Area Emergency**
Loss of **all** offsite and **all** onsite AC power capability, Table S-1, to emergency 4.16KV buses NB01 and NB02 for ≥ 15 min. (Note 1)	

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

Table S-1 AC Power Sources
Offsite: • Safeguards XFMR A or B via ESF LTC XFMR XNB01 • Startup XFMR XMR01 via ESF LTC XFMR XNB02 • Main XFMR XMA01 backfed via UAT XFMR XMA02 (only if already aligned) **Onsite:** • EDG NE01 • EDG NE02

Mode Applicability:

1 - Power Operation, 2 - Startup, 3 - Hot Standby, 4 - Hot Shutdown

Definition(s):

None

Basis:

Deleted: Callaway

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

The 4.16KV AC System provides the power requirements for operation and safe shutdown of the plant. The essential switchgear are buses NB01 and NB02 (ref. 1).

NB01 supplies power to Load Group 1 (Red Train) safety related loads and NB02 supplies power to Load Group 2 (Yellow Train) safety related loads. Each bus has two sources of offsite power. One source is from 13.8 KV safeguards transformer A or B via ESF Load Tap Changing (LTC) transformer XNB01 and the other source is from the startup transformer XMR01 via ESF LTC transformer XNB02. Transformer XNB01 is the normal supply to bus NB01; XNB02 is the normal supply to bus NB02 (ref. 1, 2 3).

In addition, NB01 and NB02 each have an emergency diesel generator (onsite power supply) which supply electrical power to the bus automatically in the event that the preferred source becomes unavailable (ref. 1).

Another method to obtain offsite power is by backfeeding the emergency buses through the main transformer XMA01 and unit auxiliary transformer XMR02. However, this is only done during cold shutdown unless nuclear safety considerations require it to be done during hot shutdown when no other power sources are available (ref. 4).

Additional sources of offsite power are available from diesel generators such as the Alternate Emergency Power Supply (AEPS) or portable generation sources. Credit can be taken for these sources if they are capable of carrying an NB bus and are aligned within 15 minutes (ref. 5).

The interval begins when both offsite and onsite AC power capability are lost.

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.

Callaway Basis Reference(s):

1. E-21001(Q) Main Single Line Diagram (Electrical Distribution Diagram)
2. FSAR Site Addenda Section 8.2.1.2
3. FSAR Section 8.3.1
4. OTS-MA-00001-R011 Main Step-Up Transformer Backfeed – IPTE
5. ECA-0.0 Loss of All AC Power
6. NEI 99-01 SS1

Deleted: The 15-minute interval was selected as a threshold to exclude transient or momentary power losses.

Deleted: NEI 99-01 Basis:¶

ATTACHMENT 1
EAL Bases

Category: S –System Malfunction

Subcategory: 1 – Loss of Emergency AC Power

Initiating Condition: Prolonged loss of **all** offsite and **all** onsite AC power to emergency buses

EAL:

SG1.1	**General Emergency**

Loss of **all** offsite and **all** onsite AC power capability, Table S-1, to emergency 4.16KV buses NB01 and NB02

AND EITHER:

- Restoration of at least one emergency bus in < 4 hours is **not** likely (Note 1)
- CSFST Core Cooling RED Path conditions met

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

Table S-1 AC Power Sources
Offsite:
• Safeguards XFMR A or B via ESF LTC XFMR XNB01
• Startup XFMR XMR01 via ESF LTC XFMR XNB02
• Main XFMR XMA01 backfed via UAT XFMR XMA02 (only if already aligned)
Onsite:
• EDG NE01
• EDG NE02

Mode Applicability:

1 - Power Operation, 2 - Startup, 3 - Hot Standby, 4 - Hot Shutdown

Definition(s):

None

Basis:

Deleted: Callaway

This EAL is indicated by the extended loss of all offsite and onsite AC power capability to 4.16KV emergency buses NB01 and NB02 either for greater then the Callaway Station Blackout (SBO) coping analysis time (4 hrs.) (ref. 1, 2) 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. 3).

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

The 4.16KV AC System provides the power requirements for operation and safe shutdown of the plant. The essential switchgear are buses NB01 and NB02 (ref. 1).

NB01 supplies power to Load Group 1 (Red Train) safety related loads and NB02 supplies power to Load Group 2 (Yellow Train) safety related loads. Each bus has two sources of offsite power. One source is from 13.8 KV safeguards transformer A or B via ESF Load Tap Changing (LTC) transformer XNB01 and the other source is from the startup transformer XMR01 via ESF LTC transformer XNB02. Transformer XNB01 is the normal supply to bus NB01; XNB02 is the normal supply to bus NB02 (ref. 4, 5 6).

In addition, NB01 and NB02 each have an emergency diesel generator (onsite power supply) which supply electrical power to the bus automatically in the event that the preferred source becomes unavailable (ref. 4).

Another method to obtain offsite power is by backfeeding the emergency buses through the main transformer XMA01 and unit auxiliary transformer XMR02. However, this is only done during cold shutdown unless nuclear safety considerations require it to be done during hot shutdown when no other power sources are available (ref. 7).

Additional sources of offsite power are available from diesel generators such as the Alternate Emergency Power Supply (AEPS) or portable generation sources. Credit can be taken for these sources if they are capable of carrying an NB bus and are aligned within 15 minutes (ref. 8).

Four hours is the station blackout coping time (ref 1, 2).

Indication of continuing core cooling degradation must be based on fission product barrier monitoring with particular emphasis on Emergency Coordinator judgment as it relates to imminent Loss of fission product barriers and degraded ability to monitor fission product barriers. Indication of continuing core cooling degradation is manifested by CSFST Core Cooling RED PATH conditions being met (ref. 3). Specifically, Core Cooling RED PATH conditions exist if either core exit T/Cs are reading greater than or equal to 1200ºF or core exit T/Cs are reading greater than or equal to 706ºF with RCS subcooling less than or equal to 30ºF [50ºF], and RVLIS indication is less than or equal to that specified based on the number of RCPs running (ref. 3).

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.

Deleted: NEI-9901 Basis:¶

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.

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.

Callaway Basis Reference(s):

1. FSAR Section 8.3A.5
2. BO-01 Station Blackout (SBO) Coping Duration, sh 1
3. CSF-1 Critical Safety Function Status Trees (CSFST) Figure 2, Core Cooling
4. E-21001(Q) Main Single Line Diagram (Electrical Distribution Diagram)
5. FSAR Site Addenda Section 8.2.1.2
6 FSAR Section 8.3.1
7. OTS-MA-00001-R011 Main Step-Up Transformer Backfeed – IPTE
8. ECA-0.0 Loss of All AC Power
9. NEI 99-01 SG1

Category: S –System Malfunction

Subcategory: 1 – Loss of Emergency AC Power

Initiating Condition: Loss of **all** AC and vital DC power sources for 15 minutes or longer

EAL:

SG1.2	**General Emergency**

Loss of **all** offsite and **all** onsite AC power capability, Table S-1, to emergency 4.16KV buses NB01 and NB02 for ≥ 15 min.

AND

Loss of **all** 125 VDC power based on battery bus voltage indications < 107 VDC on **all** vital DC buses NK01, NK03 (Division 1) and NK02, NK04 (Division 2) for ≥ 15 min.

(Note 1)

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

Table S-1 AC Power Sources
Offsite: • Safeguards XFMR A or B via ESF LTC XFMR XNB01 • Startup XFMR XMR01 via ESF LTC XFMR XNB02 • Main XFMR XMA01 backfed via UAT XFMR XMA02 (only if already aligned) **Onsite:** • EDG NE01 • EDG NE02

Mode Applicability:

1 - Power Operation, 2 - Startup, 3 - Hot Standby, 4 - Hot Shutdown

Definition(s):

None

Basis:

Deleted: Callaway

This EAL is indicated by the loss of all offsite and onsite emergency AC power capability to 4.16KV emergency buses NB01 and NB02 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.

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

The 4.16KV AC System provides the power requirements for operation and safe shutdown of the plant. The essential switchgear are buses NB01 and NB02 (ref. 1).

NB01 supplies power to Load Group 1 (Red Train) safety related loads and NB02 supplies power to Load Group 2 (Yellow Train) safety related loads. Each bus has two sources of offsite power. One source is from 13.8 KV safeguards transformer A or B via ESF Load Tap Changing (LTC) transformer XNB01 and the other source is from the startup transformer XMR01 via ESF LTC transformer XNB02. Transformer XNB01 is the normal supply to bus NB01; XNB02 is the normal supply to bus NB02 (ref. 1, 2 3).

In addition, NB01 and NB02 each have an emergency diesel generator (onsite power supply) which supply electrical power to the bus automatically in the event that the preferred source becomes unavailable (ref. 1).

Another method to obtain offsite power is by backfeeding the emergency buses through the main transformer XMA01 and unit auxiliary transformer XMR02. However, this is only done during cold shutdown unless nuclear safety considerations require it to be done during hot shutdown when no other power sources are available (ref. 4).

Additional sources of offsite power are available from diesel generators such as the Alternate Emergency Power Supply (AEPS) or portable generation sources. Credit can be taken for these sources if they are capable of carrying an NB bus and are aligned within 15 minutes (ref. 5).

The vital DC buses are the following 125 VDC Class 1E buses (ref. 6):

Division 1:	Division 2:
• NK01	• NK02
• NK03	• NK04

There are four, 60 cell, lead-calcium storage batteries (NK11, NK12, NK13 and NK14) that supplement the output of the battery chargers. They supply DC power to the distribution buses when AC power to the chargers is lost or when transient loads exceed the 300 amp capacity of the battery chargers.

Due to the load distribution on each of the 125VDC buses, the four batteries for each bus do not have the same rating. All four of the 125VDC buses supply inverters for 120VAC NN bus power as well as control power for various safety related systems. NK01 and NK04 supply additional DC loads such as diesel field flashing, breaker control power, main control board power and emergency lighting. These loads are not supplied by the other two buses, NK02 and NK03. For this reason, batteries NK11 and NK14 require additional capacity. Each battery is designed to have sufficient stored energy to supply the required emergency loads for 240 minutes following a loss of AC power (station blackout) (ref. 8, 9, 10).

Minimum DC bus voltage is 107.0 VDC (ref. 9, 10). Bus voltage may be obtained from the following instruments (ref. 6):

- NK EI-1 (NK01)
- NK EI-2 (NK02)
- NK EI-3 (NK03)
- NK EI-4 (NK04)

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,

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containment heat removal/pressure control, spent fuel heat removal and the ultimate heat sink. A loss of vital DC power compromises the ability to monitor and control SAFETY SYSTEMS. A sustained loss of both emergency AC and vital DC power will lead to multiple challenges to fission product barriers.

Fifteen minutes was selected as a threshold to exclude transient or momentary power losses. The 15-minute emergency declaration clock begins at the point when both EAL thresholds are met.

Callaway Basis Reference(s):

1. E-21001(Q) Main Single Line Diagram (Electrical Distribution Diagram)
2. FSAR Site Addenda Section 8.2.1.2
3. FSAR Section 8.3.1
4. OTS-MA-00001-R011 Main Step-Up Transformer Backfeed – IPTE
5. ECA-0.0 Loss of All AC Power
6. E-21010(Q) DC Single Line Diagram
7. FSAR Tables 8.3-1, -2, -3
8. FSAR Section 8.3.2
9. Calculation NK-10, NK System DC Voltage Drop
10. FSAR Table 8.3A-1 III.B
11. NEI 99-01 SG8

Category: S – System Malfunction

Subcategory: 2 – Loss of Vital DC Power

Initiating Condition: Loss of **all** vital DC power for 15 minutes or longer

EAL:

SS2.1	**Site Area Emergency**
Loss of **all** 125 VDC power based on battery bus voltage indications < 107 VDC on **all** vital DC buses NK01, NK03 (Division 1) and NK02, NK04 (Division 2) for ≥ 15 min. (Note 1)	

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

Mode Applicability:

1 - Power Operation, 2 - Startup, 3 - Hot Standby, 4 - Hot Shutdown

Definition(s):

None

Basis:

Deleted: Callaway

The vital DC buses are the following 125 VDC Class 1E buses (ref. 1):

Division 1:	Division 2:
• NK01	• NK02
• NK03	• NK04

There are four, 60 cell, lead-calcium storage batteries (NK11, NK12, NK13 and NK14) that supplement the output of the battery chargers. They supply DC power to the distribution buses when AC power to the chargers is lost or when transient loads exceed the 300 amp capacity of the battery chargers.

Due to the load distribution on each of the 125VDC buses, the four batteries for each bus do not have the same rating. All four of the 125VDC buses supply inverters for 120VAC NN bus power as well as control power for various safety related systems. NK01 and NK04 supply additional DC loads such as diesel field flashing, breaker control power, main control board power and emergency lighting. These loads are not supplied by the other two buses, NK02 and NK03. For this reason, batteries NK11 and NK14 require additional capacity. Each battery is designed to have sufficient stored energy to supply the required emergency loads for 240 minutes following a loss of AC power (station blackout) (ref. 2, 3, 4).

Minimum DC bus voltage is 107.0 VDC (ref. 4, 5). Bus voltage may be obtained from the following instruments (ref. 6):

- NK EI-1 (NK01)
- NK EI-2 (NK02)
- NK EI-3 (NK03)
- NK EI-4 (NK04)

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.

Deleted: NEI 99-01 Basis:¶

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.

Callaway Basis Reference(s):

1. E-21010(Q) DC Single Line Diagram
2. FSAR Tables 8.3-1, -2, -3
3. FSAR Section 8.3.2
4. Calculation NK-10, NK System DC Voltage Drop
5. FSAR Table 8.3A-1 III.B
6. ECA-0.0 Loss of All AC Power
7. NEI 99-01 SS8

Category: S – System Malfunction

Subcategory: 3 – Loss of Control Room Indications

Initiating Condition: UNPLANNED loss of Control Room indications for 15 minutes or longer

EAL:

SU3.1	**Unusual Event**
An UNPLANNED event results in the inability to monitor one or more Table S-2 parameters from within the Control Room for ≥ 15 min. (Note 1)	

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

Table S-2 Safety System Parameters
• Reactor power • RCS level • RCS pressure • Core Exit T/C temperature • Level in at least one S/G • Auxiliary or emergency feed flow in at least one S/G

Mode Applicability:

1 - Power Operation, 2 - Startup, 3 - Hot Standby, 4 - Hot Shutdown

Definition(s):

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

Basis:

Deleted: Callaway

SAFETY SYSTEM parameters listed in Table S-1 are monitored in the Control Room through a combination of hard control panel indicators as well as computer based information systems. The Plant Computer, which displays SPDS required information, serves as a redundant compensatory indicator which may be utilized in lieu of normal Control Room indicators (ref. 1, 2).

Deleted: NEI 99-01 Basis:

This IC addresses the difficulty associated with monitoring normal plant conditions without the ability to obtain SAFETY SYSTEM parameters from within the Control Room. This condition is a precursor to a more significant event and represents a potential degradation in the level of safety of the plant.

As used in this EAL, an "inability to monitor" means that values for one or more of the listed parameters cannot be determined from within the Control Room. This situation would require a loss of all of the Control Room sources for the given parameter(s). For example, the reactor power level cannot be determined from any analog, digital and recorder source within the Control Room.

An event involving a loss of plant indications, annunciators and/or display systems is evaluated in accordance with 10 CFR 50.72 (and associated guidance in NUREG-1022) to determine if an NRC event report is required. The event would be reported if it significantly impaired the capability to perform emergency assessments. In particular, emergency assessments necessary to implement abnormal operating procedures, emergency operating procedures, and emergency plan implementing procedures addressing emergency classification, accident assessment, or protective action decision-making.

This EAL is focused on a selected subset of plant parameters associated with the key safety functions of reactivity control, core cooling and RCS heat removal. The loss of the ability to determine one or more of these parameters from within the Control Room is considered to be more significant than simply a reportable condition. In addition, if all indication sources for one or more of the listed parameters are lost, then the ability to determine the values of other SAFETY SYSTEM parameters may be impacted as well. For example, if the value for reactor vessel level cannot be determined from the indications and recorders on a main control board, the SPDS or the plant computer, the availability of other parameter values may be compromised as well.

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

Escalation of the emergency classification level would be via IC SA3.

Callaway Basis Reference(s):

1. UFSAR Section 7.5 Safety-Related Display Instrumentation
2. OTO-RJ-00001 Loss of Plant Computer
3. NEI 99-01 SU2

Category: S – System Malfunction

Subcategory: 3 – Loss of Control Room Indications

Initiating Condition: UNPLANNED loss of Control Room indications for 15 minutes or longer with a significant transient in progress

EAL:

SA3.1 Alert

An UNPLANNED event results in the inability to monitor one or more Table S-2 parameters from within the Control Room for ≥ 15 min. (Note 1)

AND

Any significant transient is in progress, Table S-3

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

Table S-2 Safety System Parameters
• Reactor power • RCS level • RCS pressure • Core Exit T/C temperature • Level in at least one S/G • Auxiliary or emergency feed flow in at least one S/G

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

Mode Applicability:

1 - Power Operation, 2 - Startup, 3 - Hot Standby, 4 - Hot Shutdown

Definition(s):

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

Basis:

Deleted: Callaway

SAFETY SYSTEM parameters listed in Table S-1 are monitored in the Control Room through a combination of hard control panel indicators as well as computer based information systems. The Plant Computer, which displays SPDS required information, serves as a redundant compensatory indicator which may be utilized in lieu of normal Control Room indicators (ref. 1, 2).

Significant transients are listed in Table S-2 and include response to automatic or manually initiated functions such as reactor trips, runbacks involving greater than or equal to 25% thermal power change, electrical load rejections of greater than 25% full electrical load or ECCS (SI) injection actuations.

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.

Deleted: NEI 99-01 Basis:¶

As used in this EAL, an “inability to monitor” means that values for one or more of the listed parameters cannot be determined from within the Control Room. This situation would require a loss of all of the Control Room sources for the given parameter(s). For example, the reactor power level cannot be determined from any analog, digital and recorder source within the Control Room.

An event involving a loss of plant indications, annunciators and/or display systems is evaluated in accordance with 10 CFR 50.72 (and associated guidance in NUREG-1022) to determine if an NRC event report is required. The event would be reported if it significantly impaired the capability to perform emergency assessments. In particular, emergency assessments necessary to implement abnormal operating procedures, emergency operating procedures, and emergency plan implementing procedures addressing emergency classification, accident assessment, or protective action decision-making.

This EAL is focused on a selected subset of plant parameters associated with the key safety functions of reactivity control, core cooling and RCS heat removal. The loss of the ability to determine one or more of these parameters from within the Control Room is considered to be more significant than simply a reportable condition. In addition, if all indication sources for one or more of the listed parameters are lost, then the ability to determine the values of other SAFETY SYSTEM parameters may be impacted as well. For example, if the value for reactor vessel level cannot be determined from the indications and recorders on a main control board, the SPDS or the plant computer, the availability of other parameter values may be compromised as well.

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

Escalation of the emergency classification level would be via ICs FS1 or IC RS1

Callaway Basis Reference(s):

1. UFSAR Section 7.5 Safety-Related Display Instrumentation
2. OTO-RJ-00001 Loss of Plant Computer
3. NEI 99-01 SA2

ATTACHMENT 1
EAL Bases

Category: S – System Malfunction

Subcategory: 4 – RCS Activity

Initiating Condition: Reactor coolant activity greater than Technical Specification allowable limits

EAL:

SU4.1	**Unusual Event**
Sample analysis indicates RCS activity > Technical Specification Section 3.4.16 limits	

Mode Applicability:

1 - Power Operation, 2 - Startup, 3 - Hot Standby, 4 - Hot Shutdown

Definition(s):

None

Basis:

Deleted: Callaway

The specific iodine activity is limited to either ≤ 60 µCi/gm Dose Equivalent I-131 or ≤ 1.0 µCi/gm Dose Equivalent I-131 for a > 48 hr continuous period. The specific Xe-133 activity is limited to ≤ 225 µCi/gm Dose Equivalent Xe-133 (ref 1, 2).

Deleted: XE

Deleted: NEI 99-01 Basis:¶

This IC addresses a reactor coolant activity value that exceeds an allowable limit specified in Technical Specifications. This condition is a precursor to a more significant event and represents a potential degradation of the level of safety of the plant.

Escalation of the emergency classification level would be via ICs FA1 or the Recognition Category R ICs.

Callaway Basis Reference(s):

1. Callaway Technical Specifications section 3.4.16 RCS Specific Activity
2. OTO-BB-00005 High Coolant Activity
3. NEI 99-01 SU3

Category: S – System Malfunction

Subcategory: 5 – RCS Leakage

Initiating Condition: RCS leakage for 15 minutes or longer

EAL:

SU5.1 Unusual Event

RCS unidentified or pressure boundary leakage > 10 gpm for ≥ 15 min.

OR

RCS identified leakage > 25 gpm for ≥ 15 min.

OR

Leakage from the RCS to a location outside containment > 25 gpm for ≥ 15 min. (Note 1)

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

Mode Applicability:

1 - Power Operation, 2 - Startup, 3 - Hot Standby, 4 - Hot Shutdown

Definition(s):

None

Basis:

Deleted: Callaway

Manual or computer-based methods of performing an RCS inventory balance are normally used to determine RCS leakage. The Personal Computer (PC) is preferred method of calculating RCS leak rate. When the PC is used, plant status information and all calculations are generated by the OSPBB9 software program. When the PC software is not available, procedural guidance is available to perform the manual RCS inventory balance (ref. 1).

Identified leakage includes

- Leakage such as that from pump seals or valve packing (except reactor coolant pump (RCP) seal water injection or leakoff), that is captured and conducted to collection systems or a sump or collecting tank, or
- Leakage into the containment atmosphere from sources that are both specifically located and known either not to interfere with the operation of leakage detection systems or not to be pressure boundary leakage, or
- RCS leakage through a steam generator to the secondary system (ref. 2).

Unidentified leakage is all leakage (except RCP seal water injection or leakoff) that is not identified leakage (ref. 2).

Pressure Boundary leakage is leakage (except SG leakage) through a nonisolable fault in an RCS component body, pipe wall, or vessel wall (ref. 2)

RCS leakage outside of the containment that is not considered identified or unidentified leakage per Technical Specifications includes leakage via interfacing systems such as RCS to the Component Cooling Water, or systems that directly see RCS pressure outside containment

such as Chemical & Volume Control System, Nuclear Sampling system and Residual Heat Removal system (when in the shutdown cooling mode) (ref. 3, 4)

Escalation of this EAL to the Alert level is via Category F, Fission Product Barrier Degradation, EAL FA1.1.

This IC addresses RCS leakage which may be a precursor to a more significant event. In this case, RCS leakage has been detected and operators, following applicable procedures, have been unable to promptly isolate the leak. This condition is considered to be a potential degradation of the level of safety of the plant.

Deleted: NEI 99-01 Basis:¶

The first and second EAL conditions are focused on a loss of mass from the RCS due to “unidentified leakage", "pressure boundary leakage" or "identified leakage” (as these leakage types are defined in the plant Technical Specifications). The third condition addresses an RCS mass loss caused by an UNISOLABLE leak through an interfacing system. These conditions thus apply to leakage into the containment, a secondary-side system (e.g., steam generator tube leakage) or a location outside of containment.

The leak rate values for each condition were selected because they are usually observable with normal Control Room indications. Lesser values typically require time-consuming calculations to determine (e.g., a mass balance calculation). The first condition uses a lower value that reflects the greater significance of unidentified or pressure boundary leakage.

The release of mass from the RCS due to the as-designed/expected operation of a relief valve does not warrant an emergency classification. An emergency classification would be required if a mass loss is caused by a relief valve that is not functioning as designed/expected (e.g., a relief valve sticks open and the line flow cannot be isolated).

The 15-minute threshold duration allows sufficient time for prompt operator actions to isolate the leakage, if possible.

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

Callaway Basis Reference(s):

1. OSP-BB-00009 RCS Inventory Balance
2. Callaway Technical Specifications Definitions section 1.1
3. UFSAR Section 5.2.5.2.1 Intersystem Leakage
4. OTO-BB-00003-R014 Excess RCS Leakage
5. NEI 99-01 SU4

Category: S – System Malfunction

Subcategory: 6 – RTS Failure

Initiating Condition: Automatic or manual trip fails to shut down the reactor

EAL:

SU6.1 Unusual Event

An automatic trip did **not** shut down the reactor as indicated by reactor power ≥ 5% after **any** RTS setpoint is exceeded

AND

A subsequent automatic trip or manual trip action taken at the reactor control consoles (SB-HS-1 or SB-HS-42) is successful in shutting down the reactor as indicated by reactor power < 5% (Note 8)

Note 8: A manual trip action is any operator action, or set of actions, which causes the control rods to be rapidly inserted into the core, and **does not** include manually driving in control rods or implementation of boron injection strategies.

Mode Applicability:

1 - Power Operation

Definition(s):

None

Basis:

Deleted: Callaway

The first condition of this EAL identifies the need to cease critical reactor operations by actuation of the automatic Reactor Trip System (RTS) trip function. A reactor trip is automatically initiated by the RTS when certain continuously monitored parameters exceed predetermined setpoints (ref. 1).

Following a successful reactor trip, rapid insertion of the control rods occurs. Nuclear power promptly drops to a fraction of the original power level and then decays to a level several decades less with a negative startup rate. The reactor power drop continues until reactor power reaches the point at which the influence of source neutrons on reactor power starts to be observable. A predictable post-trip response from an automatic reactor trip signal should therefore consist of a prompt drop in reactor power as sensed by the nuclear instrumentation and a lowering of power into the source range. A successful trip has therefore occurred when there is sufficient rod insertion from the trip of RTS to bring the reactor power below the immediate shutdown decay heat level of 5% (ref. 2, 3, 4).

For the purposes of emergency classification, successful manual trip actions are those which can be quickly performed from the reactor control console; SB-HS-1 on Panel RL003 or SB-HS-42 on Panel RL006. Reactor shutdown achieved by use of other trip actions specified in FR-S.1 Response to Nuclear Power Generation/ATWS (such as opening PG19 and PG20 supply breakers, depressing manual pushbutton on turbine control panel, emergency boration or manually driving control rods) do not constitute a successful manual trip (ref. 4).

Following any automatic RTS trip signal, E-0 (ref. 2) and /FR-S.1 (ref. 4) prescribe insertion of redundant manual trip signals to back up the automatic RTS trip function and ensure reactor shutdown is achieved. Even if the first subsequent manual trip signal inserts all control rods to the full-in position immediately after the initial failure of the automatic trip, the lowest level of classification that must be declared is an Unusual Event (ref. 4).

A reactor trip resulting from actuation of the ATWS Mitigation System Actuation Circuitry (AMSAC) logic that results in full insertion of control rods and diminishing neutron flux is considered a successful reactor trip. AMSAC automatically initiates auxiliary feedwater and a turbine trip under conditions indicative of an Anticipated Transient Without Scram (ATWS) event (ref. 5).

In the event that the operator identifies a reactor trip is imminent and initiates a successful manual reactor trip before the automatic RTS trip setpoint is reached, no declaration is required. The successful manual trip of the reactor before it reaches its automatic trip setpoint or reactor trip signals caused by instrumentation channel failures do not lead to a potential fission product barrier loss. However, if subsequent manual reactor trip actions fail to reduce reactor power below 5%, the event escalates to the Alert under EAL SA6.1.

If by procedure, operator actions include the initiation of an immediate manual trip following receipt of an automatic trip signal and there are no clear indications that the automatic trip failed (such as a time delay following indications that a trip setpoint was exceeded), it may be difficult to determine if the reactor was shut down because of automatic trip or manual actions. If a subsequent review of the trip actuation indications reveals that the automatic trip did not cause the reactor to be shut down, then consideration should be given to evaluating the fuel for potential damage, and the reporting requirements of 50.72 should be considered for the transient event.

This IC addresses a failure of the RTS to initiate or complete an automatic or manual reactor trip that results in a reactor shutdown, and either a subsequent operator manual action taken at the reactor control consoles or an automatic trip is successful in shutting down the reactor. This event is a precursor to a more significant condition and thus represents a potential degradation of the level of safety of the plant.

Deleted: NEI 99-01 Basis:¶

Following the failure on an automatic reactor trip, operators will promptly initiate manual actions at the reactor control consoles to shutdown the reactor (e.g., initiate a manual reactor trip). If these manual actions are successful in shutting down the reactor, core heat generation will quickly fall to a level within the capabilities of the plant's decay heat removal systems.

If an initial manual reactor trip is unsuccessful, operators will promptly take manual action at another location(s) on the reactor control consoles to shutdown the reactor (e.g., initiate a manual reactor trip) using a different switch). Depending upon several factors, the initial or subsequent effort to manually trip the reactor, or a concurrent plant condition, may lead to the generation of an automatic reactor trip signal. If a subsequent manual or automatic trip is successful in shutting down the reactor, core heat generation will quickly fall to a level within the capabilities of the plant's decay heat removal systems.

A manual action at the reactor control consoles is any operator action, or set of actions, which causes the control rods to be rapidly inserted into the core (e.g., initiating a manual reactor trip). This action does not include manually driving in control rods or implementation of boron

injection strategies. Actions taken at back-panels or other locations within the Control Room, or any location outside the Control Room, are not considered to be "at the reactor control consoles".

The plant response to the failure of an automatic or manual reactor trip will vary based upon several factors including the reactor power level prior to the event, availability of the condenser, performance of mitigation equipment and actions, other concurrent plant conditions, etc. If subsequent operator manual actions taken at the reactor control consoles are also unsuccessful in shutting down the reactor, then the emergency classification level will escalate to an Alert via IC SA6. Depending upon the plant response, escalation is also possible via IC FA1. Absent the plant conditions needed to meet either IC SA6 or FA1, an Unusual Event declaration is appropriate for this event.

A reactor shutdown is determined in accordance with applicable Emergency Operating Procedure criteria.

Should a reactor trip signal be generated as a result of plant work (e.g., RTS setpoint testing), the following classification guidance should be applied.

- If the signal causes a plant transient that should have included an automatic reactor trip and the RTS fails to automatically 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.

Callaway Basis Reference(s):

1. Callaway Technical Specifications section 3.3.1 Reactor Trip System (RTS) Instrumentation
2. E-0 Reactor Trip or Safety Injection
3. F-0 Critical Safety Function Status Trees - Subcriticality
4. FR-S.1 Response to Nuclear Power Generation/ATWS
5. FSAR Section 7.7.1
6 NEI 99-01 SU5

Category: S – System Malfunction

Subcategory: 6 – RTS Failure

Initiating Condition: Automatic or manual trip fails to shut down the reactor

EAL:

SU6.2 Unusual Event

A manual trip did **not** shut down the reactor as indicated by reactor power ≥ 5% after **any** manual trip action was initiated

AND

A subsequent automatic trip or manual trip action taken at the reactor control console (SB-HS-1 or SB-HS-42) is successful in shutting down the reactor as indicated by reactor power < 5% (Note 8)

Note 8: A manual trip action is any operator action, or set of actions, which causes the control rods to be rapidly inserted into the core, and **does not** include manually driving in control rods or implementation of boron injection strategies.

Mode Applicability:

1 - Power Operation

Definition(s):

None

Basis:

Deleted: Callaway

This EAL addresses a failure of a manually initiated trip in the absence of having exceeded an automatic RTS trip setpoint and a subsequent automatic or manual trip is successful in shutting down the reactor (reactor power < 5%). (ref. 1).

Following a successful reactor trip, rapid insertion of the control rods occurs. Nuclear power promptly drops to a fraction of the original power level and then decays to a level several decades less with a negative startup rate. The reactor power drop continues until reactor power reaches the point at which the influence of source neutrons on reactor power starts to be observable. A predictable post-trip response from an automatic reactor trip signal should therefore consist of a prompt drop in reactor power as sensed by the nuclear instrumentation and a lowering of power into the source range. A successful trip has therefore occurred when there is sufficient rod insertion from the trip of RTS to bring the reactor power below the immediate shutdown decay heat level of 5% (ref. 2, 3, 4).

For the purposes of emergency classification, successful manual trip actions are those which can be quickly performed from the reactor control console; SB-HS-1 on Panel RL003 or SB-HS-42 on Panel RL006. Reactor shutdown achieved by use of other trip actions specified in FR-S.1 Response to Nuclear Power Generation/ATWS (such as opening PG19 and PG20 supply breakers, depressing manual pushbutton on turbine control panel, emergency boration or manually driving control rods) do not constitute a successful manual trip (ref. 4).

Following the failure of any manual trip signal, E-0 (ref. 2) and FR-S.1 (ref. 4) prescribe insertion of redundant manual trip signals to back up the RTS 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. 4).

A reactor trip resulting from actuation of the ATWS Mitigation System Actuation Circuitry (AMSAC) logic that results in full insertion of control rods and diminishing neutron flux is considered a successful reactor trip. AMSAC automatically initiates auxiliary feedwater and a turbine trip under conditions indicative of an Anticipated Transient Without Scram (ATWS) event (ref. 5).

If both subsequent automatic and subsequent manual reactor trip actions in the Control Room fail to reduce reactor power below the power associated with the safety system design (< 5%) following a failure of an initial manual trip, the event escalates to an Alert under EAL SA6.1

This IC addresses a failure of the RTS to initiate or complete an automatic or manual reactor trip that results in a reactor shutdown, and either a subsequent operator manual action taken at the reactor control consoles or an automatic trip is successful in shutting down the reactor. This event is a precursor to a more significant condition and thus represents a potential degradation of the level of safety of the plant.

Deleted: NEI 99-01 Basis:¶

Following the failure on an automatic reactor trip, operators will promptly initiate manual actions at the reactor control consoles to shutdown the reactor (e.g., initiate a manual reactor trip). If these manual actions are successful in shutting down the reactor, core heat generation will quickly fall to a level within the capabilities of the plant's decay heat removal systems.

If an initial manual reactor trip is unsuccessful, operators will promptly take manual action at another location(s) on the reactor control consoles to shutdown the reactor (e.g., initiate a manual reactor trip) using a different switch). Depending upon several factors, the initial or subsequent effort to manually trip the reactor, or a concurrent plant condition, may lead to the generation of an automatic reactor trip signal. If a subsequent manual or automatic trip is successful in shutting down the reactor, core heat generation will quickly fall to a level within the capabilities of the plant's decay heat removal systems.

A manual action at the reactor control consoles is any operator action, or set of actions, which causes the control rods to be rapidly inserted into the core (e.g., initiating a manual reactor trip). This action does not include manually driving in control rods or implementation of boron injection strategies. Actions taken at back-panels or other locations within the Control Room, or any location outside the Control Room, are not considered to be "at the reactor control consoles".

The plant response to the failure of an automatic or manual reactor trip will vary based upon several factors including the reactor power level prior to the event, availability of the condenser, performance of mitigation equipment and actions, other concurrent plant conditions, etc. If subsequent operator manual actions taken at the reactor control consoles are also unsuccessful in shutting down the reactor, then the emergency classification level will escalate to an Alert via IC SA6. Depending upon the plant response, escalation is also possible via IC FA1. Absent the plant conditions needed to meet either IC SA6 or FA1, an Unusual Event declaration is appropriate for this event.

A reactor shutdown is determined in accordance with applicable Emergency Operating Procedure criteria.

Should a reactor trip signal be generated as a result of plant work (e.g., RTS setpoint testing), the following classification guidance should be applied.

- If the signal causes a plant transient that should have included an automatic reactor trip and the 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.

Callaway Basis Reference(s):

1. Callaway Technical Specifications section 3.3.1 Reactor Trip System (RTS) Instrumentation
2. E-0 Reactor Trip or Safety Injection
3. F-0 Critical Safety Function Status Trees - Subcriticality
4. FR-S.1 Response to Nuclear Power Generation/ATWS
5. FSAR Section 7.7.1
6. NEI 99-01 SU5

Category: S – System Malfunction

Subcategory: 2 – RTS Failure

Initiating Condition: Automatic or manual trip fails to shut down the reactor and subsequent manual actions taken at the reactor control consoles are not successful in shutting down the reactor

EAL:

SA6.1	**Alert**

An automatic or manual trip fails to shut down the reactor as indicated by reactor power ≥ 5%

AND

Manual trip actions taken at the reactor control console (SB-HS-1 or SB-HS-42) are **not** successful in shutting down the reactor as indicated by reactor power ≥ 5% (Note 8)

Note 8: A manual trip action is any operator action, or set of actions, which causes the control rods to be rapidly inserted into the core, and **does not** include manually driving in control rods or implementation of boron injection strategies.

Mode Applicability:

1 - Power Operation

Definition(s):

None

Basis:

Deleted: Callaway

This EAL addresses any automatic or manual reactor trip signal that fails to shut down the reactor (reactor power < 5%) followed by a subsequent manual trip that fails to shut down the reactor to an extent the reactor is producing energy in excess of the heat load for which the safety systems were designed (ref. 1).

For the purposes of emergency classification, successful manual trip actions are those which can be quickly performed from the reactor control console; SB-HS-1 on Panel RL003 or SB-HS-42 on Panel RL006. Reactor shutdown achieved by use of other trip actions specified in FR-S.1 Response to Nuclear Power Generation/ATWS (such as opening PG19 and PG20 supply breakers, depressing manual pushbutton on turbine control panel, emergency boration or manually driving control rods) do not constitute a successful manual trip (ref. 4).

A reactor trip resulting from actuation of the ATWS Mitigation System Actuation Circuitry (AMSAC) logic that results in full insertion of control rods and diminishing neutron flux is considered a successful reactor trip. AMSAC automatically initiates auxiliary feedwater and a turbine trip under conditions indicative of an Anticipated Transient Without Scram (ATWS) event (ref. 5).

5% rated power is a minimum reading on the power range scale that indicates continued power production. It also approximates the decay heat which the shutdown systems were designed to remove and is indicative of a condition requiring immediate response to prevent

subsequent core damage. Below 5%, plant response will be similar to that observed during a normal shutdown. Nuclear instrumentation can be used to determine if reactor power is greater than 5 % power (ref. 3, 4).

Escalation of this event to a Site Area Emergency would be under EAL SS6.1 or Emergency Coordinator judgment.

This IC addresses a failure of the RPS to initiate or complete an automatic or manual reactor trip that results in a reactor shutdown, and subsequent operator manual actions taken at the reactor control consoles to shutdown the reactor are also unsuccessful. This condition represents an actual or potential substantial degradation of the level of safety of the plant. An emergency declaration is required even if the reactor is subsequently shutdown by an action taken away from the reactor control consoles since this event entails a significant failure of the RTS.

Deleted: NEI 99-01 Basis:¶

A manual action at the reactor control console is any operator action, or set of actions, which causes the control rods to be rapidly inserted into the core (e.g., initiating a manual reactor trip). This action does not include manually driving in control rods or implementation of boron injection strategies. If this action(s) is unsuccessful, operators would immediately pursue additional manual actions at locations away from the reactor control console (e.g., locally opening breakers). Actions taken at backpanels or other locations within the Control Room, or any location outside the Control Room, are not considered to be “at the reactor control console”.

The plant response to the failure of an automatic or manual reactor trip will vary based upon several factors including the reactor power level prior to the event, availability of the condenser, performance of mitigation equipment and actions, other concurrent plant conditions, etc. If the failure to shut down the reactor is prolonged enough to cause a challenge to the core cooling or RCS heat removal safety functions, the emergency classification level will escalate to a Site Area Emergency via IC SS6. Depending upon plant responses and symptoms, escalation is also possible via IC FS1. Absent the plant conditions needed to meet either IC SS6 or FS1, an Alert declaration is appropriate for this event.

It is recognized that plant responses or symptoms may also require an Alert declaration in accordance with the Recognition Category F ICs; however, this IC and EAL are included to ensure a timely emergency declaration.

A reactor shutdown is determined in accordance with applicable Emergency Operating Procedure criteria.

Callaway Basis Reference(s):

1. Callaway Technical Specifications section 3.3.1 Reactor Trip System (RTS) Instrumentation
2. E-0 Reactor Trip or Safety Injection
3. F-0 Critical Safety Function Status Trees - Subcriticality
4. FR-S.1 Response to Nuclear Power Generation/ATWS
5. FSAR Section 7.7.1
6. NEI 99-01 SA5

Category: S – System Malfunction

Subcategory: 2 – RTS Failure

Initiating Condition: Inability to shut down the reactor causing a challenge to core cooling or RCS heat removal

EAL:

SS6.1 Site Area Emergency

An automatic or manual trip fails to shut down the reactor as indicated by reactor power ≥ 5%

AND

All actions to shut down the reactor are **not** successful as indicated by reactor power ≥ 5%

AND EITHER:

- CSFST Core Cooling RED Path conditions met
- CSFST Heat Sink RED Path conditions met

Mode Applicability:

1 - Power Operation

Definition(s):

None

Basis:

Deleted: Callaway

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 opening PG19 and PG20 supply breakers, depressing manual pushbutton on turbine control panel, emergency boration or manually driving control rods) are also credited as a successful manual trip provided reactor power can be reduced below 5% before indications of an extreme challenge to either core cooling or heat removal exist (ref. 1, 4).

5% rated power is a minimum reading on the power range scale that indicates continued power production. It also approximates the decay heat which the shutdown systems were designed to remove and is indicative of a condition requiring immediate response to prevent subsequent core damage. Below 5%, plant response will be similar to that observed during a

normal shutdown. Nuclear instrumentation can be used to determine if reactor power is greater than 5 % power (ref. 1, 4).

Indication of continuing core cooling degradation is manifested by CSFST Core Cooling RED PATH conditions being met (ref. 2). Specifically, Core Cooling RED PATH conditions exist if either core exit T/Cs are reading greater than or equal to 1200ºF or core exit T/Cs are reading greater than or equal to 706ºF with Reactor Vessel Lower Range level less than or equal to that specified based on the number of RCPs running (ref. 2).

Indication of inability to adequately remove heat from the RCS is manifested by CSFST Heat Sink RED PATH conditions being met (ref. 2). Specifically, Heat Sink RED PATH conditions exist if narrow range level in at least on steam generator is not greater than or equal to 7% (25% ACC) and total feedwater flow to the steam generators is less than or equal to 285,000 lbm/hr. (ref. 3).

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.

Deleted: NEI 99-01 Basis:¶

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.

Callaway Basis Reference(s):

1. CSF-1 Critical Safety Function Status Trees – Figure 1 Subcriticality
2. CSF-1 Critical Safety Function Status Tress – Figure 2 Core Cooling
3. CSF-1 Critical Safety Function Status Tress – Figure 3 Heat Sink
4. FR-S.1 Response to Nuclear Power Generation/ATWS
5. NEI 99-01 SS5

Category: S – System Malfunction

Subcategory: 7 – Loss of Communications

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

EAL:

SU7.1 Unusual Event

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

OR

Loss of **all** Table S-4 ORO communication methods

OR

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

Table S-4 Communication Methods			
System	**Onsite**	**ORO**	**NRC**
Gaitronics	X		
Plant Radios	X		
Plant Emergency Dedicated Phones	X		
Plant Telephone System	X	X	X
ENS (Red Phone) Line		X	X
Back-Up Radio System		X	
Sentry Notification System		X	

Mode Applicability:

1 - Power Operation, 2 - Startup, 3 - Hot Standby, 4 - Hot Shutdown

Definition(s):

None

Basis:

Onsite/offsite communications include one or more of the systems listed in Table S-4 (ref. 1, 2).

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1. Gai-tronics system

   The Gai-tronics system provides six separate independent communication channels--one general page, one Control Room page and four party lines. Communication between parties within the plant can be easily and quickly established by using the general page channel. Communication between parties in the plant and the Control Room can be easily and quickly established using the Control Room page channel. The party line channel is normally used after the page call is completed. As many as four party lines may communicate simultaneously. The portion of the PA system connecting the fuel transfer area in the Containment, the spent fuel area and new fuel handling area in the fuel building, and the control room can be isolated from the remainder of the PA system from the control room. This permits extended use of the fuel handling communications system without disruption to the remainder of the system.

2. Plant Radios

   A six channel 800 MHZ trunked radio system for overall plant site area coverage reaches out as far as the intake structure. This two-way radio system provides communications for operating purposes with plant radio-equipped vehicles and plant hand-held portable radios. These systems are for use during normal operation or during a plant emergency. This radio system is available on the Control Room radio consoles, on the security radio consoles, on the EOF radio console, and the TSC radio console. This system is also in the field monitoring team vehicles and is used to communicate during emergencies.

3. Plant Emergency Dedicated Phones

   Three independent telephone systems are available for communications between the Emergency Response Facilities: the Technical Assessment Bridge Line, the Dose Assessment Bridge Line and the Emergency Management Bridge Line. Each system operates independently from the other systems and allows for conference calls between the members of that bridge line group

4. Plant telephone system

   The telephone system consists of digital automatic switchboard (DPBX) equipment and telephone stations. The DPBX is provided with redundant processors for reliability. The telephone stations are located throughout the power block, in the main control room, in the various buildings around the site, in the security building, and in the service building where the administrative offices are located. For emergency use, unlisted telephone numbers are provided for direct access to the outside local public telephone system.

5. ENS (Red Phone) line

   The NRC Emergency Notification System (ENS) is an FTS telephone used for official communications with NRC Headquarters. The NRC Headquarters has the capability to patch into the NRC Regional offices. The primary purpose of this phone is to provide a reliable method for the initial notification of the NRC and to maintain continuous communications with the NRC after initial notification. ENS telephones are located in the Control Room, TSC and EOF.

6. Back-Up Radio System (BURS)

   The Back-up Radio System is a communication link between the Callaway Plant and offsite emergency response agencies. The primary use of this system is the back up notification of offsite agencies and the coordination of offsite activities during a radio logic a1 emergency. The system uses 800 MHz radios. There are radio control base: units in the Plant Control Room, TSC and EOF, as well as each county EOC and the State EOC. The backup to this system is the commercial touchtone telephone system Notifications may also be initiated through the Callaway County/City of Fulton EOC via the Security radio.

7. Sentry Notification System

   A computerized notification system linked between the Callaway Plant, the State Emergency Management Agency and the four (4) EPZ risk counties. It allows the Communicator to fill out a notification form on screen and transmit the data simultaneously. Notifications on Sentry can be initiated from the Control Room, the Emergency Operations Facility (EOF), or the Technical Support Center (TSC).

This EAL is the hot condition equivalent of the cold condition EAL CU5.1.

Deleted: NEI 99-01 Basis:¶

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

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

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

The second EAL condition addresses a total loss of the communications methods used to notify all OROs of an emergency declaration. The OROs referred to here are the State, Callaway, Gasconade, Montgomery and Osage County EOCs

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

Callaway Basis Reference(s):

1. Callaway Plant Radiological Emergency Response Plan (RERP), Section 7.2
2. FSAR Section 9.5.2
3. NEI 99-01 SU6

Category: S – System Malfunction

Subcategory: 8 – Containment Failure

Initiating Condition: Failure to isolate containment or loss of containment pressure control.

EAL:

SU8.1	**Unusual Event**
Any penetration is not isolated within 15 min. of a VALID containment isolation signal	
OR	
Containment pressure > 27 psig with < one full train of containment depressurization equipment operating per design for ≥ 15 min. (Note 9)	
(Note 1)	

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

Note 9: One Containment Spray System train and one Containment Cooling System train comprise one full train of depressurization equipment.

Mode Applicability:

1 - Power Operation, 2 - Startup, 3 - Hot Standby, 4 - Hot Shutdown

Definition(s):

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

Basis:

Deleted: Callaway

The Containment Spray System consists of two separate trains of equal capacity, each capable of meeting the design bases requirement. Each train includes a containment spray pump, spray headers, nozzles, valves, and piping. The refueling water storage tank (RWST) supplies borated water to the Containment Spray System during the injection phase of operation. In the recirculation mode of operation, Containment Spray pump suction is transferred from the RWST to the Containment sumps (ref. 2).

The Containment Cooling System consists of two trains of Containment cooling, each of sufficient capacity to supply 100% of the design cooling requirement. Each train of two fan units is supplied with cooling water from a separate train of essential service water (ESW). Air is drawn into the coolers through the fan and discharged to the steam generator compartments, pressurizer compartment, and instrument tunnel, and outside the secondary shield in the lower areas of containment. During normal operation, all four fan units may be operating. In post accident operation following an actuation signal, the Containment Cooling System fans are designed to start automatically in slow speed if not already running (ref. 3).

The Containment pressure setpoint (27 psig, ref. 4, 5, 6) is the pressure at which the equipment should actuate and begin performing its function. The design basis accident analyses and evaluations assume the loss of one Containment Spray System train and one Containment Cooling System train (ref. 7). Consistent with the design requirement, “one full train of depressurization equipment” is therefore defined to be the availability of one train of each system. If less than this equipment is operating and Containment pressure is above the actuation setpoint, the threshold is met.

This EAL addresses a failure of one or more containment penetrations to automatically isolate (close) when required by an actuation signal. It also addresses an event that results in high containment pressure with a concurrent failure of containment pressure control systems. Absent challenges to another fission product barrier, either condition represents potential degradation of the level of safety of the plant.

Deleted: NEI 99-01 Basis:¶

For the first condition, the containment isolation signal must be generated as the result on an off-normal/accident condition (e.g., a safety injection or high containment pressure); a failure resulting from testing or maintenance does not warrant classification. The determination of containment and penetration status – isolated or not isolated – should be made in accordance with the appropriate criteria contained in the plant AOPs and EOPs. The 15-minute criterion is included to allow operators time to manually isolate the required penetrations, if possible.

The second condition addresses a condition where containment pressure is greater than the setpoint at which containment energy (heat) removal systems are designed to automatically actuate, and less than one full train of equipment is capable of operating per design. The 15-minute criterion is included to allow operators time to manually start equipment that may not have automatically started, if possible. The inability to start the required equipment indicates that containment heat removal/depressurization systems (e.g., containment sprays or ice condenser fans) are either lost or performing in a degraded manner.

This event would escalate to a Site Area Emergency in accordance with IC FS1 if there were a concurrent loss or potential loss of either the Fuel Clad or RCS fission product barriers.

Callaway Basis Reference(s):

1. FSAR Section 6.2.2
2. FSAR Section 6.2.2.1.2.1
3. FSAR Section 6.2.2.2.2
4. CSF-1 Critical Safety Function Status Trees (CSFST) Figure 5, Containment
5. FR-Z.1 Response to High Containment Pressure
6. Technical Specifications Table 3.3.2-1
7. Technical Specifications B3.6.6
8. NEI 99-01 SU7

Category: S – System Malfunction

Subcategory: 9 – Hazardous Event Affecting Safety Systems

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

EAL:

SA9.1 Alert

The occurrence of **any** Table S-5 hazardous event

AND EITHER:

- Event damage has caused indications of degraded performance in at least one train of a SAFETY SYSTEM needed for the current operating mode
- The event has caused VISIBLE DAMAGE to a SAFETY SYSTEM component or structure needed for the current operating mode

Table S-5 Hazardous Events
• Seismic event (earthquake) • Internal or external FLOODING event • High winds or tornado strike • FIRE • EXPLOSION • Other events with similar hazard characteristics as determined by the Emergency Coordinator

Mode Applicability:

1 - Power Operation, 2 - Startup, 3 - Hot Standby, 4 - Hot Shutdown

Definition(s):

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

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

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

SAFETY SYSTEM - A system required for safe plant operation, cooling down the plant and/or placing it in the cold shutdown condition, including the ECCS. These are typically systems

classified as safety-related (as defined in 10CFR50.2):

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

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

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

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

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

Basis:

Deleted: Callaway

- Annunciator 98D, OBE will illuminate if the seismic instrument detects ground motion in excess of the OBE threshold. OTO-SG-00001, Seismic Event provides the guidance for determining if an OBE earthquake threshold is exceeded and any required response actions (ref. 1).
- Internal FLOODING may be caused by events such as component failures, equipment misalignment, or outage activity mishaps (ref. 2).

Deleted: <#>The significance of seismic events are discussed under EAL HU2.1 (ref. 1).¶

- External flooding may be due to high lake level. Callaway plant grade elevation is 840.0 ft MSL. (ref. 3).
- Seismic Category I structures are analyzed to withstand a sustained, design wind velocity of at least 100 mph. (ref. 4).
- Areas containing functions and systems required for safe shutdown of the plant are identified by fire area (ref. 5).
- An explosion that degrades the performance of a SAFETY SYSTEM train or visibly damages a SAFETY SYSTEM component or structure would be classified under this EAL.

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

Deleted: NEI 99-01 Basis:¶

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

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

Escalation of the emergency classification level would be via IC FS1 or RS1.

Callaway Basis Reference(s):

1. OTO-SG-00001 Seismic Event
2. IPE Section 3.4.2.3 Results of the Vulnerability Screening
3. UFSAR Section 3.4 Water Level (Flood) Design Table 3.4-1 PMF, Groundwater, Reference, and Actual Plant Elevations
4. UFSAR Section 3.3.1.1 Design Wind Loadings
5. UFSAR Appendix 9.5B Fire Hazard Analysis and Site Combustion Restrictions
6. NEI 99-01 SA9

Category F – Fission Product Barrier Degradation

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

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

A. Fuel Clad (FC): The Fuel Clad Barrier consists of the cladding material that contains the fuel pellets.
B. Reactor Coolant System (RCS): The RCS Barrier includes the RCS primary side and its connections up to and including the pressurizer safety and relief valves, and other connections up to and including the primary isolation valves.
C. Containment (CMT): The Containment Barrier includes the containment building and connections up to and including the outermost containment isolation valves. This barrier also includes the main steam, feedwater, and blowdown line extensions outside the containment building up to and including the outermost secondary side 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 Callaway design and operating characteristics.
- As used in this category, the term RCS leakage encompasses not just those types defined in Technical Specifications but also includes the loss of RCS mass to any location– inside the primary containment, an interfacing system, or outside of the primary containment. The release of liquid or steam mass from the RCS due to the as-designed/expected operation of a relief valve is not considered to be RCS leakage.
- At the Site Area Emergency level, EAL users should maintain cognizance of how far present conditions are from meeting a threshold that would require a General Emergency declaration. For example, if the Fuel Clad and RCS fission product barriers were both lost, then there should be frequent assessments of containment radioactive inventory and integrity. Alternatively, if both the Fuel Clad and RCS fission product barriers were potentially lost, the Emergency Coordinator would have more assurance that there was no immediate need to escalate to a General Emergency.

Category: Fission Product Barrier Degradation

Subcategory: N/A

Initiating Condition: **Any** loss or **any** potential loss of either Fuel Clad or RCS

EAL:

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

Mode Applicability:

1 - Power Operation, 2 - Startup, 3 - Hot Standby, 4 - Hot Shutdown

Definition(s):

None

Basis:

Deleted: Callaway

Fuel Clad, RCS and Containment comprise the fission product barriers. Table F-1 (Attachment 2) lists the fission product barrier thresholds, bases and references.

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

Callaway Basis Reference(s):

Deleted: NEI 99-01 Basis:¶
None¶

1. NEI 99-01 FA1

Category: Fission Product Barrier Degradation

Subcategory: N/A

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

EAL:

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

Mode Applicability:

1 - Power Operation, 2 - Startup, 3 - Hot Standby, 4 - Hot Shutdown

Definition(s):

None

Basis:

Deleted: Callaway

Fuel Clad, RCS and Containment comprise the fission product barriers. Table F-1 (Attachment 2) lists the fission product barrier thresholds, bases and references.

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

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

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

Callaway Basis Reference(s):

Deleted: NEI 99-01 Basis:¶
None¶

1. NEI 99-01 FS1

Category: Fission Product Barrier Degradation

Subcategory: N/A

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

EAL:

FG1.1	**General Emergency**
Loss of **any** two barriers **AND** Loss or potential loss of third barrier (Table F-1)	

Mode Applicability:

1 - Power Operation, 2 - Startup, 3 - Hot Standby, 4 - Hot Shutdown

Definition(s):

None

Basis:

Deleted: Callaway

Fuel Clad, RCS and Containment comprise the fission product barriers. Table F-1 (Attachment 2) lists the fission product barrier thresholds, bases and references.

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

- Loss of Fuel Clad, RCS and Containment barriers
- Loss of Fuel Clad and RCS barriers with potential loss of Containment barrier
- Loss of RCS and Containment barriers with potential loss of Fuel Clad barrier
- Loss of Fuel Clad and Containment barriers with potential loss of RCS barrier

Callaway Basis Reference(s):

Deleted: NEI 99-01 Basis:¶
None¶

1. NEI 99-01 FG1

ATTACHMENT 2
Fission Product Barrier Loss/Potential Loss Matrix and Bases

Introduction

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

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

A. RCS or SG Tube Leakage
B. Inadequate Heat removal
C. CMT Radiation / RCS Activity
D. CMT Integrity or Bypass
E. Emergency Coordinator Judgment

Each category occupies a row in Table F-1 thus forming a matrix defined by the categories. The intersection of each row with each Loss/Potential Loss column forms a cell in which one or more fission product barrier thresholds appear. If NEI 99-01 does not define a threshold for a barrier Loss/Potential Loss, the word “None” is entered in the cell.

Thresholds are assigned sequential numbers within each Loss and Potential Loss column beginning with number one. In this manner, a threshold can be identified by its category title and number. For example, the first Fuel Clad barrier Loss in Category A would be assigned “FC Loss A.1,” the third Containment barrier Potential Loss in Category C would be assigned “CMT P-Loss C.3,” etc.

If a cell in Table F-1 contains more than one numbered threshold, each of the numbered thresholds, if exceeded, signifies a Loss or Potential Loss of the barrier. It is not necessary to exceed all of the thresholds in a category before declaring a barrier Loss/Potential Loss.

Subdivision of Table F-1 by category facilitates association of plant conditions to the applicable fission product barrier Loss and Potential Loss thresholds. This structure promotes a systematic approach to assessing the classification status of the fission product barriers.

When equipped with knowledge of plant conditions related to the fission product barriers, the EAL-user first scans down the category column of Table F-1, locates the likely category and then reads across the fission product barrier Loss and Potential Loss thresholds in that category to determine if a threshold has been exceeded. If a threshold has not been exceeded, the EAL-user proceeds to the next likely category and continues review of the thresholds in the new category

If the EAL-user determines that any threshold has been exceeded, by definition, the barrier is lost or potentially lost – even if multiple thresholds in the same barrier column are exceeded, only that one barrier is lost or potentially lost. The EAL-user must examine each of the three fission product barriers to determine if other barrier thresholds in the category are lost or potentially lost. For example, if containment radiation is sufficiently high, a Loss of the Fuel Clad and RCS barriers and a Potential Loss of the Containment barrier can occur. Barrier

ATTACHMENT 2

Fission Product Barrier Loss/Potential Loss Matrix and Bases

Losses and Potential Losses are then applied to the algorithms given in EALs FG1.1, FS1.1, and FA1.1 to determine the appropriate emergency classification.

In the remainder of this Attachment, the Fuel Clad barrier threshold bases appear first, followed by the RCS barrier and finally the Containment barrier threshold bases. In each barrier, the bases are given according category Loss followed by category Potential Loss beginning with Category A, then B,…, E.

Table F-1 Fission Product Barrier Threshold Matrix						
	Fuel Clad (FC) Barrier		Reactor Coolant System (RCS) Barrier		Containment (CMT) Barrier	
Category	Loss	Potential Loss	Loss	Potential Loss	Loss	Potential Loss
A **RCS or SG Tube Leakage**	None	None	1. An automatic or manual ECCS (SI) actuation required by **EITHER:** • UNISOLABLE RCS leakage • SG tube RUPTURE	1. Operation of a standby charging pump is required by **EITHER:** • UNISOLABLE RCS leakage • SG tube RUPTURE 2. CSFST Integrity-RED Path conditions met	1. A leaking or RUPTURED SG is FAULTED outside of containment	None
B **Inadequate Heat Removal**	1. CSFST Core Cooling-RED Path conditions met	1. CSFST Core Cooling-ORANGE Path conditions met 2. CSFST Heat Sink-RED Path conditions met **AND** Heat sink is required	None	1. CSFST Heat Sink-RED Path conditions met **AND** Heat sink is required	None	1. CSFST Core Cooling-RED Path conditions met **AND** Restoration procedures **not** effective within 15 min. (Note 1)
C **CMT Radiation / RCS Activity**	1. Containment radiation > 2.80E+03 R/hr on GT-RE-59 (591) or GT-RE-60 (601) 2. Dose equivalent I-131 coolant activity > 300 µCi/gm 3. CVCS letdown radiation > 2.50E+01 µCi/ml on SJ-RE-01 (016)	None	1. Containment radiation > 6.40E+00 R/hr on GT-RE-59 (591) or GT-RE-60 (601)	None	None	1. Containment radiation > 8.06E+04 R/hr on GT-RE-59 (591) or GT-RE-60 (601)
D **CMT Integrity or Bypass**	None	None	None	None	1. Containment isolation is required **AND EITHER:** • Containment integrity has been lost based on Emergency Coordinator judgment • UNISOLABLE pathway from Containment to the environment exists 2. Indications of RCS leakage outside of Containment	1. CSFST Containment-RED Path conditions met 2. Containment hydrogen concentration ≥ 4% 3. Containment pressure > 27 psig with < one full train of depressurization equipment operating per design for > 15 min. (Note 1, 9)
E **EC Judgment**	1. **Any** condition in the opinion of the Emergency Coordinator that indicates loss of the fuel clad barrier	1. **Any** condition in the opinion of the Emergency Coordinator that indicates potential loss of the fuel clad barrier	1. **Any** condition in the opinion of the Emergency Coordinator that indicates loss of the RCS barrier	1. **Any** condition in the opinion of the Emergency Coordinator that indicates potential loss of the RCS barrier	1. **Any** condition in the opinion of the Emergency Coordinator that indicates loss of the Containment barrier	1. **Any** condition in the opinion of the Emergency Coordinator that indicates potential loss of the Containment barrier

Barrier: Fuel Clad

Category: A. RCS or SG Tube Leakage

Degradation Threat: Loss

Threshold:

None

Barrier: Fuel Clad

Category: A. RCS or SG Tube Leakage

Degradation Threat: Potential Loss

Threshold:

None

Barrier: Fuel Clad

Category: B. Inadequate Heat Removal

Degradation Threat: Loss

Threshold:

1. CSFST Core Cooling-RED Path conditions met

Definition(s):

None

Basis:

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 SPDS display on the Plant Computer (ref. 1).

Deleted: Plant-Specific¶

This reading indicates temperatures within the core are sufficient to cause significant superheating of reactor coolant.

Deleted: Generic¶

Callaway Basis Reference(s):

1. CSF-1 Critical Safety Function Status Trees
2. FR-C.1 Response to Inadequate Core Cooling
3. FR-C.2 Response to Degraded Core Cooling
4. NEI 99-01 Inadequate Heat Removal Fuel Clad Loss 2.A

Barrier: Fuel Clad

Category: B. Inadequate Heat Removal

Degradation Threat: Potential Loss

Threshold:

1. CSFST Core Cooling-**ORANGE** Path conditions met

Definition(s):

None

Basis:

Critical Safety Function Status Tree (CSFST) Core Cooling-ORANGE path indicates subcooling has been lost and that some fuel clad damage may potentially occur. The CSFSTs are normally monitored using the SPDS display on the Plant Computer (ref. 1).

This reading indicates a reduction in reactor vessel water level sufficient to allow the onset of heat-induced cladding damage.

Callaway Basis Reference(s):

1. CSF-1 Critical Safety Function Status Trees
2. FR-C.1 Response to Inadequate Core Cooling
3. FR-C.2 Response to Degraded Core Cooling
4. NEI 99-01 Inadequate Heat Removal Fuel Clad Loss 2.A

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Barrier: Fuel Clad

Category: B. Inadequate Heat Removal

Degradation Threat: Potential Loss

Threshold:

2. CSFST Heat Sink-RED Path conditions met **AND** Heat sink is required

Definition(s):

None

Basis:

In combination with RCS Potential Loss B.1, meeting this threshold results in a Site Area Emergency.

Deleted: Plant-Specific¶

Critical Safety Function Status Tree (CSFST) Heat Sink-RED path indicates the ultimate heat sink function is under extreme challenge and that some fuel clad damage may potentially occur (ref. 1).

The CSFSTs are normally monitored using the 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 EOP. For example, FR-H.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 either RCS temperature is greater than 350ºF or RCS pressure is greater than 360 psig. 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).

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.

Deleted: Generic¶

Callaway Basis Reference(s):

1. CSF-1 Critical Safety Function Status Trees Figure 3 Heat Sink
2. FR-H.1 Response to Loss of Secondary Heat Sink
3. NEI 99-01 Inadequate Heat Removal Fuel Clad Loss 2.B

Barrier: Fuel Clad

Category: C. CMT Radiation / RCS Activity

Degradation Threat: Loss

Threshold:

1. Containment radiation > 2.80E+03 R/hr on GT-RE-59 (591) or GT-RE-60 (601)

Definition(s):

None

Basis:

Containment radiation monitor readings greater than 2.8E+03 R/hr (ref. 1) indicate the release of reactor coolant, with elevated activity indicative of fuel damage, into the Containment. The reading is derived assuming the instantaneous release and dispersal of the reactor coolant noble gas and iodine inventory associated with a concentration of 300 µCi/cc dose equivalent I-131 into the Containment atmosphere. Reactor coolant concentrations of this magnitude are several times larger than the maximum concentrations (including iodine spiking) allowed within technical specifications and are therefore indicative of fuel damage (approximately 5% clad failure depending on core inventory and RCS volume). This value is higher than that specified for RCS barrier Loss #3.

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Monitors used for this fission product barrier loss threshold are the Containment High Range Radiation Monitors GT-RE-59 (Panel RM-11 channel 591) and GT-RE-60 (Panel RM-11 channel 601). The threshold value of 2.8E+03 R/hr is the HI-HI (RED) alarm setpoint (ref. 2).

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.

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

Callaway Basis Reference(s):

1. EPCI-9801 pg 27
2. OTA-SP-RM011 Radiation Monitor Control Panel RM-11
2. NEI 99-01 CMT Radiation / RCS Activity Fuel Clad Loss 3.A

Barrier: Fuel Clad

Category: C. CMT Radiation / RCS Activity

Degradation Threat: Loss

Threshold:

2. Dose equivalent I-131 coolant activity > 300 µCi/cc

Definition(s):

None

Basis:

Dose Equivalent Iodine (DEI) is determined by Chemistry procedure CDP-ZZ-08100, Post Accident Sampling Guidelines (ref. 1).

Deleted: Plant-Specific¶

Elevated reactor coolant activity represents a potential degradation in the level of safety of the plant and a potential precursor of more serious problems. The threshold dose equivalent I-131 concentration is well above that expected for iodine spikes and corresponds to about 2% to 5% fuel clad damage. When reactor coolant activity reaches this level the Fuel Clad barrier is considered lost. (ref. 2).

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

Deleted: Generic¶

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

Callaway Basis Reference(s):

1. CDP-ZZ-08100 Post Accident Sampling Guidelines
2. NEI 99-01 CMT Radiation / RCS Activity Fuel Clad Loss 3.B

Barrier: Fuel Clad

Category: C. CMT Radiation / RCS Activity

Degradation Threat: Loss

Threshold:

3. CVCS letdown radiation > 2.50E+01 µCi/ml on SJ-RE-01 (016)

Definition(s):

None

Basis:

The normal Chemical and Volume Control System (CVCS) charging and letdown flow path allows purification of the reactor coolant and control of the RCS volume while maintaining a continuous feed and bleed flow between the RCS and the CVCS. Reactor coolant is first "letdown" from the RCS through a regenerative heat exchanger, which minimizes heat losses from the RCS. Additional cooling takes place in a letdown heat exchanger that acts as the heat sink for the system. Downstream of the letdown heat exchanger pressure control valve and upstream of the mixed bed demineralizers, the letdown stream passes by radiation monitor SJ-RE-01, which will warn of fission products in the letdown coolant if a fuel element failure occurs. The monitor is located in the Primary Sample Sink Room.

Deleted: Plant-Specific¶

The CVCS letdown monitor SJ-RE-01 provides indication in the Control Room on Panel RM-11 channel 016 with a range of 1.7E-03 to 1.7E+03 µCi/ml (ref. 2, 3). The HI-HI (RED) alarm is 5E0 + background + (background x 0.05) (ref. 4) and represents a total fuel clad failure in excess of 1% in 30 minutes (ref. 2, 3). Five times this alarm setpoint corresponds to approximately 5% fuel clad failure. 5% clad failure is also the basis for the coolant activity and Containment radiation Fuel Clad loss thresholds.

Callaway Basis Reference(s):

Deleted: Generic¶
None¶

1. FSAR Section 9.3.4.2
2. FSAR Table 11.5-1
3. OTA-SP-RM011 Radiation Monitor Control Panel RM-11
4. HPCI-05-02 Gaseous and Liquid Radiation Monitor Setpoints Rev. 0, Note 11
5. NEI 99-01 Other Indications Fuel Clad Loss 5.A

Barrier: Fuel Clad

Category: C. CMT Radiation / RCS Activity

Degradation Threat: Potential Loss

Threshold:

None

Barrier: Fuel Clad

Category: D. CMT Integrity or Bypass

Degradation Threat: Loss

Threshold:

None

Barrier: Fuel Clad

Category: D. CMT Integrity or Bypass

Degradation Threat: Potential Loss

Threshold:

None

Barrier: Fuel Clad

Category: E. Emergency Coordinator Judgment

Degradation Threat: Loss

Threshold:

1. **Any** condition in the opinion of the Emergency Coordinator that indicates loss of the Fuel Clad barrier

Definition(s):

None

Basis:

The Emergency Coordinator judgment threshold addresses any other factors relevant to determining if the Fuel Clad barrier is lost. Such a determination should include imminent barrier degradation, barrier monitoring capability and dominant accident sequences.

- Imminent barrier degradation exists if the degradation will likely occur within relatively short period of time based on a projection of current safety system performance. The term "imminent" refers to recognition of the inability to reach safety acceptance criteria before completion of all checks.
- Barrier monitoring capability is decreased if there is a loss or lack of reliable indicators. This assessment should include instrumentation operability concerns, readings from portable instrumentation and consideration of offsite monitoring results.
- Dominant accident sequences lead to degradation of all fission product barriers and likely entry to the EOPs. The Emergency Coordinator should be mindful of the Loss of AC power (Station Blackout) and ATWS EALs to assure timely emergency classification declarations.

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

Callaway Basis Reference(s):

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

Deleted: Plant-Specific¶

Deleted: Generic¶

Barrier: Fuel Clad

Category: E. Emergency Coordinator Judgment

Degradation Threat: Potential Loss

Threshold:

1. **Any** condition in the opinion of the Emergency Coordinator that indicates potential loss of the Fuel Clad barrier

Basis:

The Emergency Coordinator judgment threshold addresses any other factors relevant to determining if the Fuel Clad barrier is potentially lost. Such a determination should include imminent barrier degradation, barrier monitoring capability and dominant accident sequences.

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- Imminent barrier degradation exists if the degradation will likely occur within relatively short period of time based on a projection of current safety system performance. The term "imminent" refers to recognition of the inability to reach safety acceptance criteria before completion of all checks.
- Barrier monitoring capability is decreased if there is a loss or lack of reliable indicators. This assessment should include instrumentation operability concerns, readings from portable instrumentation and consideration of offsite monitoring results.
- Dominant accident sequences lead to degradation of all fission product barriers and likely entry to the EOPs. The Emergency Coordinator should be mindful of the Loss of AC power (Station Blackout) and ATWS EALs to assure timely emergency classification declarations.

This threshold addresses any other factors that are to be used by the Emergency Coordinator in determining whether the Fuel Clad barrier is potentially lost. The Emergency Coordinator should also consider whether or not to declare the barrier potentially lost in the event that barrier status cannot be monitored.

Deleted: Generic¶

Callaway Basis Reference(s):

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

Barrier: Reactor Coolant System

Category: A. RCS or SG Tube Leakage

Degradation Threat: Loss

Threshold:

1. An automatic or manual ECCS (SI) actuation required by **EITHER:**
    - UNISOLABLE RCS leakage
    - SG tube RUPTURE

Definition(s):

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

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

Basis:

ECCS (SI) actuation is caused by (ref. 1):

- Pressurizer low pressure < 1849 psig
- Steamline low pressure < 615 psig
- Containment high pressure > 3.5 psig

This threshold is based on an UNISOLABLE RCS leak of sufficient size to require an automatic or manual actuation of the Emergency Core Cooling System (ECCS). This condition clearly represents a loss of the RCS Barrier.

This threshold is applicable to unidentified and pressure boundary leakage, as well as identified leakage. It is also applicable to UNISOLABLE RCS leakage through an interfacing system. The mass loss may be into any location – inside containment, to the secondary-side (i.e., steam generator tube leakage) or outside of containment.

A steam generator with primary-to-secondary leakage of sufficient magnitude to require a safety injection is considered to be RUPTURED. If a RUPTURED steam generator is also FAULTED outside of containment, the declaration escalates to a Site Area Emergency since the Containment Barrier Loss threshold 1.A will also be met.

Callaway Basis Reference(s):

1. E-0 Reactor Trip or Safety Injection
2. E-3 Steam Generator Tube Rupture
3. NEI 99-01 RCS or SG Tube Leakage Reactor Coolant System Loss 1.A

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Barrier: Reactor Coolant System

Category: A. RCS or SG Tube Leakage

Degradation Threat: Potential Loss

Threshold:

1. Operation of a standby charging pump is required by **EITHER:**
   - UNISOLABLE RCS leakage
   - SG tube leakage

Definition(s):

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

Basis:

This threshold is based on the inability to maintain liquid inventory within the RCS by normal operation of the Chemical and Volume Control System (CVCS). The CVCS includes three charging pumps: one Normal Charging Pump with a flow capacity of 130 gpm, and two centrifugal charging pumps each with a flow capacity of 150 gpm (ref. 1). Approximately 12 gpm of charging flow bypasses the RCS due to leakage through the RCP seals; thus, the Normal Charging Pump can deliver 130 gpm – 12 gpm = 118 gpm (rounded to 120 gpm for readability) (ref. 2). A second charging pump being required is indicative of a substantial RCS leak.

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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.

Deleted: Generic¶

This threshold is applicable to unidentified and pressure boundary leakage, as well as identified leakage. It is also applicable to UNISOLABLE RCS leakage through an interfacing system. The mass loss may be into any location – inside containment, to the secondary-side (i.e., steam generator tube leakage) or outside of containment.

If a leaking steam generator is also FAULTED outside of containment, the declaration escalates to a Site Area Emergency since the Containment Barrier Loss threshold 1.A will also be met.

Callaway Basis Reference(s):

1. UFSAR Table 9.3-9
2. UFSAR Section 9.3.4 Chemical and Volume Control System
3. E-3 Steam Generator Tube Rupture
4. NEI 99-01 RCS or SG Tube Leakage Reactor Coolant System Potential Loss 1.A

Barrier: Reactor Coolant System

Category: A. RCS or SG Tube Leakage

Degradation Threat: Potential Loss

Threshold:

2. CSFST Integrity-RED Path conditions met

Definition(s):

None

Basis:

The "Potential Loss" threshold is defined by the CSFST Reactor Coolant Integrity - RED path. CSFST RCS Integrity - Red Path plant conditions and associated PTS Limit Curve A indicates an extreme challenge to the safety function when plant parameters are to the left of the limit curve following excessive RCS cooldown under pressure (ref. 1, 2).

This condition indicates an extreme challenge to the integrity of the RCS pressure boundary due to pressurized thermal shock – a transient that causes rapid RCS cooldown while the RCS is in Mode 3 or higher (i.e., hot and pressurized).

Deleted: Generic¶

Deleted: Generic¶

Callaway Basis Reference(s):

1. CSF-1 Critical Safety Function Status Trees Figure 4 Integrity and 4a Limit A Curve
2. FR-P.1 Response to Imminent Pressurized Thermal Shock Condition
3. NEI 99-01 RCS or SG Tube Leakage Reactor Coolant System Potential Loss 1.B

Barrier: Reactor Coolant System

Category: B. Inadequate Heat Removal

Degradation Threat: Loss

Threshold:

None

Barrier: Reactor Coolant System

Category: B. Inadequate Heat Removal

Degradation Threat: Potential Loss

Threshold:

1. CSFST Heat Sink-RED path conditions met

   AND

   Heat sink is required

Definition(s):

None

Basis:

In combination with FC Potential Loss B.2, meeting this threshold results in a Site Area Emergency.

Deleted: Plant-Specific¶

Critical Safety Function Status Tree (CSFST) Heat Sink-RED path indicates the ultimate heat sink function is under extreme challenge and that some fuel clad damage may potentially occur (ref. 1).

The CSFSTs are normally monitored using the 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 EOP. For example, FR-H.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 either RCS temperature is greater than 350ºF or RCS pressure is greater than 360 psig. 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).

Generic

This condition indicates an extreme challenge to the ability to remove RCS heat using the steam generators (i.e., loss of an effective secondary-side heat sink). This condition represents a potential loss of the RCS Barrier. In accordance with EOPs, there may be unusual accident conditions during which operators intentionally reduce the heat removal capability of the steam generators; during these conditions, classification using threshold is not warranted.

Meeting this threshold results in a Site Area Emergency because this threshold is identical to Fuel Clad Barrier Potential Loss threshold B.2; both will be met. This condition warrants a Site Area Emergency declaration because inadequate RCS heat removal may result in fuel heat-up sufficient to damage the cladding and increase RCS pressure to the point where mass will be lost from the system.

Callaway Basis Reference(s):

1. CSF-1 Critical Safety Function Status Trees Figure 3 Heat Sink
2. FR-H.1 Response to Loss of Secondary Heat Sink
3. NEI 99-01 Inadequate Heat Removal RCS Loss 2.B

Barrier: Reactor Coolant System

Category: C. CMT Radiation/ RCS Activity

Degradation Threat: Loss

Threshold:

1. Containment radiation > 6.40E+00 R/hr on GT-RE-59 (591) or GT-RE-60 (601)

Definition(s):

N/A

Basis:

Containment radiation monitor readings greater than 6.4 E+00 R/hr (ref. 1) indicate the release of reactor coolant to the Containment. The readings assume the instantaneous release and dispersal of the reactor coolant noble gas and iodine inventory associated with normal operating concentrations (i.e., within Technical Specifications) into the Containment atmosphere. Because of the very high fuel clad integrity, only small amounts of noble gases would be dissolved in the primary coolant.

Deleted: Plant-Specific¶

Monitors used for this fission product barrier loss threshold are the Containment High Range Radiation Monitors GT-RE-59 (Panel RM-11 channel 591) and GT-RE-60 (Panel RM-11 channel 601). The threshold value of 6.4 E+00 R/hr is the HI (YELLOW) alarm setpoint (ref. 2).

The radiation monitor reading corresponds to an instantaneous release of all reactor coolant mass into the containment, assuming that reactor coolant activity equals Technical Specification allowable limits. This value is lower than that specified for Fuel Clad Barrier Loss threshold C.1 since it indicates a loss of the RCS Barrier only.

Deleted: Generic¶

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

Callaway Basis Reference(s):

1. EPCI-9801 pg 27
2. OTA-SP-RM011 Radiation Monitor Control Panel RM-11
3. NEI 99-01 CMT Radiation / RCS Activity RCS Loss 3.A

Barrier: Reactor Coolant System

Category: C. CMT Radiation/ RCS Activity

Deleted: B

Degradation Threat: Potential Loss

Threshold:

None

Barrier: Reactor Coolant System

Category: D. CMT Integrity or Bypass

Degradation Threat: Loss

Threshold:

None

Barrier: Reactor Coolant System

Category: D. CMT Integrity or Bypass

Degradation Threat: Potential Loss

Threshold:

None

Barrier: Reactor Coolant System

Category: E. Emergency Coordinator Judgment

Degradation Threat: Loss

Threshold:

1. **Any** condition in the opinion of the Emergency Coordinator that indicates loss of the RCS barrier

Definition(s):

None

Basis:

The Emergency Coordinator judgment threshold addresses any other factors relevant to determining if the RCS barrier is lost. Such a determination should include imminent barrier degradation, barrier monitoring capability and dominant accident sequences.

Deleted: Plant-Specific¶

- Imminent barrier degradation exists if the degradation will likely occur within relatively short period of time based on a projection of current safety system performance. The term "imminent" refers to recognition of the inability to reach safety acceptance criteria before completion of all checks.
- Barrier monitoring capability is decreased if there is a loss or lack of reliable indicators. This assessment should include instrumentation operability concerns, readings from portable instrumentation and consideration of offsite monitoring results.
- Dominant accident sequences lead to degradation of all fission product barriers and likely entry to the EOPs. The Emergency Coordinator should be mindful of the Loss of AC power (Station Blackout) and ATWS EALs to assure timely emergency classification declarations.

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

Deleted: Generic¶

Callaway Basis Reference(s):

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

Barrier: Reactor Coolant System

Category: E. Emergency Coordinator Judgment

Degradation Threat: Potential Loss

Threshold:

1. **Any** condition in the opinion of the Emergency Coordinator that indicates potential loss of the RCS barrier

Definition(s):

None

Basis:

The Emergency Coordinator judgment threshold addresses any other factors relevant to determining if the RCS barrier is potentially lost. Such a determination should include imminent barrier degradation, barrier monitoring capability and dominant accident sequences.

- Imminent barrier degradation exists if the degradation will likely occur within relatively short period of time based on a projection of current safety system performance. The term "imminent" refers to recognition of the inability to reach safety acceptance criteria before completion of all checks.
- Barrier monitoring capability is decreased if there is a loss or lack of reliable indicators. This assessment should include instrumentation operability concerns, readings from portable instrumentation and consideration of offsite monitoring results.
- Dominant accident sequences lead to degradation of all fission product barriers and likely entry to the EOPs. The Emergency Coordinator should be mindful of the Loss of AC power (Station Blackout) and ATWS EALs to assure timely emergency classification declarations.

This threshold addresses any other factors that may be used by the Emergency Coordinator in determining whether the RCS Barrier is potentially lost. The Emergency Coordinator should also consider whether or not to declare the barrier potentially lost in the event that barrier status cannot be monitored.

Callaway Basis Reference(s):

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

Deleted: Plant-Specific¶

Deleted: Generic¶

Deleted: Director

Barrier: Containment

Category: A. RCS or SG Tube Leakage

Degradation Threat: Loss

Threshold:

1. A leaking or RUPTURED SG is FAULTED outside of containment

Definition(s):

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

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

Basis:

This threshold addresses a leaking or RUPTURED Steam Generator (SG) that is also FAULTED outside of containment. The condition of the SG, whether leaking or RUPTURED, is determined in accordance with the thresholds for RCS Barrier Potential Loss A.1 and Loss A.1, respectively. This condition represents a bypass of the containment barrier.

Deleted: Plant-Specific¶
None.¶
Generic¶

FAULTED is a defined term within the NEI 99-01 methodology; this determination is not necessarily dependent upon entry into, or diagnostic steps within, an EOP. For example, if the pressure in a steam generator is decreasing uncontrollably (part of the FAULTED definition) and the FAULTED steam generator isolation procedure is not entered because EOP user rules are dictating implementation of another procedure to address a higher priority condition, the steam generator is still considered FAULTED for emergency classification purposes.

The FAULTED criterion establishes an appropriate lower bound on the size of a steam release that may require an emergency classification. Steam releases of this size are readily observable with normal Control Room indications. The lower bound for this aspect of the containment barrier is analogous to the lower bound criteria specified in IC SU4 for the fuel clad barrier (i.e., RCS activity values) and IC SU5 for the RCS barrier (i.e., RCS leak rate values).

This threshold also applies to prolonged steam releases necessitated by operational considerations such as the forced steaming of a leaking or RUPTURED steam generator directly to atmosphere to cooldown the plant, or to drive an auxiliary (emergency) feed water pump. These types of conditions will result in a significant and sustained release of radioactive steam to the environment (and are thus similar to a FAULTED condition). The inability to isolate the steam flow without an adverse effect on plant cooldown meets the intent of a loss of containment.

Steam releases associated with the expected operation of a SG power operated relief valve or safety relief valve do not meet the intent of this threshold. Such releases may occur intermittently for a short period of time following a reactor trip as operators process through emergency operating procedures to bring the plant to a stable condition and prepare to initiate a plant cooldown. Steam releases associated with the unexpected operation of a valve (e.g., a stuck-open safety valve) do meet this threshold.

Following an SG tube leak or rupture, there may be minor radiological releases through a secondary-side system component (e.g., air ejectors, glad seal exhausters, valve packing, etc.). These types of releases do not constitute a loss or potential loss of containment but should be evaluated using the Recognition Category R ICs.

The ECLs resulting from primary-to-secondary leakage, with or without a steam release from the FAULTED SG, are summarized below.

	Affected SG is FAULTED Outside of Containment?	
P-to-S Leak Rate	**Yes**	**No**
Less than or equal to 25 gpm	No classification	No classification
Greater than 25 gpm	Unusual Event per SU5.1	Unusual Event per SU5.1
Requires operation of a standby charging (makeup) pump (*RCS Barrier Potential Loss*)	Site Area Emergency per FS1.1	Alert per FA1.1
Requires an automatic or manual ECCS (SI) actuation (*RCS Barrier Loss*)	Site Area Emergency per FS1.1	Alert per FA1.1

There is no Potential Loss threshold associated with RCS or SG Tube Leakage.

Callaway Basis Reference(s):

1. E-2 Faulted Steam Generator Isolation
2. E-3 Steam Generator Tube Rupture
3. NEI 99-01 RCS or SG Tube Leakage Containment Loss 1.A

Barrier: Containment

Category: A. RCS or SG Tube Leakage

Degradation Threat: Potential Loss

Threshold:

None

Barrier: Containment

Category: B. Inadequate heat Removal

Degradation Threat: Loss

Threshold:

None

Barrier: Containment

Category: B. Inadequate heat Removal

Degradation Threat: Potential Loss

Threshold:

1. CSFST Core Cooling-RED path conditions met **AND** Restoration procedures **not** effective within 15 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.

Definition(s):

None

Basis:

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 SPDS display on the Plant Computer (ref. 1).

Deleted: Plant-Specific¶

The function restoration procedures are those emergency operating procedures that address the recovery of the core cooling critical safety functions. The procedure is considered effective if the temperature is decreasing or if the vessel water level is increasing (ref. 1, 2, 3).

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 (ref. 1), Fuel Clad barrier is also lost.

This threshold addresses any other factors that may be used by the Emergency Coordinator in determining whether the RCS Barrier is potentially lost. The Emergency Director should also consider whether or not to declare the barrier potentially lost in the event that barrier status cannot be monitored.

Deleted: Generic¶

Callaway Basis Reference(s):

1. CSF-1 Critical Safety Function Status Trees Figure 2 Core Cooling
2. FR-C.1 Response to Inadequate Core Cooling
3. FR-C.2 Response to Degraded Core Cooling
4. NEI 99-01 Inadequate Heat Removal Containment Potential Loss 2.A

Barrier: Containment

Category: C. CMT Radiation/RCS Activity

Degradation Threat: Loss

Threshold:

None

Barrier: Containment

Category: C. CMT Radiation/RCS Activity

Degradation Threat: Potential Loss

Threshold:

1. Containment radiation > 8.06E+04 R/hr on GT-RE-59 (591) or GT-RE-60 (601)

Definition(s):

None

Basis:

Containment radiation monitor readings greater than 8.06 E+04 R/hr (ref. 1) indicate significant fuel damage well in excess of that required for loss of the RCS barrier and the Fuel Clad barrier.

Deleted: Plant-Specific¶

The readings are higher than that specified for Fuel Clad barrier Loss C.1 and RCS barrier Loss C.1. Containment radiation readings at or above the Containment barrier Potential Loss threshold, therefore, signify a loss of two fission product barriers and Potential Loss of a third, indicating the need to upgrade the emergency classification to a General Emergency.

Monitors used for this fission product barrier loss threshold are the Containment High Range Radiation Monitors GT-RE-59 (Panel RM-11 channel 591) and GT-RE-60 (Panel RM-11 channel 601) (ref. 2).

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.

Deleted: Generic¶

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.

Callaway Basis Reference(s):

1. EPCI-9801 pg 27
2. OTA-SP-RM011 Radiation Monitor Control Panel RM-11
3. NEI 99-01 CMT Radiation / RCS Activity Containment Potential Loss 3.A

Barrier: Containment

Category: D. CMT Integrity or Bypass

Degradation Threat: Loss

Threshold:

1. Containment isolation is required

   AND EITHER:

   - Containment integrity has been lost based on Emergency Coordinator judgment
   - UNISOLABLE pathway from containment to the environment exists

Definition(s):

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

Basis:

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.

Deleted: Plant-Specific¶
None¶
Generic¶

First Threshold – Containment integrity has been lost, i.e., the actual containment atmospheric leak rate likely exceeds that associated with allowable leakage (or sometimes referred to as design leakage). Following the release of RCS mass into containment, containment pressure will fluctuate based on a variety of factors; a loss of containment integrity condition may (or may not) be accompanied by a noticeable drop in containment pressure. Recognizing the inherent difficulties in determining a containment leak rate during accident conditions, it is expected that the Emergency Coordinator will assess this threshold using judgment, and with due consideration given to current plant conditions, and available operational and radiological data (e.g., containment pressure, readings on radiation monitors outside containment, operating status of containment pressure control equipment, etc.).

Refer to the middle piping run of Figure 1. Two simplified examples are provided. One is leakage from a penetration and the other is leakage from an in-service system valve. Depending upon radiation monitor locations and sensitivities, the leakage could be detected by any of the four monitors depicted in the figure.

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.

Callaway Basis Reference(s):

1. NEI 99-01 CMT Integrity or Bypass Containment Loss 4.A

Barrier: Containment

Category: D. CMT Integrity or Bypass

Degradation Threat: Loss

Threshold:

2. Indications of RCS leakage outside of containment

Definition(s):

None

Basis:

ECA-1.2 LOCA Outside Containment (ref. 1) provides instructions to identify and isolate a LOCA outside of the containment. Potential RCS leak pathways outside containment include (ref. 1, 2):

- Residual Heat Removal
- Safety Injection
- Chemical & Volume Control
- RCP seals
- PZR/RCS Loop sample lines

Containment sump, temperature, pressure and/or radiation levels will increase if reactor coolant mass is leaking into the containment. If these parameters have not increased, then the reactor coolant mass may be leaking outside of containment (i.e., a containment bypass sequence). Increases in sump, temperature, pressure, flow and/or radiation level readings outside of the containment may indicate that the RCS mass is being lost outside of containment.

Unexpected elevated readings and alarms on radiation monitors with detectors outside containment should be corroborated with other available indications to confirm that the source is a loss of RCS mass outside of containment. If the fuel clad barrier has not been lost, radiation monitor readings outside of containment may not increase significantly; however, other unexpected changes in sump levels, area temperatures or pressures, flow rates, etc. should be sufficient to determine if RCS mass is being lost outside of the containment.

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.

Callaway Basis Reference(s):

1. ECA-1.2 LOCA Outside Containment

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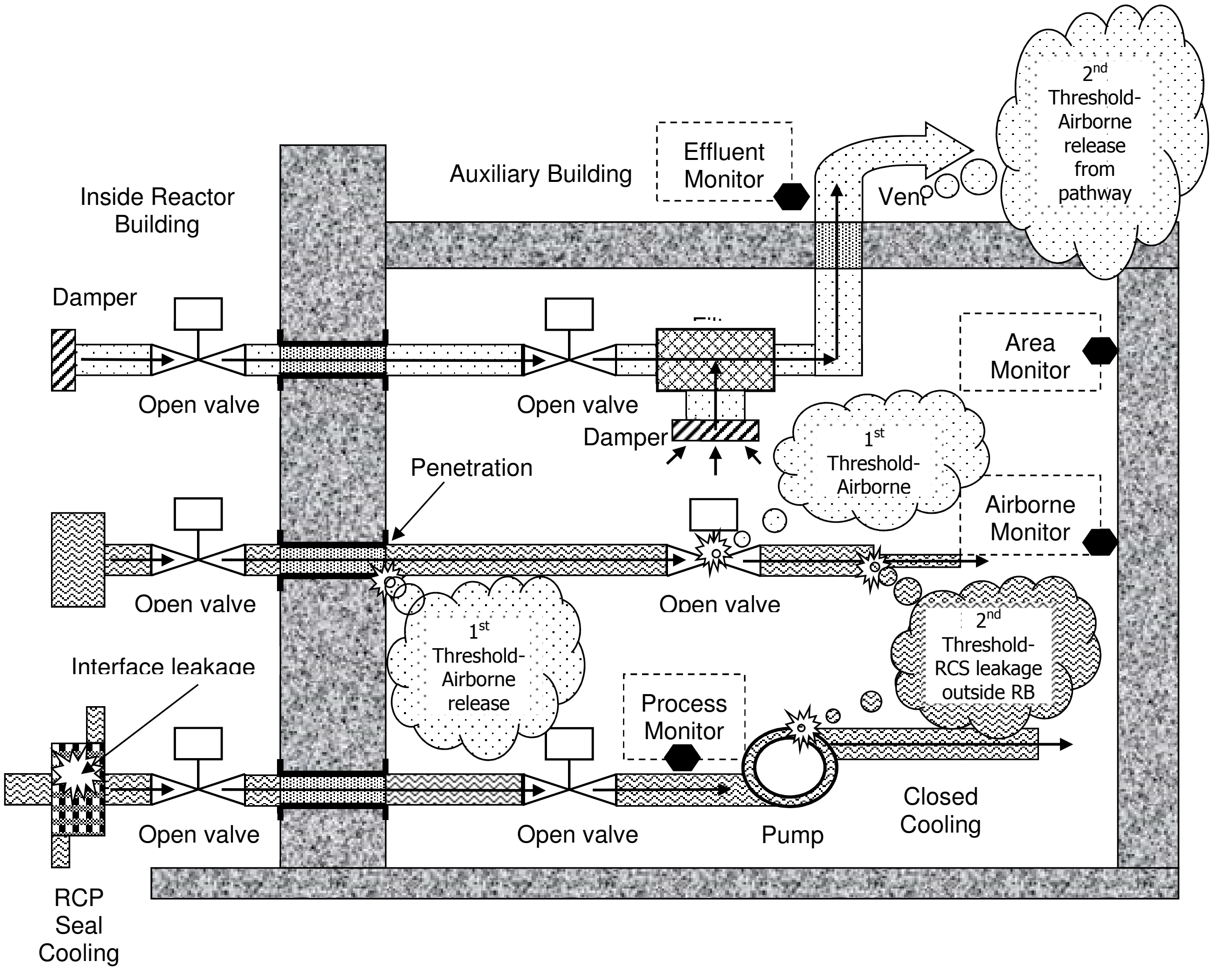
Deleted: Generic¶

2. E-1 Loss of Reactor or Secondary Coolant
3. NEI 99-01 CMT Integrity or Bypass Containment Loss

Figure 1: Containment Integrity or Bypass Examples



ATTACHMENT 2
Fission Product Barrier Loss/Potential Loss Matrix and Bases

Barrier: Containment

Category: D. CMT Integrity or Bypass

Degradation Threat: Potential Loss

Threshold:

1.	CSFST Containment-RED path conditions met

Definition(s):

None

Basis:

Critical Safety Function Status Tree (CSFST) Containment-RED path is entered if containment pressure is greater than or equal to 48 psig and represents an extreme challenge to safety function. The CSFSTs are normally monitored using the SPDS display on the Plant Computer (ref. 1).

Deleted: Plant-Specific¶

48 psig is the containment pressure that is expected to occur following a design basis Loss of Coolant Accident (LOCA) (ref. 2) and is the pressure used to define CSFST Containment Red Path conditions.

If containment pressure exceeds the pressure that is expected to occur following a design basis Loss of Coolant Accident (LOCA), 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.

Deleted: Generic¶

Callaway Basis Reference(s):

1. CSF-1 Critical Safety Function Status Trees Containment Figure 5
2. Calc No. 392.2 XX-95 Callaway Containment Parameters EOP Action Values, Setpoint ID T.03
3. NEI 99-01 CMT Integrity or Bypass Containment Potential Loss 4.A

Barrier: Containment

Category: D. CMT Integrity or Bypass

Degradation Threat: Potential Loss

Threshold:

2. Containment hydrogen concentration ≥ 4%

Definition(s):

None

Basis:

Following a design basis accident, hydrogen gas may be generated inside the containment by reactions such as zirconium metal with water, corrosion of materials of construction and radiolysis of aqueous solution in the core and sump. (ref. 1).

Deleted: Plant-Specific¶

Callaway is equipped with a Hydrogen Control System (HCS) which serves to limit or reduce combustible gas concentrations in the Containment. The HCS is an engineered safety feature with redundant hydrogen recombiners, hydrogen mixing system, hydrogen monitoring subsystem, and a backup hydrogen purge subsystem. The HCS is designed to maintain the Containment hydrogen concentration below 4% by volume (ref. 1).

HCS operation is prescribed by EOPs if Containment hydrogen concentration should reach 0.5% by volume (ref. 4). If the Potential Loss threshold is reached or exceeded, the primary means of controlling Containment hydrogen concentration must have failed to perform its design function or has otherwise been inadequate in mitigating the hydrogen generation rate. For either case, continued hydrogen production may yield a flammable hydrogen concentration and a consequent threat to Containment integrity.

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.

Two Containment hydrogen monitors (GS AI-10 and GS AI-19) with a range of 0% to 10% provide indication on Control Room Panel RL020 and ERFIS (ref. 3). The hydrogen monitors require a 2 hour warmup period when starting from the OFF position and 15 minutes when starting from STANDBY (ref. 4, 5). If an actual hydrogen concentration measurement is unavailable, CA-3 (ref. 6) may be used to estimate the Containment atmosphere hydrogen concentration.

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.

Deleted: Generic¶

Callaway Basis Reference(s):

1. UFSAR Section 6.2 Containment Systems

2. FR-Z.4 Response to High Containment Hydrogen Concentration
3. FSAR Table 7A-3 (Sheet 32 Data Sheet 6.4)
4. OTN-GS-00001 Containment Hydrogen Control System
5. Calc No. 392.2 XX-95 Callaway Containment Parameters EOP Action Values, Setpoint ID T101 & T102
6. CA-3 Hydrogen Flammability in Containment
7. NEI 99-01 CMT Integrity or Bypass Containment Potential Loss 4.B

Barrier: Containment

Category: D. CMT Integrity or Bypass

Degradation Threat: Potential Loss

Threshold:

3.	Containment pressure > 27 psig with < one full train of containment depressurization equipment operating per design for ≥ 15 min. (Note 1, 9)

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

Note 9: One Containment Spray System train and one Containment Cooling System train comprise one full train of depressurization equipment.

Definition(s):

None

Basis:

The Containment Spray System consists of two separate trains of equal capacity, each capable of meeting the design bases requirement. Each train includes a containment spray pump, spray headers, nozzles, valves, and piping. The refueling water storage tank (RWST) supplies borated water to the Containment Spray System during the injection phase of operation. In the recirculation mode of operation, Containment Spray pump suction is transferred from the RWST to the Containment sumps (ref. 2).

Deleted: Plant-Specific¶

The Containment Cooling System consists of two trains of Containment cooling, each of sufficient capacity to supply 100% of the design cooling requirement. Each train of two fan units is supplied with cooling water from a separate train of essential service water (ESW). Air is drawn into the coolers through the fan and discharged to the steam generator compartments, pressurizer compartment, and instrument tunnel, and outside the secondary shield in the lower areas of containment. During normal operation, all four fan units may be operating. In post accident operation following an actuation signal, the Containment Cooling System fans are designed to start automatically in slow speed if not already running (ref. 3).

The Containment pressure setpoint (27 psig, ref. 4, 5, 6) is the pressure at which the equipment should actuate and begin performing its function. The design basis accident analyses and evaluations assume the loss of one Containment Spray System train and one Containment Cooling System train (ref. 7). Consistent with the design requirement, "one full train of depressurization equipment" is therefore defined to be the availability of one train of each system. If less than this equipment is operating and Containment pressure is above the actuation setpoint, the threshold is met.

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

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condenser fans, etc., but not including containment venting strategies) are either lost or performing in a degraded manner.

Callaway Basis Reference(s):

1. FSAR Section 6.2.2
2. FSAR Section 6.2.2.1.2.1
3. FSAR Section 6.2.2.2.2
4. CSF-1 Critical Safety Function Status Trees (CSFST) Figure 5, Containment
5. FR-Z.1 Response to High Containment Pressure
6. Technical Specifications Table 3.3.2-1
7. Technical Specifications B3.6.6
8. NEI 99-01 CMT Integrity or Bypass Containment Potential Loss 4.C

Barrier: Containment

Category: E. Emergency Coordinator Judgment

Degradation Threat: Loss

Threshold:

1. **Any** condition in the opinion of the Emergency Coordinator that indicates loss of the Containment barrier

Definition(s):

None

Basis:

The Emergency Coordinator judgment threshold addresses any other factors relevant to determining if the Primary Containment barrier is lost. Such a determination should include imminent barrier degradation, barrier monitoring capability and dominant accident sequences.

Deleted: Plant-Specific¶

- Imminent barrier degradation exists if the degradation will likely occur within relatively short period of time based on a projection of current safety system performance. The term “imminent” refers to recognition of the inability to reach safety acceptance criteria before completion of all checks.
- Barrier monitoring capability is decreased if there is a loss or lack of reliable indicators. This assessment should include instrumentation operability concerns, readings from portable instrumentation and consideration of offsite monitoring results.
- Dominant accident sequences lead to degradation of all fission product barriers and likely entry to the EOPs. The Emergency Coordinator should be mindful of the Loss of AC power (Station Blackout) and ATWS EALs to assure timely emergency classification declarations.

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

Deleted: Generic¶

Callaway Basis Reference(s):

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

ATTACHMENT 2
Fission Product Barrier Loss/Potential Loss Matrix and Bases

Barrier: Containment

Category: E. Emergency Coordinator Judgment

Degradation Threat: Potential Loss

Threshold:

1. **Any** condition in the opinion of the Emergency Coordinator that indicates potential loss of the Containment barrier

Definition(s):

None

Basis:

The Emergency Coordinator judgment threshold addresses any other factors relevant to determining if the Primary Containment barrier is potentially lost. Such a determination should include imminent barrier degradation, barrier monitoring capability and dominant accident sequences.

Deleted: Plant-Specific¶

- Imminent barrier degradation exists if the degradation will likely occur within relatively short period of time based on a projection of current safety system performance. The term "imminent" refers to recognition of the inability to reach safety acceptance criteria before completion of all checks.
- Barrier monitoring capability is decreased if there is a loss or lack of reliable indicators. This assessment should include instrumentation operability concerns, readings from portable instrumentation and consideration of offsite monitoring results.
- Dominant accident sequences lead to degradation of all fission product barriers and likely entry to the EOPs. The Emergency Coordinator should be mindful of the Loss of AC power (Station Blackout) and ATWS EALs to assure timely emergency classification declarations.

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

Deleted: Generic¶

Callaway Basis Reference(s):

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

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SU4.2	SU3	2

Attachment 4 to

ULNRC-06230

Corrected Callaway NEI 99-01 Revision 6 EAL Wall Charts (Information Only)

3 Pages

	GENERAL EMERGENCY	SITE AREA EMERGENCY	ALERT	UNUSUAL EVENT

R — Abnormal Rad Levels / Rad Effluent

1 — Rad Effluent

General Emergency — Release of gaseous radioactivity resulting in offsite dose greater than 1,000 mrem TEDE or 5,000 mrem thyroid CDE

RG1.1 | 1 | 2 | 3 | 4 | 5 | 6 | DEF |
Reading on **any** Table R-1 effluent radiation monitor > column "GE" for ≥ 15 min. (Notes 1, 2, 3, 4)

RG1.2 | 1 | 2 | 3 | 4 | 5 | 6 | DEF |
Dose assessment using actual meteorology indicates doses > 1000 mrem TEDE or 5000 mrem thyroid CDE at or beyond the SITE BOUNDARY (Note 4)

RG1.3 | 1 | 2 | 3 | 4 | 5 | 6 | DEF |
Field survey results indicate **EITHER** of the following at or beyond the SITE BOUNDARY:
- Closed window dose rates > 1000 mR/hr expected to continue for ≥ 60 min.
- Analyses of field survey samples indicate thyroid CDE > 5000 mrem for 60 min. of inhalation.

(Notes 1, 2)

Site Area Emergency — Release of gaseous radioactivity resulting in offsite dose greater than 100 mrem TEDE or 500 mrem thyroid CDE

RS1.1 | 1 | 2 | 3 | 4 | 5 | 6 | DEF |
Reading on **any** Table R-1 effluent radiation monitor > column "SAE" for ≥ 15 min. (Notes 1, 2, 3, 4)

RS1.2 | 1 | 2 | 3 | 4 | 5 | 6 | DEF |
Dose assessment using actual meteorology indicates doses > 100 mrem TEDE or 500 mrem thyroid CDE at or beyond the SITE BOUNDARY (Note 4)

RS1.3 | 1 | 2 | 3 | 4 | 5 | 6 | DEF |
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)

Alert — Release of gaseous or liquid radioactivity resulting in offsite dose greater than 10 mrem TEDE or 50 mrem thyroid CDE

RA1.1 | 1 | 2 | 3 | 4 | 5 | 6 | DEF |
Reading on **any** Table R-1 effluent radiation monitor > column "ALERT" for ≥ 15 min. (Notes 1, 2, 3, 4)

RA1.2 | 1 | 2 | 3 | 4 | 5 | 6 | DEF |
Dose assessment using actual meteorology indicates doses > 10 mrem TEDE or 50 mrem thyroid CDE at or beyond the SITE BOUNDARY (Note 4)

RA1.3 | 1 | 2 | 3 | 4 | 5 | 6 | DEF |
Analysis of a liquid effluent sample indicates a concentration or release rate that would result in doses > 10 mrem TEDE or 50 mrem thyroid CDE at or beyond the SITE BOUNDARY for 60 min. of exposure (Notes 1, 2)

RA1.4 | 1 | 2 | 3 | 4 | 5 | 6 | DEF |
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)

Unusual Event — Release of gaseous or liquid radioactivity greater than 2 times the ODCM limits for 60 minutes or longer

RU1.1 | 1 | 2 | 3 | 4 | 5 | 6 | DEF |
Reading on **any** Table R-1 effluent radiation monitor > column "UE" for ≥ 60 min. (Notes 1, 2, 3)

RU1.2 | 1 | 2 | 3 | 4 | 5 | 6 | DEF |
Sample analyses for a gaseous or liquid release indicates a concentration or release rate > 2 x ODCM limits for > 60 min. (Notes 1, 2)

2 — Irradiated Fuel Event

General Emergency — Spent fuel pool level cannot be restored to at least the top of the fuel racks for 60 minutes or longer

RG2.1 | 1 | 2 | 3 | 4 | 5 | 6 | DEF |
Spent fuel pool level cannot be restored to at least 2022 ft. 1.25 in. (Level 3) for ≥ 60 min. (Note 1)

Site Area Emergency — Spent fuel pool level at the top of the fuel racks

RS2.1 | 1 | 2 | 3 | 4 | 5 | 6 | DEF |
Lowering of spent fuel pool level to 2022 ft. 1.25 in. (Level 3)

Alert — Significant lowering of water level above, or damage to, irradiated fuel

RA2.1 | 1 | 2 | 3 | 4 | 5 | 6 | DEF |
Uncovery of irradiated fuel in the REFUELING PATHWAY

RA2.2 | 1 | 2 | 3 | 4 | 5 | 6 | DEF |
Damage to irradiated fuel resulting in a release of radioactivity from the fuel as indicated by **any** of the following:
- Hi-Hi Alarm on Fuel Building exhaust monitors (GG-RE-27 or 28)
- Manipulator crane radiation monitor (SD-RE-41) >100 mR/hr
- Fuel Pool Bridge Crane OR Spent Fuel Pool Area radiation monitor (SD-RE-37 or 38) > 30 mR/hr

RA2.3 | 1 | 2 | 3 | 4 | 5 | 6 | DEF |
Lowering of spent fuel pool level to 2031 ft. 1.25 in. (Level 2)

Unusual Event — Unplanned loss of water level above irradiated fuel

RU2.1 | 1 | 2 | 3 | 4 | 5 | 6 | DEF |
UNPLANNED water level drop in the REFUELING PATHWAY as indicated by low water level alarm or indication (EC LI-0039A, EC LI-0039B, local observation of SFP level)
AND
UNPLANNED rise in corresponding area radiation levels as indicated by **any** Table R-2 radiation monitors

Table R-1 Effluent Monitor Classification Thresholds

	Release Point	Monitor	GE	SAE	Alert	UE
Gaseous	Unit Vent	GT-RE-21B	6.59E+7 μCi/sec	6.59E+6 μCi/sec	6.59E+5 μCi/sec	2 X Hi-Hi alarm
Gaseous	ASD Monitors (A/B/C/D)	AB-RE-111/ 112/113/114	12 mR/hr	1.2 mR/hr	-----	-----
Gaseous	TD AFW Steam Discharge	FC-RE-385	163 mR/hr	16.3 mR/hr	1.6 mR/hr	-----
Gaseous	Radwaste Bldg Vent	GH-RE-10B	-----	-----	-----	2 X Hi-Hi alarm
Liquid	Liquid Radwaste Discharge	HB-RE-18	-----	-----	-----	2 X Hi-Hi alarm

3 — Area Radiation Levels

General Emergency: None

Site Area Emergency:

Table R-2 Fuel Building & Containment Area Radiation Monitors

Fuel Building:
- SD-RE-34, Cask Handle Area Radiation
- SD-RE-35, New Fuel Storage Area Radiation
- SD-RE-36, New Fuel Storage Area Radiation
- SD-RE-37, Fuel Pool Bridge Crane Radiation
- SD-RE-38, Spent Fuel Pool Area Radiation

Containment:
- SD-RE-40, Personnel Access Hatch Area
- SD-RE-41, Manipulator Crane Radiation Monitor
- SD-RE-42, Containment Building Radiation
- GT-RE-59 Containment High Area Radiation Monitor
- GT-RE-60 Containment High Area Radiation Monitor

Alert — Radiation levels that IMPEDE access to equipment necessary for normal plant operations, cooldown or shutdown

RA3.1 | 1 | 2 | 3 | 4 | 5 | 6 | DEF |
Dose rate > 15 mR/hr in **EITHER** of the following areas:
- Control Room (SD-RE-33)
- Central Alarm Station (by survey)

E — ISFSI

1 — Confinement Boundary

General Emergency: None

Alert: None

Unusual Event — Damage to a loaded cask CONFINEMENT BOUNDARY

EU1.1 | 1 | 2 | 3 | 4 | 5 | 6 | DEF |
Damage to a loaded cask CONFINEMENT BOUNDARY as indicated by an on-contact radiation reading > **EITHER** of the following:
- 60 mrem/hr (gamma + neutron) on the top of the closure lid of the overpack
- 7,000 mrem/hr (gamma + neutron) on the side of the transfer cask

H — Hazards

1 — Security

General Emergency — HOSTILE ACTION resulting in loss of physical control of the facility

HG1.1 | 1 | 2 | 3 | 4 | 5 | 6 | DEF |
A HOSTILE ACTION is occurring or has occurred within the PROTECTED AREA as reported by the Security Shift Supervisor
AND EITHER of the following has occurred:
- **Any** of the following safety functions cannot be controlled or maintained
  - Reactivity control
  - Core cooling
  - RCS heat removal

  OR
- Damage to spent fuel has occurred or is IMMINENT

Site Area Emergency — HOSTILE ACTION within the PROTECTED AREA

HS1.1 | 1 | 2 | 3 | 4 | 5 | 6 | DEF |
A HOSTILE ACTION is occurring or has occurred within the PROTECTED AREA as reported by the Security Shift Supervisor

Alert — HOSTILE ACTION within the OWNER CONTROLLED AREA or airborne attack threat within 30 minutes

HA1.1 | 1 | 2 | 3 | 4 | 5 | 6 | DEF |
A HOSTILE ACTION is occurring or has occurred within the OWNER CONTROLLED AREA as reported by the Security Shift Supervisor
OR
A validated notification from NRC of an aircraft attack threat within 30 min. of the site

Unusual Event — Confirmed SECURITY CONDITION or threat

HU1.1 | 1 | 2 | 3 | 4 | 5 | 6 | DEF |
A SECURITY CONDITION that does **not** involve a HOSTILE ACTION as reported by Security Shift Supervisor
OR
Notification of a credible security threat directed at the site
OR
A validated notification from the NRC providing information of an aircraft threat

2 — Seismic Event

General Emergency: None

Site Area Emergency: None

Alert: None

Unusual Event — Seismic event greater than OBE level

HU2.1 | 1 | 2 | 3 | 4 | 5 | 6 | DEF |
Seismic event > OBE as indicated by Seismic Activity, Annunciator 98D

3 — Natural or Tech. Hazard

Site Area Emergency: None

Alert: None

Unusual Event — Hazardous event

HU3.1 | 1 | 2 | 3 | 4 | 5 | 6 | DEF |
A tornado strike within the PROTECTED AREA

HU3.2 | 1 | 2 | 3 | 4 | 5 | 6 | DEF |
Internal room or area FLOODING of a magnitude sufficient to require manual or automatic electrical isolation of a SAFETY SYSTEM component needed for the current operating mode

HU3.3 | 1 | 2 | 3 | 4 | 5 | 6 | DEF |
Movement of personnel within the PROTECTED AREA is IMPEDED due to an offsite event involving hazardous materials (e.g., an offsite chemical spill or toxic gas release)

HU3.4 | 1 | 2 | 3 | 4 | 5 | 6 | DEF |
A hazardous event that results in on-site conditions sufficient to prohibit the plant staff from accessing the site via personal vehicles (Note 7)

Notes

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

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

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

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

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

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

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

Note 8: A manual trip action is any operator action, or set of actions, which causes the control rods to be rapidly inserted into the core, and **does not** include manually driving in control rods or implementation of boron injection strategies

Note 9: One Containment Spray System train and one Containment Cooling System train comprise one full train of depressurization equipment

Note 10: Begin monitoring hot condition EALs concurrently for any new event or condition not related to the loss of decay heat removal

4 — Fire

Site Area Emergency: None

Table H-1 Fire Areas
- Area 5
- Containment
- Aux Feed Pump Rooms
- Auxiliary Building
- Diesel Generator Building
- UHS Cooling Tower
- ESW Pumphouse
- Control Building/ Communications Corridor
- RWST
- Fuel Building

Unusual Event — FIRE potentially degrading the level of safety of the plant

HU4.1 | 1 | 2 | 3 | 4 | 5 | 6 | DEF |
A FIRE is **not** extinguished within 15 min. of **any** of the following FIRE detection indications (Note 1):
- Report from the field (i.e., visual observation)
- Receipt of multiple (more than 1) fire alarms or indications
- Field verification of a single fire alarm

AND
The FIRE is located within **any** Table H-1 area

HU4.2 | 1 | 2 | 3 | 4 | 5 | 6 | DEF |
Receipt of a single fire alarm (i.e., **no** other indications of a FIRE)
AND
The fire alarm is indicating a FIRE within **any** Table H-1 area
AND
The existence of a FIRE is **not** verified within 30 min. of alarm receipt (Note 1)

HU4.3 | 1 | 2 | 3 | 4 | 5 | 6 | DEF |
A FIRE within the plant PROTECTED AREA **not** extinguished within 60 min. of the initial report, alarm or indication (Note 1)

HU4.4 | 1 | 2 | 3 | 4 | 5 | 6 | DEF |
A FIRE within the plant PROTECTED AREA that requires firefighting support by an offsite fire response agency to extinguish

5 — Control Room Evacuation

General Emergency: None

Site Area Emergency — Inability to control a key safety function from outside the Control Room

HS5.1 | 1 | 2 | 3 | 4 | 5 | 6 | |
An event has resulted in plant control being transferred from the Control Room to the Auxiliary Shutdown Panel (ASP)
AND
Control of **any** of the following key safety functions is **not** reestablished within 15 min. (Note 1):
- Reactivity control (Mode 1, 2 and 3 **only**)
- Core cooling
- RCS heat removal

Alert — Control Room evacuation resulting in transfer of plant control to alternate locations

HA5.1 | 1 | 2 | 3 | 4 | 5 | 6 | DEF |
An event has resulted in plant control being transferred from the Control Room to the Auxiliary Shutdown Panel (ASP)

Unusual Event: None

6 — Judgment

General Emergency — Other conditions existing that in the judgment of the Emergency Coordinator warrant declaration of General Emergency

HG6.1 | 1 | 2 | 3 | 4 | 5 | 6 | DEF |
Other conditions exist which in the judgment of the Emergency Coordinator indicate that events are in progress or have occurred which involve actual or IMMINENT substantial core degradation or melting with potential for loss of containment integrity or HOSTILE ACTION that results in an actual loss of physical control of the facility. Releases can be reasonably expected to exceed EPA Protective Action Guideline exposure levels offsite for more than the immediate site area.

Site Area Emergency — Other conditions existing that in the judgment of the Emergency Coordinator warrant declaration of Site Area Emergency

HS6.1 | 1 | 2 | 3 | 4 | 5 | 6 | DEF |
Other conditions exist which in the judgment of the Emergency Coordinator indicate that events are in progress or have occurred which involve actual or likely major failures of plant functions needed for protection of the public or HOSTILE ACTION that results in intentional damage or malicious acts, (1) toward site personnel or equipment that could lead to the likely failure of or, (2) that prevent effective access to equipment needed for the protection of the public. Any releases are not expected to result in exposure levels which exceed EPA Protective Action Guideline exposure levels beyond the SITE BOUNDARY.

Alert — Other conditions existing that in the judgment of the Emergency Coordinator warrant declaration of an Alert

HA6.1 | 1 | 2 | 3 | 4 | 5 | 6 | DEF |
Other conditions exist which, in the judgment of the Emergency Coordinator, indicate that events are in progress or have occurred which involve an actual or potential substantial degradation of the level of safety of the plant or a security event that involves probable life threatening risk to site personnel or damage to site equipment because of HOSTILE ACTION. Any releases are expected to be limited to small fractions of the EPA Protective Action Guideline exposure levels.

Unusual Event — Other conditions existing that in the judgment of the Emergency Coordinator warrant declaration of a UE

HU6.1 | 1 | 2 | 3 | 4 | 5 | 6 | DEF |
Other conditions exist which in the judgment of the Emergency Coordinator indicate that events are in progress or have occurred which indicate a potential degradation of the level of safety of the plant or indicate a security threat to facility protection has been initiated. No releases of radioactive material requiring offsite response or monitoring are expected unless further degradation of SAFETY SYSTEMS occurs

Modes:

1	2	3	4	5	6	DEF
Power Operation	Startup	Hot Standby	Hot Shutdown	Cold Shutdown	Refueling	Defueled

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ALL CONDITIONS

Prepared for Ameren by: Operations Support Services, Inc. – 7/2/15

	GENERAL EMERGENCY	SITE AREA EMERGENCY	ALERT	UNUSUAL EVENT

S System Malfunct.

1 Loss of Emergency AC Power

GENERAL EMERGENCY

Prolonged loss of **all** offsite and **all** onsite AC power to emergency buses

SG1.1 | 1 | 2 | 3 | 4 |

Loss of **all** offsite and **all** onsite AC power capability, Table S-1, to emergency 4.16KV buses NB01 and NB02

AND EITHER:

- Restoration of at least one emergency bus in < 4 hours is **not** likely (Note 1)
- CSFST Core Cooling-**RED** Path conditions met

SITE AREA EMERGENCY

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

SS1.1 | 1 | 2 | 3 | 4 |

Loss of **all** offsite and **all** onsite AC power capability, Table S-1, to emergency 4.16KV buses NB01 and NB02 for ≥ 15 min. (Note 1)

ALERT

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

SA1.1 | 1 | 2 | 3 | 4 |

AC power capability, Table S-1, to emergency 4.16KV buses NB01 and NB02 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

UNUSUAL EVENT

Loss of **all** offsite AC power capability to emergency buses for 15 minutes or longer

SU1.1 | 1 | 2 | 3 | 4 |

Loss of **all** offsite AC power capability, Table S-1, to emergency 4.16KV buses NB01 and NB02 for ≥ 15 min. (Note 1)

Table S-1 AC Power Supplies

Offsite:

- Safeguards XFMR A or B via ESF LTC XFMR XNB01
- Startup XFMR XMR01 via ESF LTC XFMR XNB02
- Main XFMR XMA01 backfed via UAT XFMR XMA02 (only if already aligned)

Onsite:

- EDG NE01
- EDG NE02

2 Loss of Vital DC Power

GENERAL EMERGENCY

Loss of **all** AC and vital DC power sources for 15 minutes or longer

SG1.2 | 1 | 2 | 3 | 4 |

Loss of **all** offsite and **all** onsite AC power capability, Table S-1, to emergency 4.16KV buses NB01 and NB02 for ≥ 15 min.

AND

Loss of **all** 125 VDC power based on battery bus voltage indications < 107 VDC on **all** vital DC buses NK01, NK03 (Division 1) and NK02, NK04 (Division 2) for ≥ 15 min. (Note 1)

SITE AREA EMERGENCY

Loss of **all** vital DC power for 15 minutes or longer

SS2.1 | 1 | 2 | 3 | 4 |

Loss of **all** 125 VDC power based on battery bus voltage indications < 107 VDC on **all** vital DC buses NK01, NK03 (Division 1) and NK02, NK04 (Division 2) for ≥ 15 min. (Note 1)

ALERT

None

UNUSUAL EVENT

None

3 Loss of Control Room Indications

GENERAL EMERGENCY

None

SITE AREA EMERGENCY

Table S-2 Safety System Parameters

- Reactor power
- RCS level
- RCS pressure
- Core Exit T/C temperature
- Level in at least one S/G
- Auxiliary or emergency feed flow in at least one S/G

ALERT

UNPLANNED loss of Control Room indications for 15 minutes or longer with a significant transient in progress

SA3.1 | 1 | 2 | 3 | 4 |

An UNPLANNED event results in the inability to monitor one or more Table S-2 parameters from within the Control Room for ≥ 15 min. (Note 1)

AND

Any significant transient is in progress, Table S-3

UNUSUAL EVENT

UNPLANNED loss of Control Room indications for 15 minutes or longer

SU3.1 | 1 | 2 | 3 | 4 |

An UNPLANNED event results in the inability to monitor one or more Table S-2 parameters from within the Control Room for ≥ 15 min. (Note 1)

4 RCS Activity

GENERAL EMERGENCY

None

SITE AREA EMERGENCY

Table S-3 Significant Transients

- Reactor trip
- Runback ≥ 25% thermal power
- Electrical load rejection > 25% electrical load
- ECCS actuation

ALERT

None

UNUSUAL EVENT

Reactor coolant activity greater than Technical Specification allowable limits

SU4.1 | 1 | 2 | 3 | 4 |

Sample analysis indicates RCS activity > Technical Specification Section 3.4.16 limits

5 RCS Leakage

GENERAL EMERGENCY

None

SITE AREA EMERGENCY

None

ALERT

None

UNUSUAL EVENT

RCS leakage for 15 minutes or longer

SU5.1 | 1 | 2 | 3 | 4 |

RCS unidentified or pressure boundary leakage > 10 gpm for ≥ 15 min.

OR

RCS identified leakage > 25 gpm for ≥ 15 min.

OR

Leakage from the RCS to a location outside containment > 25 gpm for ≥ 15 min.

(Note 1)

6 RTS Failure

GENERAL EMERGENCY

None

SITE AREA EMERGENCY

Inability to shut down the reactor causing a challenge to core cooling or RCS heat removal

SS6.1 | 1 |

An automatic or manual trip fails to shut down the reactor as indicated by reactor power ≥ 5%

AND

All actions to shut down the reactor are **not** successful as indicated by reactor power ≥ 5%

AND EITHER:

- CSFST Core Cooling-**RED** Path conditions met
- CSFST Heat Sink-**RED** Path conditions met

ALERT

Automatic or manual trip fails to shut down the reactor and subsequent manual actions taken at the reactor control consoles are not successful in shutting down the reactor

SA6.1 | 1 |

An automatic or manual trip fails to shut down the reactor as indicated by reactor power ≥ 5%

AND

Manual trip actions taken at the reactor control console (SB-HS-1 or SB-HS-42) are **not** successful in shutting down the reactor as indicated by reactor power ≥ 5% (Note 8)

UNUSUAL EVENT

Automatic or manual trip fails to shut down the reactor

SU6.1 | 1 |

An automatic trip did **not** shut down the reactor as indicated by reactor power ≥ 5% after **any** RTS setpoint is exceeded

AND

A subsequent automatic trip or manual trip action taken at the reactor control consoles (SB-HS-1 or SB-HS-42) is successful in shutting down the reactor as indicated by reactor power < 5% (Note 8)

SU6.2 | 1 |

A manual trip did **not** shut down the reactor as indicated by reactor power ≥ 5% after **any** manual trip action was initiated

AND

A subsequent automatic trip or manual trip action taken at the reactor control console (SB-HS-1 or SB-HS-42) is successful in shutting down the reactor as indicated by reactor power < 5% (Note 8)

7 Loss of Comm.

GENERAL EMERGENCY

Notes

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

Note 8: A manual action is any operator action, or set of actions, which causes the control rods to be rapidly inserted into the core, and **does not** include manually driving in control rods or implementation of boron injection strategies

Note 9: One Containment Spray System train and one Containment Cooling System train comprise one full train of depressurization equipment

SITE AREA EMERGENCY

None

ALERT

Table S-4 Communications Methods

System	Onsite	ORO	NRC
Gaitronics	X		
Plant Radios	X		
Plant Emergency Dedicated Phones	X		
Plant Telephone System	X	X	X
ENS (Red Phone) Line		X	X
Back-Up Radio System		X	
Sentry Notification System		X	

UNUSUAL EVENT

Loss of all onsite or offsite communications capabilities

SU7.1 | 1 | 2 | 3 | 4 |

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

OR

Loss of **all** Table S-4 ORO communication methods

OR

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

8 CMT Isolation Failure

GENERAL EMERGENCY

(See Notes above)

SITE AREA EMERGENCY

None

ALERT

None

UNUSUAL EVENT

Failure to isolate containment or loss of containment pressure control

SU8.1 | 1 | 2 | 3 | 4 |

Any penetration is not isolated within 15 min. of a VALID containment isolation signal

OR

Containment pressure > 27 psig with < one full train of containment depressurization equipment operating per design for ≥ 15 min. (Note 9)

(Note 1)

9 Hazardous Event Affecting Safety Systems

GENERAL EMERGENCY

None

SITE AREA EMERGENCY

Table S-5 Hazardous Events

- Seismic event (earthquake)
- Internal or external FLOODING event
- High winds or tornado strike
- FIRE
- EXPLOSION
- Other events with similar hazard characteristics as determined by the Emergency Coordinator

ALERT

Hazardous event affecting a SAFETY SYSTEM needed for the current operating mode

SA9.1 | 1 | 2 | 3 | 4 |

The occurrence of **any** Table S-5 hazardous event

AND EITHER:

- Event damage has caused indications of degraded performance in at least one train of a SAFETY SYSTEM needed for the current operating mode
- The event has caused VISIBLE DAMAGE to a SAFETY SYSTEM component or structure needed for the current operating mode

UNUSUAL EVENT

None

F Fission Product Barrier Degradation

GENERAL EMERGENCY

FG1.1 | 1 | 2 | 3 | 4 |

Loss of **any** two barriers

AND

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

SITE AREA EMERGENCY

FS1.1 | 1 | 2 | 3 | 4 |

Loss or potential loss of **any** two barriers (Table F-1)

ALERT

FA1.1 | 1 | 2 | 3 | 4 |

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

UNUSUAL EVENT

None

Table F-1 Fission Product Barrier Matrix

Category	Fuel Clad (FC) Barrier: Loss	Fuel Clad (FC) Barrier: Potential Loss	Reactor Coolant System (RCS) Barrier: Loss	Reactor Coolant System (RCS) Barrier: Potential Loss	Containment (CMT) Barrier: Loss	Containment (CMT) Barrier: Potential Loss
A RCS or SG Tube Leakage	None	None	1. An automatic or manual ECCS (SI) actuation required by **EITHER:** • UNISOLABLE RCS leakage • SG tube RUPTURE	1. Operation of a standby charging pump is required by **EITHER:** • UNISOLABLE RCS leakage • SG tube leakage 2. CSFST Integrity-**RED** Path conditions met	1. A leaking or RUPTURED SG is FAULTED outside of containment	None
B Inadequate Heat Removal	1. CSFST Core Cooling-**RED** Path conditions met	1. CSFST Core Cooling-**ORANGE** Path conditions met 2. CSFST Heat Sink-**RED** Path conditions met **AND** Heat sink required	None	1. CSFST Heat Sink-**RED** Path conditions met **AND** Heat sink required	None	1. CSFST Core Cooling-**RED** Path conditions met **AND** Restoration procedures **not** effective within 15 min. (Note1)
C CMT Radiation / RCS Activity	1. Containment radiation > 2.80E+03 R/hr on GT-RE-59 (591) or GT-RE-60 (601) 2. Dose equivalent I-131 coolant activity > 300 µCi/cc 3. CVCS letdown radiation > 2.50E+01 µCi/ml on SJ-RE-01 (016)	None	1. Containment radiation > 6.40E+00 R/hr on GT-RE-59 (591) or GT-RE-60 (601)	None	None	1. Containment radiation > 8.06E+04 R/hr on GT-RE-59 (591) or GT-RE-60 (601
D CMT Integrity or Bypass	None	None	None	None	1. Containment isolation is required **AND EITHER:** • Containment integrity has been lost based on Emergency Coordinator judgment • UNISOLABLE pathway from containment to the environment exists 2. Indications of RCS leakage outside of containment	1. CSFST Containment-**RED** Path conditions met 2. Containment hydrogen concentration ≥ 4% 3. Containment pressure > 27 psig with < one full train of Containment depressurization equipment operating per design for ≥ 15 min. (Note 1, 9)
E Judgment	1. **Any** condition in the opinion of the Emergency Coordinator that indicates loss of the Fuel Clad barrier	1. **Any** condition in the opinion of the Emergency Coordinator that indicates potential loss of the Fuel Clad barrier	1. **Any** condition in the opinion of the Emergency Coordinator that indicates loss of the RCS barrier	1. **Any** condition in the opinion of the Emergency Coordinator that indicates potential loss of the RCS barrier	1. **Any** condition in the opinion of the Emergency Coordinator that indicates loss of the Containment barrier	1. **Any** condition in the opinion of the Emergency Coordinator that indicates potential loss of the Containment barrier

Modes:

1	2	3	4	5	6	DEF
Power Operation	Startup	Hot Standby	Hot Shutdown	Cold Shutdown	Refueling	Defueled

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HOT CONDITIONS
RCS > 200°F)

C Cold SD/ Refueling System Malfunct.

	GENERAL EMERGENCY	SITE AREA EMERGENCY	ALERT	UNUSUAL EVENT

1 RCS Level

GENERAL EMERGENCY
Loss of RCS inventory affecting fuel clad integrity with Containment challenged

CG1.1 Modes: 5, 6

RVLIS Pumps Off < 65% (Top of Fuel) for > 30 min. (Note 1)
AND
Any Containment Challenge indication, Table C-2

CG1.2 Modes: 5, 6

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 C-1 sump/tank level of sufficient magnitude to indicate core uncovery
- Manipulator crane radiation monitor SD-RE-41 > 10,000 mR/hr
- Erratic Source Range Monitor indication

AND
Any Containment Challenge indication, Table C-2

Table C-2 Containment Challenge Indications
- CONTAINMENT CLOSURE **not** established (Note 6)
- Containment hydrogen concentration ≥ 4%
- Unplanned rise in Containment pressure

SITE AREA EMERGENCY
Loss of RCS inventory affecting core decay heat removal capability

CS1.1 Modes: 5, 6

With CONTAINMENT CLOSURE **not** established, RVLIS Pumps Off < 72%

CS1.2 Modes: 5, 6

With CONTAINMENT CLOSURE established, RVLIS Pumps Off < 65% (Top of Fuel)

CS1.3 Modes: 5, 6

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 C-1 sump/tank level of sufficient magnitude to indicate core uncovery
- Manipulator crane radiation monitor SD-RE-41 > 10,000 mR/hr
- Erratic Source Range Monitor indication

ALERT
Loss of RCS inventory

CA1.1 Modes: 5, 6

Loss of RCS inventory as indicated by Reactor Vessel level < bottom of RCS hot leg ID (RVLIS Pumps Off < 73% or BBLI-53 A/B at 0 inches)

CA1.2 Modes: 5, 6

RCS water level cannot be monitored for ≥ 15 min. (Note 1)
AND EITHER
- UNPLANNED increase in **any** Table C-1 Sump / Tank level
- Visual observation of UNISOLABLE RCS leakage

Table C-1 Sumps/Tanks
- Containment Sumps
- Containment Normal Sumps
- Containment Instrument Sump
- PRT
- RCDT
- Auxiliary Building Sump

UNUSUAL EVENT
UNPLANNED loss of RCS inventory for 15 minutes or longer

CU1.1 Modes: 5, 6

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

CU1.2 Modes: 5, 6

RCS water level cannot be monitored
AND EITHER:
- UNPLANNED increase in **any** Table C-1 sump/ tank level due to a loss of RCS inventory
- Visual observation of UNISOLABLE RCS leakage

2 Loss of Emergency AC Power

GENERAL EMERGENCY
None

SITE AREA EMERGENCY
Table C-3 AC Power Supplies

Offsite:
- Safeguards XMFR A or B via ESF LTC XMFR XNB01
- Startup XMFR XMR01 via ESF LTC XMFR XNB02
- Main XMFR XMA01 backfed via UAT XMFR XMA02 (only if already aligned)

Onsite:
- EDG NE01
- EDG NE02

ALERT
Loss of **all** offsite and **all** onsite AC power to emergency buses for greater than 15 minutes

CA2.1 Modes: 5, 6, DEF

Loss of **all** offsite and **all** onsite AC power capability, Table C-3, to emergency 4.16KV buses NB01 and NB02 for ≥ 15 min. (Note 1)

UNUSUAL EVENT
Loss of all but one AC power source to emergency buses for 15 minutes or longer

CU2.1 Modes: 5, 6, DEF

AC power capability, Table C-3, to emergency 4.16KV buses NB01 and NB02 reduced to a single power source for ≥ 15 min. (Note 1)
AND
Any additional single power source failure will result in loss of **all** AC power to SAFETY SYSTEMS

3 RCS Temp.

GENERAL EMERGENCY
None

SITE AREA EMERGENCY
Table C-4 RCS Reheat Duration Thresholds

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

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

ALERT
Inability to maintain plant in cold shutdown

CA3.1 Modes: 5, 6

UNPLANNED increase in RCS temperature to > 200°F for > Table C-4 duration (Notes 1, 10)
OR
UNPLANNED RCS pressure increase > 10 psig (This EAL does not apply during water-solid plant conditions)

UNUSUAL EVENT
UNPLANNED increase in RCS temperature

CU3.1 Modes: 5, 6

UNPLANNED increase in RCS temperature to > 200°F (Note 10)

CU3.2 Modes: 5, 6

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

4 Loss of Vital DC Power

GENERAL EMERGENCY
None

SITE AREA EMERGENCY
None

UNUSUAL EVENT
Loss of required DC power for 15 minutes or longer

CU4.1 Modes: 5, 6

< 107 VDC bus voltage indications on Technical Specification **required** 125 VDC buses for ≥ 15 min. (Note 1)

5 Loss of Comm.

GENERAL EMERGENCY
None

SITE AREA EMERGENCY
None

ALERT
Table C-5 Communications Methods

System	Onsite	ORO	NRC
Gaitronics	X		
Plant Radios	X		
Plant Emergency Dedicated Phones	X		
Plant Telephone System	X	X	X
ENS (Red Phone) Line		X	X
Back-Up Radio System		X	
Sentry Notification System		X	

UNUSUAL EVENT
Loss of **all** onsite or offsite communications capabilities

CU5.1 Modes: 5, 6, DEF

Loss of **all** Table C-5 onsite communication methods
OR
Loss of **all** Table C-5 ORO communication methods
OR
Loss of **all** Table C-5 NRC communication methods

6 Hazardous Event Affecting Safety Systems

GENERAL EMERGENCY
None

SITE AREA EMERGENCY
Table C-6 Hazardous Events
- Seismic event (earthquake)
- Internal or external FLOODING event
- High winds or tornado strike
- FIRE
- EXPLOSION
- Other events with similar hazard characteristics as determined by the Emergency Coordinator

ALERT
Hazardous event affecting a SAFETY SYSTEM needed for the current operating mode

CA6.1 Modes: 5, 6

The occurrence of **any** Table C-6 hazardous event
AND EITHER:
- Event damage has caused indications of degraded performance in at least one train of a SAFETY SYSTEM needed for the current operating mode
- The event has caused VISIBLE DAMAGE to a SAFETY SYSTEM component or structure needed for the current operating mode

UNUSUAL EVENT
None

Notes

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

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

Note 10: Begin monitoring hot condition EALs concurrently for any new event or condition not related to the loss of decay heat removal

Modes:

1	2	3	4	5	6	DEF
Power Operation	Startup	Hot Standby	Hot Shutdown	Cold Shutdown	Refueling	Defueled

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COLD CONDITIONS (RCS ≤ 200°F)

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