**10 CFR 50.90**

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102-07255-MLL/JF/DCE
May 12, 2016

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

References:

1. Arizona Public Service Company (APS) Letter No. 102-07135, *Palo Verde Nuclear Generating Station, Units 1, 2, and 3, and Independent Spent Fuel Storage Installation Request for NRC Approval of Proposed Changes to PVNGS Emergency Action Levels,* dated October 9, 2015 [Agencywide Documents Access and Management System (ADAMS) Accession No. ML15293A335]
2. NRC Staff Email, *Palo Verde Nuclear Generating Station, Units 1, 2, and 3 - Official RAIs for EAL LAR*, dated March 23, 2016 [ADAMs Accession No. ML16083A250]

Dear Sirs:

Subject: **Palo Verde Nuclear Generating Station**
**Units 1, 2, and 3 and Independent Spent Fuel Storage Installation**
**Docket Nos. STN 50-528, 50-529, 50-530, and 72-44**
**Renewed Operating License Nos. NPF-41, NPF-51, NPF-74**
**Response to Request for Additional Information Regarding Proposed Changes to PVNGS Emergency Action Levels**

On October 9 , 2015, Arizona Public Service Company (APS) submitted a request to revise Palo Verde Nuclear Generating Station (PVNGS) Emergency Action Levels (EALs) in accordance with the provisions of Part 50, Appendix E, Section IV.B.2 and 50.90 of Title 10 of the Code of Federal Regulations (10 CFR) (Reference 1). APS proposed to change the EALs from a scheme based on Nuclear Energy Institute (NEI) 99-01, Revision 5, *Methodology for Development of Emergency Action Levels*, to a scheme provided in the subsequent Revision 6.

NRC staff determined that additional information is required and provided a request for additional information (RAI) by an NRC email (Reference 2). Attachment A of the enclosure to this letter provides the APS response to the NRC RAI. Attachment C of the enclosure to this letter provides an updated copy of the EAL Technical Basis.

This response letter also includes new proposals to deviate from NEI 99-01, Revision 6, EALs HG 1.1 and HS 6.1 that are not related to RAI questions. These changes are itemized in Attachment B of the enclosure to this letter, which lists adjustments to EALs not specifically related to the RAI questions and provides corresponding justifications. Attachment B also itemizes other minor changes to the EALs not related to the RAI questions.

Updated wallcharts are provided in Attachment D of the Enclosure, for information.

By copy of this letter, this license amendment request is being forwarded to the Arizona Radiation Regulatory Agency (ARRA) pursuant to 10 CFR 50.91(b)(1).

No commitments are being made by this letter. Should you need further information regarding this submittal, please contact Jeffery Fearn, Emergency Preparedness Manager, at (623) 393-5045.

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

Executed on May 12, 2016
(Date)

Sincerely,

Lacal, Maria L(Z06149)
Digitally signed by Lacal, Maria L(Z06149)
DN: cn=Lacal, Maria L(Z06149)
Date: 2016.05.12 18:10:48 -07'00'

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Enclosure: Response to Request for Additional Information

| | | |
|---|---|---|
| cc: | M. L. Dapas | NRC Region IV Regional Administrator |
| | S. P. Lingam | NRC NRR Project Manager for PVNGS |
| | M. M. Watford | NRC NRR Project Manager |
| | C. A. Peabody | NRC Senior Resident Inspector for PVNGS |
| | A. V. Godwin | Arizona Radiation Regulatory Agency (ARRA) |
| | T. Morales | Arizona Radiation Regulatory Agency (ARRA) |

# Enclosure

# Response to Request for Additional Information

**<u>Summary Description</u>**

On October 9, 2015, Arizona Public Service Company (APS) submitted a request to revise Palo Verde Nuclear Generating Station (PVNGS) Emergency Action Levels (EALs) in accordance with the provisions of Part 50, Appendix E, Section IV.B.2 and 50.90 of Title 10 of the Code of Federal Regulations (10 CFR) (Reference 1). APS proposed to change the emergency action levels from a scheme based on Nuclear Energy Institute (NEI) 99-01, Revision 5, *Methodology for Development of Emergency Action Levels*, to a scheme provided in the subsequent Revision 6.

NRC staff determined that additional information is required and provided a request for additional information (RAI) by an NRC email (Reference 2). Attachment A of this enclosure provides the APS response to the NRC RAI. Attachment C of this enclosure provides an updated copy of the EAL Technical Basis.

This response letter also includes new proposals to deviate from NEI 99-01, Revision 6, EALs HG 1.1 and HS 6.1 that are not related to RAI questions. These changes are itemized in Attachment B of this enclosure, which lists adjustments to EALs not specifically related to the RAI questions and provides corresponding justifications. Attachment B also itemizes other minor changes to the EALs not related to the RAI questions.

Updated wallcharts are provided in Attachment D of this enclosure.

The responses to the RAIs and other changes to EALs identified in this enclosure do not affect the regulatory analysis, including the no significant hazards and environmental considerations provided in the enclosure to Reference 1.

**Attachments**

Attachment A - Response to RAI Questions
Attachment B - Summary of EAL Changes Not Associated with RAI Responses
Attachment C - Updated EAL Technical Bases (Clean Copy)
Attachment D - Updated EAL Wall Charts (Information Only)

**References**

1. Arizona Public Service Company (APS) Letter No. 102-07135, *Palo Verde Nuclear Generating Station, Units 1, 2, and 3, and Independent Spent Fuel Storage Installation Request for NRC Approval of Proposed Changes to PVNGS Emergency Action Levels,* dated October 9, 2015 [Agencywide Documents Access and Management System (ADAMS) Accession No. ML15293A335]

2. NRC Staff Email, *Palo Verde Nuclear Generating Station, Units 1, 2, and 3 - Official RAIs for EAL LAR*, dated March 23, 2016 [ADAMs Accession No. ML16083A250]

# Attachment A - Response to RAI Questions

The table below provides responses to the NRC staff RAI questions.

| RAI # PV- | SECTION/ IC/EAL | Question | PVNGS Response |
|---|---|---|---|
| 1 | 2.5 | Section 2.5, "Technical Bases Information," describes the bases as "a plant-specific basis section that provides PVNGS-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." EAL decision-makers may be confused between these two sections when the information appears to be inconsistent. Please explain the reasoning for two sections rather than one basis section that is specific to the plant and includes the applicable generic information, or revise accordingly. | The PVNGS site specific and NEI 99-01 generic bases sections have been combined into a single bases section for each EAL. Section 2.5, "Technical Bases Information," has been revised accordingly. |
| 2 | RU1 | NEI EAL AU1(2) is not included in the proposed EAL scheme. Please provide justification for not including this in the proposed EAL scheme, or revise the EAL accordingly per endorsed guidance. | PVNGS does not use discharge permits to establish or set gaseous effluent monitor release setpoints. Planned gaseous effluents are monitored by the Plant Vent effluent monitor.<br><br>PVNGS is a no-liquid release plant and has no liquid effluent release pathways. The PVNGS Offsite Dose Calculation Manual does not address liquid release pathways for this reason. See Section 1.1 of the PVNGS ODCM.<br><br>Therefore, generic EAL AU1(2) is not applicable to PVNGS. |
| 3 | 4.3 | Section 4.3, "Instrumentation Used for EALs," to NEI 99-01, Revision 6, states (in part): *"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 proposed 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.<br><br>For example: Plant Vent RU-143 CH-1 >1.22E-02 uCi/cc<br>Fuel Building RU-146 CH-1 >1.13E-01 uCi/cc | PVNGS has confirmed that setpoints and indications used in the proposed EAL scheme are within the calibrated range(s) of the stated instrumentation and that the resolution of the instrumentation is appropriate for the setpoint or indication. |

| RAI # PV- | SECTION/ IC/EAL | Question | PVNGS Response |
|---|---|---|---|
| 4 | 5.0 | Section 5.0, "Definitions, Acronyms, & Abbreviations," does not contain a definition for OWNER CONTROLLED AREA, even though it is used in the current PVNGS Emergency Plan and EALs. Please provide justification for its removal, or revise accordingly to include. | The term OWNER CONTROLLED AREA (OCA) is not used within the proposed PVNGS NEI 99-01, Revision 6, based EAL scheme. NEI 99-01, Revision 6, Appendix B, Definitions, provides the following developer note for the definition of OWNER CONTROLLED AREA:<br>*"This term is typically taken to mean the site property owned by, or otherwise under the control of, the licensee. In some cases, it may be appropriate for a licensee to define a smaller area with a perimeter closer to the plant Protected Area perimeter (e.g., a site with a large OCA where some portions of the boundary may be a significant distance from the Protected Area). In these cases, developers should consider using the boundary defined by the Restricted or Secured Owner Controlled Area (ROCA/SOCA). The area and boundary selected for scheme use must be consistent with the description of the same area and boundary contained in the Security Plan."*<br>Due to the size of the PVNGS OCA, PVNGS has a defined SECURITY OWNER CONTROLLED AREA (SOCA). The term is synonymous with the generic term "Secured Owner Controlled Area." PVNGS has replaced OCA with SOCA. The definition of the PVNGS SOCA is consistent with the PVNGS Security Plan and the generic guidance developer note regarding the definition of the OWNER CONTROLLED AREA (OCA). |
| 5 | RA1.2<br>RA1.3<br>RS1.2<br>RG1.2 | NEI basis statement, *"classification based on effluent monitor readings assumes that a release path to the environment is established. If the effluent flow past the monitor is known to have stopped…"* is included in the bases for EALs RA1.2, RA1.3, RS1.2, and RG1.2. Since these EALs are independent of radiation monitors, the statement is not applicable and may be confusing, resulting in a delay in classification. Please remove that statement from the bases, or explain how it is applicable. | The cited bases statement was deleted from EALs RA1.2, RA1.3, RS1.2 and RG1.2. |

| RAI # PV- | SECTION/ IC/EAL | Question | PVNGS Response |
|---|---|---|---|
| 6 | RU2.1 | EAL RU2.1 includes level indication (visual or Refueling Water Level Indicating System (RWLIS)). The PVNGS Basis references a remote spent fuel pool (SFP) level indication, PCN-LSHL-3. Please explain why PCN-LSHL-3 is not included in this EAL, or revise accordingly. If PCN-LSHL-3 can only be read in the vicinity of the SFP, please clarify the basis description stating, *"The SFP level is remotely monitored by..."* as this could be interpreted as remote from the SFP or remote from the Control Room. | The EAL RU2.1 bases was revised to clarify both local and remote SFP level alarms and indications:<br>*"The SFP is locally monitored in the Fuel Building by Level indicators PCN-LIT-3/5 on PCNE02. These level indicating transmitters also initiate local panel alarms via level switches PCN-LSHL-3/PCN-LSL-5 on low and low low SFP level respectively. The alarms are also located on PCNE02 and annunciate a general Control Room alarm on window "FUEL POOL CLG SYS TRBL" indicating an alarm is in on the local panel.*<br>*Level is also indicated in the Control Room visually via digital camera feed and in the back panel area on panel PCN-E015 by a digital level indicator. This Control Room indication does not have associated annunciation."* |
| 7 | RU2.1 | For EAL RU2.1, please explain why the word "Unplanned" was removed from the alarm thresholds as it appears in endorsed guidance, or revise accordingly. | The defined term UNPLANNED was added to the cited EAL RU2.1 threshold to conform with the endorsed guidance. |
| 8 | RA2.2 | EAL RA2.2 is based on receiving a high alarm on listed radiation monitors. Please verify that the listed radiation monitors would be expected to alarm high for damage to irradiated fuel or provide an expected range indication for any radiation monitors not expected to reach the high alarm setpoint. | Per the referenced abnormal operating procedure, 40AO-9ZZ22, *Fuel Damage*, the listed radiation monitors would be expected to alarm high for a damaged irradiated fuel event (Containment or Fuel Building, as applicable). |
| 9 | RS2.1<br>RG2.1 | For EALs RS2.1 and RG2.1, please explain the statement, *"…(includes a 1 ft. 10 inches instrument indication margin),"* as it relates to this EAL threshold. | The actual elevation of the top of the spent fuel racks is 114 ft. 2 inches. However, Level 3 has been set at 116 ft. to account for up to a 1 ft. 10 inches instrument error. For clarification, the bases has been revised to read:<br>*"includes a 1 ft. 10 inches instrument indication error margin)"* |
| 10 | RA3.1 | For EAL RA3.1, please add "by survey" to "Central Alarm Station (CAS)" to ensure that this is clearly acceptable for this particular location and appears on the EAL wallboard, or provide | EAL RA3.1 was revised to add "(by survey)" to CAS. |

| RAI # PV- | SECTION/ IC/EAL | Question | PVNGS Response |
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| | | justification for not including. | |
| 11 | RA3.2<br>HA5.1 | For EALs RA3.2 and HA5.1, please add a note to the basis section to remind users that this EAL is intended to be applicable to ALL Operating Modes but that limiting the Operating Modes to those determined to be necessary for the scope of this EAL is acceptable. However, if plant and/or system design is changed, such that other areas for other Operating Modes become applicable, then it is expected that these areas be added to the table for the applicable Operating Mode(s). | The following note was added to the EALs RA3.2 and HA5.1 bases to ensure RA3.2 and HA5.1 mode applicability remains consistent with Table R-2/H-2 Room/Area mode applicability:<br>"***NOTE***: *EAL RA3.2 [HA5.1] mode applicability has been limited to the applicable modes identified in Table R-2 [H-2] Safe Operation & Shutdown Rooms/Areas. If due to plant operating procedure or plant configuration changes, the applicable plant modes specified in Table R-2 [H-2] are changed, a corresponding change to Attachment 3 'Safe Operation & Shutdown Areas Tables R-2 & H-2 Bases' and to EAL RA3.2 [HA5.1] mode applicability is required."* |
| 12 | CU1.1 | For EAL CU1.1, please incorporate the information in the PVNGS Basis related to the limitations with the RWLIS [Refueling Water Level Indicating System] as a note to the EAL to ensure it is captured on the EAL wallboards, or provide justification for not including. | Added Note 10 to EAL CU1.1 that reads:<br>*Variations in RCS boron concentration, temperature and Containment Temperature from those used in RWLIS calibration will induce indication errors. Refer to Operator Assistance Program RWLIS_Spreadsheet.xls.*<br>The basis provides a similar statement related to this note. The referenced spreadsheet is qualified software. The spreadsheet correlates RWLIS indications to actual plant conditions. |
| 13 | CU1.2<br>CA1.2<br>CS1.1<br>CG1.1<br>RCS Loss A.1<br>RCS PLoss A.1<br>CMT Loss D.1 | For EALs CU1.2, CA1.2, CS1.1, CG1.1, RCS [Reactor Coolant System]-Loss 1, RCS-Potential Loss 1 and Containment-Loss 1, please explain the statement, *"A RCS Leak is considered unisolable if the leak cannot be isolated within 15-minutes,"* as this did not come from the endorsed guidance, or revise accordingly. | The cited statement was deleted from EALs CU1.2, CA1.2, CS1.1, CG1.1, RCS Loss A.1, RCS Potential Loss A.1 and Containment Loss D.1 bases. |

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| 14 | CU4.1 | For EAL CU4.1, please explain why "vital" was removed from the EAL as in endorsed guidance, or revise accordingly. | The word "vital" was added to CU4.1 to conform with the endorsed guidance. |
| 15 | CA1.1 | For EAL CA1.1, please explain why the proposed value of 101 ft., 6 in. is different than the current EAL threshold of 101 ft. 4 in. (no basis for this level in current EAL), or revise accordingly. | The current PVNGS equivalent EAL to CA1.1 is based on NEI 99-01, Revision 5, which specifies the Alert threshold water level to be equivalent to the bottom of the RCS hotleg (101 ft. 4 inches). NEI 99-01, Revision 6, specifies the Alert threshold water level to be equivalent to the minimum level for continued operation of a residual heat removal pump in the shutdown cooling mode (101 ft. 6 inches).<br><br>Therefore the proposed EAL CA1.1 is consistent with the generic guidance of NEI 99-01, Revision 6. |
| 16 | CA3.1 | For EAL CA3.1, please confirm that the engineering unit specified for capturing the reactor coolant system pressure increase is correct. Current EAL has units in "psi [pounds per square inch]," while the proposed EAL has "psia [pounds per square inch absolute]." | The specified units (psia) are consistent with the units as read in the control room on the specified instruments RCA PI-103, RCB-PI-104, RCC-PI-105, and RCD-PI-106. |
| 17 | CA2.1 | For EAL CA2.1, the PVNGS basis states (in part): *"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 [alternating current] power to the emergency buses."* This conflicts with the included NEI 99-01 basis, which states (in part): "*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."* Please revise the basis to eliminate this discrepancy, or provide further justification for this inconsistency. | The cited bases statement was deleted to eliminate the discrepancy. |
| 18 | CA2.1<br>SS1.1<br>SG1.2 | For EALs CA2.1, SS1.1 and SG1.2, per the Developer Notes in endorsed guidance, the licensee may establish the capability to power an essential bus from an alternate power supply during the | The reference to Table C-3/S-1 AC power source tables was deleted from EALs CA2.1, SS1.1 and SG1.2.<br><br>No other information was included in the basis that limits |

| RAI # PV- | SECTION/ IC/EAL | Question | PVNGS Response |
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| | | additional time that may be potentially available. Please remove the table from these EALs as the intent is to capture events of concern when the busses have no power, regardless of the source, or provide further basis for retaining. In addition, please remove any information in the basis section that limits consideration of any other sources. | consideration of other sources, other than the 15 minutes timeliness restrictions of the SBOGs. |
| 19 | CA6.1<br>SA9.1 | CA6.1 and SA9.1 PVNGS basis (first three bulleted items) provides details of seismic events, flooding and high winds. These descriptions could cause a delay in declaration or a failure to declare an Alert due to attempting to determine if the cause of the failure meets the description in the basis (e.g., high wind less than or in excess of 105 mph). Please delete these descriptions or provide a justification for including this information. | The bulleted items are included to provide site-specific background information to the EAL end-user relative to the types of hazardous events listed in Tables C-6 and S-5. This information does not alter the intent or scope of the classifiable hazardous events listed in the tables and does not constitute criteria used for classification. The bulleted items have been moved to the end of the bases discussion of EAL CA6.1 and SA9.1 to de-emphasize and eliminate the stated concern. This information will not cause a delay or failure to appropriately classify an event. |
| 20 | CS1.1<br>CG1.1 | For EALs CS1.1 and CG1.1, please explain why the other EALs from the endorsed guidance cannot be implemented at PVNGS, as they are in the current EALs. Please explain why the staff should consider removing these EALs from the PVNGS scheme, or revise accordingly. Also, explain why RVLMS [Reactor Vessel Level Monitoring system] levels are not used. | The current PVNGS EALs are based on NEI 99-01, Revision 5. In the absence of developer guidance in Revision 5, for the PVNGS equivalent of generic EAL CS1 (Example EALs # 1 and #2) and CG1 (Example EAL #1), PVNGS used the minimum measurable RCS level (bottom of the hot leg) in lieu of the specified thresholds of 6 inches below the bottom of the hot leg and top of active fuel (TAF). This was because at PVNGS, RCS water level cannot be measured directly below the elevation of the last RVLMS sensor which is less than two inches below the bottom of the RCS hot leg and well above the top of active fuel (TAF).<br><br>EALs are consistent with the NEI 99-01, Revision 6, generic developers guidance: *If the design and operation of water level instrumentation is such that this level value cannot be determined at any time during Cold Shutdown or Refueling modes, then do not include EAL #1 (#2) (classification will be accomplished in accordance with EAL #3).* Therefore, CS1 (Example EALs #1 and #2) and CG1 (Example EAL #1) are not implemented. |

| RAI # PV- | SECTION/ IC/EAL | Question | PVNGS Response |
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| 21 | CS1.1<br>CG1.1 | PVNGS EALs CS1.1 and CG1.1 include the definition of "unisolable" and the Basis statement, *"a RCS leak should be considered unisolable if the leak cannot be isolated within 15 minutes."* These EALs do not refer to unisolable RCS leaks. Please remove the references to unisolable leakage or provide an explanation of why it is desired. | The definition of "unisolable" was deleted from EALs CS1.1 and CG1.1 bases. The cited bases were revised consistent with the response to RAI-13. |
| 22 | CG1.1 | For EALs CG1.1 and Containment-Potential Loss 2 (for the hydrogen concentration), please explain why the values for the noted radiation monitor and for the containment hydrogen concentration are different than what is currently in place. The radiation monitor value is currently 10,000 mR [milli-Roentgen]/hr with 9,000 mR/hr proposed. The containment hydrogen concentration is currently 4.5% with 4% proposed. | EAL CG1.1 uses radiation monitor RU-33 as an alternative indication of possible core uncovery in the refueling mode. The upper range of RU-33 is 10,000 mR/hr. The expected dose rate due to core uncovery is likely beyond this upper range. Consistent with NEI 99-01, Revision 6, developer notes, a value corresponding to 90% of the instrument range has been selected to ensure an accurate monitor reading is available:<br><br>*"It is recognized that the condition described by this IC may result in a radiation value beyond the operating or display range of the installed radiation monitor. In those cases, EAL values should be determined with a margin sufficient to ensure that an accurate monitor reading is available. For example, an EAL monitor reading might be set at 90% to 95% of the highest accurate monitor reading."*<br><br>The containment hydrogen concentration was revised to reflect the current PVNGS threshold value of 4.5%.<br><br>The emergency operating procedure, 40EP-9EO03, *Loss of Coolant Accident*, uses 4.5% as an acceptance criterion for performance of the Containment Combustible Gas Control Safety Function Status Check. Hydrogen concentration exceeding 4.5% indicates the hydrogen recombiners are not operating satisfactorily.<br><br>Minor editorial changes to the basis discussion were made to insert the above justification. |

| RAI # PV- | SECTION/ IC/EAL | Question | PVNGS Response |
|---|---|---|---|
| 23 | HU1.1<br>HA1.1<br>HS1.1<br>HG1.1 | For EALs HU1.1, HA1.1, HS1.1 and HG1.1, please explain how the term "security team" can be synonymous with "security supervision," as the intent of the EAL is to ensure an individual specifically trained on communicating with the Control Room, through hostile action-based drills, is tasked with this responsibility, or revise accordingly. | EALs HU1.1, HA1.1, and HS1.1 and associated bases were revised to replace the term "Security Team" with "Security Shift Supervision."<br>EAL HG 1.1 is deleted, refer to Attachment B of this Enclosure. |
| 24 | HU2.1 | For EAL HU2.1, please explain: (1) how timely the process is for using the "Modified-Mercalli Intensity Scale," and (2) what the relationship is between that and the 0.1g OBE [operating basis earthquake] value for PVNGS; or revise accordingly. Consider use of the alternate wording provided in Developer Notes, with a preplanned confirmation. | EAL HU2.1 was revised to read:<br>"*Seismic event > OBE as indicated on Control Panel A-J-SMN-C01*"<br>Deleted bases reference to the Modified-Mercalli Intensity Scale. |
| 25 | HA1.1 | For EAL HA1.1, please explain why the staff should consider a change from the current "owner controlled area" to the proposed "secured owner controlled area." Please elaborate on the distinction between the two terms, as well as the basis for why consideration should be given to restrict the area of concern for these EALs, or revise accordingly. In addition, by adding this term to the NEI 99-01 Basis section of these EALs, it is implied that this term is used in NEI 99-01, which it is not. | See response to RAI-4. |
| 26 | HS6.1 | Proposed EAL HS6.1 allows 45 minutes to gain control of key safety functions. The existing EAL HS2 allows 15 minutes to gain control of key safety functions. The proposed basis states (in part), *"Auxiliary Feedwater can be initiated as late as 45 minutes after reactor trip, per the PVNGS Fire Protection Analysis."* Please provide evidence supporting establishment of reactivity and RCS inventory control as late as 45 minutes, instead of the expected 15 minutes, or revise the EAL accordingly. | EAL HS6.1 and associated bases were revised to reflect a 15 minute threshold for establishment of safety functions.<br>EAL HS6.1 was also revised to change mode applicability. See Attachment B of this enclosure. |
| 27 | SU1.1 | EAL SU1.1 is based on a loss of offsite AC power. The basis referenced in Table S-1 includes onsite AC power sources. Consider modifications to the table in the basis, and the wording | Table S-1 provides a clearly identified list of credited onsite and offsite emergency bus AC power sources. This common table is referenced by EALs SU1.1 and SA1.1. This precludes the need for separate onsite and off-site AC power source tables. EAL |

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| | | of the EAL and the EAL wallboard, to prevent possible confusion. | validation activities evaluated whether the tables created confusion among the staff and concluded the wording of the table would provide sufficient clarity. Therefore, no change is proposed. |
| 28 | SU3.1<br>SA3.1 | For EALs SU3.1 and SA3.1, please add a note Table S-2 for the Auxiliary feed flow to capture that downcomer flow instruments can be credited for this indication, or provide basis for not including. | Note 11 was added to Table S-2 that reads:<br>"*Downcomer flow instruments are also credited for auxiliary feed flow indication*" |
| 29 | SU5.1 | For EAL SU5.1, please consider deleting references to RCS Loss and Potential Loss thresholds as they are not applicable to this EAL or explain how they are applicable. | The bases discussion related to reactor coolant pump high pressure seal cooler (RCP HPSC) leaks and RCS loss or potential loss thresholds were deleted from the EAL SU5.1 bases. |
| 30 | FPB FC<br>PLoss A.1 | For Fission Product Barrier (FPB) Fuel Clad Category A -- Potential Loss 1: RVLMS < 21% plenum, Pant Specific basis indicates this level corresponds to 4 inches above the fuel alignment pin. PVNGS Updated Final Safety Analysis Report Appendix 18B states that the RVLMS measures reactor vessel water levels above the fuel alignment plate. Please revise the basis accordingly. | The FPB Fuel Clad Potential Loss A.1 bases were revised to read: "alignment plate" |
| 31 | FPB FC<br>Loss B.1<br>PLoss B.1 | FPB Fuel Clad Category B -- Loss 1 and Potential Loss 1 contain the undefined term "Rep CETs [representative core exit thermocouples]." Please provide a definition for Rep CET. | A definition of Rep CET has been added to the bases of FPB Fuel Clad Loss B.1, Fuel Clad Potential Loss B.1, Containment Potential Loss B.1, EAL SS6.1 and EAL SG1.1:<br>*"Rep CET (Representative Core Exit Temperature) is a calculated temperature value generated by the Qualified Safety Parameter Display System (QSPDS). The QSPDS CET processing function generates a representative temperature based on a statistical analysis of thermocouples monitoring the reactor coolant temperature at the top of selected fuel assemblies."* |
| 32 | FPB RCS<br>PLoss A.1 | FPB Reactor Coolant System -- Potential Loss 1 is not consistent with endorsed guidance, which states the threshold is based on maintaining pressurizer level within limits by operation of **a** | The PVNGS FPB RCS Potential Loss A.1 wording is consistent with the generic NEI 99-01 Rev. 6 guidance: |

| RAI # PV- | SECTION/ IC/EAL | Question | PVNGS Response |
|---|---|---|---|
| | | normally used charging pump (emphasis added). Additionally, the NEI 99-01 Change Summary includes the following: *"The RCS P-Loss leak rate threshold has been simplified - instead of quantifying the leak rate (i.e., determining if the leak rate is greater than a pump capacity), the new threshold requires classification if operation of a standby charging (make-up) pump is required. This action would be directed by an AOP [abnormal operating procedure]/EOP [emergency operating procedure] in response to indications that unisolable RCS leakage, or SG [steam generator] tube leakage, is beyond the capacity of one charging pump (e.g., letdown is isolated and pressurizer level continues to decrease)."* Please revise this threshold and basis, or provide additional justification for the proposed change as written. | NEI 99-01 Rev. 6 wording:<br>"*Operation of a standby charging (makeup) pump is required by EITHER…*"<br>PVNGS wording:<br>*"With letdown isolated, operation of the standby charging pump is required by EITHER…"*<br>The change of the cited "**a**" to "**the**" is because PVNGS has three charging pumps, two of which are normally running. Therefore the threshold is met when the standby (3rd) pump must be operated to maintain level with letdown isolated.<br>The PVNGS (CE System 80) Charging and Letdown subsystems of the Chemical and Volume Control system normally operates with two of three positive displacement charging pumps running (designated always and normally running). There is normally one charging pump in standby, available to be placed into service by operator demand or an automatic start signal generated from the Pressurizer Level Control System. The nominal capacity of each Charging Pump is 44 gallons per minute (gpm). Two charging pumps are required to be operating to support nominal letdown flow cooling requirements. With less than two charging pumps in operation, letdown temperature will rise rapidly due to the reduction of charging cooling flow in the letdown regenerative heat exchanger. Letdown will isolate if the system is in automatic due to high temperature. Letdown (nominally 77 gpm) takes some time to arrive at a new value in response to a level change in the Pressurizer. These level changes can be produced by a change in RCS temperature or a leak.<br>The abnormal operating procedure 40AO-9ZZ03, *Excessive RCS Leakrate*, directs starting of the 3rd (standby) charging pump if additional makeup is required. |

| RAI # PV- | SECTION/ IC/EAL | Question | PVNGS Response |
|---|---|---|---|
| | | | Once all available charging pumps are running, the procedure directs isolation of letdown to eliminate this as a possible leakage path. If the standby charging pump is still required to maintain level with letdown isolated then the threshold is met for the RCS potential loss. This threshold allows for a clear line between abnormal leakage and the transient response caused by much smaller RCS leaks/temperature transients on the system. It has historically provided a clear line to support the EAL threshold without the delay of measuring a physical leak rate from instrumentation influenced by the transient. Recognizing the PVNGS design operates two charging pumps during normal operation, the proposed EAL is consistent with the objectives of the endorsed guidance. The need to continue operation of the standby (makeup) pump (i.e., the third charging pump at PVNGS) with indication of unisolable RCS leakage (e.g., letdown isolated) for PVNGS is an appropriate threshold. |
| 33 | FPB RCS PLoss B.2 | FPB Fuel Clad Category B -- Potential Loss 2, and FPB Reactor Coolant System Category B -- Potential Loss 1, "RCS heat removal cannot be established", are not consistent with NEI 99-01, Revision 6. The guidance states, (in part), *"enter the site-specific parameters and values that define an extreme challenge to the ability to remove heat from the RCS via the steam generators."* Please provide the parameters and values in the EALs which indicate that RCS heat removal cannot be established, or explain how decision makers could consistently reach the conclusion that heat removal cannot be established. | PVNGS is a Combustion-Engineering (C-E) System 80 design and implements the CE Owners Group EOP scheme, which provides different evaluations and success sequences for accident response than the WOG EOP scheme. The PVNGS EOPs direct operators to maintain Safety Function Status Checks (SFSCs), one of which is RCS Heat Removal. The RCS heat removal SFSC is established by maintaining adequate RCS subcooling (≥ 24ºF) with an effective RCS heat removal method (steam generators). There are no other site-specific parameters or values other than RCS subcooling used to define inadequate heat removal capability via steam generators. |
| 34 | FPB CMT PLoss B.1 | Concerning FPB Containment Category B -- Potential Loss 1, the additional condition representing an imminent core melt sequence is included in the current PVNGS EAL scheme:<br>a. Rep CET greater than 700 'F.<br>AND | PVNGS' C-E based EOPs provide different evaluations and success sequences for accident response than the WOG EOP scheme. The cited current EAL FPB threshold was implemented as an equivalent of Westinghouse CSFST Core Cooling Red Path in the absence of design specific guidance for the C-E System 80 design. The site-specific criterion for entry into core cooling |

| RAI # PV- | SECTION/ IC/EAL | Question | PVNGS Response |
|---|---|---|---|
| | | b. RVLMS less than 21% plenum.<br>AND<br>c. Restoration not effective within 15 minutes.<br>Please explain why this is not applicable to the proposed FPB Containment Category B -- Potential Loss 1, or revise accordingly. | restoration procedures is Rep CETs > 1200 ºF.<br>This is consistent with the generic developer notes that state that a reading of 1200 ºF on the CETs can be used. |

<table>
<tr><th>RAI #<br>PV-</th><th>SECTION/<br>IC/EAL</th><th>Question</th><th>PVNGS Response</th></tr>
<tr>
<td>35</td>
<td>FPB CMT D</td>
<td>Concerning Plant Specific Basis for FPB Containment Category D items below:<br><br>a. References to RCS Loss are unnecessary and could be confusing. Potential Loss encompasses the conditions in each of the bulleted statements;<br><br>b. Where leak rates are greater than or equal to the Potential Loss threshold, a fifteen minute time limit for leak isolation is not consistent with endorsed guidance, and<br><br>c. The fourth bulleted sentence is incomplete.<br><br>Please modify the basis to address the bullet items above, or provide additional justifications for the basis as written.</td>
<td>The cited bases discussion related to RCP HPSC leaks and RCS loss or potential loss thresholds was deleted from the Containment Loss D.1 and D.2 bases.<br><br>The following bases guidance was added to Containment Loss D.2 to be consistent with the bases for Containment Loss A.1:<br><br>
<table>
<tr><th></th><th colspan="2">RCS Leakage<br>Outside of Containment?</th></tr>
<tr><th>RCS Leak Rate</th><th>Yes</th><th>No</th></tr>
<tr><td>Less than or equal to 25 gpm</td><td>No classification</td><td>No classification</td></tr>
<tr><td>Greater than 25 gpm</td><td>Unusual Event per SU5.1</td><td>Unusual Event per SU5.1</td></tr>
<tr><td>Requires operation of the standby charging (makeup) pump (RCS Barrier Potential Loss)</td><td>Site Area Emergency per FS1.1</td><td>Alert per FA1.1</td></tr>
<tr><td>Requires an automatic or manual ECCS (SIAS) actuation (RCS Barrier Loss)</td><td>Site Area Emergency per FS1.1</td><td>Alert per FA1.1</td></tr>
</table>
The third item in the above table was changed to “the standby charging (makeup) pump” from “a standby charging (makeup) pump” in Containment Loss A.1 and D.2 consistent with the discussion provided in RAI-32.<br><br>The bases of Containment Loss D.1 and D.2 have been revised to clarify and align the Figure 1 Containment Integrity and Bypass Examples with the PVNGS NC system configuration.</td>
</tr>
</table>

# Attachment B - Summary of EAL Changes Not Associated with RAI Responses

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

| EAL # | Tech Basis Change? | Description |
|---|---|---|
| General | No | Minor typographical and format corrections. These are not deviations. |
| SU8.1<br>CMT Potential Loss D.3 | Yes | The references to “ice condenser fans” were deleted from the bases. PVNGS does not have containment ice condenser fans. This is not a deviation. |
| HG1.1 | Yes | APS proposes to delete EAL HG1.1.<br>There are several other Initiating Conditions (ICs) that are redundant with this IC (See description table below), and are better suited to ensure timely and effective emergency declarations. In addition, the development of new spent fuel pool level EALs, as a result of NRC Order EA-12-051, clarified the intended emergency classification level for spent fuel pool level events. This deviation is justified because:<br>1. Hostile Action in the Protected Area is bounded by ICs HS1 and HS7. Hostile Action resulting in a loss of physical control is bound by EAL HG7, as well as any event that may lead to radiological releases to the public in excess of Environmental Protection Agency (EPA) Protective Action Guides (PAGs).<br>a. If, for whatever reason, the Control Room must be evacuated, and control of safety functions (e.g., reactivity control, core cooling, and RCS heat removal) cannot be reestablished, then IC HS6 would apply, as well as IC HS7 if desired by the EAL decision-maker. Therefore, this part of HG1 is redundant and unnecessary.<br>b. Also, as stated above, any event (including Hostile Action) that could reasonably be expected to have a release exceeding EPA PAGs would be bound by IC HG7. Therefore, this part of HG1 is redundant and unnecessary.<br>c. From a Hostile Action perspective, ICs HS1, HS7 and HG7 are appropriate. Therefore, this part of HG1 is redundant and unnecessary.<br>d. From a loss of physical control perspective, ICs HS6, HS7 and HG7 are appropriate. Therefore, this part of HG1 |

| EAL # | Tech Basis Change? | Description |
|---|---|---|
| | | is redundant and unnecessary.<br>2. Any event which causes a loss of spent fuel pool level will be bounded by ICs RA2, RS2 and RG2, regardless of whether it was based upon a Hostile Action or not. Therefore, this part of HG1 is redundant and unnecessary.<br>3. An event that leads to a radiological release will be bounded by ICs RU1, RA1, RS1 and RG1. Events that lead to radiological releases in excess of EPA PAGs will be bounded by EALs RG1 and HG7. Therefore, this part of HG1 is redundant and unnecessary.<br>ICs RU1, RA1, RS1, RA2, RS2, RG2, RS1, RG1, HS1, HS6, HS7 and HG7 have been implemented consistent with NEI 99-01, Revision 6, and thus HG1 is adequately bounded as described above. Once exception is noted: A deviation for HS 6.1 is described below, however, that exception does not affect the rationale to delete EAL HG1.1.<br>This is an acceptable deviation from the generic NEI 99-01, Revision 6. |
| HS6.1 | Yes | The defueled mode was deleted from the EAL applicability. Control of the cited safety functions is not important for a defueled reactor as there is no energy source in the reactor vessel or RCS.<br>The mode applicability for the reactivity control safety function has been limited to Modes 1, 2, and 3. Control of the reactivity control safety function is expected to be maintained in Modes 4, 5, and 6 while transferring control of the plant from the control room to the remote shutdown panel. Maintenance of the reactivity control safety function, achieved by inserted control element assemblies (CEAs) and / or maintenance of sufficient reactor coolant system boron concentration are prerequisites for entry into Modes 3 and 4 following a plant shutdown (Reference procedure 40DP-9EO02, *Reactor Trip*, and Technical Specifications 3.1.1, *SHUTDOWN MARGIN (SDM) - Reactor Trip Breakers Open*, and 3.1.2, *SHUTDOWN MARGIN (SDM) - Reactor Trip Breakers Closed* ). The transfer of plant control to the remote shutdown panel cannot credibly result in a change to CEA position and RCS boron concentration, therefore, there is no need to re-establish the reactivity control safety function at the remote shutdown panel for Modes 4, 5, and 6.<br>This is an acceptable deviation from the generic NEI 99-01, Revision 6. |

# Attachment C - Updated EAL Technical Bases (Clean Copy)

**EP-0901**

**Appendix B**

**Revision 0**

**TABLE OF CONTENTS**

## 1.0 PURPOSE

This document provides an explanation and rationale for each Emergency Action Level (EAL) included in the EAL Upgrade Project for Palo Verde Nuclear Generating Station (PVNGS). Decision-makers responsible for implementation of EP-0901, *Classifications*, may use this document as a technical reference in support of EAL interpretation. This information may assist the Emergency Coordinator in making classifications, particularly those involving judgment or multiple events. The basis information may also be useful in training and for explaining event classifications to off-site officials.

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

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

## 2.0 DISCUSSION

### 2.1 Background

EALs are the plant-specific indications, conditions or instrument readings that are utilized to classify emergency conditions defined in the PVNGS Emergency Plan.

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

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

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

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

## 2.2 Fission Product Barriers

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

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

The primary fission product barriers are:

A. Fuel Clad (FC): The FC 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 (CTMT): The CTMT 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 emergency classification level (ECL) from Alert to a Site Area Emergency or a General Emergency

## 2.3 Fission Product Barrier Classification Criteria

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

*Alert:*

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

*Site Area Emergency:*

*Loss or potential loss of any two barriers*

*General Emergency:*

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

## 2.4 EAL Organization

The PVNGS EAL scheme includes the following features:

- Division of the EAL set into three broad groups:
  - EALs applicable under <u>any</u> plant operating modes – This group would be reviewed by the EAL-user any time emergency classification is considered.
  - EALs applicable only under <u>hot</u> operating modes – This group would only be reviewed by the EAL-user when the plant is in 1 - Power Operation, 2 - Startup, 3 - Hot Standby or 4 - Hot Shutdown mode.
  - EALs applicable only under <u>cold</u> operating modes – This group would only be reviewed by the EAL-user when the plant is in 5 - Cold Shutdown, 6 - 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 user for a given plant condition, reduces user reading burden and, thereby, facilitates timely 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 PVNGS EAL categories are aligned to and represent the NEI 99-01 ”Recognition Categories.” Subcategories are used in the PVNGS scheme as necessary to further divide the EALs of a category into logical sets of possible emergency classification thresholds. The PVNGS 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<br>2 – **I**rradiated Fuel Event<br>3 – Area Radiation Levels |
| H – **H**azards and Other Conditions Affecting Plant Safety | 1 – Security<br>2 – Seismic Event<br>3 – Natural or Technological Hazard<br>4 – Fire<br>5 – Hazardous Gas<br>6 – Control Room Evacuation<br>7 – Emergency Coordinator Judgment |
| E – ISFSI | 1 – Confinement Boundary |
| **Hot Conditions:** | |
| S – **S**ystem Malfunction | 1 – Loss of Emergency AC Power<br>2 – Loss of Vital DC Power<br>3 – Loss of Control Room Indications<br>4 – RCS Activity<br>5 – RCS Leakage<br>6 – RPS Failure<br>7 – Loss of Communications<br>8 – Containment Failure<br>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<br>2 – Loss of Emergency AC Power<br>3 – RCS Temperature<br>4 – Loss of Vital DC Power<br>5 – Loss of Communications<br>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, E and F) and EAL subcategory.  A summary explanation

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

Category Letter & Title

Subcategory Number & Title

Initiating Condition (IC)

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

EAL Identifier (enclosed in rectangle)

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

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

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

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

Classification (enclosed in rectangle):

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

EAL (enclosed in rectangle)

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

Mode Applicability

One or more of the following plant operating conditions comprise the mode to which each EAL is applicable: 1 - Power Operation, 2 - Startup, 3 – Hot Standby, 4 - Hot Shutdown, 5 - Cold Shutdown, 6 - Refueling, DEF - 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 basis section that provides PVNGS-relevant information concerning the EAL as well as a a description of the rationale for the EAL as provided in NEI 99-01, Rev. 6.

PVNGS Basis Reference(s):

Site-specific source documentation from which the EAL is derived

**2.6 Operating Mode Applicability (ref. 4.1.6)**

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

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

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

4 Hot Shutdown
$K_{eff} < 0.99$ and average coolant temperature 350ºF > $T_{avg}$ > 210 ºF and all reactor vessel head closure bolts fully tensioned

5 Cold Shutdown
$K_{eff} < 0.99$ and average coolant temperature $\leq$ 210ºF and all reactor vessel head closure bolts fully tensioned

6 Refueling
One or more reactor vessel head closure bolts 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.

# 3.0 GUIDANCE ON MAKING EMERGENCY CLASSIFICATIONS

## 3.1 General Considerations

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

### 3.1.1 Classification Timeliness

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

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.

### 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. 4.1.9).

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

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

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

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

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

#### 3.2.8 Retraction of an Emergency Declaration

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

## 4.0 REFERENCES

### 4.1 Developmental

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

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

4.1.3 NUREG-1022, *Event Reporting Guidelines: 10 CFR 50.72 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 Technical Specifications Table 1.1-1, *Modes*

4.1.7 Procedure 40EP-9EO10, *LM-Containment Evacuation and Closure*, Appendix 249

4.1.8 Procedure Writers Manual PVNGS Plant Procedure Writers Manual

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

4.1.10 *PVNGS Emergency Plan*

4.1.11 Procedure 40DP-9ZZ30, *Reduced Inventory Operations*

4.1.12 Procedure 20DP-0SK49, *Security Integrated Response Plan (Proprietary Information)*

### 4.2 Implementing

4.2.1 Procedure, EP-0901, *Classifications*

4.2.2 PVNGS-TO-NEI 99-01, Rev. 6, EAL CROSS-REFERENCE

4.2.3 PVNGS EAL Matrix

## 5.0 DEFINITIONS, ABBREVIATIONS, & ACRONYMS

### 5.1 Definitions (ref. 4.1.1 except as noted)

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

**Alert**

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

**Confinement Boundary**

The barrier(s) between spent fuel and the environment once the spent fuel is processed for dry storage. As related to the PVNGS ISFSI, Confinement Boundary is defined as the Transportable Storage Canister (TSC) for the NAC-UMS.

**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 PVNGS, Containment Closure is established when the requirements of procedure 40EP-9EO10, *LM-Containment Evacuation and Closure*, Appendix 249, for containment closure are met (ref. 4.1.7).

**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 PVNGS 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 or feedwater leak on the secondary side of sufficient size to cause an uncontrolled drop in steam generator pressure or the steam generator to become completely depressurized.

**Fire**

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

**Fission Product Barrier Threshold**

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

**Flooding**

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

**General Emergency**

Events are in progress or have occurred which involve actual or imminent substantial core degradation or melting with potential for loss of containment integrity or hostile 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 PVNGS 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 PVNGS. Non-terrorism-based EALs should be used to address such activities (i.e., this may include violent acts between individuals in the owner controlled area).

**Hostile Force**

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

**Imminent**

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

**Impede(d)**

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

**Independent Spent Fuel Storage Installation (ISFSI)**

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

**Initiating Condition (IC)**

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

**Intrusion**

The act of entering without authorization. Discovery of a bomb in a specified area is indication of intrusion into that area by a hostile force.

**Maintain**

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

**Projectile**

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

**Plant or ISFSI Protected Area**

An area, located within the PVNGS Exclusion Area Boundary, encompassed by physical barriers and to which access is controlled per 10 CFR 73.55. The PVNGS Power Plant Protected Area and the ISFSI Protected Area are two Protected Areas located within the PVNGS OWNER CONTROLLED AREA (ref 4.1.10).

**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, pressurizer manway and safeties installed).

**Reduced Inventory**

Plant condition when fuel is in the reactor vessel and Reactor Coolant System level is less than or equal to the 111 foot elevation (ref. 4.1.11).

**Refueling Pathway**

The reactor refueling pool, fuel storage 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 10 CFR 50.2):

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

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

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

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

**Security Owner Controlled Area (SOCA)**

An area encompassed by physical barriers to which access is controlled. (ref 4.1.12).

**Security Condition**

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

**Site Area Emergency**

Events are in progress or have occurred which involve actual or likely major failures of plant functions needed for protection of the public or hostile 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**

The boundary of a reactor site beyond which the land or property is not owned, leased, or otherwise controlled by the licensee (ref. 4.1.10).

**Unisolable**

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

**Unplanned**

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

**Unusual Event**

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

**Valid**

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

**Visible Damage**

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

### 5.2 Abbreviations/Acronyms

°F ........ Degrees Fahrenheit
° ........ Degrees
AC ........ Alternating Current
AOP ........ Abnormal Operating Procedure
ATWS ........ Anticipated Transient Without Scram
CET ........ Core Exit Thermocouple
CDE ........ Committed Dose Equivalent
CFR ........ Code of Federal Regulations
CR ........ Control Room
CSFST ........ Critical Safety Function Status Tree
CTMT ........ Containment
DBA ........ Design Basis Accident
DC ........ Direct Current
DEF ........ Defueled
DG ........ Diesel Generator
EAL ........ Emergency Action Level
ECCS ........ Emergency Core Cooling System
ECL ........ Emergency Classification Level
EOC ........ Emergency Operations Center
EOF ........ Emergency Operations Facility
EOP ........ Emergency Operating Procedure
EPA ........ Environmental Protection Agency
EPABX ........ Electronic Private Automatic Branch Exchange
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
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 ...................................................................... Off-site Dose Calculation Manual
ORO ..................................................................................... Offsite Response Organization
OSC ...................................................................................... Operations Support Center
PA .................................................................................................................. Protected Area
PAG ...................................................................................... Protective Action Guideline
PPS ........................................................................................ Plant Protection System
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
Rep CET ........................................................ Representative Core Exit Thermocouple
RETS ..................................................... Radiological Effluent Technical Specifications
RFAT ........................................................................... Radiological Field Assessment Team
R(P)V ............................................................................... Reactor (Pressure) Vessel
RVLIS ........................................................... Reactor Vessel Level Indicating System
RVLMS ...................................................... Reactor Vessel Level Monitoring System
RWLIS ...................................................... Refueling Water Level Indicating System
RWT ............................................................................... Refueling Water Storage Tank
SAR ............................................................................................. Safety Analysis Report
SBO ........................................................................................................ Station Blackout
SBOG ............................................................................... Station Blackout Generator
SCBA.................................................................... Self-Contained Breathing Apparatus
SG ........................................................................................................ Steam Generator
SI ............................................................................................................... Safety Injection
SIAS ............................................................................ Safety Injection Actuation System

SOCA ........................................................................Security Owner Controlled Area
SPDS........................................................................ Safety Parameter Display System
SRO...................................................................................... Senior Reactor Operator
STSC.......................................................................Satellite Technical Support Center
SUT ............................................................................................Startup Transformer
TEDE....................................................................... Total Effective Dose Equivalent
TOAF .............................................................................................. Top of Active Fuel
TSC .......................................................................................Technical Support Center
UFSAR .........................................................Updated Final Safety Analysis Report
WOG............................................................................ Westinghouse Owners Group

## 6.0 PVNGS-TO-NEI 99-01, Rev. 6, EAL CROSS-REFERENCE

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

| PVNGS | 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 |
| RA2.1 | AA2 | 1 |
| RA2.2 | AA2 | 2 |
| RA2.3 | AA2 | 3 |
| RA3.1 | AA3 | 1 |
| RA3.2 | AA3 | 2 |
| RS1.1 | AS1 | 1 |
| RS1.2 | AS1 | 2 |
| RS1.3 | AS1 | 3 |
| RS2.1 | AS2 | 1 |
| RG1.1 | AG1 | 1 |
| RG1.2 | AG1 | 2 |
| RG1.3 | AG1 | 3 |
| RG2.1 | AG2 | 1 |
| CU1.1 | CU1 | 1 |
| CU1.2 | CU1 | 2 |

| PVNGS | 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 | 3 |
| CG1.1 | 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 |
| HU4.2 | HU4 | 2 |
| HU4.3 | HU4 | 3 |
| HU4.4 | HU4 | 4 |

| PVNGS | NEI 99-01, Rev. 6 | |
|---|---|---|
| **EAL** | **IC** | **Example EAL** |
| HU7.1 | HU7 | 1 |
| HA1.1 | HA1 | 1, 2 |
| HA5.1 | HA5 | 1 |
| HA6.1 | HA6 | 1 |
| HA7.1 | HA7 | 1 |
| HS1.1 | HS1 | 1 |
| HS6.1 | HS6 | 1 |
| HS7.1 | HS7 | 1 |
| HG7.1 | HG7 | 1 |
| SU1.1 | SU1 | 1 |
| SU3.1 | SU2 | 1 |
| SU4.1 | SU3 | 1 |
| SU4.2 | SU3 | 2 |
| SU5.1 | SU4 | 1, 2, 3 |
| 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 |
| SS2.1 | SS8 | 1 |
| SS6.1 | SS5 | 1 |

| PVNGS | NEI 99-01, Rev. 6 | |
|---|---|---|
| **EAL** | **IC** | **Example EAL** |
| SG1.1 | SG1 | 1 |
| SG1.2 | SG8 | 1 |
| EU1.1 | EU1 | 1 |

## 7.0 ATTACHMENTS

Attachment 1 - Emergency Action Level Technical Bases

Attachment 2 - Fission Product Barrier Loss/Potential Loss Matrix and Basis

Attachment 3 - Safe Operation & Shutdown Rooms Tables R-2 & H-2 Bases

Attachment 4 - Palo Verde Safety System List

**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. Radioactivity release through degradation of fission product barriers is not always apparent via non-radiological symptoms. Therefore, direct indication of elevated radiological effluents or area radiation levels are appropriate symptoms for emergency classification.

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

Events of this category pertain to the following subcategories:

1. Radiological Effluent

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

2. Irradiated Fuel Event

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

3. Area Radiation Levels

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

**Category:** R – Abnormal Rad Levels / Rad Effluent

**Subcategory:** 1 – Radiological Effluent

**Initiating Condition:** Release of gaseous radioactivity > 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 minutes (Notes 1, 2, 3)

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

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

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

**Table R-1 Effluent Monitor Classification Thresholds**

| Release Point | | Monitor | GE | SAE | Alert | UE |
|---|---|---|---|---|---|---|
| Gaseous | Plant Vent Low | RU-143 Ch 1 | ---- | ---- | 1.04E-02 µCi/cc | 1.22E-03 µCi/cc |
| Gaseous | Plant Vent High | RU-144 Ch 1 | 1.04E+00 µCi/cc | 1.04E-01 µCi/cc | ---- | ---- |
| Gaseous | Fuel Building Low | RU-145 Ch1 | ---- | ---- | ---- | 1.13E-02 µCi/cc |
| Gaseous | Fuel Building High | RU-146 Ch 1 | ---- | 3.50E+00 µCi/cc | 3.50E-01 µCi/cc | ---- |
| Gaseous | Fuel Building High | RU-146 Ch 2 | 3.50E+01 µCi/cc | ---- | ---- | ---- |

**Mode Applicability:**

All

**Definition(s):**

None

**Basis:**

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

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 radiological release, monitored or un-monitored.

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 effluent pathways.

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

**PVNGS Basis Reference(s):**

1. *Offsite Dose Calculation Manual Palo Verde Nuclear Generating Station Units 1, 2 and 3*
2. Letter 102-05894-DCM/CJS, Dated 9/15/08, *PVNGS Units 1, 2, 3, and ISFSI Docket Nos. 50-528, 50-529, 50-530, and 72-44 Proposed PVNGS Emergency Plan Change to Implement NEI 99-01, Revision 5, Emergency Action Levels (EALs) Attachment 1 Radiological Calculations*
3. NEI 99-01, AU1

**Category:** R – Abnormal Rad Levels / Rad Effluent

**Subcategory:** 1 – Radiological Effluent

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

**EAL:**

| **RU1.2** | **Unusual Event** |
|---|---|
| Sample analysis for a gaseous release indicates a concentration or release rate > 2 x ODCM limits for ≥ 60 minutes (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:**

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 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 releases that are detected by sample analyses or environmental surveys, particularly on unmonitored pathways.

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

**PVNGS Basis Reference(s):**

1. *Offsite Dose Calculation Manual Palo Verde Nuclear Generating Station Units 1, 2 and 3*
2. NEI 99-01, AU1

**Category:** R – Abnormal Rad Levels / Rad Effluent

**Subcategory:** 1 – Radiological Effluent

**Initiating Condition:** Release of gaseous 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 minutes (Notes 1, 2, 3, 4)

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

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

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

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

**Table R-1 Effluent Monitor Classification Thresholds**

| Release Point | | Monitor | GE | SAE | Alert | UE |
|---|---|---|---|---|---|---|
| Gaseous | Plant Vent Low | RU-143 Ch 1 | ---- | ---- | 1.04E-02 μCi/cc | 1.22E-03 μCi/cc |
| | Plant Vent High | RU-144 Ch 1 | 1.04E+00 μCi/cc | 1.04E-01 μCi/cc | ---- | ---- |
| | Fuel Building Low | RU-145 Ch1 | ---- | ---- | ---- | 1.13E-02 μCi/cc |
| | Fuel Building High | RU-146 Ch 1 | ---- | 3.50E+00 μCi/cc | 3.50E-01 μCi/cc | ---- |
| | | RU-146 Ch 2 | 3.50E+01 μCi/cc | ---- | ---- | ---- |

**Mode Applicability:**

All

**Definition(s):**

None

**Basis:**

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

- 10 mRem TEDE
- 50 mRem CDE Thyroid

The column “ALERT” gaseous effluent release values in Table R-1 correspond to calculated doses of 1% (10% of the SAE thresholds) 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 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.

**PVNGS Basis Reference(s):**

1. Letter 102-05894-DCM/CJS, Dated 9/15/08, *PVNGS Units 1, 2, 3, and ISFSI Docket Nos. 50-528, 50-529, 50-530, and 72-44, Proposed PVNGS Emergency Plan Change to Implement NEI 99-01, Revision 5, Emergency Action Levels (EALs) Attachment 1 Radiological Calculations*
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 10 mrem TEDE or 50 mrem thyroid CDE

**EAL:**

**RA1.2 Alert**

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

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

**Mode Applicability:**

All

**Definition(s):**

*SITE BOUNDARY* - The boundary of a reactor site beyond which the land or property is not owned, leased, or otherwise controlled by the licensee.

**Basis:**

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

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

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

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

**PVNGS Basis Reference(s):**

1. Procedure EP-0903, *Accident Assessment*
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 10 mrem TEDE or 50 mrem thyroid CDE |

**EAL:**

**RA1.3 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 minutes
- Analyses of field survey samples indicate thyroid CDE > 50 mrem for 60 minutes 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* - The boundary of a reactor site beyond which the land or property is not owned, leased, or otherwise controlled by the licensee.

**Basis:**

Procedure EP-0904, *ERO/ERF Activation and Operation*, provides guidance for emergency or post-accident radiological environmental monitoring (ref. 1).

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

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

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

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

**PVNGS Basis Reference(s):**

1. Procedure EP-0904, *ERO/ERF Activation and Operation*
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 minutes (Notes 1, 2, 3, 4)

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

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

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

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

**Table R-1 Effluent Monitor Classification Thresholds**

| Release Point | | Monitor | GE | SAE | Alert | UE |
|---|---|---|---|---|---|---|
| Gaseous | Plant Vent Low | RU-143 Ch 1 | ---- | ---- | 1.04E-02 μCi/cc | 1.22E-03 μCi/cc |
| | Plant Vent High | RU-144 Ch 1 | 1.04E+00 μCi/cc | 1.04E-01 μCi/cc | ---- | ---- |
| | Fuel Building Low | RU-145 Ch1 | ---- | ---- | ---- | 1.13E-02 μCi/cc |
| | Fuel Building High | RU-146 Ch 1 | ---- | 3.50E+00 μCi/cc | 3.50E-01 μCi/cc | ---- |
| | | RU-146 Ch 2 | 3.50E+01 μCi/cc | ---- | ---- | ---- |

**Mode Applicability:**

All

**Definition(s):**

None

**Basis:**

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

- 100 mRem TEDE
- 500 mRem CDE Thyroid

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

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.

**PVNGS Basis Reference(s):**

1. Letter 102-05894-DCM/CJS, Dated 9/15/08, *PVNGS Units 1, 2, 3, and ISFSI Docket Nos. 50-528, 50-529, 50-530, and 72-44 Proposed PVNGS Emergency Plan Change to Implement NEI 99-01, Revision 5, Emergency Action Levels (EALs) Attachment 1 Radiological 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.

**Mode Applicability:**

All

**Definition(s):**

*SITE BOUNDARY* - The boundary of a reactor site beyond which the land or property is not owned, leased, or otherwise controlled by the licensee.

**Basis:**

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

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

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

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

**PVNGS Basis Reference(s):**

1. Procedure EP-0903, *Accident Assessment*
2. NEI 99-01, AS1

**Category:** R – Abnormal Rad Levels / Rad Effluent

**Subcategory:** 1 – Radiological Effluent

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

**EAL:**

**RS1.3 Site Area Emergency**

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

- Closed window dose rates > 100 mR/hr expected to continue for ≥ 60 minutes
- Analyses of field survey samples indicate thyroid CDE > 500 mrem for 60 minutes 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* - The boundary of a reactor site beyond which the land or property is not owned, leased, or otherwise controlled by the licensee.

**Basis:**

Procedure EP-0904, *ERO/ERF Activation and Operation*, 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.

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.

**PVNGS Basis Reference(s):**

1. Procedure EP-0904, *ERO/ERF Activation and Operation*
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 minutes (Notes 1, 2, 3, 4)

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

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

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

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

**Table R-1 Effluent Monitor Classification Thresholds**

| | Release Point | Monitor | GE | SAE | Alert | UE |
|---|---|---|---|---|---|---|
| Gaseous | Plant Vent Low | RU-143 Ch 1 | ---- | ---- | 1.04E-02 µCi/cc | 1.22E-03 µCi/cc |
| Gaseous | Plant Vent High | RU-144 Ch 1 | 1.04E+00 µCi/cc | 1.04E-01 µCi/cc | ---- | ---- |
| Gaseous | Fuel Building Low | RU-145 Ch1 | ---- | ---- | ---- | 1.13E-02 µCi/cc |
| Gaseous | Fuel Building High | RU-146 Ch 1 | ---- | 3.50E+00 µCi/cc | 3.50E-01 µCi/cc | ---- |
| Gaseous | Fuel Building High | RU-146 Ch 2 | 3.50E+01 µCi/cc | ---- | ---- | ---- |

**Mode Applicability:**

All

**Definition(s):**

None

**Basis:**

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

- 1000 mRem TEDE
- 5000 mRem CDE Thyroid

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

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

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

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

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

**PVNGS Basis Reference(s):**

1. Letter 102-05894-DCM/CJS, Dated 9/15/08, *PVNGS Units 1, 2, 3, and ISFSI Docket Nos. 50-528, 50-529, 50-530, and 72-44 Proposed PVNGS Emergency Plan Change to Implement NEI 99-01, Revision 5, Emergency Action Levels (EALs) Attachment 1 Radiological 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.

**Mode Applicability:**

All

**Definition(s):**

*SITE BOUNDARY* - The boundary of a reactor site beyond which the land or property is not owned, leased, or otherwise controlled by the licensee.

**Basis:**

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

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

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

**PVNGS Basis Reference(s):**

1. Procedure EP-0903, *Accident Assessment*
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:**

**RG1.3 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 minutes
- Analyses of field survey samples indicate thyroid CDE > 5,000 mrem for 60 minutes 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* - The boundary of a reactor site beyond which the land or property is not owned, leased, or otherwise controlled by the licensee.

**Basis:**

Procedure EP-0904, *ERO/ERF Activation and Operation*, 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.

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.

**PVNGS Basis Reference(s):**

1. Procedure EP-0904, *ERO/ERF Activation and Operation*
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 (PCN-E02) or level indication (installed plant indicator/camera or RWLIS)

**AND**

UNPLANNED alert alarm on **any** of the following corresponding radiation monitors:

- RU-16 Containment Operating Level Area
- RU-17 Incore Instrument Area (when installed)
- RU-19 New Fuel Area
- RU-31 Spent Fuel Pool Area
- RU-33 Refueling Machine Area (when installed)

**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 pool, fuel storage pool and fuel transfer canal comprise the refueling pathway.

**Basis:**

The low water level alarm in this EAL refers to the Fuel Pool low level alarm (procedure 40AL-9PC01, *Fuel Pool Cooling and Cleanup Local Alarm Panel PCN-E02 Responses*) (ref. 1). During the fuel transfer phase of refueling operations, the fuel transfer canal is normally in communication with the fuel storage 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. (ref. 1, 2).

The SFP is locally monitored in the Fuel Building by Level indicators PCN-LIT-3/5 on PCNE02. These level indicating transmitters also initiate local panel alarms via level switches PCN-LSHL-3/PCN-LSL-5 on low and low low SFP level respectively. The alarms are also located on PCNE02 and annunciate a general Control Room alarm on window “FUEL POOL CLG SYS TRBL” indicating an alarm is in on the local panel.

Level is also indicated in the Control Room visually via digital camera feed and in the back panel area on panel PCN-E015 by a digital level indicator. This Control Room indication does not have associated annunciation.

Technical Specification LCO 3.7.14 (ref. 3) requires at least 23 ft of water above the Fuel Storage Pool storage racks. Technical Specification LCO 3.9.6 (ref. 4) 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 listed radiation monitors are those expected to see increased area radiation levels as a result of a loss of REFUELING PATHWAY inventory (ref. 2). Increasing radiation indications on these monitors in the absence of indications of decreasing REFUELING PATHWAY level are not classifiable under this EAL. The Alert alarms are set very low (3 X normal background) and would promptly alert operators of any increase in area radiation (ref. 5).

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.

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 with Recognition Category C during the Cold Shutdown and Refueling modes.

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

**PVNGS Basis Reference(s):**

1. Procedure 40AL-9PC01, *Fuel Pool Cooling and Cleanup Local Alarm Panel PCN-E02 Responses*
2. Procedure 40AO-9ZZ23, *Loss of SFP Level or Cooling*
3. Technical Specification LCO 3.7.14, *Fuel Storage Pool Water Level*
4. Technical Specification LCO 3.9.6, *Refueling Water Level – Fuel Assemblies*
5. Design Basis Manual – Radiation Monitoring System
6. 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 | |

**Mode Applicability:**

All

**Definition(s):**

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

**Basis:**

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

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

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

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

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

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

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

**PVNGS Basis Reference(s):**

1. Procedure 40AO-9ZZ23, *Loss of SFP Level or Cooling*
2. NEI 99-01, AA2

**Category:** R – Abnormal Rad Levels / Rad Effluent

**Subcategory:** 2 – Irradiated Fuel Event

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

**EAL:**

**RA2.2 Alert**

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

- RU-16 Containment Operating Level Area
- RU-17 Incore Instrument Area (when installed)
- RU-19 New Fuel Area
- RU-31 Spent Fuel Pool Area
- RU-33 Refueling Machine Area (when installed)
- RU-37/38 Containment Purge Exhaust Area
- RU-143 Plant Vent
- RU-145 Fuel Building Vent

**Mode Applicability:**

All

**Definition(s):**

None

**Basis:**

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

This IC addresses events that have caused 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.

**PVNGS Basis Reference(s):**

1. Design Basis Manual – Radiation Monitoring System
2. Procedure 40AO-9ZZ22, *Fuel Damage*
3. NEI 99-01, AA2

**Category:** R – Abnormal Rad Levels / Rad Effluent

**Subcategory:** 2 – Irradiated Fuel Event

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

**EAL:**

**RA2.3 Alert**

Spent fuel pool level ≤ 125 ft. (Level 2)

**Mode Applicability:**

All

**Definition(s):**

None

**Basis:**

For PVNGS, Level 2, which corresponds to 10 ft. above the top of the fuel racks in the SFP (9 ft. based on instrument indication margin), is an indicated level of 125 ft. (ref. 2).

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.

**PVNGS Basis Reference(s):**

1. Letter 102-06728, dated July 11, 2013, and Adams Accession #13199A033, *Response to Request for Additional Information for the PVNGS Overall Integrated Plan in Response to March 12, 2012, Commission Order Modifying License with Regard to Reliable Spent Fuel Pool Level Instrumentation (Order Number EA-12-051)*
2. Evaluation 4512970
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**

Spent fuel pool level ≤ 116 ft. (Level 3)

**Mode Applicability:**

All

**Definition(s):**

None

**Basis:**

For PVNGS, Level 3, which corresponds to 0 ft. above the top of the fuel racks in the SFP, is an indicated level of 116 ft. (includes a 1 ft.10 inches instrument indication error margin) (ref. 2).

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

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

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

**PVNGS Basis Reference(s):**

1. Letter 102-06728, dated July 11, 2013 and Adams Accession #13199A033, *Response to Request for Additional Information for the PVNGS Overall Integrated Plan in Response to March 12, 2012 Commission Order Modifying License with Regard to Reliable Spent Fuel Pool Level Instrumentation (Order Number EA-12-051)*
2. Evaluation 4512970
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 116 ft. (Level 3) for ≥ 60 minutes (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:**

For PVNGS, Level 3, which corresponds to 0 ft. above the top of the fuel racks in the SFP, is an indicated level of 116 ft. (includes a 1 ft.10 inches instrument indication error margin) (ref. 2).

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.

**PVNGS Basis Reference(s):**

1. Letter 102-06728, dated July 11, 2013 and Adams Accession #13199A033, *Response to Request for Additional Information for the PVNGS Overall Integrated Plan in Response to March 12, 2012 Commission Order Modifying License with Regard to Reliable Spent Fuel Pool Level Instrumentation (Order Number EA-12-051)*
2. Evaluation 4512970
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

**OR**

Central Alarm Station (CAS) (by survey)

**Mode Applicability:**

All

**Definition(s):**

None

**Basis:**

Areas that meet this threshold include the Control Room and the Central Alarm Station (CAS). The Radiation Monitoring System monitors the Control room for area radiation (ref. 1). If unavailable local radiation surveys can be performed. 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 monitor that may be used to assess this EAL threshold. Therefore this threshold must be assessed via local radiation survey for the CAS (ref. 1).

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.

**PVNGS Basis Reference(s):**

1. Design Basis Manual – *Radiation Monitoring System*
2. NEI 99-01, AA3

**Category:** R – Abnormal Rad Levels / Rad Effluent

**Subcategory:** 3 – Area Radiation Levels

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

**EAL:**

**RA3.2 Alert**

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

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

| **Table R-2 Safe Operation & Shutdown Rooms** | |
|---|---|
| **Room** | **Mode Applicability** |
| Control Building 100 ft. Class DC Equipment Room C | 4, 5 |
| Control Building 100 ft. Class DC Equipment Room D | 4, 5 |

**Mode Applicability:**

4 – Hot Shutdown, 5 – Cold Shutdown

**Definition(s):**

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

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

**Basis:**

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

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

This IC addresses elevated radiation levels in certain plant rooms/areas sufficient to preclude or impede personnel from performing actions necessary to maintain normal plant operation, or to perform a normal plant cooldown and shutdown. As such, it represents an actual or

potential substantial degradation of the level of safety of the plant. The Emergency Coordinator should consider the cause of the increased radiation levels and determine if another IC may be applicable.

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

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

- The plant is in an operating mode different than the mode specified for the affected room/area (i.e., entry is not required during the operating mode in effect at the time of the elevated radiation levels). For example, the plant is in Mode 1 when the radiation increase occurs and the procedures used for normal operation, cooldown and shutdown do not require entry into the affected room until Mode 4.
- The increased radiation levels are a result of a planned activity that includes compensatory measures which address the temporary inaccessibility of a room or area (e.g., radiography, spent filter or resin transfer, etc.).
- The action for which room/area entry is required is of an administrative or record keeping nature (e.g., normal rounds or routine inspections).
- The access control measures are of a conservative or precautionary nature and would not actually prevent or impede a required action.

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

**NOTE**: EAL RA3.2 mode applicability has been limited to the applicable modes identified in Table R-2 Safe Operation & Shutdown Rooms/Areas. If due to plant operating procedure or plant configuration changes, the applicable plant modes specified in Table R-2 are changed, a corresponding change to Attachment 3 'Safe Operation & Shutdown Areas Tables R-2 & H-2 Bases' and to EAL RA3.2 mode applicability is required.

**PVNGS Basis Reference(s):**

1. Attachment 3 - Safe Operation & Shutdown Areas Tables R-3 & H-2 Bases
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 environment for there to be a significant environmental effect resulting from an accident involving the dry storage of spent nuclear fuel.

A Notification of 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 canister CONFINEMENT BOUNDARY as indicated by an on-contact radiation reading on the surface of a loaded spent fuel cask greater than **any** of the following:

- 100 mrem/hr (neutron + gamma) on the side of the cask
- 100 mrem/hr (neutron + gamma) on the top of the cask
- 200 mrem/hr (neutron + gamma) at the air inlets or outlets

**Mode Applicability:**

All

**Definition(s):**

*CONFINEMENT BOUNDARY* - The barrier(s) between spent fuel and the environment once the spent fuel is processed for dry storage. As related to the PVNGS ISFSI, Confinement Boundary is defined as the Transportable Storage Canister (TSC) for the NAC-UMS.

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

**Basis:**

The PVNGS ISFSI utilizes the NAC-UMS dry spent fuel storage system for dry spent fuel storage.

The system consist of a Transportable Storage Canister (TSC) and concrete Vertical Concrete Cask (VCC). The TSC is the CONFINEMENT BOUNDARY. The TSC is welded and designed to provide confinement of all radionuclides under normal, off-normal and accident conditions (ref. 1, 2).

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

The values shown represent 2 times the limits specified in the ISFSI Certificate of Compliance (C of C) Technical Specification for radiation external to a loaded TSC for a NAC-UMS canister (ref. 1)

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 ISFSI C of C 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.

**PVNGS Basis Reference(s):**

1. *USNRC Certificate of Compliance for NAC International’s UMS Spent Fuel Storage Casks No. 1015, Amendment 5 ,Appendix A,Technical Specifications for the NAC-UMS System*
2. NEI 99-01, E-HU1

**Category C – Cold Shutdown / Refueling System Malfunction**

EAL Group: Cold Conditions (RCS temperature ≤ 210ºF); 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 minutes (Notes 1, 10) | |

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

Note 10: Variations in RCS boron concentration, temperature and Containment Temperature from those used in RWLIS calibration will induce indication errors. Refer to *Operator Assistance Program RWLIS_Spreadsheet.xls*.

**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:**

With the plant in Cold Shutdown, RCS water level is normally maintained above the partial drain condition of 10% pressurizer level (117 ft. RWLIS W.R.) (ref. 1). However, if RCS level is being controlled below the pressurizer partial drain setpoint, or if level is being maintained in a designated band in the reactor vessel it is the inability to maintain level above the low end of the designated control band due to a loss of inventory resulting from a leak in the RCS that is the concern.

With the plant in Refueling mode, RCS water level is normally maintained at or above the reactor vessel flange (Technical Specification LCO 3.9.6 requires at least 23 ft. of water above the top of the reactor vessel flange in the refueling pool during refueling operations) (ref. 2).

Procedure 40OP-9ZZ16, *RCS Drain Operations,* provides direction regarding variations in RCS boron concentration, temperature and Containment Temperature from those used in RWLIS calibration will induce indication errors, which are addressed by a controlled program, *Operator Assistance Program RWLIS_Spreadsheet.xls* (ref. 1).

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.

**PVNGS Basis Reference(s):**

1. Procedure 40OP-9ZZ16, *RCS Drain Operations*
2. Technical Specification LCO 3.9.6, *Refueling Water Level – Fuel Assemblies*
3. NEI 99-01, CU1

**Category:** C – Cold Shutdown / Refueling System Malfunction

**Subcategory:** 1 – RCS Level

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

**EAL:**

**CU1.2 Unusual Event**

RCS level cannot be monitored

**AND EITHER**

- UNPLANNED increase in **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<br>• Reactor Cavity Sump<br>• Auxiliary Building Sumps<br>• CVCS Holdup Tank<br>• Reactor Drain Tank<br>• Refueling Water Tank<br>• Equipment Drain Tank |

**Mode Applicability:**

5 - Cold Shutdown, 6 – Refueling

**Definition(s):**

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

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

**Basis:**

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

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

In this EAL, all water level indication is unavailable and the RCS inventory loss must be detected by indirect leakage indications (Table C-1). 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 significant leakage from systems connected to the RCS that cannot be isolated could also be indicative of a loss of RCS inventory (ref. 1, 2).

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

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

This EAL addresses a condition where all means to determine 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.

**PVNGS Basis Reference(s):**

1. Procedure 40AO-9ZZ02, *Excessive RCS Leakrate*
2. Procedure 40OP-9ZZ16, *RCS Drain Operations*
3. 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 RCS level < 101 ft. 6 in.
(RWLIS NR RCN-LI-752A/RCN-LR-752)

**Mode Applicability:**

5 - Cold Shutdown, 6 – Refueling

**Definition(s):**

None

**Basis:**

RCS water level, as indicated on RWLIS narrow range (RCN-LI-752A or RCN-LR-752), of 101 ft 6 in., corresponds to 2 inches above the RCS Hot Leg centerline and is the lowest level for continued operation of normal shutdown cooling (SDC) (ref. 1).

The inability to restore and maintain level after reaching this setpoint infers a failure of the RCS barrier.

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

For this EAL, a lowering of RCS water level below 101 ft. 6 in. indicates that operator actions have not been successful in restoring and maintaining RCS water level. The heat-up rate of the coolant will increase as the available water inventory is reduced. A continuing decrease in water level will lead to core uncovery.

Although related, this EAL is concerned with the loss of RCS inventory and not the potential concurrent effects on systems needed for decay heat removal (e.g., loss of a 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.

**PVNGS Basis Reference(s):**

1. Procedure 40OP-9ZZ16, *RCS Drain Operations*
2. NEI 99-01, CA1

**Category:** C – Cold Shutdown / Refueling System Malfunction

**Subcategory:** 1 – RCS Level

**Initiating Condition:** Loss of RCS inventory

**EAL:**

**CA1.2 Alert**

RCS level cannot be monitored for ≥ 15 minutes (Note 1)

**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

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<br>• Reactor Cavity Sump<br>• Auxiliary Building Sumps<br>• CVCS Holdup Tank<br>• Reactor Drain Tank<br>• Refueling Water Tank<br>• Equipment Drain Tank |

**Mode Applicability:**

5 - Cold Shutdown, 6 – Refueling

**Definition(s):**

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

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

**Basis:**

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

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

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

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

For this EAL, 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.

**PVNGS Basis Reference(s):**

1. Procedure 40AO-9ZZ02, *Excessive RCS Leakrate*
2. Procedure 40OP-9ZZ16, *RCS Drain Operations*
3. 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**

RCS level cannot be monitored for ≥ 30 minutes (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
- RU-33 ≥ 9,000 mR/hr (when installed)
- Erratic Excore Monitor indication

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

| Table C-1 Sumps / Tanks |
|---|
| • Containment Sumps<br>• Reactor Cavity Sump<br>• Auxiliary Building Sumps<br>• CVCS Holdup Tank<br>• Reactor Drain Tank<br>• Refueling Water Tank<br>• Equipment Drain Tank |

**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:**

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.

The bottom of the RWLIS indication is 99' 7". If level lowers less than 99' 7" then level would not be able to be monitored. If RWLIS is not in service then when RVLMS is < 21 % plenum level (Detector #8) level would not be able to be monitored.

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

Sump or tank level increases should be of a magnitude that correlates to a volume sufficient to indicate fuel has been uncovered or uncovery is imminent.

The Reactor Vessel inventory loss may be detected by the refueling machine area radiation monitor or erratic Excore 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 (10,000 mR/hr) refueling machine area radiation monitor (RU-33) indication. A threshold value of 90% of scale has been selected as an on-scale indicator (ref. 3, 4).

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. 5).

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.

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

**PVNGS Basis Reference(s):**

1. Procedure 40AO-9ZZ02, *Excessive RCS Leakrate*
2. Procedure 40OP-9ZZ16, *RCS Drain Operations*
3. UFSAR Table 11.5-1, Continuous Process and Effluent Radiation Monitoring
4. UFSAR Section 11.5.2.1.5.4, Refueling Area Monitor
5. Nuclear Safety Analysis Center (NSAC), 1980, *Analysis of Three Mile Island - Unit 2 Accident*, NSAC-1
6. 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**

RCS level cannot be monitored for ≥ 30 minutes (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
- RU-33 ≥ 9,000 mR/hr (when installed)
- Erratic Excore Monitor indication

**AND**

**Any** Containment Challenge indication, Table C-2

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

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

| Table C-1 Sumps / Tanks |
|---|
| • Containment Sumps<br>• Reactor Cavity Sump<br>• Auxiliary Building Sumps<br>• CVCS Holdup Tank<br>• Reactor Drain Tank<br>• Refueling Water Tank<br>• Equipment Drain Tank |

| Table C-2 Containment Challenge Indications |
|---|
| • CONTAINMENT CLOSURE **not** established (Note 6)<br>• Containment hydrogen concentration ≥ 4.5%<br>• Unplanned rise in containment pressure |

**Mode Applicability:**

5 - Cold Shutdown, 6 – Refueling

**Definition(s):**

*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 PVNGS, Containment Closure is established when the requirements of procedure 40EP-9EO10, *LM-Containment Evacuation and Closure*, Appendix 249, for 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.

**Basis:**

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.

The bottom of the RWLIS indication is 99' 7". If level lowers less than 99' 7" then level would not be able to be monitored. If RWLIS is not in service then when RVLMS is < 21 % plenum level (Detector #8) level would not be able to be monitored.

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

Sump or tank level increases should be of a magnitude that correlates to a volume sufficient to indicate fuel has been uncovered or uncovery is imminent.

The Reactor Vessel inventory loss may be detected by the refueling machine area radiation monitor or erratic Excore 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 (10,000 mR/hr) refueling machine area radiation monitor (RU-33) indication. A threshold value of 90% of scale has been selected as an on-scale indicator (ref. 3, 4).

Post-TMI accident studies indicate 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. 5).

Three conditions are associated with a challenge to Containment integrity:

1. CONTAINMENT CLOSURE not established - The status of Containment closure is tracked if plant conditions change that could raise the risk of a fission product release as a result of a loss of decay heat removal (ref. 6). 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 ≥ 4.5% - The 4.5% hydrogen concentration threshold represents the Hydrogen Recombiners Function Failure Indication (ref. 11) and is the acceptance criteria for the PVNGS Safety Function Status Check for LOCA, Containment Combustible Gas Control (ref.7, 8, 10,). PVNGS 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. 8). Two containment hydrogen monitors have a range of 0% to 10% (ref. 8, 9). Since the hydrogen monitoring system may be out of service in Modes 5 and 6, alternative means of determining hydrogen concentration may be required if the Emergency Coordinator believes conditions exist that may cause hydrogen generation inside containment.
3. UNPLANNED rise in containment pressure - An unplanned pressure rise in containment while in cold shutdown or refueling modes can threaten Containment Closure capability and thus containment potentially cannot be relied upon as a barrier to fission product release.

This 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*.

**PVNGS Basis Reference(s):**

1. Procedure 40AO-9ZZ02, *Excessive RCS Leakrate*
2. Procedure 40OP-9ZZ16, *RCS Drain Operations*
3. UFSAR Table 11.5-1, Continuous Process and Effluent Radiation Monitoring
4. UFSAR Section 11.5.2.1.5.4, Refueling Area Monitor
5. Nuclear Safety Analysis Center (NSAC), 1980, *Analysis of Three Mile Island - Unit 2 Accident,* NSAC-1
6. Procedure 40EP-9EO10, *LM-Containment Evacuation and Closure,* Appendix 249
7. Procedure 40DP-9AP08, *Loss of Coolant Accident Technical Guideline*
8. UFSAR Section 1.2.4.2, Additional PVNGS Engineered Safety Features
9. UFSAR Table 6.2.5-1, Combustible Gas Control System Design Parameters
10. Procedure 40EP-9EO03, *Loss of Coolant Accident*
11. Nuclear Fuel Management Analysis Calculation TA-13-C00-2000-001*, EOP Setpoint Document*
12. NEI 99-01, CG1

**Category:** C – Cold Shutdown / Refueling System Malfunction

**Subcategory:** 2 – Loss of Emergency AC Power

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

**EAL:**

**CU2.1 Unusual Event**

AC power capability, Table C-3, to emergency 4.16KV buses PBA-S03 and PBB-S04 reduced to a single power source for ≥ 15 minutes (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:**<br>• SUT (normal)<br>• SUT (alternate)<br>• SBOG #1 (if already aligned)<br>• SBOG #2 (if already aligned)<br>**Onsite:**<br>• DG A<br>• DG B |

**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 10 CFR 50.2):

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

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

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

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

**Basis:**

For emergency classification purposes, “capability” means that an AC power source is available to and capable of powering the emergency bus(es) within 15 min, whether or not the buses are currently 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 PBA-S03 and PBB-S04 are the emergency (essential) buses. PBA-S03 supplies power to Train A safety related loads and PBB-S04 supplies power to Train B safety related loads. Each bus has two normal sources of offsite power. Each source is from one of three 13.8 KV Startup Transformers (SUT) via its normal and alternative ESF Service Transformer NAN-X03 and the alternate supply to PBB-S04 or NAN-X04. Transformer NAN-X03 is the normal supply to bus PBA-S03 and the alternate supply to PBB-S04; Transformer NAN-X04 is the normal supply to bus PBB-S04 and the alternate supply to PBA-S03 (ref. 1).

In addition, PBA-S03 and PBB-S04 each have an emergency diesel generator (DG A & DG B) which supply electrical power to the bus automatically in the event that the preferred source becomes unavailable (ref. 1).

Additional alternate offsite AC power sources are the two redundant 13.8KV SBO gas turbine generators (SBOG #1 & SBOG #2). However, these sources can only be credited if already aligned, that is, capable of powering one or more emergency bus within 15 minutes. Each SBOG is rated at approximately 3.4 MW and can supply the shutdown SAFETY SYSTEM loads in Modes 5, 6 and Defueled.

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

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 EOP 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 fed from an SBOG.
- A loss of emergency power sources (e.g., onsite diesel generators) with a single train of emergency buses being fed from an offsite power source.

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

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

**PVNGS Basis Reference(s):**

1. Drawing 13-E-MAA-001, *Main Single Line Diagram*
2. UFSAR Section 8.3.1, AC Power Systems
3. Procedure 40AO-9ZZ12, *Degraded Electrical Power*
4. UFSAR Section 1.2.10.3.9, Alternate AC Power System
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 to emergency 4.16KV buses PBA-S03 and PBB-S04 for ≥ 15 minutes (Note 1) | |

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

**Mode Applicability:**

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

**Basis:**

For emergency classification purposes, “capability” means that an AC power source is available to and capable of powering the emergency bus(es) within 15 min, whether or not the buses are currently powered from it.

4.16KV buses PBA-S03 and PBB-S04 are the emergency (essential) buses. PBA-S03 supplies power to Train A safety related loads and PBB-S04 supplies power to Train B safety related loads. Each bus has two normal sources of offsite power. Each source is from one of three 13.8 KV Startup Transformers (SUT) via its normal and alternative ESF Service Transformer NAN-X03 or NAN-X04. Transformer NAN-X03 is the normal supply to bus PBA-S03 and the alternate supply to PBB-S04; Transformer NAN-X04 is the normal supply to bus PBB-S04 and the alternate supply to PBA-S03 (ref. 1).

In addition, PBA-S03 and PBB-S04 each have an emergency diesel generator (DG A & DG B) which supply electrical power to the bus automatically in the event that the preferred source becomes unavailable (ref. 1).

Additional alternate offsite AC power sources include, but not limited to, the two redundant 13.8KV SBO gas turbine generators (SBOG #1 & SBOG #2). However, these sources can only be credited if already aligned, that is, capable of powering one or more emergency bus within 15 minutes. Each SBOG is rated at approximately 3.4 MW and can supply the shutdown SAFETY SYSTEM loads in Modes 5, 6 and Defueled.

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.

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.

**PVNGS Basis Reference(s):**

1. Drawing 13-E-MAA-001, *Main Single Line Diagram*
2. UFSAR Section 8.3.1, AC Power Systems
3. Procedure 40AO-9ZZ12, *Degraded Electrical Power*
4. UFSAR Section 1.2.10.3.9, Alternate AC Power System
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 > 210°F | |

**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:**

Several instruments are capable of providing indication of RCS temperature with respect to the Technical Specification cold shutdown temperature limit (210°F, ref. 1). These include cold leg ($T_{cold}$) temperature indications, hot leg ($T_{hot}$) temperature indications with RCPs running, CETs and SDC Heat Exchanger inlet temperature indications (ref. 2, 3).

However, if Shutdown Cooling (SDC) flow is lost, then the normal temperature elements used to monitor RCS temperature are not accurate indicators of RCS temperature. The CETs are the design instruments for these conditions. For some periods of time the CETs may not be available. The current practices concerning determining time to boil can be used in the evaluation of these EALs. Without CET indication and with a loss of SDC flow the following guidance should be used (ref. 4):

- Use the predetermined "time to boil" data for evaluating these EALs. This approach reflects the relatively small numerical difference between the typical Technical Specification cold shutdown temperature limit of 210°F and the boiling temperature of RCS water with the plant in Mode 5 or 6.
- Alternately, the Control Room staff may use a procedure or user aid to determine when RCS temperature will likely exceed 210°F given the actual plant conditions (e.g., using a heat-up curve).

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

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

This EAL involves a loss of decay heat removal capability, or an addition of heat to the RCS in excess of that which can currently be removed, such that reactor coolant temperature cannot be maintained below the cold shutdown temperature limit specified in Technical Specifications.

During this condition, there is no immediate threat of fuel damage because the core decay heat load has been reduced since the cessation of power operation.

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

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

**PVNGS Basis Reference(s):**

1. Technical Specifications Table 1.1-1, *Modes*
2. Procedure 40OP-9ZZ03, *Reactor Startup*
3. Procedure 40ST-9RC01, *RCS and Pressurizer Heatup and Cooldown Rates*
4. Safety Analysis Operational Data Book
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 minutes (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:**

Several instruments are capable of providing indication of RCS temperature with respect to the Technical Specification cold shutdown temperature limit (210°F, ref. 1). These include cold leg ($T_{cold}$) temperature indications, hot leg ($T_{hot}$) temperature indications with RCPs running, CETs and SDC Heat Exchanger inlet temperature indications (ref. 2, 3).

Several instruments are capable of providing indication of RCS level including pressurizer level, RWLIS, RVLMS and local monitor (gauge glass) (ref. 4).

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

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

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

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

**PVNGS Basis Reference(s):**

1. Technical Specification Table 1.1-1, *Modes*
2. Procedure 40OP-9ZZ03, *Reactor Startup*
3. Procedure 40ST-9RC01, *RCS and Pressurizer Heatup and Cooldown Rates*
4. Procedure 40OP-9ZZ16, *RCS Drain Operations*
5. NEI 99-01, CU3

**Category:** C – Cold Shutdown / Refueling System Malfunction

**Subcategory:** 3 – RCS Temperature

**Initiating Condition:** Inability to maintain plant in cold shutdown

**EAL:**

**CA3.1 Alert**

UNPLANNED increase in RCS temperature to > 210°F for > Table C-4 duration (Note 1)

**OR**

UNPLANNED RCS pressure increase > 10 psia (This criterion does not apply during water-solid plant conditions)

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

<table>
<tr><th colspan="3">Table C-4: RCS Heat-up Duration Thresholds</th></tr>
<tr><th>RCS Status</th><th>CONTAINMENT CLOSURE Status</th><th>Heat-up Duration</th></tr>
<tr><td>Intact (but not REDUCED INVENTORY)</td><td>N/A</td><td>60 minutes.*</td></tr>
<tr><td rowspan="2">Not intact<br>OR<br>REDUCED INVENTORY</td><td>Established</td><td>20 minutes.*</td></tr>
<tr><td>Not Established</td><td>0 minutes.</td></tr>
<tr><td colspan="3">* If an RCS heat removal system is in operation within this time frame and RCS temperature is being reduced, the EAL is not applicable.</td></tr>
</table>

**Mode Applicability:**

5 - Cold Shutdown, 6 – Refueling

**Definition(s):**

*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 PVNGS, Containment Closure is established when the requirements of procedure 40EP-9EO10, *LM-Containment Evacuation and Closure*, Appendix 249, for 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 less than or equal to the 111 foot elevation.

**Basis:**

Several instruments are capable of providing indication of RCS temperature with respect to the Technical Specification cold shutdown temperature limit (210°F, ref. 1). These include cold leg ($T_{cold}$) temperature indications, hot leg ($T_{hot}$) temperature indications with RCPs running, CETs and SDC Heat Exchanger inlet temperature indications (ref. 2, 3).

However, if Shutdown Cooling (SDC) flow is lost, then the normal temperature elements used to monitor RCS temperature are not accurate indicators of RCS temperature. The CETs are the design instruments for these conditions. For some periods of time the CETs may not be available. The current practices concerning determining time to boil can be used in the evaluation of these EALs. Without CET indication and with a loss of SDC flow the following guidance should be used (ref. 4):

- Use the predetermined "time to boil" data for evaluating these EALs. This approach reflects the relatively small numerical difference between the typical Technical Specification cold shutdown temperature limit of 210°F and the boiling temperature of RCS water with the plant in Mode 5 or 6.
- Alternately, the Control Room staff may use a procedure or user aid to determine when RCS temperature will likely exceed 210°F given the actual plant conditions (e.g., using a heat-up curve).

RCS pressure instruments RCA PI-103, RCC-PI-105, RCD-PI-106 and RCB-PI-104 are capable of measuring pressure to less than 10 psia (ref. 3).

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.

**PVNGS Basis Reference(s):**

1. Technical Specification Table 1.1-1, *Modes*
2. Procedure 40OP-9ZZ03, *Reactor Startup*
3. Procedure 40ST-9RC01, *RCS and Pressurizer Heatup and Cooldown Rates*
4. Safety Analysis Operational Data Book
5. NEI 99-01, CA3

**Category:** C – Cold Shutdown / Refueling System Malfunction

**Subcategory:** 4 – Loss of Vital DC Power

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

**EAL:**

| **CU4.1** | **Unusual Event** |
|---|---|
| Indicated voltage is < 112VDC on vital DC buses required by Technical Specifications for ≥ 15 minutes (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:**

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

| Train A: | Train B: |
|---|---|
| • PKA-M41 | • PKB-M42 |
| • PKC-M43 | • PKD-M44 |

There are four, 60 cell, lead-calcium storage batteries (PKA-F11, PKC-F13, PKB-F12 and PKD-F14) 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 capacity of the battery chargers (ref. 1).

All four of the 125VDC buses supply inverters for 120VAC PN bus power as well as control power for various safety related systems. Each battery is designed to have sufficient stored energy to supply the required emergency loads for 120 minutes following a loss of AC power to the chargers (ref. 2).

Minimum DC bus voltage is 112 VDC (ref. 3).

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.

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.

**PVNGS Basis Reference(s):**

1. Drawing 01-E-PKA-001, *Main Single Line Diagram 125V DC Class 1E and 120VAC Vital Inst Power System*
2. UFSAR Section 8.3.2, DC Power Systems
3. Calculation 01-EC-PK-0207, *DC Battery Sizing and Minimum Voltage*
4. 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 Offsite Response Organization (ORO) communication methods

**OR**

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

**Table C-5 Communication Methods**

| System | Onsite | ORO | NRC |
|---|---|---|---|
| PBX | X | X | X |
| Plant Page | X | | |
| Two-Way Radio | X | | |
| FTS (ENS) | | | X |
| Telephone Ringdown Circuits (NAN) | | X | |
| Cellular Phones | | X | X |

**Mode Applicability:**

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

**Definition(s):**

None

**Basis:**

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

1. PBX

   Onsite emergency telephone lines are divided among three onsite EPABX switches. Each EPABX switch is provided with a backup battery for reliability.

This system will function during emergencies as it does during normal operations. Telephones have the capability of trunk access (via local provider) and the APS owned private communications system which provides direct dial capabilities to the entire APS voice system via the company owned private communications system. The PVNGS telephone EPABX Systems through which all PVNGS telephone calls pass, are equipped with uninterruptible power supplies (battery chargers and batteries) and dedicated priority switching to ensure the reliability of the telephone system. The PVNGS EPABXs are the primary links for PVNGS phones. There are also administratively dedicated lines for the CR, STSC, TSC, EOF and OSC.

2. Plant (Area) Paging

   The area paging system provides a reliable means of notifying and providing instructions to onsite personnel. Access to this system is through the EPABX system telephones by use of dedicated numbers.

3. Two-Way Radios

   PVNGS operates a trunked radio system, with separate talk groups available for departments such as Operations, Security, Fire Protection, Radiation Protection, Emergency Preparedness, the Water Reclamation Facility, etc. This system includes base station consoles at various locations and emergency facilities throughout the site. Some of the radios used during emergencies are portable radios at various site locations, mobile radios in the RFAT vehicles and base station consoles at the TSC, EOF, Unit OSCs, Unit STSCs and Unit Control Rooms. PVNGS Fire Protection also maintains radios that are used to contact the air ambulance service to provide landing instructions.

4. FTS (ENS)

   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.

5. Telephone Ringdown Circuits (NAN)

   These voice circuits serve as a primary communications link for providing technical information to offsite agencies, public information communications and the communication of protective action recommendations to offsite authorities.

6. Cellular Phones

   Each STSC, the TSC and EOF have a cellular phone to provide additional independent lines of communication.

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.

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 and Maricopa County EOCs.

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

**PVNGS Basis Reference(s):**

1. *PVNGS Emergency Plan*, Section 7.2 Communications Systems
2. UFSAR Section 9.5.2, Communication Systems
3. NEI 99-01, CU5

**Category:** C – Cold Shutdown / Refueling System Malfunction

**Subcategory:** 6 – Hazardous Event Affecting Safety Systems

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

**EAL:**

**CA6.1 Alert**

The occurrence of **any** Table C-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)<br>• Internal or external FLOODING event<br>• High winds or tornado strike<br>• FIRE<br>• EXPLOSION<br>• Other events with similar hazard characteristics as determined by the Shift Manager |

**Mode Applicability:**

5 - Cold Shutdown, 6 - Refueling

**Definition(s):**

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

*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 10 CFR 50.2):

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

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

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

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

*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:**

Refer to Attachment 4 for a list of Palo Verde SAFETY SYTEMS (ref. 5).

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

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

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

- The significance of seismic events are discussed under EAL HU2.1. Annunciator 7C14A, SEISMIC OCCURRENCE will illuminate if the seismic instrument detects ground motion in excess of the seismic EVENT trigger threshold (ref. 1).
- Internal FLOODING may be caused by events such as component failures, equipment misalignment, or outage activity mishaps.
- High winds in excess of design (105 mph) or tornado strikes can cause significant structural damage (ref. 4).
- Areas containing functions and systems required for safe shutdown of the plant are identified by fire area (ref. 2).
- 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.

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

**PVNGS Basis Reference(s):**

1. Procedure 40AO-9ZZ21, *Acts of Nature*
2. UFSAR Table 3-2.1, Quality Classification of Structures, Systems and Components
3. UFSAR Section 2.4.2.2.1, Offsite Flood Design Considerations
4. UFSAR Section 2.3.1.2.3, Extreme Winds
5. Attachment 4 - Palo Verde Safety Systems
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 Plant 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 Plant Protected Area or which may affect operability of equipment needed for safe shutdown

5. Hazardous Gas

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

6. Control Room Evacuation

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

7. Emergency 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 Supervision

**OR**

Notification of a credible security threat directed at the site

**OR**

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

**Mode Applicability:**

All

**Definition(s):**

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

*HOSTILE ACTION* - An act toward PVNGS 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 PVNGS. 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:**

This EAL is based on the *PVNGS Security Plan, Training and Qualification Plan, Safeguards Contingency Plan and Independent Spent Fuel Storage Installation Security Program* (ref. 1).

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

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

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

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

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

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 PVNGS Security Plan (ref. 1).

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

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

**PVNGS Basis Reference(s):**

1. *PVNGS Security Plan, Training and Qualification Plan, Safeguards Contingency Plan and Independent Spent Fuel Storage Installation Security Program (Safeguards)*
2. NEI 99-01, HU1

**Category:** H – Hazards

**Subcategory:** 1 – Security

**Initiating Condition:** Hostile action within the SECURED OWNER CONTROLLED AREA or airborne attack threat within 30 minutes

**EAL:**

**HA1.1 Alert**

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

**OR**

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

**Mode Applicability:**

All

**Definition(s):**

*HOSTILE ACTION* - An act toward PVNGS 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 PVNGS. 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).

*SECURED OWNER CONTROLLED AREA* - An area encompassed by physical barriers to which access is controlled.

**Basis:**

This IC addresses the occurrence of a HOSTILE ACTION within the SECURED 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 PLANT PROTECTED AREA, or the need to prepare the plant and staff for a potential aircraft impact.

Timely and accurate communications between the Security Shift Supervision and the Control Room is essential for proper classification of a security-related event (ref. 1).

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

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

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

The first threshold is applicable for any HOSTILE ACTION occurring, or that has occurred, in the SECURED 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 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 SECURED 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 PVNGS Security Plan (ref. 1).

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

**PVNGS Basis Reference(s):**

1. *PVNGS Security Plan, Training and Qualification Plan, Safeguards Contingency Plan and Independent Spent Fuel Storage Installation Security Program (Safeguards)*
2. NEI 99-01, HA1

**Category:** H – Hazards

**Subcategory:** 1 – Security

**Initiating Condition:** Hostile Action within the PLANT PROTECTED AREA

**EAL:**

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

**Mode Applicability:**

All

**Definition(s):**

*HOSTILE ACTION* - An act toward PVNGS 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 PVNGS. 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).

*PLANT PROTECTED AREA* - An area, located within the PVNGS Exclusion Area Boundary, encompassed by physical barriers and to which access is controlled per 10 CFR 73.55. The PVNGS Plant Protected Area and the ISFSI Protected Area are two Protected Areas located within the PVNGS OWNER CONTROLLED AREA.

**Basis:**

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

Timely and accurate communications between the Security Shift Supervision and the Control Room is essential for proper classification of a security-related event (ref. 1).

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

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

This IC does not apply to a HOSTILE ACTION directed at an ISFSI PROTECTED AREA located outside the PLANT PROTECTED AREA; such an attack should be assessed using IC HA1. It also does not apply to incidents that are accidental events, acts of civil disobedience,

or otherwise are not a HOSTILE ACTION perpetrated by a HOSTILE FORCE. Examples include the crash of a small aircraft, shots from hunters, physical disputes between employees, etc. Reporting of these types of events is adequately addressed by other EALs, or the requirements of 10 CFR 73.71 or 10 CFR 50.72.

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

**PVNGS Basis Reference(s):**

1. *PVNGS Security Plan, Training and Qualification Plan, Safeguards Contingency Plan and Independent Spent Fuel Storage Installation Security Program (Safeguards)*
2. NEI 99-01, HS1

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

**Subcategory:** 2 – Seismic Event

**Initiating Condition:** Seismic event greater than OBE levels

**EAL:**

| | |
|---|---|
| **HU2.1** | **Unusual Event** |
| Seismic event > OBE as indicated on Control Panel A-J-SMN-C01 | |

**Mode Applicability:**

All

**Definition(s):**

None

**Basis:**

Five Force Balance Accelerometer units are installed within Unit 1 structures and one is installed in the Free Field area south of Unit 1.

Peak ground motion acceleration of 0.10g horizontal or vertical is the Operating Basis Earthquake for PVNGS (ref. 1). OBE is detected and analyzed by Free Field Accelerometer Sensor #6 (AJSMNXT0006) only.

Annunciator 7C14A, SEISMIC OCCURRENCE, will illuminate if the seismic instrument detects ground motion in excess of the seismic EVENT trigger threshold (ref. 1, 2).

Unit 1 Control Panel A-J-SMN-C01 provides both red EVENT and yellow "OBE" LED indications (ref. 1, 2). Peak acceleration levels can also be determined using the graphic user interface display screen (ref. 4).

Procedure 40AO-9ZZ21, *Acts of Nature*, provides the guidance should the OBE earthquake threshold be exceeded and any required response actions (ref. 3, 4).

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 the number listed in procedure 40AO-9ZZ21. Select option #1 and inform the analyst you wish to confirm recent seismic activity in the vicinity of PVNGS. If requested, provide the analyst with the following PVNGS Unit 1 coordinates: 33º 23' 23" north latitude, 112º 51' 43" west longitude (ref. 5). Alternatively, near real-time seismic activity can be accessed via the NEIC website:*http://earthquake.usgs.gov/earthquakes/dyfi/archives.php*

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 0.10g). 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.

**PVNGS Basis Reference(s):**

1. UFSAR Section 2.5.2.7, Operating Basis Earthquakes
2. Procedure 40AL-9RK7C, *Panel C07C Alarm Response 7C14A Seismic Occurrence*
3. Procedure 40AO-9ZZ21, *Acts of Nature*
4. Procedure 79IS-9SM01, *Analysis of Seismic Event*
5 UFSAR Table 2.1-1, Containment Building Centerlines
6. NEI 99-01, HU2

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

**Subcategory:** 3 – Natural or Technology Hazard

**Initiating Condition:** Hazardous event

**EAL:**

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

**Mode Applicability:**

All

**Definition(s):**

*PLANT PROTECTED AREA* - An area, located within the PVNGS Exclusion Area Boundary, encompassed by physical barriers and to which access is controlled per 10 CFR 73.55. The PVNGS Plant Protected Area and the ISFSI Protected Area are two Protected Areas located within the PVNGS OWNER CONTROLLED AREA.

**Basis:**

Response actions associated with a tornado onsite is provided in procedure 40AO-9ZZ21, *Acts of Nature* (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 PLANT 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.

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 PLANT PROTECTED AREA.

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

**PVNGS Basis Reference(s):**

1. Procedure 40AO-9ZZ21, *Acts of Nature*
2. UFSAR Section 2.3.1.2.3, Extreme Winds
2. NEI 99-01, HU3

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

**Subcategory:** 3 – Natural or Technology Hazard

**Initiating Condition:** Hazardous event

**EAL:**

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

**Mode Applicability:**

All

**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 10 CFR 50.2):

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

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

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

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

**Basis:**

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

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

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

**PVNGS 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:**

| | |
|---|---|
| **HU3.3** | **Unusual Event** |
| Movement of personnel within the PLANT 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).

*PLANT PROTECTED AREA* - An area, located within the PVNGS Exclusion Area Boundary, encompassed by physical barriers and to which access is controlled per 10 CFR 73.55. The PVNGS Plant Protected Area and the ISFSI Protected Area are two Protected Areas located within the PVNGS OWNER CONTROLLED AREA.

**Basis:**

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

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 PLANT PROTECTED AREA.

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

**PVNGS 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.

**PVNGS Basis Reference(s):**

1. NEI 99-01, HU3

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

**Subcategory:** 4 – Fire

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

**EAL:**

**HU4.1 Unusual Event**

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

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

**AND**

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

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

| **Table H-1 Fire Areas** |
|---|
| • Containment<br>• Auxiliary Building<br>• Control Building<br>• Diesel Generator Building<br>• Diesel Generator Fuel Oil Storage Tanks<br>• Fuel Building<br>• Main Steam Support Structure<br>• Refueling Water Tank<br>• Essential Spray Pond System<br>• Condensate Storage Tank |

**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:**

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.

Table H-1 Fire Areas are based on UFSAR Table 3.2-1 Quality Classification of Structures, Systems and Components. 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.

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 multiple alarms, indication, or report was received and not the time that a subsequent verification action was performed. If only a single indication is available to the Control Room staff, the emergency declaration clock starts at the time a field report is given that validates the existence. Similarly, the fire duration clock also starts at the time of receipt of the initial multiple alarms, 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.

**PVNGS Basis Reference(s):**

1. UFSAR Table 3.2-1, Quality Classification of Structures, Systems and Components
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 minutes of alarm receipt (Note 1)

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

| **Table H-1 Fire Areas** |
|---|
| • Containment<br>• Auxiliary Building<br>• Control Building<br>• Diesel Generator Building<br>• Diesel Generator Fuel Oil Storage Tanks<br>• Fuel Building<br>• Main Steam Support Structure<br>• Refueling Water Tank<br>• Essential Spray Pond System<br>• Condensate Storage Tank |

**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:**

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 UFSAR Table 3.2-1 Quality Classification of Structures, Systems and Components. 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.

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.

<u>Basis-Related Requirements from Appendix R</u>

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.

**PVNGS Basis Reference(s):**

1. UFSAR Table 3.2-1, Quality Classification of Structures, Systems and Components
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 or ISFSI PROTECTED AREA **not** extinguished within 60 minutes 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.

*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.

*PLANT or ISFSI PROTECTED AREA* - An area, located within the PVNGS Exclusion Area Boundary, encompassed by physical barriers and to which access is controlled per 10 CFR 73.55. The PVNGS Plant Protected Area and the ISFSI Protected Area are two Protected Areas located within the PVNGS OWNER CONTROLLED 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. This basis extends to a FIRE occurring within the ISFSI PROTECTED AREA .

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

**PVNGS Basis Reference(s):**

1. NEI 99-01, HU4

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

**Subcategory:** 4 – Fire

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

**EAL:**

**HU4.4 Unusual Event**

A FIRE within the PLANT PROTECTED AREA or ISFSI 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.

*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.

*PLANT or ISFSI PROTECTED AREA* - An area, located within the PVNGS Exclusion Area Boundary, encompassed by physical barriers and to which access is controlled per 10 CFR 73.55. The PVNGS Plant Protected Area and the ISFSI Protected Area are two Protected Areas located within the PVNGS OWNER CONTROLLED 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 or ISFSI 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 Onsite Fire Department 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.

**PVNGS Basis Reference(s):**

1. NEI 99-01, HU4

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

**Subcategory:** 5 – Hazardous Gases

**Initiating Condition:** Gaseous release IMPEDING access to equipment necessary for normal plant operations, cooldown or shutdown

**EAL:**

**HA5.1 Alert**

Release of a toxic, corrosive, asphyxiant or flammable gas into **any** Table H-2 rooms

**AND**

Entry into the room is prohibited or IMPEDED (Note 5)

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

| Table H-2 Safe Operation & Shutdown Rooms | |
|---|---|
| **Room** | **Mode Applicability** |
| Control Building 100 ft. Class DC Equipment Room C | 4, 5 |
| Control Building 100 ft. Class DC Equipment Room D | 4, 5 |

**Mode Applicability:**

4 – Hot Shutdown, 5 – Cold Shutdown

**Definition(s):**

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

**Basis:**

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

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

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

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

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

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

- The plant is in an operating mode different than the mode specified for the affected room/area (i.e., entry is not required during the operating mode in effect at the time of the gaseous release). For example, the plant is in Mode 1 when the gaseous release occurs and the procedures used for normal operation, cooldown and shutdown do not require entry into the affected room until Mode 4.
- The gas release is a planned activity that includes compensatory measures which address the temporary inaccessibility of a room or area (e.g., fire suppression system testing).
- The action for which room/area entry is required is of an administrative or record keeping nature (e.g., normal rounds or routine inspections).
- The access control measures are of a conservative or precautionary nature and would not actually prevent or impede a required action.

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

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

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

**NOTE**: EAL HA5.1 mode applicability has been limited to the applicable modes identified in Table H-2 Safe Operation & Shutdown Rooms/Areas. If due to plant operating procedure or plant configuration changes, the applicable plant modes specified in Table H-2 are changed, a corresponding change to Attachment 3 'Safe Operation & Shutdown Areas Tables R-2 & H-2 Bases' and to EAL HA5 mode applicability is required.

**PVNGS Basis Reference(s):**

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

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

**Subcategory:** 6 – Control Room Evacuation

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

**EAL:**

| **HA6.1** | **Alert** |
|---|---|
| An event has resulted in plant control being transferred from the Control Room to the Remote Shutdown Panel (RSP) | |

**Mode Applicability:**

All

**Definition(s):**

None

**Basis:**

The Control Room Supervisor (CRS) determines if the Control Room is uninhabitable 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.

Procedure 40AO-9ZZ18, *Shutdown Outside the Control Room*, provides the instructions for bringing the unit to Mode 5, Cold Shutdown, if the Control Room has been determined to be uninhabitable for any reason other than fire (Ref. 1).

Procedure 40AO-9ZZ19, *Control Room Fire*, provides the instructions for bringing the unit to Mode 5, Cold Shutdown, if the Control Room has been determined to be uninhabitable due to a fire (Ref. 2).

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

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.

**PVNGS Basis Reference(s):**

1. Procedure 40AO-9ZZ18, *Shutdown Outside the Control Room*
2. Procedure 40AO-9ZZ19, *Control Room Fire*
3. NEI 99-01, HA6

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

**Subcategory:** 6 – Control Room Evacuation

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

**EAL:**

**HS6.1 Site Area Emergency**

An event has resulted in plant control being transferred from the Control Room to the Remote Shutdown Panel (RSP)

**AND**

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

- Reactivity Control (Modes 1, 2 and 3 **only**)
- Core Heat Removal
- 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:**

The Control Room Supervisor (CRS) determines if the Control Room is uninhabitable 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.

Procedure 40AO-9ZZ18, *Shutdown Outside the Control Room*, provides the instructions for tripping the unit and maintaining RCS inventory and Hot Shutdown conditions from outside the Control Room due to reasons other than fire (Ref. 1).

Procedure 40AO-9ZZ19, *Control Room Fire*, provides the instructions for tripping the unit and maintaining RCS inventory and Hot Shutdown conditions from outside the Control Room due to a fire (Ref. 2).

The intent of this EAL is to capture events in which control of the plant cannot be reestablished in a timely manner. The 15 minute time for transfer starts when the Control Room is evacuated (when CRS leaves the Control Room, not when procedures 40AO-9ZZ18 or 40AO-9ZZ19 are 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 15 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 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.

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

**PVNGS Basis Reference(s):**

1. Procedure 40AO-9ZZ18, *Shutdown Outside the Control Room*
2. Procedure 40AO-9ZZ19, *Control Room Fire*
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:**

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

**Mode Applicability:**

All

**Definition(s):**

None

**Basis:**

The Emergency Coordinator is the designated onsite individual having the responsibility and authority for implementing the *PVNGS Emergency 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 Unusual Event.

**PVNGS Basis Reference(s):**

1. *PVNGS Emergency Plan,* Section 4.2.1.1, Emergency Coordinator
2. *PVNGS Emergency Plan*, Section 4.2.1.12, 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:**

**HA7.1 Alert**

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

**Mode Applicability:**

All

**Definition(s):**

*HOSTILE ACTION* - An act toward PVNGS 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 PVNGS. 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:**

The Emergency Coordinator is the designated onsite individual having the responsibility and authority for implementing the *PVNGS Emergency 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.

**PVNGS Basis Reference(s):**

1. *PVNGS Emergency Plan,* Section 4.2.1.1, Emergency Coordinator
2. *PVNGS Emergency Plan,* Section 4.2.1.12, 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:**

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

**Mode Applicability:**

All

**Definition(s):**

*HOSTILE ACTION* - An act toward PVNGS 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 PVNGS. 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:**

The Emergency Coordinator is the designated onsite individual having the responsibility and authority for implementing the *PVNGS Emergency 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.

**PVNGS Basis Reference(s):**

1. *PVNGS Emergency Plan,* Section 4.2.1.1 Emergency Coordinator
2. *PVNGS Emergency Plan,* Section 4.2.1.12 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 exist which in the judgment of the Emergency Coordinator warrant declaration of a General Emergency

**EAL:**

**HG7.1 General Emergency**

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

**Mode Applicability:**

All

**Definition(s):**

*HOSTILE ACTION* - An act toward PVNGS 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 PVNGS. 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:**

The Emergency Coordinator is the designated onsite individual having the responsibility and authority for implementing the *PVNGS Emergency 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.

**PVNGS Basis Reference(s):**

1. *PVNGS Emergency Plan,* Section 4.2.1.1 Emergency Coordinator
2. *PVNGS Emergency Plan,* Section 4.2.1.12 Shift Manager
3. NEI 99-01, HA7

**Category S – System Malfunction**

EAL Group: Hot Conditions (RCS temperature > 210ºF); 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. RPS Failure

This subcategory includes events related to failure of the Reactor Protection System (RPS) to initiate and complete reactor trips. In the plant licensing basis, postulated failures of the RPS to complete a reactor trip comprise a specific set of analyzed events referred to as Anticipated Transient Without Scram (ATWS) events. For EAL classification, however, ATWS is intended to mean any trip failure event that does not achieve reactor shutdown. If RPS actuation fails to

assure reactor shutdown, positive control of reactivity is at risk and could cause a threat to fuel clad, RCS and containment integrity.

7. Loss of Communications

Certain events that degrade plant operator 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.

**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 PBA-S03 and PBB-S04 for ≥ 15 minutes (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:**<br>• SUT (normal)<br>• SUT (alternate)<br>• SBOG #1 **AND** SBOG #2 (if already aligned)<br>**Onsite:**<br>• DG A<br>• DG B |

**Mode Applicability:**

1 - Power Operation, 2 - Startup, 3 - Hot Standby, 4 – Hot Shutdown

**Definition(s):**

None

**Basis:**

The 4.16KV AC System provides the power requirements for operation and safe shutdown of the plant. The essential switchgear are buses PBA-S03 and PBB-S04 (ref. 1).

The condition indicated by this EAL is the degradation of all offsite AC power sources such that any only onsite AC power capability exists for 15 minutes or longer.

4.16KV buses PBA-S03 and PBB-S04 are the emergency (essential) buses. PBA-S03 supplies power to Train A safety related loads and PBB-S04 supplies power to Train B safety related loads. Each bus has two normal sources of offsite power. Each source is from one of three 13.8 KV Startup Transformers (SUT) via its normal and alternative ESF Service Transformer NAN-X03 or NAN-X04. Transformer NAN-X03 is the normal supply to bus PBA-S03 and the alternate supply to PBB-S04; Transformer NAN-X04 is the normal supply to bus PBB-S04 and the alternate supply to PBA-S03 (ref. 1).

Additional alternate offsite AC power sources are the two redundant 13.8KV SBO gas turbine generators (SBOG #1 & SBOG #2). However, these sources can only be credited if already

aligned, that is, capable of powering one or more emergency bus within 15 minutes. The SBOGs can only be credited if they are running in parallel since they are not rated to supply all the SAFETY SYSTEM loads.

PBA-S03 and PBB-S04 each have an onsite emergency diesel generator (DG A & DG B) which supply electrical power to the bus automatically in the event that the preferred source becomes unavailable (ref. 1).

This IC addresses a prolonged loss of offsite power. The loss of offsite power sources renders the plant more vulnerable to a complete loss of power to AC emergency buses. This condition represents a potential reduction in the level of safety of the plant.

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

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

Escalation of the emergency classification level would be via IC SA1.

**PVNGS Basis Reference(s):**

1. Drawing 13-E-MAA-001, *Main Single Line Diagram*
2. UFSAR Section 8.3.1, AC Power Systems
3. Procedure 40AO-9ZZ12, *Degraded Electrical Power*
4. UFSAR Section 1.2.10.3.9, Alternate AC Power System
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 PBA-S03 and PBB-S04 reduced to a single power source for ≥ 15 minutes (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:**<br>• SUT (normal)<br>• SUT (alternate)<br>• SBOG #1 **AND** SBOG #2 (if already aligned)<br>**Onsite:**<br>• DG A<br>• DG B |

**Mode Applicability:**

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

**Definition(s):**

*SAFETY SYSTEM* - A system required for safe plant operation, cooling down the plant and/or placing it in the cold shutdown condition, including the ECCS. These are typically systems classified as safety-related (as defined in 10 CFR 50.2):

> Those structures, systems and components that are relied upon to remain functional during and following design basis events to assure:
>
> (1) The integrity of the reactor coolant pressure boundary;
>
> (2) The capability to shut down the reactor and maintain it in a safe shutdown condition;
>
> (3) The capability to prevent or mitigate the consequences of accidents which could result in potential offsite exposures.

**Basis:**

For emergency classification purposes, “capability” means that an AC power source is available to and capable of powering the emergency bus(es) within 15 min, whether or not the buses are currently 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 PBA-S03 and PBB-S04 (ref. 1).

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 PBA-S03 and PBB-S04 are the emergency (essential) buses. PBA-S03 supplies power to Train A safety related loads and PBB-S04 supplies power to Train B safety related loads. Each bus has two normal sources of offsite power. Each source is from one of three 13.8 KV Startup Transformers (SUT) via its normal and alternative ESF Service Transformer NAN-X03 or NAN-X04. Transformer NAN-X03 is the normal supply to bus PBA-S03 and the alternate supply to PBB-S04; Transformer NAN-X04 is the normal supply to bus PBB-S04 and the alternate supply to PBA-S03 (ref. 1).

In addition, PBA-S03 and PBB-S04 each have an emergency diesel generator (DG A & DG B) which supply electrical power to the bus automatically in the event that the preferred source becomes unavailable (ref. 1).

Additional alternate offsite AC power sources are the two redundant 13.8KV SBO gas turbine generators (SBOG #1 & SBOG #2). However, these sources can only be credited if already aligned, that is, capable of powering one or more emergency bus within 15 minutes. The SBOGs can only be credited if they are running in parallel since they are not rated to supply all the SAFETY SYSTEM loads

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.

**PVNGS Basis Reference(s):**

1. Drawing 13-E-MAA-001, *Main Single Line Diagram*
2. UFSAR Section 8.3.1, AC Power Systems
3. Procedure 40AO-9ZZ12, *Degraded Electrical Power*
4. UFSAR Section 1.2.10.3.9, Alternate AC Power System
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 to emergency 4.16KV buses PBA-S03 and PBB-S04 for ≥ 15 minutes (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:**

For emergency classification purposes, “capability” means that an AC power source is available to and capable of powering the emergency bus(es) within 15 min, whether or not the buses are currently 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 PBA-S03 and PBB-S04 (ref. 1).

4.16KV buses PBA-S03 and PBB-S04 are the emergency (essential) buses. PBA-S03 supplies power to Train A safety related loads and PBB-S04 supplies power to Train B safety related loads. Each bus has two normal sources of offsite power. Each source is from one of three 13.8 KV Startup Transformers (SUT) via its normal and alternative ESF Service Transformer NAN-X03 or NAN-X04. Transformer NAN-X03 is the normal supply to bus PBA-S03 and the alternate supply to PBB-S04; Transformer NAN-X04 is the normal supply to bus PBB-S04 and the alternate supply to PBA-S03 (ref. 1).

In addition, PBA-S03 and PBB-S04 each have an emergency diesel generator (DG A & DG B) which supply electrical power to the bus automatically in the event that the preferred source becomes unavailable (ref. 1).

Additional alternate offsite AC power sources include, but not limited to, the two redundant 13.8KV SBO gas turbine generators (SBOG #1 & SBOG #2). However, these sources can only be credited if already aligned, that is, capable of powering one or more emergency bus within 15 minutes. The SBOGs can only be credited if they are running in parallel since they are not rated to supply all the SAFETY SYSTEM loads

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.

**PVNGS Basis Reference(s):**

1. Drawing 13-E-MAA-001, *Main Single Line Diagram*
2. UFSAR Section 8.3.1, AC Power Systems
3. Procedure 40AO-9ZZ12, *Degraded Electrical Power*
4. UFSAR Section 1.2.10.3.9, Alternate AC Power System
5. Procedure 40EP-9EO08, *Blackout*
6. NEI 99-01, SS1

**Category:** S –System Malfunction

**Subcategory:** 1 – Loss of Emergency AC Power

**Initiating Condition:** Prolonged loss of **all** offsite and **all** onsite AC power to emergency buses

**EAL:**

**SG1.1 General Emergency**

Loss of **all** offsite and **all** onsite AC power capability to emergency 4.16KV buses PBA-S03 and PBB-S04

**AND EITHER:**

- Restoration of at least one emergency bus in < 4 hour is **not** likely (Note 1)
- Rep CET reading > 1200ºF

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:**

This EAL is indicated by the extended loss of all offsite and onsite AC power capability to 4.16KV emergency buses PBA-S03 and PBB-S04 either for greater then the PVNGS Station Blackout (SBO) coping analysis time (4 hrs.) (ref. 8) or that has resulted in indications of an actual loss of adequate core cooling (Rep CET > 1200 ºF) (ref. 6, 7).

For emergency classification purposes, "capability" means that an AC power source is available to and capable of powering the emergency bus(es), whether or not the buses are currently 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 PBA-S03 and PBB-S04 (ref. 1).

4.16KV buses PBA-S03 and PBB-S04 are the emergency (essential) buses. PBA-S03 supplies power to Train A safety related loads and PBB-S04 supplies power to Train B safety related loads. Each bus has two normal sources of offsite power. Each source is from one of three 13.8 KV Startup Transformers (SUT) via its normal and alternative ESF Service Transformer NAN-X03 or NAN-X04. Transformer NAN-X03 is the normal supply to bus PBA-S03 and the alternate supply to PBB-S04; Transformer NAN-X04 is the normal supply to bus PBB-S04 and the alternate supply to PBA-S03 (ref. 1).

In addition, PBA-S03 and PBB-S04 each have an emergency diesel generator (DG A & DG B) which supply electrical power to the bus automatically in the event that the preferred source becomes unavailable (ref. 1).

Additional alternate offsite AC power sources include, but no limited to, the two redundant 13.8KV SBO gas turbine generators (SBOG #1 & SBOG #2). The SBOGs can only be credited if they are running in parallel since they are not rated to supply all the SAFETY SYSTEM loads.

Rep CET (Representative Core Exit Temperature) is a calculated temperature value generated by the Qualified Safety Parameter Display System (QSPDS). The QSPDS CET processing function generates a representative temperature based on a statistical analysis of thermocouples monitoring the reactor coolant temperature at the top of selected fuel assemblies.

This IC addresses a prolonged loss of all power sources to AC emergency buses. A loss of all AC power compromises the performance of all SAFETY SYSTEMS requiring electric power including those necessary for emergency core cooling, containment heat removal/pressure control, spent fuel heat removal and the ultimate heat sink. A prolonged loss of these buses will lead to a loss of one or more fission product barriers. In addition, fission product barrier monitoring capabilities may be degraded under these conditions.

The EAL should require declaration of a General Emergency prior to meeting the thresholds for IC FG1. This will allow additional time for implementation of offsite protective actions.

Escalation of the emergency classification from Site Area Emergency will occur if it is projected that power cannot be restored to at least one AC emergency bus by the end of the analyzed station blackout coping period. Beyond this time, plant responses and event trajectory are subject to greater uncertainty and there is an increased likelihood of challenges to multiple fission product barriers.

The estimate for restoring at least one emergency bus should be based on a realistic appraisal of the situation. Mitigation actions with a low probability of success should not be used as a basis for delaying a classification upgrade. The goal is to maximize the time available to prepare for and implement, protective actions for the public.

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.

**PVNGS Basis Reference(s):**

1. Drawing13-E-MAA-001, *Main Single Line Diagram*
2. UFSAR Section 8.3.1, AC Power Systems
3. EOP Setpoint Document TA-13-C00-2000-001
4. 40AO-9ZZ12, *Degraded Electrical Power*
5. UFSAR Section 1.2.10.3.9 Alternate AC Power System
6. Procedure 40DP-9AP13, *Blackout Technical Guideline*
7. Procedure 40EP-9EO09, *Functional Recovery*
8. Core Damage Assessment User Manual
9. Evaluation 4578373, *Station Blackout Coping Analysis for Margin to Core Covery*
9. NEI 99-01, SG1

**Category:** S –System Malfunction

**Subcategory:** 1 – Loss of Emergency AC Power

**Initiating Condition:** Loss of **all** emergency AC and vital DC power sources for 15 minutes or longer

**EAL:**

**SG1.2 General Emergency**

Loss of **all** offsite and **all** onsite AC power capability to emergency 4.16KV buses PBA-S03 and PBB-S04 for ≥ 15 minutes

**AND**

Loss of 125 VDC power based on battery bus voltage indications < 112 VDC on **both** vital DC buses PKA-M41and PKB-M42 for ≥ 15 minutes (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:**

This EAL is indicated by the loss of all offsite and onsite emergency AC power capability to 4.16KV emergency buses PBA-S03 and PBB-S04 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 and capable of powering the emergency bus(es) within 15 min, whether or not the buses are currently 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 PBA-S03 and PBB-S04 (ref. 1).

The 4.16KV buses PBA-S03 and PBB-S04 are the emergency (essential) buses. PBA-S03 supplies power to Train A safety related loads and PBB-S04 supplies power to Train B safety related loads. Each bus has two normal sources of offsite power. Each source is from one of three 13.8 KV Startup Transformers (SUT) via its normal and alternative ESF Service Transformer NAN-X03 or NAN-X04. Transformer NAN-X03 is the normal supply to bus PBA-S03 and the alternate supply to PBB-S04; Transformer NAN-X04 is the normal supply to bus PBB-S04 and the alternate supply to PBA-S03 (ref. 1).

In addition, PBA-S03 and PBB-S04 each have an emergency diesel generator (DG A & DG B) which supply electrical power to the bus automatically in the event that the preferred source becomes unavailable (ref. 1). However, these sources can only be credited if already aligned, that is, power one or more emergency bus within 15 minutes.

Additional alternate offsite AC power sources include, but not limited to, the two redundant 13.8KV SBO gas turbine generators (SBOG #1 & SBOG #2). However, these sources can only be credited if already aligned, that is, capable of powering one or more emergency bus within 15 minutes. The SBOGs can only be credited if they are running in parallel since they are not rated to supply all the SAFETY SYSTEM loads

The vital DC buses are the following 125 VDC Class 1E buses (ref. 6):

| Train A: | Train B: |
|---|---|
| • PKA-M41 | • PKB-M42 |
| • PKC-M43 | • PKD-M44 |

For this EAL credit is only taken for buses PKA-M41 and PKB-M42 as these are the Train A and Train B buses that provide safety system control power.

There are four, 60 cell, lead-calcium storage batteries (PKA-F11, PKC-F13, PKB-F12 and PKD-F14) 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 capacity of the battery chargers (ref. 6).

All four of the 125VDC buses supply inverters for 120VAC PN bus power as well as control power for various safety related systems. Each battery is designed to have sufficient stored energy to supply the required emergency loads for 120 minutes following a loss of AC power to the chargers (ref. 7).

Minimum DC bus voltage is 112 VDC (ref. 8).

This IC addresses a concurrent and prolonged loss of both emergency AC and Vital DC power. A loss of all emergency AC power compromises the performance of all SAFETY SYSTEMS requiring electric power including those necessary for emergency core cooling, containment heat removal/pressure control, spent fuel heat removal and the ultimate heat sink. A loss of vital DC power compromises the ability to monitor and control SAFETY SYSTEMS. A sustained loss of both emergency AC and vital DC power will lead to multiple challenges to fission product barriers.

Fifteen minutes was selected as a threshold to exclude transient or momentary power losses. The 15-minute emergency declaration clock begins at the point when both EAL thresholds are met.

**PVNGS Basis Reference(s):**

1. Drawing 13-E-MAA-001, *Main Single Line Diagram*
2. UFSAR Section 8.3.1 AC Power Systems
3. Procedure 40AO-9ZZ12, *Degraded Electrical Power*
4. UFSAR Section 1.2.10.3.9, Alternate AC Power System
5. Procedure 40DP-9AP13, *Blackout Technical Guideline*
6. Drawing 01-E-PKA-001, *Main Single Line Diagram 125V DC Class 1E and 120VAC Vital Inst Power System*
7. UFSAR Section 8.3.2, DC Power Systems
8. Calculation 01-EC-PK-0207 DC, *Battery Sizing and Minimum Voltage*
9. 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 125 VDC power based on battery bus voltage indications < 112 VDC on **both** vital DC buses PKA-M41and PKB-M42 for ≥ 15 minutes (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:**

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

| Train A: | Train B: |
|---|---|
| • PKA-M41 | • PKB-M42 |
| • PKC-M43 | • PKD-M44 |

For this EAL credit is only taken for buses PKA-M41 and PKB-M42 as these are the Train A and Train B buses that provide safety system control power.

There are four, 60 cell, lead-calcium storage batteries (PKA-F11, PKC-F13, PKB-F12 and PKD-F14) 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 capacity of the battery chargers (ref. 1).

All four of the 125VDC buses supply inverters for 120VAC PN bus power as well as control power for various safety related systems. Each battery is designed to have sufficient stored energy to supply the required emergency loads for 120 minutes following a loss of AC power to the chargers (ref. 2).

Minimum DC bus voltage is 112 VDC (ref. 3).

This IC addresses a loss of vital DC power which compromises the ability to monitor and control SAFETY SYSTEMS. In modes above Cold Shutdown, this condition involves a major failure of plant functions needed for the protection of the public.

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

Escalation of the emergency classification level would be via ICs RG1, FG1 or SG1.

**PVNGS Basis Reference(s):**

1. Drawing 01-E-PKA-001, *Main Single Line Diagram 125V DC Class 1E and 120VAC Vital Inst Power System*
2. UFSAR Section 8.3.2, DC Power Systems
3. Calculation 01-EC-PK-0207, *DC Battery Sizing and Minimum Voltage*
4. 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 minutes (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 11: Downcomer flow instruments are also credited for auxiliary feed flow indication.

| **Table S-2 Safety System Parameters** |
|---|
| • Reactor power<br>• RCS level<br>• RCS pressure<br>• CET temperature<br>• Level in at least one S/G<br>• Auxiliary feed flow to at least one S/G (Note 11) |

**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:**

SAFETY SYSTEM parameters listed in Table S-2 are monitored in the Control Room through a combination of hard control panel indicators as well as computer based information systems. The Plant Computer serves as a redundant compensatory indicator which may be utilized in lieu of normal Control Room indicators (ref. 1, 2).

Downcomer flow instruments are also credited for auxiliary feed flow indication.

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.

**PVNGS Basis Reference(s):**

1. UFSAR Section 7.5, Safety-Related Display Instrumentation
2. UFSAR Section 18.I.D.2, Plant Safety Parameter Display System
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 minutes (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.

Note 11: Downcomer flow instruments are also credited for auxiliary feed flow indication.

| Table S-2 Safety System Parameters |
|---|
| • Reactor power<br>• RCS level<br>• RCS pressure<br>• CET temperature<br>• Level in at least one S/G<br>• Auxiliary feed flow to at least one S/G (Note 11) |

| Table S-3 Significant Transients |
|---|
| • Reactor trip<br>• Runback > 25% thermal power<br>• Electrical load rejection > 25% electrical load<br>• Reactor power cutback<br>• 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:**

SAFETY SYSTEM parameters listed in Table S-2 are monitored in the Control Room through a combination of hard control panel indicators as well as computer based information systems. The Plant Computer serves as a redundant compensatory indicator which may be utilized in lieu of normal Control Room indicators (ref. 1, 2).

Downcomer flow instruments are also credited for auxiliary feed flow indication.

Significant transients are listed in Table S-3 and include response to automatic or manually initiated functions such as reactor trips, runbacks involving greater than 25% thermal power change, electrical load rejections of greater than 25% full electrical load, reactor power cutbacks 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.

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 heat removal 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.

**PVNGS Basis Reference(s):**

1. UFSAR Section 7.5, Safety-Related Display Instrumentation
2. UFSAR Section 18.I.D.2, Plant Safety Parameter Display System
3. NEI 99-01, SA2

**Category:** S – System Malfunction

**Subcategory:** 4 – RCS Activity

**Initiating Condition:** Reactor coolant activity greater than Technical Specification allowable limits

**EAL:**

| **SU4.1** | **Unusual Event** |
|---|---|
| Letdown Monitor RU-155D reading > high alarm | |

**Mode Applicability:**

1 - Power Operation, 2 - Startup, 3 - Hot Standby, 4 - Hot Shutdown

**Definition(s):**

None

**Basis:**

A reading on the Letdown Monitor RU-155D > high alarm is indicative of coolant activity in excess of the Technical Specification RCS activity limits (ref 1, 2).

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.

**PVNGS Basis Reference(s):**

1. Technical Specification 3.4.17, *RCS Specific Activity*
2. Calculation 13-NC-CH-311, *Letdown Line PRM Dose Rates*
3. NEI 99-01, SU3

**Category:** S – System Malfunction

**Subcategory:** 4 – RCS Activity

**Initiating Condition:** Reactor coolant activity greater than Technical Specification allowable limits

**EAL:**

| **SU4.2** | **Unusual Event** |
|---|---|
| Sample analysis indicates RCS activity > Technical Specification LCO 3.4.17 limits | |

**Mode Applicability:**

1 - Power Operation, 2 - Startup, 3 - Hot Standby, 4 - Hot Shutdown

**Definition(s):**

None

**Basis:**

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 > 48 hr continuous period. The specific Xe-133 activity is limited to ≤ 550 μCi/gm Dose Equivalent XE-133 for > 48 hr continuous period. Entry into Condition C of LCO 3.4.17 meets the intent of this EAL (ref 1, 2).

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.

**PVNGS Basis Reference(s):**

1. Technical Specification 3.4.17, *RCS Specific Activity*
2. Procedure 40AO-9ZZ22, *Fuel Damage*
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 minutes

**OR**

RCS identified leakage > 25 gpm for ≥ 15 minutes

**OR**

Reactor coolant leakage to a location outside containment > 25 gpm for ≥ 15 minutes (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:**

Manual or computer-based methods of performing an RCS inventory balance are normally used to determine RCS leakage. ERFDADS is the preferred method of calculating RCS leak rate. When ERFDADS software is not available, procedural guidance is available to perform the backup and manual RCS inventory balance (ref. 1, 4, 5, 6).

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 (leakage into an intact Reactor Drain Tank is also considered identified leakage), 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)

Reactor coolant leakage outside of the containment that is not considered identified or unidentified leakage per Technical Specifications. For example: leakage via interfacing systems such as RCS to the Nuclear Cooling Water System, Essential Cooling Water System, Safety Injection System, 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).

Palo Verde specific operating experience is that a High Pressure Seal Cooler (HPSC) leak to the Nuclear Cooling Water (NC) System must be isolated to containment within 15 minutes of discovery due to the location of the NC system expansion tank and potential dose concerns on the Auxiliary Building roof.

This IC addresses RCS leakage which may be a precursor to a more significant event. In this case, RCS leakage has been detected and operators, following applicable procedures, have been unable to promptly isolate the leak. This condition is considered to be a potential degradation of the level of safety of the plant.

The first and second EAL conditions are focused on a loss of mass from the RCS due to “unidentified leakage," "pressure boundary leakage" or "identified leakage” (as these leakage types are defined in the plant Technical Specifications). The third condition addresses an RCS mass loss caused by an UNISOLABLE leak through an interfacing system. These conditions thus apply to leakage into the containment, 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.

**PVNGS Basis Reference(s):**

1. Procedure 40ST-9RC02, *ERFDADS (Preferred) Calculation of RCS Water Inventory*
2. Technical Specification, 1.1, *Definitions*
3. UFSAR Section 5.2.5.4, Intersystem Leakage
4. Procedure 40AO-9ZZ02, *Excessive RCS Leakrate*
5. Procedure 40ST-9RC05, *Manual Calculation of RCS Water Inventory*
6. Procedure 40ST-9RC08, *OAP (Backup) Calculation of RCS Water Inventory*
7. NEI 99-01, SU4

**Category:** S – System Malfunction

**Subcategory:** 6 – RPS Failure

**Initiating Condition:** Automatic or manual trip fails to shut down the reactor

**EAL:**

**SU6.1 Unusual Event**

An automatic trip did **not** shut down the reactor as indicated by reactor power > 5% after **any** RPS setpoint is exceeded

**AND**

A subsequent automatic trip or manual trip action taken at the reactor control consoles (B05 or B01) 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:**

The first condition of this EAL identifies the need to cease critical reactor operations by actuation of the automatic Reactor Protection System (RPS) trip function. A reactor trip is automatically initiated by the RPS when certain continuously monitored parameters exceed predetermined setpoints (ref. 1, 4).

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. For the purpose of emergency classification a successful trip has occurred when there is sufficient rod insertion from the trip of RPS to bring the reactor power to or below the Power Operation Mode threshold of 5% (ref. 2).

5% rated power is the Power Operation mode threshold. Below 5%, plant response will be similar to that observed during a normal shutdown. Nuclear instrumentation can be used to determine if reactor power is greater than 5 % power (ref. 1, 2).

For the purposes of emergency classification, successful manual trip actions are those which can be quickly performed from the reactor control consoles (B05 or B01). Reactor shutdown achieved by use of other trip actions do not constitute a successful manual trip (ref. 3).

Following any automatic RPS trip signal, procedure 40EP-9EO01, *Standard Post Trip Actions* (ref. 3) prescribes insertion of redundant manual trip signals to back up the automatic RPS trip function and ensure reactor shutdown is achieved if Reactivity Control acceptance criteria are not met. Even if the first subsequent manual trip signal inserts all control rods to the full-in position immediately after the initial failure of the automatic trip, the lowest level of classification that must be declared is an Unusual Event.

In the event that the operator identifies a reactor trip is imminent and initiates a successful manual reactor trip before the automatic RPS trip setpoint is reached, no declaration is required. The successful manual trip of the reactor before it reaches its automatic trip setpoint or reactor trip signals caused by instrumentation channel failures (without exceeding an RPS trip setpoint) do not lead to a potential fission product barrier loss and are thus not classifiable under this EAL. However, if subsequent manual reactor trip actions fail to reduce reactor power to or 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 RPS to initiate or complete an automatic or manual reactor trip that results in a reactor shutdown and either a subsequent operator manual action taken at the reactor control consoles or an automatic trip is successful in shutting down the reactor. This event is a precursor to a more significant condition and thus represents a potential degradation of the level of safety of the plant.

Following the failure on an automatic reactor trip, operators will promptly initiate manual actions at the reactor control consoles to shutdown the reactor (e.g., initiate a manual reactor trip). If these manual actions are successful in shutting down the reactor, core heat generation will quickly fall to a level within the capabilities of the plant’s decay heat removal systems.

If an initial manual reactor trip is unsuccessful, operators will promptly take manual action at another location(s) on the reactor control consoles to shutdown the reactor (e.g., initiate a manual reactor trip) using a different switch). Depending upon several factors, the initial or subsequent effort to manually trip the reactor, or a concurrent plant condition, may lead to the generation of an automatic reactor trip signal. If a subsequent manual or automatic trip is successful in shutting down the reactor, core heat generation will quickly fall to a level within the capabilities of the plant’s decay heat removal systems.

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.

Should a reactor trip signal be generated as a result of plant work (e.g., RPS setpoint testing), or instrument failure 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.

**PVNGS Basis Reference(s):**

1. Technical Specification 3.3.1, *Reactor Protection System (RPS) Instrumentation – Operating*
2. Technical Specification Table 1.1-1, *Modes*
3. Procedure 40EP-9EO01, *Standard Post Trip Actions*
4. UFSAR Section, 7.2.2.2 Trip Bases
5 NEI 99-01, SU5

**Category:** S – System Malfunction

**Subcategory:** 6 – RPS Failure

**Initiating Condition:** Automatic or manual trip fails to shut down the reactor

**EAL:**

**SU6.2 Unusual Event**

A manual trip did **not** shut down the reactor as indicated by reactor power > 5% after **any** manual trip action was initiated

**AND**

A subsequent automatic trip or manual trip action taken at the reactor control consoles (B05 or B01) 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:**

This EAL addresses a failure of a manually initiated trip in the absence of having exceeded an automatic RPS trip setpoint and a subsequent automatic or manual trip is successful in shutting down the reactor (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. For the purpose of emergency classification a successful trip has occurred when there is sufficient rod insertion from the manual trip to bring the reactor power to or below the Power Operation Mode threshold level of 5% (ref. 2).

5% rated power is the Power Operation mode threshold. Below 5%, plant response will be similar to that observed during a normal shutdown. Nuclear instrumentation can be used to determine if reactor power is greater than 5 % power (ref. 1, 2).

For the purposes of emergency classification, successful manual trip actions are those which can be quickly performed from the reactor control consoles (B05 or B01). Reactor shutdown achieved by use of other trip actions do not constitute a successful manual trip (ref. 3).

Following the failure of any manual trip signal, procedure 40EP-9EO01, *Standard Post Trip Actions* (ref. 3), prescribes insertion of redundant manual trip signals to back up the RPS trip

function and ensure reactor shutdown is achieved if Reactivity Control acceptance criteria are not met. 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. 3).

If both subsequent automatic and subsequent manual reactor trip actions in the Control Room fail to reduce reactor power below $\leq$ 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 RPS to initiate or complete an automatic or manual reactor trip that results in a reactor shutdown and either a subsequent operator manual action taken at the reactor control consoles or an automatic trip is successful in shutting down the reactor. This event is a precursor to a more significant condition and thus represents a potential degradation of the level of safety of the plant.

Following the failure on an automatic reactor trip, operators will promptly initiate manual actions at the reactor control consoles to shutdown the reactor (e.g., initiate a manual reactor trip). If these manual actions are successful in shutting down the reactor, core heat generation will quickly fall to a level within the capabilities of the plant's decay heat removal systems.

If an initial manual reactor trip is unsuccessful, operators will promptly take manual action at another location(s) on the reactor control consoles to shutdown the reactor (e.g., initiate a manual reactor trip) using a different switch). Depending upon several factors, the initial or subsequent effort to manually 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.

Should a reactor trip signal be generated as a result of plant work (e.g., RPS setpoint testing) or instrument failure, 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.

**PVNGS Basis Reference(s):**

1. Technical Specification 3.3.1, *Reactor Protection System (RPS) Instrumentation – Operating*
2. Technical Specification Table 1.1-1, *Modes*
3. Procedure 40EP-9EO01, *Standard Post Trip Actions*
4. UFSAR Section 7.2.2.2, Trip Bases
5. NEI 99-01, SU5

**Category:** S – System Malfunction

**Subcategory:** 2 – RPS Failure

**Initiating Condition:** Automatic or manual trip fails to shut down the reactor and subsequent manual actions taken at the reactor control consoles are not successful in shutting down the reactor

**EAL:**

**SA6.1 Alert**

An automatic or manual trip fails to shut down the reactor as indicated by reactor power > 5%

**AND**

Manual trip actions taken at the reactor control consoles (B05 or B01) 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:**

This EAL addresses any automatic or manual reactor trip signal that fails to shut down the reactor followed by a subsequent manual trip that fails to shut down the reactor to an extent the reactor is producing significant power (ref. 1, 4).

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. For the purpose of emergency classification a successful trip has occurred when there is sufficient rod insertion from the manual trip to bring the reactor power to or below 5% (ref. 2).

5% rated power is the Power Operation mode threshold. 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 (1, 2).

For the purposes of emergency classification, successful manual trip actions are those which can be quickly performed from the reactor control consoles (B05 or B01). Reactor shutdown achieved by use of other trip actions do not constitute a successful manual trip (ref. 3).

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 RPS.

A manual action at the reactor control console is any operator action, or set of actions, which causes the control rods to be rapidly inserted into the core (e.g., initiating a manual reactor trip). This action does not include manually driving in control rods or implementation of boron injection strategies. If this action(s) is unsuccessful, operators would immediately pursue additional manual actions at locations away from the reactor control console (e.g., locally opening breakers). Actions taken at 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 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.

**PVNGS Basis Reference(s):**

1. Technical Specification 3.3.1, *Reactor Trip System (RTS) Instrumentation*
2 Technical Specification Table 1.1-1, *Modes*
3. Procedure 40EP-9EO01, *Standard Post Trip Actions*
4. UFSAR Section 7.2.2.2, Trip Bases
5. NEI 99-01, SA5

**Category:** S – System Malfunction

**Subcategory:** 2 – RPS Failure

**Initiating Condition:** Inability to shut down the reactor causing a challenge to core cooling or RCS heat removal

**EAL:**

**SS6.1 Site Area Emergency**

An automatic or manual trip fails to shut down the reactor as indicated by reactor power > 5%

**AND**

**All** actions to shut down the reactor are **not** successful as indicated by reactor power > 5%

**AND EITHER**:

- Rep CET > 1200ºF
- RCS subcooling < 24 ºF

**Mode Applicability:**

1 - Power Operation

**Definition(s):**

None

**Basis:**

This EAL addresses the following:

- Any automatic reactor trip signal (ref. 1) 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 failures of both front line and backup protection systems to function in response to a plant transient, along with the continued production of heat, poses a direct threat to the Fuel Clad and RCS barriers.

Reactor shutdown achieved by use of other trip actions specified in procedure 40EP-9EO01, *Standard Post Trip Actions*, (such as opening NGN-L03B2 and NGN-L10B2 supply breakers, emergency boration or manually driving control rods) are also credited as a successful manual trip provided reactor power can be reduced to or below 5% before indications of an extreme challenge to either core cooling or heat removal exist (ref. 2, 3).

5% rated power is the Power Operation mode threshold. 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.

Indication of continuing core cooling degradation is manifested by CETs are reading greater than 1200ºF.

Rep CET (Representative Core Exit Temperature) is a calculated temperature value generated by the Qualified Safety Parameter Display System (QSPDS). The QSPDS CET processing function generates a representative temperature based on a statistical analysis of thermocouples monitoring the reactor coolant temperature at the top of selected fuel assemblies.

Indication of inability to adequately remove heat from the RCS is manifested by RCS subcooling < 24 ºF. (ref. 4).

This IC addresses a failure of the RPS to initiate or complete an automatic or manual reactor trip that results in a reactor shutdown, all subsequent operator actions to manually shutdown the reactor are unsuccessful and continued power generation is challenging the capability to adequately remove heat from the core and/or the RCS. This condition will lead to fuel damage if additional mitigation actions are unsuccessful and thus warrants the declaration of a Site Area Emergency.

In some instances, the emergency classification resulting from this IC/EAL may be higher than that resulting from an assessment of the plant responses and symptoms against the Recognition Category F ICs/EALs. This is appropriate in that the Recognition Category F ICs/EALs do not address the additional threat posed by a failure to shut down the reactor. The inclusion of this IC and EAL ensures the timely declaration of a Site Area Emergency in response to prolonged failure to shutdown the reactor.

Escalation of the emergency classification level would be via IC RG1 or FG1.

**PVNGS Basis Reference(s):**

1. Technical Specification 3.3.1, *Reactor Trip System (RTS) Instrumentation*
2 Technical Specification Table 1.1-1, *Modes*
3. Procedure 40EP-9EO01, *Standard Post Trip Actions*
4. Procedure 40EP-9EO09, *Functional Recovery*
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 Offsite Response Organization (ORO) communication methods

**OR**

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

| Table S-4 Communication Methods | | | |
|---|---|---|---|
| **System** | **Onsite** | **ORO** | **NRC** |
| PBX | X | X | X |
| Plant Page | X | | |
| Two-Way Radio | X | | |
| FTS (ENS) | | | X |
| Telephone Ringdown Circuits (NAN) | | X | |
| Cellular Phones | | X | X |

**Mode Applicability:**

1 - Power Operation, 2 - Startup, 3 - Hot Standby, 4 - Hot Shutdown

**Definition(s):**

None

**Basis:**

Onsite, offsite and NRC communications include one or more of the systems listed in Table S-4 (ref. 1, 2).

1. PBX

   Onsite emergency telephone lines are divided among three onsite EPABX switches. Each EPABX switch is provided with a backup battery for reliability.

This system will function during emergencies as it does during normal operations. Telephones have the capability of trunk access (via local provider) and the APS owned private communications system which provides direct dial capabilities to the entire APS voice system via the company owned private communications system. The PVNGS telephone EPABX Systems through which all PVNGS telephone calls pass, are equipped with uninterruptible power supplies (battery chargers and batteries) and dedicated priority switching to ensure the reliability of the telephone system. The PVNGS EPABXs are the primary links for PVNGS phones. There are also administratively dedicated lines for the CR, STSC, TSC, EOF and OSC.

2. Plant (Area) Paging

   The area paging system provides a reliable means of notifying and providing instructions to onsite personnel. Access to this system is through the EPABX system telephones by use of dedicated numbers.

3. Two-Way Radios

   PVNGS operates a trunked radio system, with separate talk groups available for departments such as Operations, Security, Fire Protection, Radiation Protection, Emergency Preparedness, the Water Reclamation Facility, etc. This system includes base station consoles at various locations and emergency facilities throughout the site. Some of the radios used during emergencies are portable radios at various site locations, mobile radios in the RFAT vehicles and base station consoles at the TSC, EOF, Unit OSCs, Unit STSCs and Unit Control Rooms. PVNGS Fire Protection also maintains radios that are used to contact the air ambulance service to provide landing instructions.

4. FTS (ENS)

   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.

5. Telephone Ringdown Circuits (NAN)

   These voice circuits serve as a primary communications link for providing technical information to offsite agencies, public information communications and the communication of protective action recommendations to offsite authorities.

6. Cellular Phones

   Each STSC, the TSC and EOF have a cellular phone to provide additional independent lines of communication.

This EAL is the hot condition equivalent of the cold condition EAL CU5.1.

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.

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 and Maricopa County EOCs.

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

**PVNGS Basis Reference(s):**

1. *PVNGS Plant Radiological Emergency Response Plan (RERP)*, Section 7.2
2. UFSAR Section 9.5.2, Communication Systems
3. NEI 99-01, SU6

**Category:** S – System Malfunction

**Subcategory:** 8 – Containment Failure

**Initiating Condition:** Failure to isolate containment or loss of containment pressure control.

**EAL:**

**SU8.1 Unusual Event**

**EITHER**:

- **Any** penetration is not closed when required within 15 minutes of a VALID isolation signal
- Containment pressure > 8.5 psig with < 4350 gpm Containment Spray flow for ≥ 15 minutes

(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):**

*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:**

Containment isolations are initiated by the Containment Isolation Actuation System (CIAS), Safety Injection Actuation Signal (SIAS), Main Steam Isolation Signal (MSIS) and Containment Spray Actuation Signal (CSAS) (ref. 1, 2).

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 (RWT) 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 RWT to the Containment sumps (ref. 3).

The Containment pressure high-high setpoint (8.5 psig) is the pressure at which the Containment Spray equipment should actuate and begin performing its function (ref. 4). Consistent with the design requirement, "one full train of depressurization equipment" is therefore defined to be the availability of one train of Containment Spray providing a minimum of 4350 gpm spray flow (ref. 5). LPSI cross-tie can be credited provided the alignment can be

made within the 15 minute threshold. 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.

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) 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.

**PVNGS Basis Reference(s):**

1. UFSAR Section 6.2.1.5.3.8, Containment Purge System
2. UFSAR Section 6.2.4, Containment Isolation System
3. UFSAR Section 6.2.2, Containment Heat Removal System
4. UFSAR Table 7.3-11A, ESFAS Setpoints and Margins to Actuation
5. Procedure 40EP-9EO01, *Standard Post Trip Actions*
6. 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)<br>● Internal or external FLOODING event<br>● High winds or tornado strike<br>● FIRE<br>● EXPLOSION<br>● Other events with similar hazard characteristics as determined by the Shift Manager |

**Mode Applicability:**

1 - Power Operation, 2 - Startup, 3 - Hot Standby, 4 - Hot Shutdown

**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 10 CFR 50.2):

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

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

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

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

*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:**

Refer to Attachment 4 for a list of Palo Verde SAFETY SYSTEMS (ref. 5)

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

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

The second condition addresses damage to a SAFETY SYSTEM component that is not in service/operation or readily apparent through indications alone, or to a structure containing 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.

- The significance of seismic events are discussed under EAL HU2.1. Annunciator 7C14A, SEISMIC OCCURRENCE will illuminate if the seismic instrument detects ground motion in excess of the seismic EVENT trigger threshold (ref. 1).
- Internal FLOODING may be caused by events such as component failures, equipment misalignment, or outage activity mishaps.
- High winds in excess of design (105 mph) or tornado strikes can cause significant structural damage (ref. 4).
- Areas containing functions and systems required for safe shutdown of the plant are identified by fire area (ref. 2).
- 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.

Escalation of the emergency classification level would be via IC FS1 or RS1.

**PVNGS Basis Reference(s):**

1. Procedure 40AO-9ZZ21, *Acts of Nature*
2. UFSAR Table 3-2.1, Quality Classification of Structures, Systems and Components
3. UFSAR Section 2.4.2.2.1, Offsite Flood Design Considerations
4. UFSAR Section 2.3.1.2.3, Extreme Winds
5. Attachment 4 - Palo Verde Safety Systems
6. NEI 99-01, SA9

**Category F – Fission Product Barrier Degradation**

EAL Group: Hot Conditions (RCS temperature > 210ºF); 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 (CTMT): 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 PVNGS design and operating characteristics.
- As used in this category, the term RCS leakage encompasses not just those types defined in Technical Specifications but also includes the loss of RCS mass to any location– inside the containment, an interfacing system, or outside of the containment. The release of liquid or steam mass from the RCS due to the as-designed/expected operation of a relief valve is not considered 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:**

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

**PVNGS Basis Reference(s):**

1. NEI 99-01, FA1

**Category:** Fission Product Barrier Degradation

**Subcategory:** N/A

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

**EAL:**

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

**Mode Applicability:**

1 - Power Operation, 2 - Startup, 3 - Hot Standby, 4 - Hot Shutdown

**Definition(s):**

None

**Basis:**

Fuel Clad, RCS and Containment comprise the fission product barriers. Table F-1 (Attachment 2) lists the fission product barrier thresholds, bases and references.

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

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

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

**PVNGS Basis Reference(s):**

1. NEI 99-01, FS1

**Category:** Fission Product Barrier Degradation

**Subcategory:** N/A

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

**EAL:**

| | |
|---|---|
| **FG1.1** | **General Emergency** |
| Loss of **any** two barriers<br>**AND**<br>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:**

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

**PVNGS Basis Reference(s):**

1. NEI 99-01, FG1

## 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. CTMT Radiation / RCS Activity

D. CTMT 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 C would be assigned "FC Loss C.1," the third Containment barrier Potential Loss in Category D would be assigned "CTMT P-Loss D.3," etc.

If a cell in Table F-1 contains more than one numbered threshold, each of the numbered thresholds, if exceeded, signifies a Loss or Potential Loss of the barrier. It is not necessary to exceed all of the thresholds in a category before declaring a barrier Loss/Potential Loss.

Subdivision of Table F-1 by category facilitates association of plant conditions to the applicable fission product barrier Loss and Potential Loss thresholds. This structure promotes a systematic approach to assessing the classification status of the fission product barriers.

When equipped with knowledge of plant conditions related to the fission product barriers, the EAL-user first scans down the category column of Table F-1, locates the likely category and then reads across the fission product barrier Loss and Potential Loss thresholds in that category to determine if a threshold has been exceeded. If a threshold has not been exceeded, the EAL-user proceeds to the next likely category and continues review of the thresholds in the new category

If the EAL-user determines that any threshold has been exceeded, by definition, the barrier is lost or potentially lost – even if multiple thresholds in the same barrier column are exceeded, only that one barrier is lost or potentially lost. The EAL-user must examine each of the three fission product barriers to determine if other barrier thresholds in the category are lost or potentially lost. For example, if containment radiation is sufficiently high, a Loss of the Fuel Clad and RCS barriers and a Potential Loss of the Containment barrier can occur. Barrier

Losses and Potential Losses are then applied to the criterion 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**

| Category | Fuel Clad (FC) Barrier | | Reactor Coolant System (RCS) Barrier | | Containment (CTMT) Barrier | |
|---|---|---|---|---|---|---|
| | **Loss** | **Potential Loss** | **Loss** | **Potential Loss** | **Loss** | **Potential Loss** |
| **A** RCS or SG Tube Leakage | None | 1. RVLMS < 21% plenum (Detector #8) | 1. An automatic or manual ECCS (SIAS) actuation required by **EITHER:**<br>• UNISOLABLE RCS leakage<br>• SG tube RUPTURE | 1. With letdown isolated, operation of the standby charging pump is required by **EITHER:**<br>• UNISOLABLE RCS leakage<br>• SG tube leakage<br>2. Pressurized thermal shock transient in excess of the upper (200ºF) subcooling P/T limit (Note 9)<br>**AND**<br>RCS pressure is rising | 1. A leaking or RUPTURED SG is FAULTED outside of containment | None |
| **B** Inadequate Heat Removal | 1. Rep CETs > 1200ºF | 1. Rep CETs > 700ºF<br>2. RCS heat removal cannot be established<br>**AND**<br>RCS subcooling < 24ºF | None | 1. RCS heat removal cannot be established<br>**AND**<br>RCS subcooling < 24ºF | None | 1. Rep CETs > 1200ºF<br>**AND**<br>Functional recovery procedures **not** effective within 15 min. (Note 1) |
| **C** CTMT Radiation / RCS Activity | 1. Containment radiation RU-148 > 2.1E+05 mR/hr **OR** RU-149 > 2.4E+05 mR/hr<br>2. Dose equivalent I-131 coolant activity > 300 µCi/gm | None | 1. Containment radiation RU-148 > 5.0E+04 mR/hr **OR** RU-149 > 5.6E+04 mR/hr | None | None | 1. Containment radiation RU-148 > 6.8E+06 mR/hr **OR** RU-149 > 7.8E+06 mR/hr |
| **D** CTMT Integrity or Bypass | None | None | None | None | 1. Containment isolation is required<br>**AND EITHER:**<br>• Containment integrity has been lost based on Emergency Coordinator judgment<br>• UNISOLABLE pathway from Containment to the environment exists<br>2. Indications of RCS leakage outside of Containment | 1. Containment pressure > 60 psig<br>2. Containment hydrogen concentration > 4.5%<br>3. Containment pressure > 8.5 psig with < 4350 gpm Containment Spray flow for ≥ 15 min. (Note 1) |
| **E** EC Judgment | 1. **Any** condition in the opinion of the Emergency Coordinator that indicates loss of the fuel clad barrier | 1. **Any** condition in the opinion of the Emergency Coordinator that indicates potential loss of the fuel clad barrier | 1. **Any** condition in the opinion of the Emergency Coordinator that indicates loss of the RCS barrier | 1. **Any** condition in the opinion of the Emergency Coordinator that indicates potential loss of the RCS barrier | 1. **Any** condition in the opinion of the Emergency Coordinator that indicates loss of the Containment barrier | 1. **Any** condition in the opinion of the Emergency Coordinator that indicates potential loss of the Containment barrier |

**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:**

| |
|---|
| 1. RVLMS < 21% plenum (Detector #8) |

**Definition(s):**

None

**Basis:**

21% plenum on RVLMS (Detector #8) is the minimum RVLMS indication above Top of Active Fuel (TOAF) which corresponds to 4 in. above the fuel alignment plate and is the last indication of inventory control (ref. 1, 2).

This reading indicates a reduction in reactor vessel water level sufficient to allow the onset of heat-induced cladding damage.

**PVNGS Basis Reference(s):**

1. Procedure 40OP-9ZZ16, *RCS Drain Operations*, Appendix M
2. Nuclear Fuel Management Analysis Calculation *TA-13-C00-2000-001, EOP Setpoint Document*
3. NEI 99-01, RCS or SG Tube Leakage Fuel Clad Potential Loss 1.A

**Barrier:** Fuel Clad

**Category:** B. Inadequate Heat Removal

**Degradation Threat:** Loss

**Threshold:**

1. Rep CETs > 1200ºF

**Definition(s):**

None

**Basis:**

Core Exit Thermocouples (CETs) are a component of Inadequate Core Cooling Instrumentation and provide an indirect indication of fuel clad temperature by measuring the temperature of the reactor coolant that leaves the core region. Although clad rupture due to high temperature is not expected for CET readings less than the threshold, temperatures of this magnitude signal significant superheating of the reactor coolant and core uncovery (ref. 1).

This reading indicates temperatures within the core are sufficient to cause significant superheating of reactor coolant.

Rep CET (Representative Core Exit Temperature) is a calculated temperature value generated by the Qualified Safety Parameter Display System (QSPDS). The QSPDS CET processing function generates a representative temperature based on a statistical analysis of thermocouples monitoring the reactor coolant temperature at the top of selected fuel assemblies.

**PVNGS Basis Reference(s):**

1. UFSAR Appendix 18B, System 80 Generic Inadequate Core Cooling Instrumentation
2. 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. Rep CETs > 700ºF

**Definition(s):**

None

**Basis:**

Core Exit Thermocouples (CETs) are a component of Inadequate Core Cooling Instrumentation and provide an indirect indication of fuel clad temperature by measuring the temperature of the reactor coolant that leaves the core region. If Rep CETs indicate > 700°F, subcooling has been lost for at least some regions of the core (ref. 1). 700°F qualifies as a condition representing a potential loss of the fuel clad barrier.

This reading indicates a reduction in reactor vessel water level sufficient to allow the onset of heat-induced cladding damage.

Rep CET (Representative Core Exit Temperature) is a calculated temperature value generated by the Qualified Safety Parameter Display System (QSPDS). The QSPDS CET processing function generates a representative temperature based on a statistical analysis of thermocouples monitoring the reactor coolant temperature at the top of selected fuel assemblies.

**PVNGS Basis Reference(s):**

1. UFSAR Appendix 18B, System 80 Generic Inadequate Core Cooling Instrumentation
2. NEI 99-01, Inadequate Heat Removal Fuel Clad Potential Loss 2.A

**Barrier:** Fuel Clad

**Category:** B. Inadequate Heat Removal

**Degradation Threat:** Potential Loss

**Threshold:**

2. RCS heat removal cannot be established
   **AND**
   RCS subcooling < 24ºF

**Definition(s):**

None

**Basis:**

In combination with RCS Potential Loss B.1, meeting this threshold results in a Site Area Emergency.

The steam generators (SGs) provide the normal means of heat transfer from the RCS to the main condenser and ultimate heat sink. Procedure 40EP-9EO03, *Loss of Coolant Accident*, requires maintenance of RCS heat removal at all times during a LOCA. Once RCS pressure and temperature are reduced, RCS heat removal can be provided by Shutdown Cooling (SDC) system. Once the SDC system is placed in service, the SG heat sink capability is no longer necessary (ref. 1).

If RCS subcooling approaches 24°F, the margin to superheated conditions is being reduced. Following an uncomplicated reactor trip, subcooling margin should be in excess of 50°F. Subcooling margin greater than 24°F ensures the fluid surrounding the core is sufficiently cooled and provides margin for reestablishing SI flow should subcooling deteriorate when SI flow is secured. Voids may exist in some parts of the RCS (e.g., Reactor Vessel head) but are permissible as long as core heat removal is maintained (ref. 2). RCS subcooling is determined using appropriate CET (natural circulation) or $T_{hot}$ (forced circulation) temperature indications. Upper head subcooling indication should not be used.

The combination of the threshold conditions indicates that RCS heat removal is under extreme challenge. This threshold addresses loss of functions required for hot shutdown with the reactor at pressure and temperature and thus a potential loss of the Fuel Clad barrier. This is also a potential loss of the RCS barrier and therefore results in at least a Site Area Emergency.

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.

**PVNGS Basis Reference(s):**

1. Procedure 40EP-9EO03, *Loss of Coolant Accident*
2. Procedure 40EP-9EO09, *Functional Recovery*
3. NEI 99-01, Inadequate Heat Removal Fuel Clad Loss 2.B

**Barrier:** Fuel Clad

**Category:** C. CTMT Radiation / RCS Activity

**Degradation Threat:** Loss

**Threshold:**

1. Containment radiation RU-148 > 2.1E+05 mR/hr **OR** RU-149 > 2.4E+05 mR/hr

**Definition(s):**

None

**Basis:**

The specified containment radiation monitor readings (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 with containment sprays operating. The values are based on calculated readings fifteen minutes after shutdown. 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 2-5% clad failure depending on core inventory and RCS volume).

Monitors used for this fission product barrier loss threshold are the Containment High Range Radiation Monitors RU-148 and RU-149 (ref. 1).

The radiation monitor reading corresponds to an instantaneous release of all reactor coolant mass into the containment, assuming that reactor coolant activity equals 300 μCi/gm dose equivalent I-131. Reactor coolant activity above this level is greater than that expected for iodine spikes and corresponds to an approximate range of 2% to 5% fuel clad damage. Since this condition indicates that a significant amount of fuel clad damage has occurred, it represents a loss of the Fuel Clad Barrier.

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

**PVNGS Basis Reference(s):**

1. Calculation 13-NC-ZY-216, *Determination of Containment Activities from High Radiation Monitors*
2. NEI 99-01, CTMT Radiation / RCS Activity Fuel Clad Loss 3.A

**Barrier:** Fuel Clad

**Category:** C. CTMT Radiation / RCS Activity

**Degradation Threat:** Loss

**Threshold:**

2. Dose equivalent I-131 coolant activity > 300 µCi/gm

**Definition(s):**

None

**Basis:**

Dose Equivalent Iodine (DEI) is determined by procedure 74ST-9RC02, *Reactor Coolant System Specific Activity Surveillance Test* (ref. 1).

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. Since this condition indicates that a significant amount of fuel clad damage has occurred, it represents a loss of the Fuel Clad Barrier (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.

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

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

**PVNGS Basis Reference(s):**

1. Procedure 74ST-9RC02, *Reactor Coolant System Specific Activity Surveillance Test*
2. NEI 99-01, CTMT Radiation / RCS Activity Fuel Clad Loss 3.B

**Barrier:** Fuel Clad

**Category:** C. CTMT Radiation / RCS Activity

**Degradation Threat:** Potential Loss

**Threshold:**

| |
|---|
| None |

**Barrier:** Fuel Clad

**Category:** D. CTMT Integrity or Bypass

**Degradation Threat:** Loss

**Threshold:**

| |
|---|
| None |

**Barrier:** Fuel Clad

**Category:** D. CTMT 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 function 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

**PVNGS Basis Reference(s):**

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

**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.

- 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 function 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.

**PVNGS 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 (SIAS) 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:**

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.

**PVNGS Basis Reference(s):**

1. Procedure 40EP-9EO01, *Reactor Trip*
2. Procedure 40EP-9EO03, *Loss of Coolant Accident*
3. Procedure 40EP-9EO04, *Steam Generator Tube Rupture*
4. NEI 99-01, RCS or SG Tube Leakage Reactor Coolant System Loss 1.A

**Barrier:** Reactor Coolant System

**Category:** A. RCS or SG Tube Leakage

**Degradation Threat:** Potential Loss

**Threshold:**

1. With letdown isolated, operation of the 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: two charging pumps are normally operating with a flow capacity of ~44 gpm each or a total of 88 gpm (ref. 1). Approximately 10 gpm of charging flow bypasses the RCS due to leakage through the RCP seals; thus, the normal charging lineup delivers 88 gpm – 10 gpm = 78 gpm (ref. 1). A third charging pump being required with letdown isolated is indicative of a substantial RCS leak.

If the standby charging pump is started in response to decreasing pressurizer level and following isolation of letdown and/or the leak pressurizer level can be subsequently maintained with just two charging pumps, this threshold is not exceeded.

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 following appropriate system isolation.

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.

**PVNGS Basis Reference(s):**

1. UFSAR Section 9.3.4, Chemical and Volume Control System
2. Procedure 40EP-9EO01, *Reactor Trip*
3. Procedure 40EP-9EO01, *Standard Post Trip Actions*
4. Procedure 40EP-9EO03, *Loss of Coolant Accident*
5. Procedure 40EP-9EO04, *Steam Generator Tube Rupture*
6. 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. Pressurized thermal shock transient in excess of the upper (200ºF) subcooling P/T limit (Note 9)

   **AND**

   RCS pressure is rising

Note 9: A pressurized thermal shock transient is defined as an UNPLANNED overcooling transient which causes RCS temperature to go below 500°F

**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:**

The "Potential Loss" threshold is defined by the upper subcooling P/T limit in combination with increasing RCS pressure which indicates an extreme challenge to the RCS barrier due to pressurized thermal shock transient. (ref. 1, 2, 3).

A pressurized thermal shock transient is defined as an unplanned overcooling transient which causes RCS temperature to go below 500°F (ref. 4).

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).

**PVNGS Basis Reference(s):**

1. Procedure 40EP-9EO05, *Excess Steam Demand*
2. Procedure 40EP-9EO09, *Functional Recovery*
3. Procedure 40EP-9EO10, *Standard Appendices Attachment 2 Figures*
4. Procedure 40DP-9AP17, *Standard Appendices Technical Guideline*
5. 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. RCS heat removal cannot be established
   **AND**
   RCS subcooling < 24ºF

**Definition(s):**

None

**Basis:**

In combination with FC Potential Loss B.1, meeting this threshold results in a Site Area Emergency.

The steam generators (SGs) provide the normal means of heat transfer from the RCS to the main condenser and ultimate heat sink. Procedure 40EP-9EO03, *Loss of Coolant Accident*, requires maintenance of RCS heat removal at all times during a LOCA. Once RCS pressure and temperature are reduced, RCS heat removal can be provided by Shutdown Cooling (SDC). Once the SDC is placed in service, the SG heat sink capability is no longer necessary (ref. 1).

If RCS subcooling approaches 24°F, the margin to superheated conditions is being reduced. Following an uncomplicated reactor trip, subcooling margin should be in excess of 50°F. Subcooling margin greater than 24°F ensures the fluid surrounding the core is sufficiently cooled and provides margin for reestablishing SI flow should subcooling deteriorate when SI flow is secured. Voids may exist in some parts of the RCS (e.g., Reactor Vessel head) but are permissible as long as core heat removal is maintained (ref. 2). RCS subcooling is determined using appropriate CET or $T_{hot}$ temperature indications. Upper head subcooling indication should not be used.

The combination of these conditions indicates the ultimate heat sink function is under extreme challenge. This threshold addresses loss of functions required for hot shutdown with the reactor at pressure and temperature and thus a potential loss of the Fuel Clad barrier. This is also a potential loss of the RCS barrier and therefore results in at least a Site Area Emergency.

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.

**PVNGS Basis Reference(s):**

1. Procedure 40EP-9EO03, *Loss of Coolant Accident*
2. Procedure 40EP-9EO09, *Functional Recovery*
3. NEI 99-01, Inadequate Heat Removal RCS Loss 2.B

**Barrier:** Reactor Coolant System

**Category:** C. CTMT Radiation/ RCS Activity

**Degradation Threat:** Loss

**Threshold:**

1. Containment radiation RU-148 > 5.0E+04 mR/hr **OR** RU-149 > 5.6E+04 mR/hr

**Definition(s):**

N/A

**Basis:**

Containment radiation monitor readings greater than the specified values (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.

The readings are derived assuming the instantaneous release and dispersal of the reactor coolant noble gas and iodine inventory associated with a concentration of 60 μCi/gm dose equivalent I-131 into the Containment atmosphere with containment sprays operating. The values are based on calculated readings fifteen minutes after shutdown.

Monitors used for this fission product barrier loss threshold are the Containment High Range Radiation Monitors RU-148 and RU-149 (ref. 1).

The radiation monitor reading corresponds to an instantaneous release of all reactor coolant mass into the containment, assuming that reactor coolant activity equals Technical Specification allowable limits. This value is lower than that specified for Fuel Clad Barrier Loss threshold C.1 since it indicates a loss of the RCS Barrier only.

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

**PVNGS Basis Reference(s):**

1. Calculation 13-NC-ZY-216, *Determination of Containment Activities from High Radiation Monitors*
2. NEI 99-01, CTMT Radiation / RCS Activity RCS Loss 3.A

**Barrier:** Reactor Coolant System

**Category:** C. CTMT Radiation/ RCS Activity

**Degradation Threat:** Potential Loss

**Threshold:**

None

**Barrier:** Reactor Coolant System

**Category:** D. CTMT Integrity or Bypass

**Degradation Threat:** Loss

**Threshold:**

| |
|---|
| None |

**Barrier:** Reactor Coolant System

**Category:** D. CTMT 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.

- 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 function 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.

**PVNGS 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 function 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.

**PVNGS Basis Reference(s):**

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

**Barrier:** Containment

**Category:** A. RCS or SG Tube Leakage

**Degradation Threat:** Loss

**Threshold:**

1. A leaking or RUPTURED SG is FAULTED outside of containment

**Definition(s):**

*FAULTED* - The term applied to a steam generator that has a steam or feedwater 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.

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. These type of condition 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 Atmospheric Dump Valve(s) 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. This includes the initial cooldown to 540ºF to isolate the ruptured SG using Atmospheric Dump Valves directed in the SGTR EOP. 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, steam traps, terry turbine exhaust, 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 (P-to-S) 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 the standby charging (makeup) pump (*RCS Barrier Potential Loss*) | Site Area Emergency per FS1.1 | Alert per FA1.1 |
| Requires an automatic or manual ECCS (SIAS) 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.

**PVNGS Basis Reference(s):**

1. Procedure 40EP-9EO01, *Reactor Trip*
2. Procedure 40EP-9EO01, *Standard Post Trip Actions*
3. Procedure 40EP-9EO03, *Loss of Coolant Accident*
4. Procedure 40EP-9EO10, *Excess Steam Demand*
5. Procedure 40EP-9EO04, *Steam Generator Tube Rupture*
6. 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

| |
|---|
| None |

**Barrier:** Containment

**Category:** B. Inadequate Heat Removal

**Degradation Threat:** Potential Loss

**Threshold:**

1. Rep CETs > 1200ºF

   **AND**

   Functional recovery procedure **not** effective within 15 minutes (Note 1)

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

**Definition(s):**

None

**Basis:**

Core Exit Thermocouples (CETs) are a component of Inadequate Core Cooling Instrumentation and provide an indirect indication of fuel clad temperature by measuring the temperature of the reactor coolant that leaves the core region. Although clad rupture due to high temperature is not expected for CET readings less than the threshold, temperatures of this magnitude signal significant superheating of the reactor coolant and core uncovery (ref. 1).

The 15 minute threshold starts when operators begin taking procedurally directed functional recovery actions.

If CET readings are greater than 1,200°F (ref. 1), the Fuel Clad barrier is also lost.

Rep CET (Representative Core Exit Temperature) is a calculated temperature value generated by the Qualified Safety Parameter Display System (QSPDS). The QSPDS CET processing function generates a representative temperature based on a statistical analysis of thermocouples monitoring the reactor coolant temperature at the top of selected fuel assemblies.

This condition represents an IMMINENT core melt sequence which, if not corrected, could lead to vessel failure and an increased potential for containment failure. For this condition to occur, there must already have been a loss of the RCS Barrier and the Fuel Clad Barrier. If implementation of a procedure(s) to restore adequate core cooling is not effective (successful) within 15 minutes, it is assumed that the event trajectory will likely lead to core melting and a subsequent challenge of the Containment Barrier.

The restoration procedure is considered “effective” if core exit thermocouple readings are decreasing and/or if reactor vessel level is increasing. Whether or not the procedure(s) will be effective should be apparent within 15 minutes. The Emergency Coordinator should escalate the emergency classification level as soon as it is determined that the procedure(s) will not be effective.

Severe accident analyses (e.g., NUREG-1150) have concluded that function restoration procedures can arrest core degradation in a significant fraction of core damage scenarios, and that the likelihood of containment failure is very small in these events. Given this, it is appropriate to provide 15 minutes beyond the required entry point to determine if procedural actions can reverse the core melt sequence.

**PVNGS Basis Reference(s):**

1. UFSAR Appendix 18B, System 80 Generic Inadequate Core Cooling Instrumentation
2. Procedure 40EP-9EO09, *Functional Recovery*
4. NEI 99-01, Inadequate Heat Removal Containment Potential Loss 2.A

**Barrier:** Containment

**Category:** C. CTMT Radiation/RCS Activity

**Degradation Threat:** Loss

**Threshold:**

| |
|---|
| None |

**Barrier:** Containment

**Category:** C. CTMT Radiation/RCS Activity

**Degradation Threat:** Potential Loss

**Threshold:**

1. Containment radiation RU-148 > 6.8E+06 mR/hr **OR** RU-149 > 7.8E+06 mR/hr

**Definition(s):**

None

**Basis:**

Containment radiation monitor readings greater than the values shown (ref. 1) indicate significant fuel damage well in excess of that required for loss of the RCS barrier and the Fuel Clad barrier.

The reading is derived assuming the instantaneous release and dispersal of the reactor coolant noble gas and iodine inventory associated with 20% clad failure into the Containment atmosphere with containment sprays operating. The values are based on calculated readings fifteen minutes after shutdown.

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 RU-148 and RU-149 (ref. 1).

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 related Fuel Clad Barrier Loss and RCS Barrier Loss thresholds.

NUREG-1228, *Source Estimations During Incident Response to Severe Nuclear Power Plant Accidents*, indicates the fuel clad failure must be greater than approximately 20% in order for there to be a major release of radioactivity requiring offsite protective actions. For this condition to exist, there must already have been a loss of the RCS Barrier and the Fuel Clad Barrier. It is therefore prudent to treat this condition as a potential loss of containment which would then escalate the ECL to a General Emergency.

**PVNGS Basis Reference(s):**

1. Calculation 13-NC-ZY-216, *Determination of Containment Activities from High Radiation Monitors*
2. NEI 99-01, CTMT Radiation / RCS Activity Containment Potential Loss 3.A

**Barrier:** Containment

**Category:** D. CTMT 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:**

Containment isolations are initiated by the Containment Isolation Actuation System (CIAS) in response to a high containment pressure signal or low pressurizer pressure below the SIAS setpoint (ref. 1, 2).

A penetration is considered isolated with at least one containment isolation valve closed. This may include a check valve if there is no indication that it has failed to close.

Palo Verde specific operating experience is that a High Pressure Seal Cooler (HPSC) leak to the Nuclear Cooling Water (NC) System must be isolated to containment within 15 minutes of discovery due to the location of the NC system expansion tank and potential dose concerns on the Auxiliary Building roof.

These thresholds address a situation where containment isolation is required and one of two conditions exists as discussed below. Users are reminded that there may be accident and release conditions that simultaneously meet both bulleted thresholds.

First Threshold – Containment integrity has been lost, i.e., the actual containment atmospheric leak rate likely exceeds that associated with allowable leakage (or sometimes referred to as design leakage). Following the release of RCS mass into containment, containment pressure will fluctuate based on a variety of factors; a loss of containment integrity condition may (or may not) be accompanied by a noticeable drop in containment pressure. Recognizing the inherent difficulties in determining a containment leak rate during accident conditions, it is expected that the Emergency Coordinator will assess this threshold using judgment and with due consideration given to current plant conditions and available operational and radiological data (e.g., containment pressure, readings on radiation monitors outside containment, operating status of containment pressure control equipment, etc.).

Refer to the middle piping run of Figure 1. Two simplified examples are provided. One is leakage from a penetration and the other is leakage from an in-service system valve. Depending upon radiation monitor locations and sensitivities, the leakage could be detected by any of the four monitors depicted in the figure.

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.

<u>Second Threshold</u> – 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. There must be a release involved to atmosphere or into another plant structure outside of Containment.

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 or atmosphere, then no threshold has been met.

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.

**PVNGS Basis Reference(s):**

1. UFSAR Section 6.2.1.5.3.8, Containment Purge System
2. UFSAR Section 6.2.4, Containment Isolation System
3. NEI 99-01, CTMT Integrity or Bypass Containment Loss 4.A

**Barrier:** Containment

**Category:** D. CTMT Integrity or Bypass

**Degradation Threat:** Loss

**Threshold:**

| | |
|---|---|
| 2. | Indications of RCS leakage outside of Containment |

**Definition(s):**

None

**Basis:**

Procedure 40AO-9ZZ02, *Excessive RCS Leakrate*, (ref. 1) provides instructions to identify and isolate a LOCA outside of the containment. Potential RCS leak pathways outside containment include (ref. 1, 2):

- Nuclear Cooling System (such as RCP high pressure seal cooler to NC system)
- Safety Injection
- Chemical & Volume Control
- RCS sample lines

Palo Verde specific operating experience is that a High Pressure Seal Cooler (HPSC) leak to the Nuclear Cooling Water (NC) System must be isolated to containment within 15 minutes of discovery due to the location of the NC system expansion tank and potential dose concerns on the Auxiliary Building roof.

| | **RCS Leakage Outside of Containment?** | |
|---|---|---|
| **RCS 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 the standby charging (makeup) pump (*RCS Barrier Potential Loss*) | Site Area Emergency per FS1.1 | Alert per FA1.1 |
| Requires an automatic or manual ECCS (SIAS) actuation (*RCS Barrier Loss*) | Site Area Emergency per FS1.1 | Alert per FA1.1 |

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.

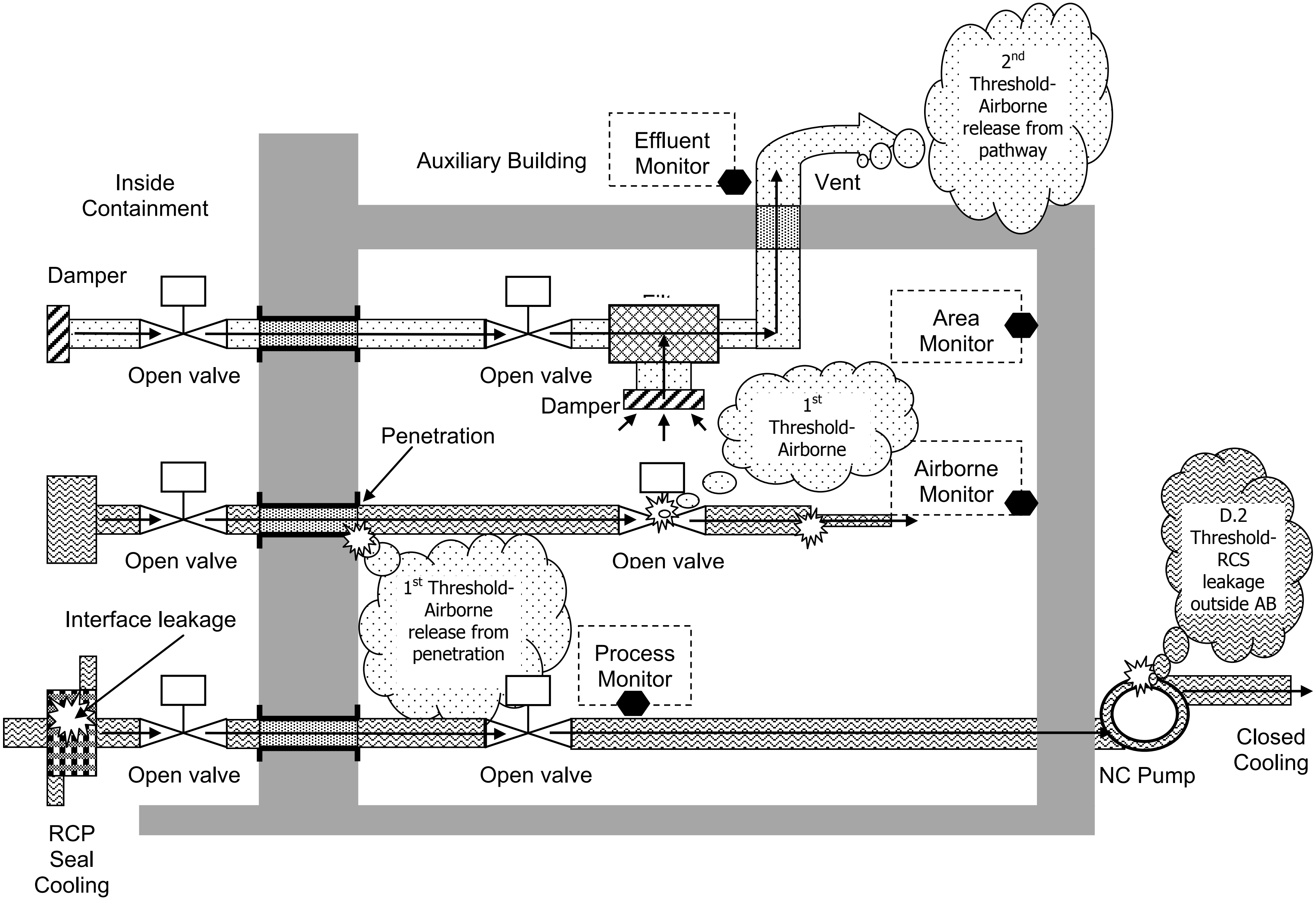
Refer to the bottom piping run of Figure 1. In this simplified example, leakage in an RCP seal cooler is allowing radioactive material to enter the Auxiliary Building and then atmosphere. The radioactivity would be detected by the Process Monitor. If the Nuclear Cooling System (NC) pump developed a leak that allowed steam/water to leak to atmosphere, then this threshold is met.

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.

**PVNGS Basis Reference(s):**

1. Procedure 40AO-9ZZ02, *Excessive RCS Leakrate*
2. Procedure 40EP-9EO03, *Loss of Coolant Accident*
3. NEI 99-01, CTMT Integrity or Bypass Containment Loss

**Figure 1: Containment Integrity or Bypass Examples**



**Barrier:** Containment

**Category:** D. CTMT Integrity or Bypass

**Degradation Threat:** Potential Loss

**Threshold:**

1. Containment pressure > 60 psig

**Definition(s):**

None

**Basis:**

60 psig is the containment design pressure (ref. 1).

If containment pressure exceeds the design pressure, there exists a potential to lose the Containment Barrier. To reach this level, there must be an inadequate core cooling condition for an extended period of time; therefore, the RCS and Fuel Clad barriers would already be lost. Thus, this threshold is a discriminator between a Site Area Emergency and General Emergency since there is now a potential to lose the third barrier.

**PVNGS Basis Reference(s):**

1. UFSAR Section 1.2.12.1, Containment Building
2. NEI 99-01, CTMT Integrity or Bypass Containment Potential Loss 4.A

**Barrier:** Containment

**Category:** D. CTMT Integrity or Bypass

**Degradation Threat:** Potential Loss

**Threshold:**

2. Containment hydrogen concentration > 4.5%

**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, 3).

PVNGS is equipped with a Containment Hydrogen Control (HP) system which serves to limit or reduce combustible gas concentrations in the Containment. The HP system is an engineered safety feature with redundant hydrogen recombiners, hydrogen mixing system, hydrogen monitoring subsystem and a backup hydrogen purge subsystem. The HP system is designed to maintain the Containment hydrogen concentration below 4% by volume (ref. 1, 2). HP system operation is prescribed by EOPs if Containment hydrogen concentration should reach 0.7% by volume (minimum detectable) (ref. 3).

The PVNGS Safety Function Status Check for LOCA, Containment Combustible Gas Control (procedure 40EP-9EO03, *Loss of Coolant Accident*), uses 4.5% as an acceptance criterion, which represents the Hydrogen Recombiner Function Failure Indication. This value should not be exceeded if the hydrogen recombiners are operating as desired.

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 monitor indicators (HPA-AI-9 and HPB-AI-10) with a range of 0% to 10% provide indication on Control Room Panel B02 (ref. 2).

The existence of an explosive mixture means, at a minimum, that the containment atmospheric hydrogen concentration is sufficient to support a hydrogen burn (e.g., 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.

**PVNGS Basis Reference(s):**

1. UFSAR Section 6.2.5, Combustible Gas Control in Containment
2. Design Basis Manual – HP Containment Hydrogen Control System
3. Procedure 40DP-9AP14, *Functional Recovery Technical Guidline,* Section 15.0 Containment Combustible Gas Control
4. NEI 99-01, CTMT Integrity or Bypass Containment Potential Loss 4.B

**Barrier:** Containment

**Category:** D. CTMT Integrity or Bypass

**Degradation Threat:** Potential Loss

**Threshold:**

| |
|---|
| 3. Containment pressure > 8.5 psig with < 4350 gpm Containment Spray flow for ≥ 15 minutes (Note 1) |

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

**Definition(s):**

None

**Basis:**

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 (RWT) 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 RWT to the Containment sumps (ref. 1).

The Containment pressure high-high setpoint (8.5 psig) is the pressure at which the Containment Spray equipment should actuate and begin performing its function (ref. 2). Consistent with the design requirement, “one full train of depressurization equipment” is therefore defined to be the availability of one train of Containment Spray providing a minimum of 4350 gpm spray flow (ref. 3). 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 but not including containment venting strategies) are either lost or performing in a degraded manner.

**PVNGS Basis Reference(s):**

1. UFSAR Section 6.2.2, Containment Heat Removal System
2. UFSAR Table 7.3-11A, ESFAS Setpoints and Margins to Actuation
3. Procedure 40EP-9EO01, *Standard Post Trip Actions*
4. NEI 99-01, CTMT 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.

- 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 function 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.

**PVNGS Basis Reference(s):**

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

**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.

- 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 function 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 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.

**PVNGS Basis Reference(s):**

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

**Background**

NEI 99-01, Revision 6 ICs AA3 and HA5 prescribe declaration of an Alert based on impeded access to rooms or areas (due to either area radiation levels or hazardous gas concentrations) where equipment necessary for normal plant operations, cooldown or shutdown is located. These areas are intended to be plant operating mode dependent. Specifically the Developers Notes For AA3 and HA5 states:

> *The 'site-specific list of plant rooms or areas with entry-related mode applicability identified' should specify those rooms or areas that contain equipment which require a manual/local action as specified in operating procedures used for normal plant operation, cooldown and shutdown. Do not include rooms or areas in which actions of a contingent or emergency nature would be performed (e.g., an action to address an off-normal or emergency condition such as emergency repairs, corrective measures or emergency operations). In addition, the list should specify the plant mode(s) during which entry would be required for each room or area.*
>
> *The list should not include rooms or areas for which entry is required solely to perform actions of an administrative or record keeping nature (e.g., normal rounds or routine inspections).*

Further, as specified in IC HA5:

> *The list need not include the Control Room if adequate engineered safety/design features are in place to preclude a Control Room evacuation due to the release of a hazardous gas. Such features may include, but are not limited to, capability to draw air from multiple air intakes at different and separate locations, inner and outer atmospheric boundaries, or the capability to acquire and maintain positive pressure within the Control Room envelope.*

**PVNGS Table R-2 and H-2 Bases**

A review of station operating procedures identified the following mode dependent in-plant actions and associated areas that are required for normal plant operation, cooldown or shutdown:

| **Location-**<br>**Safe Shutdown Area/Room** | **Modes-**<br>**1, 2** | **Modes-**<br>**3, 4 or 5** |
|---|---|---|
| LPSI Pumps A and B | SDC Equipment.<br>- *No entry required*<br>Inventory Control Equipment | Shut Down Cooling (SDC)<br>- *No entry required*<br>Inventory Control Equipment<br>- *No entry required*<br>Reactivity Control.<br>- *No entry required* |
| Containment Spray Pumps A and B | Containment Pressure Control<br>- *No entry required* | Shut Down Cooling (SDC)<br>- *No entry required*<br>Inventory Control Equipment<br>- *No entry required*<br>Reactivity Control.<br>- *No entry required* |
| HPSI Pumps A and B | Inventory Control Equipment.<br>- *No entry required*<br>Reactivity Control.<br>- *No entry required* | Inventory Control Equipment.<br>- *No entry required*<br>Reactivity Control.<br>- *No entry required* |
| Aux. Bldg 120 West Electrical Penetration Room | Electrical Power.<br>- *No entry required* | Electrical Power.<br>- *No entry required* |
| Aux. Bldg 100 East Electrical Penetration Room | Electrical Power.<br>- *No entry required* | Electrical Power.<br>- *No entry required* |
| Essential Cooling Water Pumps | Support Equipment for Habitability Control, Containment Temperature, Control and Shutdown Cooling<br>- *No entry required* | Support Equipment for Habitability Control, Containment Temperature, Control and Shutdown Cooling<br>*No entry required* |
| Control Building 100 foot 4160 Class Switchgear Room A & B | Electrical Power.<br>- *No entry required* | Electrical Power.<br>- *Entry required to access the DC equipment Rooms C and D- Modes 4 and 5* |
| Control Building 100 foot Class DC Equipment Rooms A & B | Electrical Power.<br>- *No entry required* | Electrical Power.<br>- *No entry required* |
| Control Building 100 foot Class DC Equipment Rooms C & D | Electrical Power.<br>- *No entry required* | Electrical Power.<br>- Energize LTOP Isolation Valves for SDC. Procedure 40OP-9ZZ23, *Modes 4 and 5* |
| Emergency Diesel Generators A & B | Electrical Power.<br>- *No entry required* | Electrical Power.<br>- *No entry required* |
| Emergency Diesel Generators Day Tank Rooms | Electrical Power.<br>- *No entry required* | Electrical Power.<br>- *No entry required* |
| EDG Building HVAC Room | - *No entry required* | - *No entry required* |
| Control Building 160 ft Electrical Cable Spreading | - *No entry required* | - *No entry required* |
| Control Building 120 ft Electrical Cable Spreading | - *No entry required* | - *No entry required* |
| Control Building 80 ft Essential Chiller Rooms | - *No entry required* | - *No entry required* |
| Control Building Battery Rooms | - *No entry required* | - *No entry required* |

ATTACHMENT 3

Safe Operation & Shutdown Rooms Tables R-2 & H-2 Bases

| **Location-<br>Safe Shutdown Area/Room** | **Modes-<br>1, 2** | **Modes-<br>3, 4 or 5** |
|---|---|---|
| A, B, C and D | | |
| Turbine Building Elevations | - *No entry required* | - *No entry required* |
| Main Steam Support Structure 140, 120 and 100 foot elevations | - *No entry required* | - *No entry required* |
| Aux. Feedwater Pump Room A and B | Steam Generator Heat Removal<br>- *No entry required* | Steam Generator Heat Removal<br>- *No entry required* |
| Spray Pond Pump Rooms A and B | Support Equipment for Habitability Control, Containment Temperature, Control and Shutdown Cooling<br>*No entry required* | Support Equipment for Habitability Control, Containment Temperature, Control and Shutdown Cooling<br>*No entry required* |

**<u>Table R-2 & H-2 Results</u>**

| **Table R-2 & H-2 Safe Operation & Shutdown Rooms** | |
|---|---|
| **Room** | **Mode Applicability** |
| Control Building 100 ft. Class DC Equipment Room C | 4, 5 |
| Control Building 100 ft. Class DC Equipment Room D | 4, 5 |

**<u>Plant Operating Procedures Reviewed</u>**

1. Procedure 40OP-9ZZ05, *Power Operations*
2. Procedure 40OP-9ZZ23, *Outage GOP*
3. Procedure 40OP-9ZZ10, *Mode 3 to Mode 5 Operations*
4. Procedure 40OP-9SI01, *Shutdown Cooling Initiation*

**Safety System**

A system required for safe plant operation, cooling down the plant and/or placing it in the cold shutdown condition, including the ECCS. These are typically systems classified as safety-related (as defined in 10 CFR 50.2):

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

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

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

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

The SAFETY SYSTEMS included in this definition are those included to satisfy Criteria 1, 2 or 3 of 10 CFR 50.36(c)(2)(ii). Systems included by this definition are:

**Structures - All Modes** (except as noted)

- Containment Building
- Auxiliary Building
- Diesel Building
- Fuel Building
- Spray Pond
- Control Building
- Main Steam Support Structure Mode 1-4 and Mode 5 when steam generators are required per Technical Specifications

**Modes 1-4**

- Reactor Coolant System (RC)
- Safety Injection (SI)
- Refueling Water Tank
- Containment Air Locks
- Containment Isolation Valves- except when the penetration is isolated and out of service.
- Containment Spray System (SI) Modes 1-3 and Mode 4 ≥ 385 psia
- Main Steam Safety Valves (SG) Modes 1-3
- Main Steam Isolation Valves (SG) Mode 1 and Modes 2-4 except when closed and deactivated
- Main Feedwater Isolation Valves (SG) Mode 1-4 except when closed and deactivated or isolated by another valve
- Atmospheric Dump Valves (SG) Modes 1-3, Mode 4 when Steam Generators are relied on for heat removal
- Auxiliary Feedwater System (AF) Modes 1-3, Mode 4 when Steam Generators are relied on for heat removal

- Condensate Storage Tank (CT) Modes 1-3, Mode 4 when Steam Generators are relied on for heat removal
- Essential Cooling Water System (EW)
- Essential Chill Water System (EC)
- Essential Spray Pond System (SP)
- Ultimate Heat Sink (SP)
- Control Room Essential Filtration and Ventilation (HJ)
- Engineered Safety Features Pump Room Exhaust Cleanup (HF)
- Diesel Generators (DG)
- Diesel Fuel Oil System (DF)
- DC Sources (PK)
- Class Battery Chargers (PK)
- Class Instrument Invertors (PN)
- Distribution Systems (PB, PG, PH, PK and PN)
- Shutdown Cooling System (SI) Mode 4
- Reactor Protection System (RPS)
- Engineered Safety Features Actuation System (ESFAS)
- Balance of Plant Engineered Safety Features Actuation System (BOP-ESFAS)

## **Modes 5 and 6**

- Reactor Coolant System (RC)
- Shutdown Cooling System (SI)
- Diesel Generators (DG) Normally only one train required by TS
- Diesel Fuel Oil System (DF) Normally only one train required by TS
- DC Sources (PK) Normally only one train required by TS
- Class Battery Chargers (PK) Normally only one train required by TS
- Class Instrument Invertors (PN) Normally only one train required by TS
- Distribution Systems (PB, PG, PH, PK and PN) Normally only one train required by TS
- Control Room Essential Filtration and Ventilation (HJ)
- Essential Cooling Water System (EW) Train(s) supporting Shutdown Cooling
- Essential Spray Pond System (SP) Train(s) supporting Shutdown Cooling and/or DG
- Ultimate Heat Sink (SP) Train(s) supporting Shutdown Cooling and/or DG

# Attachment D - Updated EAL Wall Charts (Information Only)

| | 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 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
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 radioactivity > 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 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 116 ft. (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
Spent fuel pool level ≤ 116 ft. (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 high alarm on **any** of the following:
- RU-16 Containment Operating Level Area
- RU-17 Incore Instrument Area (when installed)
- RU-19 New Fuel Area
- RU-31 Spent Fuel Pool Area
- RU-33 Refueling Machine Area (when installed)
- RU-37/38 Containment Purge Exhaust Area
- RU-143 Plant Vent
- RU-145 Fuel Building Vent

**RA2.3** 1 2 3 4 5 6 DEF
Spent fuel pool level ≤ 125 ft. (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 (E0204A on PCN-E02) or level indication (visual or RWLIS)
**AND**
UNPLANNED alert alarm on **any** of the following corresponding radiation monitors
- RU-16 Containment Operating Level Area
- RU-17 Incore Instrument Area (when installed)
- RU-19 New Fuel Area
- RU-31 Spent Fuel Pool Area
- RU-33 Refueling Machine Area (when installed)

**Table R-1 Effluent Monitor Classification Thresholds**

| | Release Point | Monitor | GE | SAE | Alert | UE |
|---|---|---|---|---|---|---|
| Gaseous | Plant Vent Low | RU-143 Ch 1 | ----- | ----- | 1.04E-02 μCi/cc | 1.22E-03 μCi/cc |
| Gaseous | Plant Vent High | RU-144 Ch 1 | 1.04E+00 μCi/cc | 1.04E-01 μCi/cc | ----- | ----- |
| Gaseous | Fuel Building Low | RU-145 Ch1 | ----- | ----- | ----- | 1.13E-02 μCi/cc |
| Gaseous | Fuel Building High | RU-146 Ch 1 | ----- | 3.50E+00 μCi/cc | 3.50E-01 μCi/cc | ----- |
| Gaseous | Fuel Building High | RU-146 Ch 2 | 3.50E+01 μCi/cc | ----- | ----- | ----- |

## 3 Area Radiation Levels

**GENERAL EMERGENCY** — None

**SITE AREA EMERGENCY**

**Table R-2 Safe Operation & Shutdown Rooms**

| Room | Mode Applicability |
|---|---|
| Control Building 100 ft. Class DC Equipment Room C | 4, 5 |
| Control Building 100 ft. Class DC Equipment Room D | 4, 5 |

**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
- Central Alarm Station (CAS) (by survey)

**RA3.2** 4 5
An UNPLANNED event results in radiation levels that prohibit or IMPEDE access to **any** Table R-2 room (Note 5)

# E ISFSI

## 1 Confinement Boundary

**GENERAL EMERGENCY** — None

**SITE AREA EMERGENCY** — None

**ALERT** — For an ISFSI security event refer to HA1.1

**UNUSUAL EVENT** — Damage to a loaded cask CONFINEMENT BOUNDARY

**EU1.1** 1 2 3 4 5 6 DEF
Damage to a loaded canister CONFINEMENT BOUNDARY as indicated by an on-contact radiation reading on the surface of a loaded spent fuel cask greater than **any** of the following:
- 100 mrem/hr (γ + η) on the side of the cask
- 100 mrem/hr (γ + η) on the top of the cask
- 200 mrem/hr (γ + η) at the air inlets or outlets

# H Hazards

## 1 Security

**GENERAL EMERGENCY** — None

**SITE AREA EMERGENCY** — HOSTILE ACTION within the PLANT PROTECTED AREA

**HS1.1** 1 2 3 4 5 6 DEF
A HOSTILE ACTION is occurring or has occurred within the PLANT PROTECTED AREA as reported by the Security Shift Supervision

**ALERT** — HOSTILE ACTION within the SECURED 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 SECURED OWNER CONTROLLED AREA as reported by the Security Shift Supervision
**OR**
A validated notification from NRC of an aircraft attack threat within 30 min. of the site

**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 Supervision
**OR**
Notification of a credible security threat directed at the site
**OR**
A validated notification from the NRC providing information of an aircraft threat

## 2 Seismic Event

**GENERAL EMERGENCY** — None

**SITE AREA EMERGENCY** — None

**ALERT** — Refer to CA6.1 or SA9.1 for potential escalation.

**UNUSUAL EVENT** — Seismic event greater than OBE levels

**HU2.1** 1 2 3 4 5 6 DEF
Seismic event > OBE as indicated on Control Panel A-J-SMN-C01

## 3 Natural or Tech. Hazard

**SITE AREA EMERGENCY** — None

**ALERT** — Refer to CA6.1 or SA9.1 for potential escalation.

**UNUSUAL EVENT** — Hazardous event

**HU3.1** 1 2 3 4 5 6 DEF
A tornado strike within the PLANT 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 PLANT 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:** This RCS Potential Loss threshold only applies to a pressurized thermal shock transient which is defined as an UNPLANNED overcooling transient which causes RCS temperature to go below 500°F

**Note 10:** Variations in RCS boron concentration, temperature and Containment Temperature from those used in RWLIS calibration will induce indication errors. Refer to *Operator Assistance Program RWLIS_Spreadsheet.xls.*

**Note 11:** Downcomer flow instruments are also credited for auxiliary feed flow indication

## 4 Fire

**SITE AREA EMERGENCY** — None

**ALERT** — Refer to CA6.1 or SA9.1 for potential escalation.

**Table H-1 Fire Areas**
- Containment
- Auxiliary Building
- Control Building
- Diesel Generator Building
- Diesel Generator Fuel Oil Storage Tanks
- Fuel Building
- Main Steam Support Structure
- Refueling Water Tank
- Essential Spray Pond System
- Condensate Storage Tank

**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 or ISFSI 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 or ISFSI PROTECTED AREA that requires firefighting support by an offsite fire response agency to extinguish

## 5 Hazardous Gas

**SITE AREA EMERGENCY** — None

**ALERT** — Gaseous release IMPEDING access to equipment necessary for normal plant operations, cooldown or shutdown

**HA5.1** 4 5
Release of a toxic, corrosive, asphyxiant or flammable gas into **any** Table H-2 room
**AND**
Entry into the room is prohibited or IMPEDED (Note 5)

**Table H-2 Safe Operation & Shutdown Rooms**

| Room | Mode Applicability |
|---|---|
| Control Building 100 ft. Class DC Equipment Room C | 4, 5 |
| Control Building 100 ft. Class DC Equipment Room D | 4, 5 |

## 6 Control Room Evacuation

**GENERAL EMERGENCY** — None

**SITE AREA EMERGENCY** — Inability to control a key safety function from outside the Control Room

**HS6.1** 1 2 3 4 5 6
An event has resulted in plant control being transferred from the Control Room to the Remote Shutdown Panel (RSP)
**AND**
Control of **any** of the following key safety functions is **not** re-established within 15 min. (Note 1):
- Reactivity (Modes 1, 2 and 3 **only**)
- Core cooling
- RCS heat removal

**ALERT** — Control Room evacuation resulting in transfer of plant control to alternate locations

**HA6.1** 1 2 3 4 5 6 DEF
An event has resulted in plant control being transferred from the Control Room to the Remote Shutdown Panel (RSP)

## 7 Emergency Coordinator Judgment

**GENERAL EMERGENCY** — Other conditions existing that in the judgment of the Emergency Director warrant declaration of General Emergency

**HG7.1** 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 Director warrant declaration of Site Area Emergency

**HS7.1** 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 Director warrant declaration of an Alert

**HA7.1** 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 Director warrant declaration of a UE

**HU7.1** 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 Power Operation | 2 Startup | 3 Hot Standby | 4 Hot Shutdown | 5 Cold Shutdown | 6 Refueling | DEF Defueled

EP-0901 Appendix A, Rev.[xx]
EAL Classification Matrix


**ALL CONDITIONS**

Prepared for APS by: Operations Support Services, Inc. – 5/5/15

| | 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 to emergency 4.16KV buses PBA-S03 and PBB-S04
**AND EITHER:**
- Restoration of at least one emergency bus in < 4 hours is **not** likely (Note 1)
- Rep CET reading > 1200 °F

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 to emergency 4.16KV buses PBA-S03 and PBB-S04 for ≥ 15 min.
**AND**
Loss of 125 VDC power based on battery bus voltage indications < 112 VDC on **both** vital DC buses PKA-M41 and PKB-M42 for ≥ 15 min. (Note 1)

**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 to emergency 4.16KV buses PBA-S03 and PBB-S04 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 PBA-S03 and PBB-S04 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 PBA-S03 and PBB-S04 for ≥ 15 min. (Note 1)

**Table S-1 AC Power Supplies**

**Offsite:**
- SUT (normal)
- SUT (alternate)
- SBOG #1 **AND** SBOG #2 (if already aligned)

**Onsite:**
- DG A
- DG B

## 2 Loss of Vital DC Power

**SITE AREA EMERGENCY**

Loss of **all** vital DC power for 15 minutes or longer

**SS2.1** | 1 | 2 | 3 | 4 |

Loss of 125 VDC power based on battery bus voltage indications < 112 VDC on **both** vital DC buses PKA-M41 and PKB-M42 for ≥ 15 min. (Note 1)

**ALERT**: None

**UNUSUAL EVENT**: None

## 3 Loss of Control Room Indications

**GENERAL EMERGENCY**: None

**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)

**Table S-2 Safety System Parameters**

- Reactor power
- RCS level
- RCS pressure
- CET temperature
- Level in at least one S/G
- Auxiliary feed flow to at least one S/G (Note 11)

**Table S-3 Significant Transients**

- Reactor trip
- Runback > 25% thermal power
- Electrical load rejection > 25% electrical load
- Reactor power cutback
- ECCS actuation

## 4 RCS Activity

**GENERAL EMERGENCY**: None

**ALERT**: None

**UNUSUAL EVENT**

Reactor coolant activity greater than Technical Specification allowable limits

**SU4.1** | 1 | 2 | 3 | 4 |

Letdown Monitor RU-155D reading > high alarm

**SU4.2** | 1 | 2 | 3 | 4 |

Sample analysis indicates RCS activity > Technical Specification LCO 3.4.17 limits

## 5 RCS Leakage

**GENERAL 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**
Reactor coolant leakage to a location outside containment > 25 gpm for ≥ 15 min.
(Note 1)

## 6 RPS 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**:
- Rep CET > 1200°F
- RCS subcooling < 24°F

**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 (B05 or B01) 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** RPS setpoint is exceeded
**AND**
A subsequent automatic trip or manual trip action taken at the reactor control console (B05 or B01) 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 (B05 or B01) is successful in shutting down the reactor as indicated by reactor power ≤ 5% (Note 8)

**Table S-4 Communications Methods**

| System | Onsite | ORO | NRC |
|---|---|---|---|
| PBX | X | X | X |
| Plant Page | X | | |
| Two-Way Radio | X | | |
| FTS (ENS) | | | X |
| Telephone Ringdown Circuits (NAN) | | X | |
| Cellular Phones | | X | X |

## 7 Loss of Comm.

**SITE AREA EMERGENCY**: None

**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 Offsite Response Organization (ORO) communication methods
**OR**
Loss of **all** Table S-4 NRC communication methods

## 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 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:** This RCS Potential Loss threshold only applies to a pressurized thermal shock transient which is defined as an UNPLANNED overcooling transient which causes RCS temperature to go below 500°F

**Note 11:** Downcomer flow instruments are also credited for auxiliary feed flow indication

## 8 CTMT Failure

**SITE AREA EMERGENCY**: None

**ALERT**: None

**UNUSUAL EVENT**

Failure to isolate containment or loss of containment pressure control

**SU8.1** | 1 | 2 | 3 | 4 |

**EITHER**:
- **Any** penetration is not closed when required within 15 min. of a VALID CIAS signal
- Containment pressure > 8.5 psig with < 4350 gpm Containment Spray flow for ≥ 15 min.

(Note 1)

## 9 Hazardous Event Affecting Safety Systems

**GENERAL EMERGENCY**: None

**Table S-5 Hazardous Events**

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

**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

**FG1.1** | 1 | 2 | 3 | 4 |

Loss of **any** two barriers
**AND**
Loss or potential loss of third barrier (Table F-1)

**FS1.1** | 1 | 2 | 3 | 4 |

Loss or potential loss of **any** two barriers (Table F-1)

**FA1.1** | 1 | 2 | 3 | 4 |

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

## 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 (CTMT) Barrier Loss | Containment (CTMT) Barrier Potential Loss |
|---|---|---|---|---|---|---|
| **A** RCS or SG Tube Leakage | None | 1. RVLMS < 21% plenum (Detector #8) | 1. An automatic or manual ECCS (SIAS) actuation required by **EITHER:** • UNISOLABLE RCS leakage • SG tube RUPTURE | 1. With letdown isolated, operation of the standby charging pump is required by **EITHER:** • UNISOLABLE RCS leakage • SG tube leakage 2. Pressurized thermal shock transient in excess of the upper (200°F) subcooling P/T limit (Note 9) **AND** RCS pressure is rising | 1. A leaking or RUPTURED SG is FAULTED outside of containment | None |
| **B** Inadequate Heat Removal | 1. Rep CETs > 1200ºF | 1. Rep CETs > 700ºF 2. RCS heat removal cannot be established **AND** RCS subcooling < 24ºF | None | 1. RCS heat removal cannot be established **AND** RCS subcooling < 24ºF | None | 1. Rep CETs > 1200ºF **AND** Functional recovery procedure **not** effective within 15 min. (Note 1) |
| **C** CTMT Radiation / RCS Activity | 1. Containment radiation RU-148 > 2.1E+05 mR/hr **OR** RU-149 > 2.4E+05 mR/hr 2. Dose equivalent I-131 coolant activity > 300 µCi/gm | None | 1. Containment radiation RU-148 > 5.0E+04 mR/hr **OR** RU-149 > 5.6E+04 mR/hr | None | None | 1. Containment radiation RU-148 > 6.8E+06 mR/hr **OR** RU-149 > 7.8E+06 mR/hr |
| **D** CTMT 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. Containment pressure > 60 psig 2. Containment hydrogen concentration > 4.5% 3. Containment pressure > 8.5 psig with < 4350 gpm Containment Spray flow for ≥ 15 min. (Note 1) |
| **E** Emergency Coordinator 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 | | | |
|---|---|---|---|---|---|---|
| Power Operation | Startup | Hot Standby | Hot Shutdown | Cold Shutdown | Refueling | Defueled |



**HOT CONDITIONS**
**RCS > 210°F)**

Prepared for APS by: Operations Support Services, Inc. – 5/5/15

# GENERAL EMERGENCY | SITE AREA EMERGENCY | ALERT | UNUSUAL EVENT

## C Cold SD/ Refueling System Malfunct.

### 1 RCS Level

**General Emergency** — Loss of RCS inventory affecting fuel clad integrity with Containment challenged

**CG1.1** (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
- RU-33 $\geq$ 9,000 mR/hr (when installed)
- Erratic Excore 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 $\geq$ 4.5%
- Unplanned rise in Containment pressure

**Site Area Emergency** — Loss of RCS inventory affecting core decay heat removal capability

**CS1.1** (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
- RU-33 $\geq$ 9,000 mR/hr (when installed)
- Erratic Excore Monitor indication

**Table C-1 Sumps/Tanks**

- Containment Sumps
- Reactor Cavity Sumps
- Auxiliary Building Sumps
- CVCS Holdup Tank
- Reactor Drain Tank
- Refueling Water Tank
- Equipment Drain Tank

**Alert** — Loss of RCS inventory

**CA1.1** (Modes: 5, 6)

Loss of RCS inventory as indicated by RCS level < 101 ft. 6 in. (RWLIS NR RCN-LI-752A/RCN-LR-752)

**CA1.2** (Modes: 5, 6)

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

**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

**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. (Notes 1, 10)

**CU1.2** (Modes: 5, 6)

RCS 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** — (none)

**Table C-3 AC Power Supplies**

**Offsite:**

- SUT (normal)
- SUT (alternate)
- SBOG #1 (if already aligned)
- SBOG #2 (if already aligned)

**Onsite:**

- DG A
- DG B

**Alert** — Loss of **all** offsite power and **all** onsite AC power to emergency buses for ≥ 15 min.

**CA2.1** (Modes: 5, 6, DEF)

Loss of **all** offsite and **all** onsite AC power capability to emergency 4.16KV buses PBA-S03 and PBB-S04 for ≥ 15 min.

**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 PBA-S03 and PBB-S04 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

**Table C-4 RCS Heatup 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 > 210°F for > Table C-4 duration (Note 1)

**OR**

UNPLANNED RCS pressure increase > 10 psia (This criterion 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 > 210°F

**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 min.

**CU4.1** (Modes: 5, 6)

Indicated voltage is < 112VDC on vital DC buses required by Technical Specifications for ≥ 15 min. (Note 1)

### 5 Loss of Comm.

**General Emergency** — None

**Site Area Emergency** — None

**Table C-5 Communications Methods**

| System | Onsite | ORO | NRC |
|---|---|---|---|
| PBX | X | X | X |
| Plant Page | X | | |
| Two-Way Radio | X | | |
| FTS | | | X |
| Telephone Ringdown Circuits (NAN) | | X | |
| Cellular Phones | | X | 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 Offsite Response Organization (ORO) communication methods

**OR**

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

**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:** Variations in RCS boron concentration, temperature and Containment Temperature from those used in RWLIS calibration will induce indication errors. Refer to Operator Assistance Program RWLIS_Spreadsheet.xls

### 6 Hazardous Event Affecting Safety Systems

**Table C-6 Hazardous Events**

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

**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

## DEFINITIONS

**Confinement Barrier**

The barrier(s) between spent fuel and the environment once the spent fuel is processed for dry storage. As related to the PVNGS ISFSI, Confinement Boundary is defined as the Transportable Storage Canister (TSC) for the NAC-UMS.

**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 PVNGS, Containment Closure is established when the requirements of 40EP-9EO10, *LM-Containment Evacuation and Closure*, Appendix 249, for containment closure are met.

**Explosion**

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

**Faulted**

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

**Fire**

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

**Flooding**

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

**Hostile Action**

An act toward PVNGS 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 PVNGS. 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).

**Plant or ISFSI Protected Area**

An area, located within the PVNGS Exclusion Area Boundary, encompassed by physical barriers and to which access is controlled per 10 CFR 73.55. The PVNGS Power Plant Protected Area and the ISFSI Protected Area are two Protected Areas located within the PVNGS OWNER CONTROLLED AREA.

**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, pressurizer manway and safeties installed).

**Reduced Inventory**

Plant condition when fuel is in the reactor vessel and Reactor Coolant System level is lower than or equal to the 111 foot elevation.

**Refueling Pathway**

The reactor refueling cavitypool, spent fuel storage 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.

**Secured Owner Controlled Area (SOCA)**

An area encompassed by physical barriers to which access is controlled.

**Security Condition**

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

**Site Boundary**

The boundary of a reactor site beyond which the land or property is not owned, leased, or otherwise controlled by the licensee.

**Unisolable**

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

**Unplanned**

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

**Valid**

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

**Visible Damage**

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

**Modes:**

| Power Operation | Startup | Hot Standby | Hot Shutdown | 5 Cold Shutdown | 6 Refueling | DEF Defueled |
|---|---|---|---|---|---|---|



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