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GO2-17-048 10 CFR 50.90
February 23, 2017

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

Subject: **COLUMBIA GENERATING STATION, DOCKET NO. 50-397 RESPONSE TO REQUEST FOR ADDITIONAL INFORMATION RELATED TO LICENSE AMENDMENT REQUEST TO ADOPT EMERGENCY ACTION LEVEL SCHEME PURSUANT TO NEI-99-01 REVISION 6 (MF8219)**

References:
1. Letter from W. Hettel, Energy Northwest to NRC, “Request for Amendment to Emergency Plan,” GO2-16-104 (ADAMS Accession Number ML16210A528), dated July 28, 2016.

2. Email from J. Kloss, NRC, to R. Garcia, Energy Northwest, “Columbia EAL Scheme Change, MF8219, Request for Additional Information,” GI2-17-004, ADAMS Accession Number ML 17025A061, dated January 25, 2017.

Dear Sir or Madam:

By Reference 1, Energy Northwest submitted a license amendment for Columbia Generating Station. The amendment proposes to revise the current Emergency Action Level scheme to one based upon Revision 6 to the Nuclear Energy Institute (NEI) document NEI 99-01, “Development of Emergency Action Levels for Non-Passive Reactors” (ADAMS Accession Number ML12326A805). By Reference 2, the Nuclear Regulatory Commission requested additional information related to the Energy Northwest submittal. Enclosure 1 to this letter contains the requested information, while Enclosure 2 provides an updated technical bases document.

No new commitments are being made by this letter or the enclosure. Additionally, the No Significant Hazards Consideration determination in the original submittal is not altered by the additional information provided in this response. If there are any questions or if additional information is needed, please contact Ms. L. L. Williams, Licensing Supervisor, at 509-377-8148.

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

Executed this 23 day of February, 2017.

Respectfully,

W.G. Hettel
Vice President, Operations

Enclosure: As stated

cc:
NRC Region IV Administrator
NRC NRR Project Manager
NRC Senior Resident Inspector/988C
CD Sonoda – BPA/1399
WA Horin – Winstron & Strawn

RESPONSE TO REQUEST FOR ADDITIONAL INFORMATION

NRC REQUEST No. 1a:

NEI 99-01, Revision 6, EALs, such as [AU1 (1)], typically consist of a site-specific list as an integral part of the threshold value for each EAL. The proposed Columbia EAL scheme references tables that are not included with the applicable EALs.

Please include the appropriate tables with each applicable EAL or provide explain how this formatting will not cause a potential delay or contribute to a misclassification of any EAL.

ENERGY NORTHWEST RESPONSE TO RAI 1a:

Energy Northwest has incorporated the appropriate tables from Attachment 7.3 into the applicable individual EAL bases.

NRC REQUEST No. 1b:

NEI 99-01, Revision 6, EALs include applicable notes. The proposed Columbia EAL scheme references a complete table of notes that is located at the end of the bases document rather than including the notes that are applicable with each EAL.

Please include the appropriate notes with each applicable EAL or provide explain how this formatting will not cause a potential delay or contribute to a misclassification of an EAL.

ENERGY NORTHWEST RESPONSE TO RAI 1b:

Energy Northwest has incorporated the appropriate notes from Attachment 7.3 into the applicable individual EAL bases.

NRC REQUEST No. 1c:

The proposed Columbia EAL scheme defines GE as "*General Emergency, Greater than or Equal to*." As an example of the potential for confusion, the threshold value for EAL RG1.1 states, in part: "G*T column "GE" for GE.*"

In addition to Columbia personnel, the dual usage of GE, which is typically used in emergency plans to indicate a General Emergency, could cause confusion to local, State, and NRC personnel responding to either a drill or actual emergency at Columbia.

Please revise the proposed bases document to eliminate the dual usage of the acronym of GE or explain how no confusion will occur with site, local, State or NRC personnel.

ENERGY NORTHWEST RESPONSE TO RAI 1c:

The use of the acronym “GE” to represent “General Emergency” has been eliminated in the technical bases.

NRC REQUEST No. 2:

NEI 99-01, Revision 6, Section 4.7, “*EAL/Threshold References to AOP [Abnormal Operating Procedure] and EOP [Emergency Operating Procedure] Setpoints/Criteria*,” states, “*As reflected in the generic guidance, the criteria/values used in several EALs and fission product barrier thresholds may be drawn from a plant’s AOPs and EOPs.*” The NRC staff expects that changes to AOPs and EOPs will be evaluated in accordance with the provisions of 10 CFR 50.54(q).

Please explain why this key guidance from NEI 99-01, Revision. 6, was omitted, or revise accordingly.

ENERGY NORTHWEST RESPONSE TO RAI 2:

The following has been added to Section 1.0 Purpose:

“Additionally, some criteria/values in the CGS EALs and fission product barrier thresholds are drawn from plant AOPs and EOPs. The impact of any changes to those procedures on EAL bases must be evaluated for screening in accordance with the provisions of 10 CFR 50.54(q)”

NRC REQUEST No. 3a:

Concerning the proposed Table 3, “*Effluent Monitor Classification Thresholds,”* as it relates to RU1.1, please address the following:

a. The proposed turbine building exhaust (TEA-RIS-13) value in Table 3 for an Unusual Event is 1.02E-04 µCi/cc and the proposed value for an Alert is 8.35E-04 µCi/cc. The proposed radwaste building exhaust (WEA-RIS-14) value in Table 3 for an Unusual Event is 1.98E-03 µCi/cc and the proposed value for an Alert is 3.45E-03 µCi/cc. The actual dose that corresponds to an Unusual Event, as indicated on page 5.015 of Enclosure 2, ADAMS Accession No. ML 16210A530, is 6 mrem/hr thyroid for TEA-RIS-13 and 29 mrem/hr thyroid for WEA-RIS-14.

   The NRC staff reviews proposed EALs for consistency, human factors engineering and user friendliness, and to ensure that the potential for emergency classification upgrade only when there is an increasing threat to public health and safety. The above values for the declaration of an Unusual Event and an Alert based on WEA-RIS-14 indications are less than a factor of two apart. Additionally, an Unusual Event declaration based on TEA-RIS-13 would correspond to 6 mrem/hr, and an

Unusual Event declaration based on WEA-RSI-14 would correspond to 29 mrem/hr. Both instruments indicate that the basis for the setpoint is 2 times the Offsite Dose Calculation Manual limit.

Please revise the Table 3 Unusual Event setpoint for WEA-RIS-14 to reflect a similar threat to public health and safety as TEA-RIS-13. Additionally, please revise the Table 3 Unusual Event setpoint as necessary to provide a more appropriate difference between an Unusual Event and an Alert condition or provide a more detailed explanation for the proposed Unusual Event setpoints for WEA-RIS-14.

ENERGY NORTHWEST RESPONSE TO RAI 3a:

The "actual hourly dose rate at EAL" column on page 5.015 of the EAL Technical Bases Calculation (Enclosure 2, ADAMS Accession No. ML 16210A530) reports the Unified Rascal Interface (URI) results as a check for the Alert and higher. Inclusion of dose rates for the Unusual Event is only for purposes of confirming that UE thresholds (calculated per the ODCM methodology) are below the Alert thresholds calculated via URI. In all cases, URI used an accident source term. Since the UE threshold values are calculated via the ODCM methodology (including a non-accident source term), it is not expected that the URI calculations of a UE presented in the EAL Technical Bases calculation would yield meaningful results.

The Table 3 Unusual Event threshold values are determined such that monitor readings would correlate to 2 x ODCM RFO 6.2.2.1 dose rate limits for the site boundary. Specifically this would equate to:

- 500 mrem/yr whole body (0.1 mrem/hr)
- 1500 mrem/year to any organ (0.3 mrem/hr)

Per guidance in NEI 99-01, the methodologies in the ODCM are applied to determination of effluent thresholds for EAL purposes. The factor of 20 difference between the TEA-RIS-13 and WEA-RIS-14 values is attributable to the following key differences:

1. The ODCM source term for each building is specified in FSAR Table 11.3-7. The turbine building has seven noble gas isotopes; the radwaste building has two. This leads to a factor of ~4 difference in the release rate (μCi/sec) from each building that would result in 500 mrem/yr whole body dose. The noble gases from the radwaste building source term are less dose significant, allowing for a higher curie release rate.
2. The Table 3 values are expressed in μCi/cc and as such include the difference in maximum building flow rate, which results in a factor ~5 difference in in the UE values.

When the appropriate building specific source terms and building considerations are applied, the TEA-RIS-13 and WEA-RIS-14 thresholds reported in Table 3 represent the same level of threat to public health and safety.

Regarding the difference between Unusual Event and Alert threshold values in Table 3, as noted above, the numerical values for UE and Alert are calculated via different methodologies and are selected to be representative of different dose limits (multiple of effluent ODCM limits vs percentage of EPA protective action guidelines). The proposed Table 3 threshold values have been selected in full conformance with EAL development guidelines contained in NEI 99-01, Rev 6.

NRC REQUEST No. 3b:

Calculation NE-02-09-12 – CGS Emergency Action Levels Technical Bases, Item 7 on page 5.001 of Enclosure 2 states:

> *The maximum allowed background for TEA-RIS-13 Channel 1 is 388 cpm (1.80E-5 µCi/cc). This is larger than the Alert Alarm setpoint. NE-02-08-09 [reference 80] page 5.005, TEA-RIS-13 Channel 1 and Channel 3 Alert Alarm setpoint is 9.97E-6 µCi/cc. The setpoint was established at 4.66 * background as a value reasonably above expected background to indicate that an actual release may be in progress. FSAR 11.5.2.2.1.6 states that normal activity is expected to be below detectable levels. In the absence of an actual release, the Alert Alarm setpoint represents an upper bound on background variation because a spurious alarm would alert the operators to a potentially degraded condition. A spurious alarm would result in a functionality evaluation, so there is no need for a separate procedural limit on background for Channel 1. The same conclusion applies to WEA-RIS-14.*

The NRC staff expects that licensees have equipment for determining the magnitude of and for continuously assessing the release of radioactive materials to the environment. The statement on page 5.001 of Enclosure 2 indicates that operators will have to perform a functionality evaluation to discriminate a spurious alarm from an Alert or Unusual Event condition.

Please explain how a timely and accurate assessment of RU1.1 and RA1.1 can be performed when a functionality evaluation is required to determine that the alarm is not spurious.

ENERGY NORTHWEST RESPONSE TO RAI 3b:

All alarms are treated as initially valid and response to alarms will not be delayed pending results of any functionality assessment. The following summarizes Energy Northwest procedures to ensure timely and accurate assessment of RU1.1 and RA1.1:

- Abnormal Operating Procedures are entered when an Alert Alarm (4.66 * background) is received. The operators direct all non-emergency personnel to evacuate from the affected building and monitor (continuously assess) radioactivity levels. Operators direct Health Physics to locate the areas with high airborne activity. Operators determine the source of the release and terminate it. Operators direct Chemistry to sample the affected gaseous effluent pathway. Site boundary dose rates are assessed with URI and emergency classification made.

If the ALERT alarm did not result from an actual release, a functionality evaluation would be performed and compensatory measures implemented as specified in the emergency procedures.

NRC REQUEST No. 3c:

Please explain why the proposed Unusual Event value in Table 3 for TSW-RIS-5 is 3.00E-05 µCi/cc, while the value for TSW-RIS-5 on page 5.009 of Enclosure 2 provides a value of 2.00E-05 µCi/cc, or revise accordingly.

ENERGY NORTHWEST RESPONSE TO RAI 3c:

The 2.00E-05 µCi/cc value does not include background, which is added on page 5.010 and rounded for readability to the next increment on the log scale of the instrument, yielding 3E-05 for use in Table 3.

NRC REQUEST No. 4:

NEI 99-01, Revision 6, EALs [CU2 (1)] and [SA1 (1)] state:

a. AC power capability to (site-specific emergency buses) is reduced to a single power source for 15 minutes or longer.

AND

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

The intent of the above condition ‘b’ is to focus on the inability to energize the required SAFETY SYSTEMS and not individual bus status.

For EALs CU2.1 and MA1.1 [SA1], the condition that any additional single power source will result in a loss of all AC power to SAFETY SYSTEMS was removed from the proposed EALs and replaced with “*Any additional single power source failure will result in loss of all AC power to emergency buses SM-7 and SM-8.*”

Please explain, in greater detail, why the condition, *“Any additional single power source failure will result in a loss of all AC power to SAFETY SYSTEMS,”* was removed from the proposed EAL MA1 [SA1], or revise accordingly.

ENERGY NORTHWEST RESPONSE TO RAI 4:

Energy Northwest has revised CU2.1 and MA1.1 to read:

> *"AC power capability, Table 2, to emergency buses SM-7 and SM-8 reduced to a single power source for GE 15 min. (Note 1)*
>
> *AND*
>
> *Any additional single power source failure will result in a loss of all AC power to SAFETY SYSTEMS"*

NRC REQUEST No. 5:

The first three paragraphs of the proposed Columbia CA2.1 basis could imply that only Table 2 AC power sources are acceptable to provide power to buses SM-7 and SM-8. This is not consistent with the NRC response of Emergency Preparedness Frequently Asked Question (EPFAQ) 2015-15, (ADAMS Accession No. ML16166A191) which provides that the primary point of emphasis for CA2 is a complete loss of power for 15 minutes and not a particular source of power.

Please remove the Table 2 related information from CA2.1, or explain how the information related to Table 2 cannot potentially impact a timely and accurate assessment of CA2.1.

ENERGY NORTHWEST RESPONSE TO RAI 5:

Energy Northwest has deleted the cited three bases paragraphs from CA2.1.

NRC REQUEST No. 6:

The fourth paragraph of the proposed Columbia CA6.1 and MA8.1 basis discussion states:

> *An emergency classification is required if a FIRE or EXPLOSION caused by an equipment failure damages safety system equipment that was otherwise functional or operable (i.e., equipment that was not the source/location of the failure). For example, if a FIRE or EXPLOSION resulting from the failure of a piece of safety system equipment causes damage to the other train of the affected safety system or another safety system, then an emergency declaration is required in accordance with this IC and EAL.*

The example provided in the above paragraph requires two trains of equipment to be damaged. The first train would be potentially damaged by the fire or explosion, and the second train would be damaged by the piece of safety system equipment that was on

fire or exploded. It is not the intent of CA6 and MA8 to require two trains of equipment to be damaged by an explosion or fire as declaration criteria.

Please remove the provided example from the fourth paragraph of the CA6 and MA8, or explain how this example will not potentially cause a decision maker to infer that two trains of equipment must be damage to meet the threshold value for declaration of CA6 or MA8.

ENERGY NORTHWEST RESPONSE TO RAI 6:

Energy Northwest is requesting a deviation for the definition of VISIBLE DAMAGE. Energy Northwest has deleted the cited paragraph from the CA6.1 and MA8.1 bases.

Added new Note 9 applicable to EALs CA6.1 and MA8.1 that reads:

> *"If the affected SAFETY SYSTEM (or component) was already inoperable or out of service before the event occurred, then* ***no*** *emergency classification is warranted as long as the damage was limited to this affected SAFETY SYSTEM (or component)."*

This note is consistent with guidance currently provided in NEI 99-01 Revision 6 and the NRC endorsed Alert classification language (e.g. HA5).

The definition of VISIBLE DAMAGE has been revised to read:

> *"Damage to components on two or more SAFETY SYSTEMS, or one or more structures that is readily observable without measurements, testing, or analysis. The visual impact of the damage is sufficient to cause concern regarding the operability or reliability of the affected SAFETY SYSTEMS in the area. Events that result in visible damage to the components of one SAFETY SYSTEM, and do not appear to affect the components of other SAFETY SYSTEMS, do not meet the intent of this definition as the failure of a component(s) affecting the operability of a SAFETY SYSTEM, regardless of cause, is well within the operational controls provided by a licensee's Technical Specifications and operating procedures. However, visible damage to the components of more than one SAFETY SYSTEM does meet this definition, as well as visible damage to a structure."*

This revised definition provides clarity regarding applicability of equipment failures (e.g. fires and explosion) where the resulting damage is limited to the failed safety system.

NRC REQUEST No. 7:

The NEI 99-01, Revision 6, [SG8] initiating condition is as follows: *"Loss of all AC and Vital DC power sources for 15 minutes or longer*." The threshold value for a loss of Vital DC power sources should be based on the site-specific minimum bus voltage necessary for adequate operation of safety system equipment and not a particular bus alignment.

The proposed Columbia technical basis for MG1.2 [SG8] states:

> *PPM 5.6.1 Station Blackout, directs use of DG3, DG4 or DG5 to power vital DC battery chargers. If this is already performed, this EAL would not apply (ref. 4).*

Regardless of the DC battery charger lineup, an indicated dc voltage of less than 108 volts concurrent with a loss of all offsite and onsite AC power capability for 15 minutes or longer represents an extended loss of both AC vital DC power.

Please remove the Columbia EAL technical basis reference to disregard the threshold values for MG1.2 based on how the alignment of DG3, DG4 or DG5 buses, or explain why actual DC bus voltage is not a valid indicator of the condition of the Columbia 125 VDC bus.

ENERGY NORTHWEST RESPONSE TO RAI 7:

Energy Northwest has removed the EAL technical basis reference to disregard the threshold values for MG1.2 based on the alignment of DG3, DG4 or DG5 buses.

NRC REQUEST No. 8a:

Concerning MU5.1 [SU4], please address the following:

a. The proposed Columbia technical basis for MU5.1 states:

> *Drain flow from the drywell equipment and floor drain sumps is monitored and recorded (EDR-FRS-623) on P632. The flow rates for identified and unidentified leakage in the EAL are equal to the full scale reading on EDR-FRS-623.*

Please explain how the operators can accurately assess MU5.1 using EDR-FRS-623, or revise accordingly. This explanation should address the ability of the operator to determine an RCS leak and to ensure that an instrument failure would not result in an unnecessary declaration.

ENERGY NORTHWEST RESPONSE TO RAI 8a:

The MU5.1 Technical bases incorrectly indicated that the identified leakage threshold would be full scale on the flow recorder. The identified leakage threshold for MU5.1 is 25 gpm, which is approximately 83% of full scale (30 gpm). The unidentified leakage threshold (10 gpm) is full scale on the associated flow recorder.

Energy Northwest has revised the bases statement to clarify that only unidentified leakage is full scale. The revised basis reads: "*The flow rate for unidentified leakage in the EAL is equal to the full scale reading on EDR-FRS-623 Pen 1.*"

A failed instrument can be verified by containment drywell response. As observed in the first principle simulator, a RCS leak of 10 gpm causes a drywell temperature and pressure response within six minutes. The operators are trained on delineating valid versus invalid indications with use of multiple indications verifying plant response.

EAL declarations are made when the emergency director has sufficient information to confirm the observed condition.

NRC REQUEST No. 8b:

The NEI 99-01, Revision 6, [SU4] and the proposed technical basis for MU5.1 states:

> *The leak rate values for each threshold 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 proposed Columbia technical basis for MU5.1 allows the licensee to delay declaration until the completion of analysis for RCS leakage not explicitly identified by installed instrumentation. Additionally, the proposed Columbia technical basis for MU5.1 allows the licensee to delay classification for leakage outside the containment until analysis to quantify the leak rate has been completed.

Please revise the Columbia RCS leakage basis discussion to remove the delays of the declaration clock to perform leak rate analyses, or provide site-specific leakage values that can be observed within the control room. Additionally, please explain why MU5.1 (1) for unidentified leakage of greater than or equal to (≥) 10 gpm would not be declared if the source of RCS leakage was not identified within 15 minutes while performing analyses to identify RCS leakage that is ≥ 25 gpm or revise accordingly.

ENERGY NORTHWEST RESPONSE TO RAI 8b:

RCS leakage outside of the primary containment is not directly observable in the MCR. There is no available direct MCR indication of RCS leakage outside primary containment such as a steam leak or instrument line crack in the Reactor Building. There are indirect indications of leakage such as temperature and radiation, however, these are not quantifiable into gpm for this EAL. As such, the EAL classification clock for "*Leakage from the RCS to a location outside containment GT 25 gpm for GE 15 min*" would start only after the MCR was informed, following analysis, from the field outside the MCR. This is consistent with the NRC ISG Section IV.H.4 that states that classification timeliness is based on the availability of analysis results.

NRC REQUEST No. 9:

The proposed Columbia EAL EU1.1 [E-HU1] basis states:

> *CGS has casks loaded to various amendments to the Certificate of Compliance (COC) Technical Specifications with a proposed amendment coming in 2017. The numbers above reflect the most limiting Technical Specification (TS) values (Amendment 1) and can be updated using 10 CFR 50.54(q) process, if CGS adopts a common TS amendment.*

Including this statement in the basis discussion, infers that this license amendment is subject to a pending or future amendment, thereby making this submission a linked submittal contrary to NEI 06-02, License Amendment Request (LAR) Guidelines Revision 2, October 2010. Please remove the entire statement, or modify it as necessary to clearly reflect the current status of the facility.

ENERGY NORTHWEST RESPONSE TO RAI 9:

Energy Northwest has revised the cited paragraph to read:

"Energy Northwest has casks loaded to various amendments to the Certificate of Compliance (COC) Technical Specifications. The numbers above reflect the most limiting Technical Specification (TS) values (Amendment 1)."

NRC REQUEST No. 10:

The proposed Columbia RCS barrier loss due to leak rate basis discussion states, in part: "the ruptured line cannot be isolated remotely or locally, the RCS barrier Loss threshold is met." The guidance provided in NEI 99-01, Revision 6, states, in part: "the ruptured line cannot be promptly isolated from the Control Room, the RCS barrier Loss threshold is met." The addition of the text "locally" as shown above could delay or prevent the declaration of a RCS barrier loss because a decision maker may consider a leak with the potential for local isolation to be an isolable leak even though the leak is not actually isolated, and it may not be feasible to actually isolate the leak locally.

Please remove the reference to local isolation from the basis discussion, or explain how the addition of this condition could not potentially delay or prevent classification of a loss of the RCS barrier.

ENERGY NORTHWEST RESPONSE TO RAI 10:

Energy Northwest has revised the cited bases to read:

"If it is determined that the ruptured line cannot be promptly isolated from the control room, the RCS barrier Loss threshold is met".

NRC REQUEST No. 11:

The maximum safe operating radiation levels on PPM 5.3.1, Table 24, appear to be at the upper range of the detectors as indicated on Table 12.3-1, "*Area Monitors,*" of the Columbia Final Safety Analysis Report. Considering that the containment barrier loss is based on the reactor building maximum safe operating radiation level being exceeded, it is not clear to the staff how this condition can be accurately assessed.

Please explain how the operators can accurately assess primary leakage that exceeds

maximum safe operating radiation conditions as indicated on PPM 5.3.1, Table 24, or revise accordingly.

ENERGY NORTHWEST RESPONSE TO RAI 11:

The Table 24 ARM readings come directly from Emergency Operating Procedures (EOPs). The detectors and meters are analog and the range of the indicator exceeds the maximum annotated range of 10 Rem. ARM output beyond maximum annotated range is tested during surveillances to ensure EOP Table 24 analysis can be performed.

The operators are trained and evaluated to analyze Table 24 Max Safe Radiation levels in a timely manner.

Reactor Building area temperatures and radiation levels are trended. Trending capability further provides indications that instances of exceeding the specified limits are not the result of instrument failure.

NRC REQUEST No. 12:

The proposed Columbia technical basis for the containment fission product barrier loss due to a RCS leak rate states (page 174 of 199):

> *The maximum safe operating radiation value is defined to be 10,000 mR/hr in areas other than the refueling floor. This is the maximum indication on all but the high level instruments.*

Please explain how the operators can accurately assess the Containment Fission Product Barrier for a RCS loss using radiation levels that are at the maximum indication, or revise accordingly. This explanation should address the ability of the operator to determine an RCS leak into the reactor building and to ensure that an instrument failure would not result in an unnecessary declaration.

ENERGY NORTHWEST RESPONSE TO RAI 12:

The Table 24 ARM readings come directly from EOPs. The detectors and meters are analog and the range of the indicator exceeds the maximum annotated range of 10 Rem. ARM output beyond maximum annotated range is tested during surveillances to ensure EOP Table 24 analysis can be performed.

The operators are trained and evaluated to analyze Table 24 Max Safe Radiation levels in a timely manner.

Reactor Building area temperatures and radiation levels are trended. Trending capability further provides indications that instances of exceeding the specified limits are not the result of instrument failure.

The determination of whether a primary system is discharging into the Secondary Containment is made consistent with PPM 5.3.1 Secondary Containment Control. This includes Reactor Building differential temperatures, Reactor Building exhaust plenum radiation level and increasing Reactor Building area water levels as well as increasing trends in Reactor Building area temperatures and area radiation levels.

NRC REQUEST No. 13:

The proposed Columbia technical basis for the containment fission product barrier loss due to a loss of Primary Containment Integrity or Primary Containment Bypass states (page 184 of 199):

> *If the main condenser is available with an unisolable main steam line, there may be releases through the steam jet air ejectors and gland seal exhausters. These pathways are monitored, however, and do not meet the intent of a nonisolable release path to the environment. These minor releases are assessed using the Category R, Abnormal Rad Release / Rad Effluent, EALs.*

The above statement infers that the ability to monitor an unisolable direct downstream pathway can be used as a basis to ignore the fact that the containment barrier has been lost. This logic could delay the declaration of a General Emergency for a loss of all three fission product barriers if leakage could be monitored as provided in the above example.

NEI 99-01, Revision 6, Section 2.4 “*Fission product Barrier Threshold”* states:

> *In some accident sequences, the ICs and EALs presented in the Abnormal Radiation Levels/ Radiological Effluent (A) Recognition Category will be exceeded at the same time, or shortly after, the loss of one or more fission product barriers. This redundancy is intentional as the former ICs address radioactivity releases that result in certain offsite doses from whatever cause, including events that might not be fully encompassed by fission product barriers (e.g., spent fuel pool accidents, design containment leakage following a LOCA, etc.).*

Please modify the basis discussion as necessary to ensure that fission product barriers properly assessed restoring the redundancy intended by NEI 99-01, Revision 6, or provide a justification for potentially delaying the classification of a General Emergency based on fission product barrier degradation.

ENERGY NORTHWEST RESPONSE TO RAI 13:

Energy Northwest has deleted the cited portion of the paragraph related to unisolable releases to the main condenser.

NRC REQUEST No. 14a:

EAL assessments.

Additionally, NEI 99-01, Section 4.6, "Basis Document," of Revision 6, states:

> *A Basis section should not contain information that could modify the meaning or intent of the associated IC [initiating condition] or EAL. Such information should be incorporated within the IC or EAL statements, or as an EAL Note. Information in the Basis should only clarify and inform decision-making for an emergency classification.*

a. The proposed Columbia technical bases includes the definition of intrusion (5.1.21). It does not appear that the definition of intrusion is applicable to the proposed EAL scheme.

   Please consider removing the definition of intrusion (5.1.21) to prevent potential confusion for actions related to a hostile force, or provide an explanation that the definition of intrusion is applicable to the proposed EAL scheme.

ENERGY NORTHWEST RESPONSE TO RAI 14a:

Energy Northwest has deleted the defined term "Intrusion" from Section 5.1 Definitions.

NRC REQUEST No. 14b:

The proposed Columbia technical bases include a definition for the owner controlled area (5.1.24) that is not complete.

Please provide a complete definition for the owner controlled area (5.1.24).

ENERGY NORTHWEST RESPONSE TO RAI 14b:

The definition of Owner Controlled Area is the definition as provided in Section 1.7.24 of the CGS Emergency Plan.

NRC REQUEST No. 14c:

The proposed Columbia technical basis for RG1.1 threshold 1 refers to the Table 3 column "SAE" gaseous effluent release values. Considering that this reference is in relation to calculated doses of 100% of the environmental protection agency protective action guidelines, the reference should be to the "GE (General Emergency)" values.

Please ensure that RG1.1 references that appropriate Table 3 columns or justify the reference to "SAE."

ENERGY NORTHWEST RESPONSE TO RAI 14c:

Energy Northwest has revised the cited bases “SAE” to read “General.”

NRC REQUEST No. 14d:

The proposed Columbia EALs CU1.1 and CA1.1 basis includes a discussion relative to EALs #1 and #2, followed by a discussion that applies both EALs, which is then followed by a discussion that is specific to EALs #1 and #2. There is a potential for a decision maker to assume the first discussion for EAL #1 contains all required information relative to EAL #1. As such, a decision maker could miss key basis element discussions which could result in an inaccurate and/or delayed assessment.

Please consolidate the separate EAL #1 and #2 discussions into a single EAL #1 and #2 discussion, or explain how a decision maker will not miss the key information that is contained in the second EAL #1 and #2 discussions.

ENERGY NORTHWEST RESPONSE TO RAI 14d:

Energy Northwest has reformatted the CU1.1 and CA1.1 bases to consolidate the EAL #1 and EAL #2 discussions and moved bases applicable to both thresholds prior to the individual EAL bases discussions.

NRC REQUEST No. 14e:

The proposed Columbia CS1.1 basis includes, “The difference in the specified RPV levels of CS1.1 and CS1.2 reflect that with CONTAINMENT CLOSURE established, there is a lower probability of a fission product release to the environment.”

Please replace “*CS1.1 and CS1.2*” with CS1.1 (1) and CS1.1 (2) to reflect CGS EAL numbering system. The proposed Columbia reference to CS1.2 is not appropriate as CS1.2 does not include RPV level.

ENERGY NORTHWEST RESPONSE TO RAI 14e:

Energy Northwest has replaced the cited “CS1.1 and CS1.2” with “CS1.1 (1) and CS1.1 (2).”

<u>NRC REQUEST No. 14f:</u>

The proposed Columbia technical basis for MU6.1 states:

> *By procedure, operator actions include the initiation of an immediate manual scram following receipt of an automatic scram signal. If there are no clear indications that the automatic scram failed (such as a time delay following indications that a scram setpoint was exceeded), it may be difficult to determine if the reactor was shut down because of automatic scram or manual actions. If a subsequent review of the scram actuation indications reveals that the automatic scram did not cause the reactor to be shut down, then initiate a Transitory Event Notification per EPIP 13.4.1.*

The proposed Columbia technical basis for MU6.1 appears to contain an action step that is not directly related to the assessment of MU6.1.

Please remove this apparent action step from the MU6.1 basis discussion, or explain why this step is required as basis information that could support the assessment of MU6.1.

<u>ENERGY NORTHWEST RESPONSE TO RAI 14f:</u>

Energy Northwest has removed the cited action step from the MU6.1 bases

<u>NRC REQUEST No. 14g:</u>

The proposed Columbia EALs HU2.1, MU3.1, MU6.1 and MA6.1 bases include an escalation criteria that is based on the category designation of ‘S’ instead of the proposed Columbia designation of ‘M.’

Please revise the HU2.1, MU3.1, MU6.1 and MA6.1 EAL bases escalation criteria to reflect the proposed ‘M’ EAL designation for the system malfunction category or justify using ‘S’ as a category designation.

<u>ENERGY NORTHWEST RESPONSE TO RAI 14g:</u>

Energy Northwest has corrected the cited escalation criteria to reflect the proper category “M” vs. “S.”

GO2-17-048

Enclosure 2

Revised Emergency Action Level (EAL) Bases Document

(203 pages)

PLANT PROCEDURES MANUAL	PCN#: N/A
13.1.1A	Effective Date:

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SECTION PAGE

1.0 PURPOSE

This document provides an explanation and rationale for each Emergency Action Level (EAL) included in the EAL Upgrade Project for Columbia Generating Station (CGS). It should be used to provide historical documentation for future reference and serve as a training aid. Decision-makers responsible for implementation of PPM 13.1.1, Classifying the Emergency, may (though not required) use this document as a technical reference in support of EAL interpretation.

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

Because the information in a basis document can affect emergency classification decision-making (e.g., the Emergency Director refers to it during an event), the NRC staff expects that changes to the basis document will be evaluated in accordance with the provisions of 10 CFR 50.54(q). Additionally, some criteria/values in the CGS EALs and fission product barrier thresholds are drawn from plant AOPs and EOPs. The impact of any changes to those procedures on EAL bases must be evaluated for screening in accordance with the provisions of 10 CFR 50.54(q). This Emergency Plan Implementing Procedure as identified by reference in the Emergency Plan. Changes to the EAL Scheme (Attachments 7.1, 7.2, 7.3, 7.4) require an LDCN since it is part of the Emergency Plan.

2.0 DISCUSSION

2.1 Background

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

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

2.1.3 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.

2.1.4 Subsequently, Revision 6 of NEI 99-01 has been issued which incorporates resolutions to numerous implementation issues including the NRC EAL Frequently Asked Questions (FAQs). Using NEI 99-01 Revision 6, "Methodology for the Development of Emergency Action Levels for Non-Passive Reactors,” November

2012 (ADAMS Accession Number ML12326A805) (ref. 4.1.1), CGS 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.

2.2.1 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 is the reactor coolant system pressure boundary and includes the RPV and all reactor coolant system piping out to and including the isolation valves.

c. Containment (PC): The drywell, the wetwell, their respective interconnecting paths, and other connections up to and including the outermost containment isolation valves comprise the PC barrier. Containment Barrier thresholds are used as criteria for escalation of the ECL from Alert to either a Site Area Emergency or a General Emergency using the Fission Product Barrier table.

2.3 Emergency Classification Based on Fission Product Barrier Degradation

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

2.3.1 Alert:

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

2.3.2 Site Area Emergency:

Loss or potential loss of any two barriers

2.3.3 General Emergency:

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

2.4 EAL Organization

2.4.1 The CGS EAL scheme includes the following features:

a. Division of the EAL set into three broad groups:

1) EALs applicable under all plant operating modes – This group would be reviewed by the EAL-user any time emergency classification is considered.

2) EALs applicable only under hot operating modes – This group would only be reviewed by the EAL-user when the plant is in Hot Shutdown, Startup, or Power Operations mode.

3) EALs applicable only under cold operating modes – This group would only be reviewed by the EAL-user when the plant is in Cold Shutdown, Refuel or Defueled mode.

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

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

2.4.4 Category and subcategory titles are selected to represent conditions that are operationally significant to the EAL-user. The CGS EAL categories are aligned to and represent the NEI 99-01"Recognition Categories." Subcategories are used in the CGS scheme as necessary to further divide the EALs of a category into logical sets of possible emergency classification thresholds. The CGS EAL categories and subcategories are listed in Table 2.4-1.

2.4.5 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 bases in order to obtain additional information concerning the EALs under classification consideration. The user should consult Section 3.0 and Attachments 7.1 & 7.2 of this document for such information.

Table 2.4-1 EAL Groups, Categories and Subcategories

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

2.5 Technical Bases Information

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

a. Category Letter & Title

b. Subcategory Number & Title

c. Initiating Condition (IC)

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

a. EAL Identifier (enclosed in rectangle)

1) 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:

a) First character (letter): Corresponds to the EAL category as described above (R, C, H, M, F or E)

b) Second character (letter): The emergency classification (G, S, A or U)

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

c) 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).

d) 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).

2) Classification (enclosed in rectangle):

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

3) EAL (enclosed in rectangle)

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

4) Notes

Any notes applicable to the EAL

2.6 Mode Applicability

One or more of the following plant operating conditions comprise the mode to which each EAL is applicable: 1 - Power Operations, 2 - Startup, 3 - Hot Shutdown, 4 - Cold Shutdown, 5 - Refuel, D - Defueled, or All. Additionally, unique to the ISFSI, Storage Operations. (See Section 2.10 for operating mode definitions).

~~2.7 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.~~

~~2.8~~2.7 Basis:

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

~~2.9~~2.8 CGS Basis Reference(s):

Site-specific source documentation from which the EAL is derived

~~2.10~~2.9 Operating Mode Applicability (ref. 4.1.2)

~~2.10.1~~2.9.1 Power Operations

Reactor mode switch is in RUN

~~2.10.2~~2.9.2 Startup

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

~~2.10.3~~2.9.3 Hot Shutdown

The mode switch is in SHUTDOWN, with all reactor vessel head closure bolts fully tensioned, and reactor coolant temperature is GT 200ºF

~~2.10.4~~2.9.4 Cold Shutdown

The mode switch is in SHUTDOWN, all reactor vessel head closure bolts are fully tensioned, and reactor coolant temperature is LE 200ºF

~~2.10.5~~2.9.5 Refuel

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

~~2.10.6~~2.9.6 Defueled

All reactor fuel removed from RPV. (Full core off load during refueling or extended outage).

~~2.10.7~~2.9.7 The plant operating mode that exists at the time that the event occurs (prior to any protective system or operator action being initiated in response to the condition) should be compared to the mode applicability of the EALs. If a lower or higher plant operating mode is reached before the emergency classification is made, the declaration shall be based on the mode that existed at the time the event occurred.

~~2.10.8~~2.9.8 For events that occur in Cold Shutdown or Refuel, escalation is via EALs that have Cold Shutdown or Refuel for mode applicability, even if Hot Shutdown (or a higher mode) is entered during any subsequent heat-up. In particular, the fission product barrier EALs are applicable only to events that initiate in Hot Shutdown or higher.

~~2.10.9~~2.9.9 The ISFSI related EAL EU1.1 is applicable in the Storage Operations mode as defined in the Certificate of Compliance Appendix A Section 1.1 Definitions (ref 4.1.12):

~~2.11~~2.10 Storage Operations

Storage operations include all licensed activities that are performed at the ISFSI while a Spent Fuel Storage Cask (SFSC) containing spent fuel is situated within the ISFSI perimeter. Storage Operations does not include MPC transfer between the Transfer Cask and the Overpack which begins when the MPC is lifted off the HI-TRAC bottom lid and ends when the MPC is supported from beneath by the Overpack (or the reverse).

3.0 GUIDANCE ON MAKING EMERGENCY CLASSIFICATIONS

3.1 General Considerations

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

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. 4.1.3).

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 indicator operability, condition existence, or report accuracy. For example, verification could be accomplished through an instrument channel check, response on related or redundant indicators, or direct observation by plant personnel.

An indication, report, or condition is considered to be valid when it is verified by (1) an instrument channel check, or (2) indications on related or redundant indicators, or (3)

by direct observation by plant personnel, such that doubt related to indicator operability, the condition existence, or the report accuracy is removed. Implicit in this definition is the need for timely assessment. The validation of indications should be completed in a manner that supports timely emergency declaration.

3.1.3 Imminent Conditions

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

3.1.4 Planned vs. Unplanned Events

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

3.1.5 Classification Based on Analysis

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

3.1.6 Emergency Director Judgment

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

3.2 <u>Classification Methodology</u>

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. 4.1.3).

3.2.1 Classification of Multiple Events and Conditions

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

- If an Alert EAL and a Site Area Emergency EAL are met, a Site Area Emergency should be declared.

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

- If two Alert EALs are met, an Alert should be declared.

c. 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. 4.1.5).

3.2.2 Consideration of Mode Changes During Classification

a. 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.

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

3.2.3 Classification of Imminent Conditions

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

3.2.4 Emergency Classification Level Upgrading and Downgrading

a. 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.

b. As noted above, guidance concerning classification of rapidly escalating events or conditions is provided in RIS 2007-02 (ref. 4.1.5).

3.2.5 Classification of Short-Lived Events

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

3.2.6 Classification of Transient Conditions

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

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

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

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

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

a successful corrective action prior to the Emergency Director completing the review and steps necessary to make the emergency declaration. This provision is included to ensure that any public protective actions resulting from the emergency classification are truly warranted by the plant conditions.

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

a. 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.

b. In these cases, no emergency declaration is warranted; however, the guidance contained in NUREG-1022 (ref. 4.1.6) is applicable. Specifically, the event should be reported to the NRC in accordance with 10 CFR § 50.72 (ref. 4.1.5) 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. 4.1.6).

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 Technical Specifications Table 1.1-1 Modes

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

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

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

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

4.1.7 HI-2002444, Holtec International Final Safety Analysis Report for the HI-STORM 100 Cask System, USNRC Docket No. 72-1014, Chapter 7, Confinement

4.1.8 PPM 1.20.3, Outage Risk Management

4.1.9 Deleted

4.1.10 10 § CFR 50.73 License Event Report System

4.1.11 M570, General Arrangement Plan - El. 572 ft - 0 in. and El. 606 ft - 10 1/2 in. - Reactor Building

4.1.12 Certificate of Compliance No. 1014 Appendix A Technical Specifications for the HI-STORM 100 Cask System Section 1.1 Definitions

4.1.13 SWP-PRO-03, Procedure Writer's Manual

4.1.14 CGS Physical Security Plan

4.1.15 CGS Graphics Plant Drawing 902118-P

4.1.16 Energy Northwest Columbia Generating Station Offsite Dose Calculation Manual, Amendment 52

4.2 Implementing

4.2.1 PPM 13.1.1, Classifying the Emergency

4.2.2 Emergency Plan Columbia Generating Station

4.2.3 Columbia Generating Station NEI 99-01 Revision 6 EAL Comparison Matrix

4.2.4 PPM 13.1.1B, EAL Hot Matrix

4.2.5 PPM 13.1.1C, EAL Cold Matrix

5.0 DEFINITIONS, ACRONYMS & ABBREVIATIONS

5.1 Definitions (ref. 4.1.1 except as noted)

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

5.1.1 ALERT

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

5.1.2 CAN/CANNOT BE MAINTAINED ABOVE/BELOW

The value of an identified parameter is/is not able to be held within the specified limit. The determination requires an evaluation of system performance and availability in relation to parameter values and trends. An instruction prescribing action when a parameter cannot be maintained above or below a specified limit neither requires nor prohibits anticipatory action-depending upon plant conditions, the action may be taken as soon as it is determined that the limit will ultimately be exceeded, or delayed until the limit is actually reached. Once the parameter does exceed the limit, however, the action must be performed; it may not be delayed while attempts are made to restore the parameter to within the desired control band.

5.1.3 CAN/CANNOT BE RESTORED ABOVE/BELOW

The value of an identified parameter is/is not able to be brought within the specified limit. The determination requires an evaluation of system performance and availability in relation to parameter values and trends. An instruction prescribing action when a value cannot be restored and maintained above or below a specified limit does not require immediate action simply because the current values is outside the range, but does not permit extended operation beyond the limit; the action must be taken as soon as it is apparent that the specified range cannot be attained.

5.1.4 CONFINEMENT BOUNDARY

The barrier(s) between spent fuel and the environment once the spent fuel is processed for dry storage. As related to the CGS ISFSI, Confinement Boundary is defined as the Multi-Purpose Canister (MPC) (ref. 4.1.7).

5.1.5 CONTAINMENT CLOSURE

The procedurally defined conditions or actions taken to secure Containment (Primary or Secondary) and its associated structures, systems, and components as a functional barrier to fission product release under shutdown conditions. A functional barrier is one which mitigates offsite release during an event. Containment Closure requires a functional barrier (not necessarily Technical Specification Operable; the appropriate structures, systems, and components are functional) to exist at the time of an event. The site cannot rely on contingency methods to establish a functional barrier after the event has started. In Mode 4 either a functional Primary Containment or a functional Secondary Containment is sufficient to mitigate offsite release. In Mode 5, a functional Secondary Containment is sufficient to mitigate offsite release. Therefore, Containment Closure is met in Mode 4 with either a functional Primary

Containment or a functional Secondary Containment. Containment Closure is met in Mode 5 with a functional Secondary Containment.

5.1.6 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 CGS to recommend protective actions for the general public to offsite planning agencies.

5.1.7 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.

5.1.8 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)
- General Emergency ~~(GE)~~

5.1.9 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.

5.1.10 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.

5.1.11 FISSION PRODUCT BARRIER THRESHOLD

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

5.1.12 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.

5.1.13 GENERAL EMERGENCY

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

5.1.14 HOSTAGE

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

5.1.15 HOSTILE ACTION

An act toward CGS or its personnel that includes the use of violent force to destroy equipment, take hostages, and/or intimidate Energy Northwest 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 CGS. 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).

5.1.16 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.

5.1.17 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.

5.1.18 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).

5.1.19 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.

5.1.20 INITIATING CONDITION

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

~~5.1.21 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.~~

~~5.1.22~~5.1.21 MAINTAIN

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

~~5.1.23~~5.1.22 NORMAL LEVELS

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

~~5.1.24~~5.1.23 OWNER CONTROLLED AREA

The area that Energy Northwest maintains industrial and process control of (ref. 4.2.2).

~~5.1.25~~5.1.24 PROJECTILE

An object directed toward CGS that could cause concern for its continued operability, reliability, or personnel safety.

~~5.1.26~~5.1.25 PROTECTED AREA

An area located within the OWNER CONTROLLED AREA which contains the Columbia Generating Station power block and is surrounded by chain link fence (ref. 4.2.2).

~~5.1.27~~5.1.26 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).

~~5.1.28~~5.1.27 REFUELING PATHWAY

Reactor cavity and spent fuel pool comprise the Refuel Pathway (ref. 4.1.11).

~~5.1.29~~5.1.28 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:

a. The integrity of the reactor coolant pressure boundary;

b. The capability to shut down the reactor and maintain it in a safe shutdown condition;

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

~~5.1.30~~5.1.29 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.

~~5.1.31~~5.1.30 SITE AREA EMERGENCY

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

~~5.1.32~~5.1.31 SITE BOUNDARY

1950-meter radius around the plant as depicted in Figure 3-1 of the CGS ODCM (ref. 4.1.16). The key-hole area between the river and this radius is not within the Site Boundary.

~~5.1.33~~5.1.32 UNISOLABLE

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

~~5.1.34~~5.1.33 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.

~~5.1.35~~5.1.34 UNUSUAL EVENT

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

~~5.1.36~~5.1.35 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.

~~5.1.37~~5.1.36 VISIBLE DAMAGE

Damage to components on two or more SAFETY SYSTEM trains, or one or more structures, that is readily observable without measurements, testing, or analysis. The visual impact of the damage is sufficient to cause concern regarding the operability or reliability of the affected SAFETY SYSTEM trains in the area. Events that result in visible damage to the components of one SAFETY SYSTEM train, and do not appear to affect the components of other SAFETY SYSTEM trains, do not meet the intent of this definition as the failure of a component(s) affecting the operability of one SAFETY SYSTEM train, regardless of cause, is well within the operational controls provided by a licensee's Technical Specifications and Operating Procedures. However, visible damage to the components of more than one SAFETY SYSTEM train does meet this definition, as well as visible damage to a structure.~~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
APRM	Average Power Range Meter

ARI	Automatic Rod Insertion
ATWS	Anticipated Transient Without Scram
BWR	Boiling Water Reactor
BWROG	Boiling Water Reactor Owners Group
CDE	Committed Dose Equivalent
CFR	Code of Federal Regulations
cpm	counts per minute
cps	counts per second
DBA	Design Basis Accident
DC	Direct Current
EAL	Emergency Action Level
ECCS	Emergency Core Cooling System
ECL	Emergency Classification Level
EOF	Emergency Operations Facility
EOP	Emergency Operating Procedure
EPA	Environmental Protection Agency
EPG	Emergency Procedure Guideline
EPIP	Emergency Plan Implementing Procedure
ESF	Engineered Safety Feature
FAA	Federal Aviation Administration
FBI	Federal Bureau of Investigation
FEMA	Federal Emergency Management Agency
FSAR	Final Safety Analysis Report
GDS	Graphic Display System
GE	~~General Emergency,~~ Greater than or Equal to
gm	Gram
GT	Greater Than
HCTL	Heat Capacity Temperature Limit
HPCS	High Pressure Core Spray
HOO	NRC Headquarters Operations Officer
IC	Initiating Condition
IDLH	Immediately Dangerous to Life and Health
IPEEE	Individual Plant Examination of External Events (Generic Letter 88-20)

ISFSI	Independent Spent Fuel Storage Installation
K_{eff}	Effective Neutron Multiplication Factor
LCO	Limiting Condition of Operation
LE	Less than or Equal to
LER	Licensee Event Report
LFL	Lower Flammability Limit
LOCA	Loss of Coolant Accident
LPCS	Low Pressure Core Spray
LT	Less Than
LWR	Light Water Reactor
MPC	Maximum Permissible Concentration/Multi-Purpose Canister
µCi	Micro Curie
MSCRWL	Minimum Steam Cooling RPV Water Level
MSCP	Minimum Steam Cooling Pressure
MSIV	Main Steam Isolation Valve
MSL	Main Steam Line
mR	milliRoentgen
MW	Megawatt
NEI	Nuclear Energy Institute
NESP	National Environmental Studies Project
NORAD	North American Aerospace Defense Command
NPP	Nuclear Power Plant
NRC	Nuclear Regulatory Commission
NSSS	Nuclear Steam Supply System
OBE	Operating Basis Earthquake
OCA	Owner Controlled Area
ODCM	Off-site Dose Calculation Manual
ORO	Offsite Response Organization
PPM	Plant Procedure Manual
PMU	Panel Meter Unit
PRA/PSA	Probabilistic Risk Assessment / Probabilistic Safety Assessment
PRM	Process Radiation Monitor
PWR	Pressurized Water Reactor

PSIG	Pounds per Square Inch Gauge
PSP	Pressure Suppression Pressure
R	Roentgen
RB	Reactor Building
RCC	Reactor Building Closed Cooling
RCIC	Reactor Core Isolation Cooling
RCS	Reactor Coolant System
Rem	Roentgen Equivalent Man
RHR	Residual Heat Removal
RPS	Reactor Protection System
RPV	Reactor Pressure Vessel
RWCU	Reactor Water Cleanup
SGT	Stand-By Gas Treatment
SBO	Station Blackout
SDSP	Shutdown Safety Plan
SLC	Standby Liquid Control
SPDS	Safety Parameter Display System
SRO	Senior Reactor Operator
SSC	Structure, System or Component
SW	Service Water
TEA	Turbine Exhaust Air
TEDE	Total Effective Dose Equivalent
TAF	Top of Active Fuel
TSC	Technical Support Center
TSW	Plant Service Water
WEA	Waste Exhaust Air

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

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

CGS EAL	NEI 99-01 Rev. 6	
	IC	Example EAL
RU1.1	AU1	1, 2, 3
RU2.1	AU2	1
RA1.1	AA1	1, 2
RA1.2	AA1	3
RA1.3	AA1	4
RA2.1	AA2	1
RA2.2	AA2	2
RA2.3	AA2	3
RA3.1	AA3	1, 2
RS1.1	AS1	1, 2
RS1.2	AS1	3
RS2.1	AS2	1
RG1.1	AG1	1, 2
RG1.2	AG1	3
RG2.1	AG2	1
CU1.1	CU1	1, 2
CU2.1	CU2	1
CU3.1	CU3	1
CU3.2	CU3	2
CU4.1	CU4	1
CU5.1	CU5	1, 2, 3
CA1.1	CA1	1, 2
CA2.1	CA2	1

CGS EAL	NEI 99-01 Rev. 6	
	IC	Example EAL
CA3.1	CA3	1, 2
CA6.1	CA6	1
CS1.1	CS1	1, 2
CS1.2	CS1	3
CG1.1	CG1	1
CG1.2	CG1	2
FA1.1	FA1	1
FS1.1	FS1	1
FG1.1	FG1	1
HU1.1	HU1	1, 2 3
HU2.1	HU2	1
HU3.1	HU3	1, 5
HU3.2	HU3	2
HU3.3	HU3	3, 4
HU4.1	HU4	1
HU4.2	HU4	2
HU4.3	HU4	3, 4
HU7.1	HU7	1
HA1.1	HA1	1, 2
HA5.1	HA5	1
HA6.1	HA6	1
HA7.1	HA7	1
HS1.1	HS1	1
HS6.1	HS6	1
HS7.1	HS7	1
HG7.1	HG7	1

CGS EAL	NEI 99-01 Rev. 6	
	IC	Example EAL
MU1.1	SU1	1
MU3.1	SU2	1
MU4.1	SU3	1
MU4.2	SU3	2
MU5.1	SU4	1, 2, 3
MU6.1	SU5	1, 2
MU7.1	SU6	1, 2, 3
MA1.1	SA1	1
MA3.1	SA2	1
MA6.1	SA5	1
MA8.1	SA9	1
MS1.1	SS1	1
MS2.1	SS8	1
MS6.1	SS5	1
MG1.1	SG1	1
MG1.2	SG8	1
EU1.1	E-HU1	1

7.0 ATTACHMENTS

7.1 Emergency Action Level Technical Bases

7.2 Fission Product Barrier Matrix and Bases

7.3 Notes and Tables

7.4 Safe Operation & Shutdown Areas Table 9 Bases

7.5 Columbia Generating Station Emergency Classification Chart Distribution

ATTACHMENT 7.1: EAL Technical Bases

Category R – Abnormal Rad Release / Rad Effluent

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

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

At lower levels, abnormal radioactivity releases may be indicative of a failure of containment systems or precursors to more significant releases. At higher release rates, offsite radiological conditions may result which require offsite protective actions. Elevated area radiation levels in the 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 Release / Rad Effluent

Subcategory: 1 – Radiological Effluent

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

EAL:

RU1.1 Unusual Event

(1) Reading on any Table 3 effluent radiation monitor GT column "UE" for GE 60 min.

OR

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

(Notes 1, 2, 3)

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

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

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

Table 3 Effluent Monitor Classification Thresholds

	Release Point	Monitor	General	SAE	Alert	UE
Gaseous	Reactor Building Exhaust	PRM-RE-1B (I)	----	----	----	6.00E+03 cps
Gaseous	Reactor Building Exhaust	PRM-RE-1C (H)	2.00E+04 cps	2.00E+03 cps	4.00E+02 cps	----
Gaseous	Turbine Building Exhaust	TEA-RIS-13	8.35E-02 µCi/cc	8.35E-03 µCi/cc	8.35E-04 µCi/cc	1.02E-04 µCi/cc
Gaseous	Radwaste Building Exhaust	WEA-RIS-14	3.45E-01 µCi/cc	3.45E-02 µCi/cc	3.45E-03 µCi/cc	1.98E-03 µCi/cc
Liquid	Radwaste Effluent	FDR-RIS-606	----	----	----	2 X HI-HI alarm
Liquid	TSW Effluent	TSW-RIS-5	----	----	----	3.00E-05 µCi/cc
Liquid	Service Water Process A Service Water Process B	SW-RIS-604 SW-RIS-605	----	----	----	1.00E+02 cps 1.00E+02 cps

Mode Applicability:

1	2	3	4	5	def

Basis:

Per NEI 99-01, this EAL addresses normally occurring continuous radioactivity releases from monitored gaseous or liquid effluent pathways and planned batch releases from releases from non-continuous release pathways. The column "UE" gaseous release values in Table 3 represent two times the appropriate ODCM release rate limits associated with the specified monitors (ref. 1, 2, 3, 4).

The Radwaste Effluent monitor (FDR-RIS-606) Hi-Hi alarm is established per a discharge permit and should be multiplied by 2 to determine the effluent threshold.

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

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

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

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

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

Threshold #1 - This threshold addresses normally occurring continuous radioactivity releases from monitored gaseous or liquid effluent pathways as well as radioactivity releases that cause effluent radiation monitor readings to exceed 2 times the limit established by a radioactivity discharge permit. This EAL may also be associated with planned batch releases from non-continuous release pathways (e.g., radwaste, waste gas).

Threshold #2 - This threshold addresses uncontrolled gaseous or liquid releases that are detected by sample analyses or environmental surveys, particularly on unmonitored pathways (e.g., spills of radioactive liquids into storm drains, heat exchanger leakage in river water systems, etc.).

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

CGS Basis Reference(s):

1. CGS Offsite Dose Calculation Manual (ODCM)
2. Calculation NE-02-09-12 Revision 3
3. 16.10.1 Radioactive Liquid Waste Discharge to the River
4. FSAR Section 11.5 Process and Effluent Radiological Monitoring and Sampling
5. NEI 99-01 AU1

ATTACHMENT 7.1: EAL Technical Bases

Category: R – Abnormal Rad Release / Rad Effluent

Subcategory: 1 – Radiological Effluent

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

EAL:

RA1.1 Alert

(1) Reading on any Table 3 effluent radiation monitor GT column "ALERT" for GE 15 min.

OR

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

(Notes 1, 2, 3, 4)

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

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

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

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

Table 3 Effluent Monitor Classification Thresholds

	Release Point	Monitor	General	SAE	Alert	UE
Gaseous	Reactor Building Exhaust	PRM-RE-1B (I)	----	----	----	6.00E+03 cps
Gaseous	Reactor Building Exhaust	PRM-RE-1C (H)	2.00E+04 cps	2.00E+03 cps	4.00E+02 cps	----
Gaseous	Turbine Building Exhaust	TEA-RIS-13	8.35E-02 µCi/cc	8.35E-03 µCi/cc	8.35E-04 µCi/cc	1.02E-04 µCi/cc
Gaseous	Radwaste Building Exhaust	WEA-RIS-14	3.45E-01 µCi/cc	3.45E-02 µCi/cc	3.45E-03 µCi/cc	1.98E-03 µCi/cc
Liquid	Radwaste Effluent	FDR-RIS-606	----	----	----	2 X HI-HI alarm
Liquid	TSW Effluent	TSW-RIS-5	----	----	----	3.00E-05 µCi/cc
Liquid	Service Water Process A Service Water Process B	SW-RIS-604 SW-RIS-605	----	----	----	1.00E+02 cps 1.00E+02 cps

Mode Applicability:

1	2	3	4	5	def

Basis:

Threshold #1

The pre-calculated effluent monitor values presented in Table 3 should be used for emergency classification assessments **only** until the results from a dose assessment using actual meteorology are available.

This EAL address gaseous radioactivity releases, that for whatever reason, cause effluent radiation monitor readings corresponding to SITE BOUNDARY doses that exceed either (ref. 1, 2):

- 10 mRem TEDE
- 50 mRem CDE Thyroid

The column “ALERT” gaseous effluent release values in Table 3 correspond to calculated doses of 1% of the EPA Protective Action Guidelines (TEDE or CDE Thyroid) (ref. 1).

Threshold #2

Dose assessments are performed by computer-based methods (ref. 3).

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

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

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

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

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

CGS Basis Reference(s):

1. Calculation NE-02-09-12 Revision 3
2. FSAR Section 11.5 Process and Effluent Radiological Monitoring and Sampling
3. PPM 13.8.1 Emergency Dose Projection System Operations
4. NEI 99-01 AA1

ATTACHMENT 7.1: EAL Technical Bases

Category: R – Abnormal Rad Release / Rad Effluent

Subcategory: 1 – Radiological Effluent

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

EAL:

RA1.2 Alert

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

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

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

Mode Applicability:

1	2	3	4	5	def

Basis:

For a radiological water release, the calculated effluent concentration from a field team sample is compared to the emergency action level (ref. 1, 2, 3).

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

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

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

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

CGS Basis Reference(s):

1. FSAR Section 11.5 Process and Effluent Radiological Monitoring and Sampling
2. PPM 13.9.1 Environmental Field Monitoring Operations
3. PPM 13.9.5 Environmental Sample Collection
4. NEI 99-01 AA1

Category: R – Abnormal Rad Release / Rad Effluent

Subcategory: 1 – Radiological Effluent

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

EAL:

RA1.3 Alert

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

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

(Notes 1, 2)

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

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

Mode Applicability:

1	2	3	4	5	def

Basis:

Plant procedures, provides guidance for emergency or post-accident radiological environmental monitoring (ref. 1).

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

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.

CGS Basis Reference(s):

1. PPM 13.9.1 Environmental Field Monitoring Operations
2. NEI 99-01 AA1

ATTACHMENT 7.1: EAL Technical Bases

Category: R – Abnormal Rad Release / 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

(1) Reading on any Table 3 effluent radiation monitor GT column "SAE" for GE 15 min.

OR

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

(Notes 1, 2, 3, 4)

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

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

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

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

Table 3 Effluent Monitor Classification Thresholds

	Release Point	Monitor	General	SAE	Alert	UE
Gaseous	Reactor Building Exhaust	PRM-RE-1B (I)	----	----	----	6.00E+03 cps
Gaseous	Reactor Building Exhaust	PRM-RE-1C (H)	2.00E+04 cps	2.00E+03 cps	4.00E+02 cps	----
Gaseous	Turbine Building Exhaust	TEA-RIS-13	8.35E-02 µCi/cc	8.35E-03 µCi/cc	8.35E-04 µCi/cc	1.02E-04 µCi/cc
Gaseous	Radwaste Building Exhaust	WEA-RIS-14	3.45E-01 µCi/cc	3.45E-02 µCi/cc	3.45E-03 µCi/cc	1.98E-03 µCi/cc
Liquid	Radwaste Effluent	FDR-RIS-606	----	----	----	2 X HI-HI alarm
Liquid	TSW Effluent	TSW-RIS-5	----	----	----	3.00E-05 µCi/cc
Liquid	Service Water Process A Service Water Process B	SW-RIS-604 SW-RIS-605	----	----	----	1.00E+02 cps 1.00E+02 cps

Mode Applicability:

1	2	3	4	5	def

Basis:

Threshold #1

The pre-calculated effluent monitor values presented in Table 3 should **only** be used for emergency classification assessments until the results from a dose assessment using actual meteorology are available.

This EAL address gaseous radioactivity releases, that for whatever reason, cause effluent radiation monitor readings corresponding to SITE BOUNDARY doses that exceed either (ref. 1, 2):

- 100 mRem TEDE
- 500 mRem CDE Thyroid

The column "SAE" gaseous effluent release values in Table 3 correspond to calculated doses of 10% of the EPA Protective Action Guidelines (TEDE or CDE Thyroid) (ref. 1).

Threshold #2

Dose assessments are performed by computer-based methods (ref. 3).

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.

CGS Basis Reference(s):

1. Calculation NE-02-09-12 Revision 3
2. FSAR Section 11.5 Process and Effluent Radiological Monitoring and Sampling
3. PPM 13.8.1 Emergency Dose Projection System Operations
4. NEI 99-01 AS1

Category: R – Abnormal Rad Release / 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

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

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

(Notes 1, 2)

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

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

Mode Applicability:

1	2	3	4	5	def

Basis:

Plant procedures provide 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.

CGS Basis Reference(s):

1. PPM 13.9.1 Environmental Field Monitoring Operations
2. NEI 99-01 AS1

ATTACHMENT 7.1: EAL Technical Bases

Category: R – Abnormal Rad Release / 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

(1) Reading on <u>any</u> Table 3 effluent radiation monitor GT column "GENERAL" for GE 15 min.

OR

(2) Dose assessment using actual meteorology indicates doses GT 1,000 mrem TEDE or GT 5,000 mrem thyroid CDE at or beyond the SITE BOUNDARY

(Notes 1, 2, 3, 4)

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

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

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

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

Table 3 Effluent Monitor Classification Thresholds

	Release Point	Monitor	General	SAE	Alert	UE
Gaseous	Reactor Building Exhaust	PRM-RE-1B (I)	----	----	----	6.00E+03 cps
Gaseous	Reactor Building Exhaust	PRM-RE-1C (H)	2.00E+04 cps	2.00E+03 cps	4.00E+02 cps	----
Gaseous	Turbine Building Exhaust	TEA-RIS-13	8.35E-02 µCi/cc	8.35E-03 µCi/cc	8.35E-04 µCi/cc	1.02E-04 µCi/cc
Gaseous	Radwaste Building Exhaust	WEA-RIS-14	3.45E-01 µCi/cc	3.45E-02 µCi/cc	3.45E-03 µCi/cc	1.98E-03 µCi/cc
Liquid	Radwaste Effluent	FDR-RIS-606	----	----	----	2 X HI-HI alarm
Liquid	TSW Effluent	TSW-RIS-5	----	----	----	3.00E-05 µCi/cc
Liquid	Service Water Process A Service Water Process B	SW-RIS-604 SW-RIS-605	----	----	----	1.00E+02 cps 1.00E+02 cps

Mode Applicability:

1	2	3	4	5	def

Basis:

Theshold #1

The pre-calculated effluent monitor values presented in Table 3 should **only** be used for emergency classification assessments until the results from a dose assessment using actual meteorology are available.

This EAL address gaseous radioactivity releases, that for whatever reason, cause effluent radiation monitor readings corresponding to SITE BOUNDARY doses that exceed either (ref. 1, 2):

- 1000 mRem TEDE
- 5000 mRem CDE Thyroid

The column "~~SAE~~GENERAL" gaseous effluent release values in Table 3 correspond to calculated doses of 100% of the EPA Protective Action Guidelines (TEDE or CDE Thyroid) (ref. 1).

Threshold #2

Dose assessments are performed by computer-based methods (ref. 3).

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.

CGS Basis Reference(s):

1. Calculation NE-02-09-12 Revision 3
2. FSAR Section 11.5 Process and Effluent Radiological Monitoring and Sampling
3. PPM 13.8.1 Emergency Dose Projection System Operations
4. NEI 99-01 AG1

Category: R – Abnormal Rad Release / 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

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

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

(Notes 1, 2)

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

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

Mode Applicability:

1	2	3	4	5	def

Basis:

Plant procedures provide 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.

CGS Basis Reference(s):

1. PPM 13.9.1 Environmental Field Monitoring Operations
2. NEI 99-01 AG1

Category: R – Abnormal Rad Release / 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 EITHER of the following:

- SFP level LE 22.3 ft.
- SFP low level alarm

AND

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

- ARM-RIS-1 Reactor Building Fuel Pool Area
- ARM-RIS-2 Reactor Building Fuel Pool Area
- ARM-RIS-34 Reactor Building Elevation 606

Mode Applicability:

1	2	3	4	5	def

Basis:

The spent fuel pool is designed to maintain the water level in the pool above the top of active fuel providing cooling for the fuel bundles. The fuel pool low level alarm is actuated by level switch FP-LS-4A when fuel pool water level drops below 605' 5-1/2". SFP level is can be determined by FPC-LI-21, FPC-LIT-21A, FPC-LIT-21B or local indication (ref. 1, 2, 3).

This EAL is applicable for conditions in which irradiated fuel is being transferred to and from the RPV and spent fuel pool as well as for spent fuel pool drain down events.

ARM-RIS-1 and ARM-RIS-2 are located in the fuel pool area of the 606' elevation of the Reactor

Building. ARM-RIS-34 is located on the east side of the 606' elevation of the Reactor Building (ref. 4).

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

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

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

CGS Basis Reference(s):

1. PPM 4.626.FPC1-2.2 (4.626.FPC2-2.2) Fuel Pool Level High/Low
2. PPM 4.627.FPC2-2.2 (4.627.FPC2-2.2) Fuel Pool Level High/Low
3. ABN-FPC-LOSS Loss of Fuel Pool Cooling
4. FSAR Table 12.3-1 Area Monitors
5. NEI 99-01 AU2

ATTACHMENT 7.1: EAL Technical Bases

Category: R – Abnormal Rad Release / 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:

1	2	3	4	5	def

Basis:

The spent fuel pool is designed to maintain the water level in the pool above the top of active fuel providing cooling for the fuel bundles.

This EAL is applicable for conditions in which irradiated fuel is being transferred to and from the RPV and spent fuel pool as well as for spent fuel pool drain down events.

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

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

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

This EAL escalates from RU2.1 in that the loss of level, in the affected portion of the REFUEL 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 REFUEL 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 Refuel modes.

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

CGS Basis Reference(s):

1. ABN-FPC-LOSS Loss of Fuel Pool Cooling
2. NEI 99-01 AA2

ATTACHMENT 7.1: EAL Technical Bases

Category: R – Abnormal Rad Release / Rad Effluent

Subcategory: 2 – Irradiated Fuel Event

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

EAL:

RA2.2 Alert

Damage to irradiated fuel resulting in a release of radioactivity

AND

High alarm on any of the following radiation monitors:

- ARM-RIS-1 Reactor Building Fuel Pool Area
- ARM-RIS-2 Reactor Building Fuel Pool Area
- ARM-RIS-34 Reactor Building Elevation 606
- REA-RIS-609A-D Rx Bldg Vent

Mode Applicability:

1	2	3	4	5	def

Basis:

ARM-RIS-1 and ARM-RIS-2 are located in the fuel pool area of the 606' elevation of the Reactor Building. ARM-RIS-34 is located on the east side of the 606' elevation of the Reactor Building (Ref. 1). The ARM alarm setpoints are controlled by procedure.

REA-RIS-609A-D are the Reactor Building Exhaust Plenum radiation monitors. This system monitors the radiation level of the reactor building ventilation system exhaust plenum prior to its discharge from the building into the elevated release duct. A high radioactivity level in the exhaust system could be due to fission gases from damaged or leaking spent fuel or an accident (ref. 2). Actuation of the High-High alarm actuates a Secondary Containment isolation and starts SGT (ref. 3).

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

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

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

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

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

CGS Basis Reference(s):

1. CGS FSAR Table 12.3-1 Area Monitors

2. FSAR Section 11.5.2.1.2 Reactor Building Exhaust Plenum Radiation Monitoring System
3. PPM 4.602.A5-1.4 Reactor Building Exh Plenum Rad Hi-Hi
4. NEI 99-01 AA2

Category: R – Abnormal Rad Release / Rad Effluent

Subcategory: 2 – Irradiated Fuel Event

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

EAL:

RA2.3 **Alert**
Lowering of spent fuel pool level to 10 ft

Mode Applicability:

1	2	3	4	5	def

Basis:

The spent fuel pool is designed to maintain the water level in the pool above the top of irradiated fuel and thus providing cooling for the fuel assemblies. SFP level can be determined by FPC-LIT-21A, FPC-LIT-21B, FPC-LI-21 or local indication. Instrument "reference zero" is the top of the spent fuel pool racks (ref. 1).

The spent fuel pool is equipped with primary and backup guided wave radar probes to measure pool level. The range is continuous from the high pool level elevation to the top of the spent fuel racks. Post-Fukushima order EA-12-051 required the installation of reliable SFP level indication (FPC-LIT-21A and FPC-LIT-21B) capable of identifying SFP level providing personnel shielding (Level 2: 9.8 ft [rounded to 10 ft.]) (ref. 1).

This EAL addresses events that have caused imminent or actual damage to an irradiated fuel assembly, or a significant lowering of water level within the spent fuel pool. 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.

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 Refuel modes.

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

Escalation of the emergency classification level would be via ICs RS1or RS2).

CGS Basis Reference(s):

1. IMDS for FPC-LIT-21A/21B
2. NEI 99-01 AA2

ATTACHMENT 7.1: EAL Technical Bases

Category: R – Abnormal Rad Release / Rad Effluent

Subcategory: 2 – Irradiated Fuel Event

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

EAL:

RS2.1 Site Area Emergency

Lowering of spent fuel pool level to 0.5 ft

Mode Applicability:

1	2	3	4	5	def

Basis:

The spent fuel pool is designed to maintain the water level in the pool above the top of irradiated fuel and thus providing cooling for the fuel assemblies. SFP level can be determined by FPC-LIT-21A, FPC-LIT-21B, FPC-LI-21 or local indication. Instrument “reference zero” is the top of the spent fuel pool racks (ref. 1).

The spent fuel pool is equipped with primary and backup guided wave radar probes to measure pool level. The range is continuous from the high pool level elevation to the top of the spent fuel racks. Post-Fukushima order EA-12-051 required the installation of reliable SFP level indication (FPC-LIT-21A and FPC-LIT-21B) capable of identifying SFP level near top of the fuel racks (Level 3: 0.4 ft [rounded to 0.5 ft]) (ref. 1).

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

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

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

CGS Basis Reference(s):

1. IMDS for FPC-LIT-21A/21B
2. NEI 99-01 AS2

ATTACHMENT 7.1: EAL Technical Bases

Category: R – Abnormal Rad Release / Rad Effluent

Subcategory: 2 – Irradiated Fuel Event

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

EAL:

RG2.1 General Emergency

Spent fuel pool level cannot be restored to at least 0.5 ft for GE 60 min. (Note 1)

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

Mode Applicability:

1	2	3	4	5	def

Basis:

The spent fuel pool is designed to maintain the water level in the pool above the top of irradiated fuel and thus providing cooling for the fuel assemblies. SFP level can be determined by FPC-LIT-21A, FPC-LIT-21B, FPC-LI-21 or local indication. Instrument “reference zero” is the top of the spent fuel pool racks (ref. 1).

The spent fuel pool is equipped with primary and backup guided wave radar probes to measure pool level. The range is continuous from the high pool level elevation to the top of the spent fuel racks. Post-Fukushima order EA-12-051 required the installation of reliable SFP level indication (FPC-LIT-21A and FPC-LIT-21B) capable of identifying SFP level near top of the fuel racks (Level 3: 0.4 ft [rounded to 0.5 ft]). (ref. 1).

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

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

CGS Basis Reference(s):

1. IMDS for FPC-LIT-21A/21B
2. NEI 99-01 AG2

Category: R – Abnormal Rad Release / 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

(1) Dose rates GT 15 mR/hr in Control Room (ARM-RIS-19) or CAS (by survey)

OR

(2) An UNPLANNED event results in radiation levels that prohibit or IMPEDE access to any Table 9 rooms or areas (Note 5)

Table 9 Safe Operation & Shutdown Areas	
Room/Area	**Mode Applicability**
RW 467' Radwaste Control Room (RHR flush to RW tanks)	3
RW 467' Vital Island (RHR-V-9 disconnect)	3
RB 422' B RHR Pump Rm (local pump temperatures)	3
RB 454' B RHR Pump Rm (operate RHR-V-85B)	3

Mode Applicability:

1	2	3	4	5	def

Basis:

Threshold #1

The CGS Control Room requires continuous occupancy because of its importance to assure safe plant operations and control of site security functions (Central Alarm Station).

Control Room ARM (ARM-RIS-19) measures area radiation in a range of 1 to 10^4 mR/hr (ref. 1).

Threshold #2

The list of plant rooms or areas with entry-related mode applicability identified specify those rooms or areas that contain equipment which require a manual/local action as specified in operating procedures used for normal plant operation, cooldown and shutdown. Rooms or areas in which actions of a contingent or emergency nature would be performed (e.g., an action to address an off-normal or emergency condition such as emergency repairs, corrective measures or emergency operations) are not included. In addition, the list specifies the plant mode(s) during which entry would be required for each room or area (ref. 2).

This IC addresses elevated radiation levels in certain plant rooms/areas sufficient to preclude or impede personnel from performing actions necessary to maintain normal plant operation, or to perform

a normal plant cooldown and shutdown. As such, it represents an actual or potential substantial degradation of the level of safety of the plant. The Emergency Director should consider the cause of the increased radiation levels and determine if another IC may be applicable.

For threshold #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.

CGS Basis Reference(s):

1. FSAR Table 12.3-1 Area Monitors
2. Attachment 7.4 Safe Operation & Shutdown Rooms/Areas Tables 9 Bases
3. NEI 99-01 AA3

Category C – Cold Shutdown / Refuel System Malfunction

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

Category C EALs are directly associated with cold shutdown or Refuel 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 Refuel system malfunction EALs are based on performance capability to the extent possible with consideration given to RCS integrity, CONTAINMENT CLOSURE, and fuel clad integrity for the applicable operating modes (4 - Cold Shutdown, 5 - Refuel, D – Defueled).

The events of this category pertain to the following subcategories:

1. RPV Level

Reactor Pressure Vessel 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 4160 V 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 vital plant electrical power can compromise plant safety system operability including decay heat removal and emergency core cooling systems which may be necessary to ensure fission product barrier integrity. This category includes loss of power to or degraded voltage on the vital125 VDC 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.

ATTACHMENT 7.1: EAL Technical Bases

Category: C – Cold Shutdown / Refuel System Malfunction

Subcategory: 1 – RPV Level

Initiating Condition: UNPLANNED loss of RPV inventory

EAL:

CU1.1 Unusual Event

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

OR

(2) RPV level cannot be monitored

AND

UNPLANNED increase in any Table 1 sump or pool levels due to a loss of RPV inventory

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

Table 1 Sumps/Pool
• Any valid Hi-Hi level alarm on R-1 through R-5 sumps • EDR GE 25 GPM • FDR GE 10 GPM • Wetwell level rise • Observation of UNISOLABLE RCS leakage

Mode Applicability:

			4	5	

Basis:

~~This~~ These Cold Shutdown EALs represent~~s~~ the hot condition EAL MU5.1, in which RCS leakage is associated with Technical Specification limits. In Cold Shutdown, these limits are not applicable; hence, the use of RPV level as the parameter of concern in this EAL.

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

Refuel 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.

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

ATTACHMENT 7.1: EAL Technical Bases

EAL #1

In Mode 4 and Mode 5, prior to flood up, RPV level is monitored from -310 in. to +400 in. to ensure adequate coverage for expected and postulated conditions of RPV level. All instruments are referenced to a benchmark at 527.5 in. above the inside bottom head of the reactor vessel. This benchmark corresponds to the bottom edge of the steam dryer skirt and is the 0 in. reference indication on the RPV level instruments (ref. 1, 2, 3).

In preparation for refueling operations, level instruments are modified to provide continuous level indication from within the RPV to the refuel floor (ref. 4, 5).

The RPV level is controlled in a designated band in the reactor vessel and 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 Refuel mode, RPV water level is normally maintained at or above the reactor vessel flange (ref. 6).

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

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

EAL #2

In this EAL, all RPV level indication is unavailable and the RPV inventory loss must be detected by the leakage indications listed in Table 1. Level increases must be evaluated against other potential sources of leakage such as cooling water sources inside the drywell to ensure they are indicative of RPV leakage. Reactor Building equipment or floor drain sump level rise may be indicative of RPV inventory losses external to the primary containment from systems connected to the RPV (ref. 7, 8, 6). With RHR System operating in the Shutdown Cooling mode, an unexplained rise in wetwell level could be indicative of RHR valve misalignment or leakage (ref. 10). If the make-up rate to the RPV unexplainably rises above the pre-established rate, a loss of RPV inventory may be occurring even if the source of the leakage cannot be immediately identified. Visual observation of leakage from systems connected to the RCS in areas outside the primary containment that cannot be isolated could be indicative of a loss of RPV inventory.

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

CGS Basis Reference(s):

1. FSAR Section 7.5.1.1
2. FSAR Table 7.5-1
3. FSAR Figure 7.7-1
4. PPM 10.27.39 Refueling Reactor Vessel Level (Temporary)
5. SOP-CAVITY-FILL Reactor Cavity and Dryer Separator Pit Fill
6. Technical Specifications 3.9.6
7. FSAR Section 7.6.1.3

8. SOP-EDR-OPS Equipment Drain System Operation
9. SOP-FDR-OPS Floor Drain System Operation
10. SOP-RHR-SDC RHR Shutdown Cooling
11. NEI 99-01 CU1

ATTACHMENT 7.1: EAL Technical Bases

Category: C – Cold Shutdown / Refuel System Malfunction

Subcategory: 1 – RPV Level

Initiating Condition: Significant loss of RPV inventory

EAL:

CA1.1 Alert

(1) Loss of RPV inventory as indicated by RPV level LT -50 in.

OR

(2) RPV level cannot be monitored for GE 15 min. (Note 1)

AND

UNPLANNED increase in any Table 1 sump or pool levels due to a loss of RPV inventory

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

Table 1 Sumps/Pool
• Any valid Hi-Hi level alarm on R-1 through R-5 sumps • EDR GE 25 GPM • FDR GE 10 GPM • Wetwell level rise • Observation of UNISOLABLE RCS leakage

Mode Applicability:

			4	5	

Basis:

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

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

EAL #1

The threshold RPV level of -50 in. is the low-low ECCS (HPCS) actuation setpoint (ref. 1, 2).

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

For EAL #1, a lowering of water level below -50 in. indicates that operator actions have not been successful in restoring and maintaining RPV water level. The heat-up rate of the coolant will increase as the available water inventory is reduced. A continuing decrease in water level will lead to core uncovery.

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

EAL #2

In this EAL, all RPV level indication is unavailable and the RPV inventory loss must be detected by the leakage indications listed in Table 1. Level increases must be evaluated against other potential sources of leakage such as cooling water sources inside the drywell to ensure they are indicative of RPV leakage. Reactor Building equipment or floor drain sump level rise may be indicative of RPV inventory losses external to the primary containment from systems connected to the RPV (ref. 3, 4). With RHR System operating in the Shutdown Cooling mode, an unexplained rise in wetwell level could be indicative of RHR valve misalignment or leakage (ref. 5). If the make-up rate to the RPV unexplainably rises above the pre-established rate, a loss of RPV inventory may be occurring even if the source of the leakage cannot be immediately identified. Visual observation of leakage from systems connected to the RCS in areas outside the primary containment that cannot be isolated could be indicative of a loss of RPV inventory.

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

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

CGS Basis Reference(s):

1. Technical Specifications Table 3.3.5.1-1
2. PPM 5.1.1 RPV Control
3. SOP-EDR-OPS Equipment Drain System Operation
4. SOP-FDR-OPS Floor Drain System Operation
5. SOP-RHR-SDC RHR Shutdown Cooling
6. NEI 99-01 CA1

ATTACHMENT 7.1: EAL Technical Bases

Category: C – Cold Shutdown / Refuel System Malfunction

Subcategory: 1 – RPV Level

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

EAL:

CS1.1 Site Area Emergency

(1) CONTAINMENT CLOSURE not established

AND

RPV level LT -129 in.

OR

(2) CONTAINMENT CLOSURE established

AND

RPV level LT -161 in.

Mode Applicability:

			4	5	

Basis:

EAL #1

The threshold RPV water level of -129 in. is the low-low-low ECCS actuation setpoint. The magnitude of this loss of water indicates that makeup systems have not been effective and may not be capable of preventing further RPV water level decrease and potential core uncovery. The inability to restore and maintain level after reaching this setpoint infers a failure of the RCS barrier and Potential Loss of the Fuel Clad barrier. (ref. 1)

EAL #2

When RPV level drops to the top of active fuel (an indicated RPV level of -161 in.), core uncovery starts to occur (ref. 2).

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

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

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

This EAL addresses concerns raised by Generic Letter 88-17, Loss of Decay Heat Removal; SECY 91-283, Evaluation of Shutdown and Low Power Risk Issues; NUREG-1449, Shutdown and Low-Power Operations 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.

CGS Basis Reference(s):

1. Technical Specifications Table 3.3.5.1-1, “Emergency Core Cooling System Instrumentation”
2. PPM 5.1.1 RPV Control
3. NEI 99-01 CS1

ATTACHMENT 7.1: EAL Technical Bases

Category: C – Cold Shutdown / Refuel System Malfunction

Subcategory: 1 – RPV Level

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

EAL:

CS1.2 Site Area Emergency

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

AND

Core uncovery is indicated by any of the following:

- UNPLANNED wetwell level rise GT 2 inches (PPM 5.2.1 entry condition)
- VALID indication of RB room flooding as identified by high level alarms (PPM 5.3.1 Table 25)
- Observation of UNISOLABLE RCS leakage outside primary containment of sufficient magnitude to indicate core uncovery

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

Mode Applicability:

			4	5	

Basis:

In this EAL, all RPV level indication is unavailable and the RPV inventory loss must be detected by the leakage indications provided. Level increases must be evaluated against other potential sources of leakage such as cooling water sources inside the drywell to ensure they are indicative of RPV leakage. Reactor Building equipment or floor drain sump level rise may be indicative of RPV inventory losses external to the primary containment from systems connected to the RPV (ref. 1, 2). With RHR System operating in the Shutdown Cooling mode, an unexplained rise in wetwell level could be indicative of RHR valve misalignment or leakage (ref. 3). If the make-up rate to the RPV unexplainably rises above the pre-established rate, a loss of RPV inventory may be occurring even if the source of the leakage cannot be immediately identified.

An UNPLANNED wetwell level increase to GT 2 inches or a VALID RB room high level alarm indicates a significant loss of RCS that could lead to core uncovery if not isolated (ref. 4, 5).

Visual observation of significant leakage from systems connected to the RCS in areas outside the primary containment that cannot be isolated could be indicative of a loss of RPV inventory sufficient to lead to core uncovery.

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

CGS Basis Reference(s):

1. SOP-EDR-OPS Equipment Drain System Operation
2. SOP-FDR-OPS Floor Drain System Operation
3. SOP-RHR-SDC RHR Shutdown Cooling
4. PPM 5.2.1 Primary Containment Control
5. PPM 5.3.1 Secondary Containment Control
6. NEI 99-01 CS1

Category: C – Cold Shutdown / Refuel System Malfunction

Subcategory: 1 – RPV Level

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

EAL:

CG1.1 General Emergency

RPV level LT -161 in. for GE 30 min. (Note 1)

AND

Any of the following indications of Containment Challenge:

- CONTAINMENT CLOSURE not established (Note 6)
- Explosive mixture inside PC (H_2 GE 6% and O_2 GE 5%)
- UNPLANNED rise in PC pressure
- RB area radiation GT any Maximum Safe Operating level (PPM 5.3.1 Table 24)

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

Mode Applicability:

			4	5	

Basis:

When RPV level drops to the top of active fuel (an indicated RPV level of -161 in.), core uncovery starts to occur (ref. 1, 2).

Four conditions are associated with a challenge to primary containment (PC) integrity:

- Containment Closure is defined as the Shutdown Safety Plan (SDSP) actions taken to secure primary or secondary containment and its associated structures, systems, and components as a functional barrier to fission product release under existing plant conditions. This definition is less restrictive than Technical Specification criteria governing Primary and Secondary Containment operability. If the Technical Specification criteria are met, therefore, Containment Closure has been established. (ref. 3, 4, 5)
- Explosive (deflagration) mixtures in the primary containment are assumed to be elevated concentrations of hydrogen and oxygen. BWR industry evaluation of hydrogen generation for development of EOPs/SAGs indicates that any hydrogen concentration above minimum detectable is not to be expected within the short term. Post-LOCA hydrogen generation primarily caused by radiolysis is a slowly evolving, long-term condition. Hydrogen concentrations that rapidly develop are most likely caused by metal-water reaction. A metal-water reaction is indicative of an accident more severe than accidents considered in the plant design basis and would be indicative, therefore, of a potential threat to primary containment integrity. Hydrogen concentration of approximately 6% is considered the global deflagration concentration limit (ref. 6).

  The specified values for this threshold are the minimum global deflagration concentration limits (6% hydrogen and 5% oxygen, ref. 5) and readily recognizable because 6% hydrogen is well

above the EOP flowchart entry condition (ref. 8). The minimum global deflagration hydrogen/oxygen concentrations (6%/5%, respectively) require intentional primary containment venting, which is defined to be a loss of the primary containment barrier.

Atmosphere samples from a minimum of two locations inside the primary containment and one location in the suppression chamber are sequentially monitored for hydrogen and oxygen percentage levels by each of two redundant analyzer systems. The analyzers are single range (0 to 30% hydrogen and 0 to 30% oxygen). Two redundant (divisional) recorders are provided in the Main Control Room CMS-O2/H2R-1 (H13-P827) and CMS-O2/H2R-2 (H13-P811). Hydrogen and oxygen concentrations can also be displayed on the plant computers (ref. 9-12)

- Any UNPLANNED rise in PC pressure in the Cold Shutdown or Refueling mode indicates Containment Closure cannot be assured and the primary containment cannot be relied upon as a barrier to fission product release.
- RB (Reactor Building) area radiation monitors should provide indication of increased release that may be indicative of a challenge to Containment Closure. The Maximum Safe Operating levels are indicative of problems in the secondary containment that are spreading. The locations into which the primary system discharge is of concern correspond to the areas addressed in Table 24 of the EOP flowcharts (ref. 13). All Table 24 Maximum Safe Operating radiation levels can be determined in the main Control Room.

If RPV level is restored and maintained above the top of active fuel before a Containment Challenge condition occurs and subsequently a Containment Challenge condition is reached, this EAL is not met.

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

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

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

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

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

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

leakage, recover inventory control/makeup equipment and/or restore level monitoring.

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

CGS Basis Reference(s):

1. Calculation NE-02-03-05 Attachment 3 Note 8
2. PPM 5.1.1 RPV Control
3. Technical Specifications 3.6.1.1
4. Technical Specifications 3.6.4.1
5. PPM 1.20.3 Outage Risk Management
6. BWROG EPG/SAG Revision 2, Sections PC/G
7. PPM 5.7.1 RPV and Primary Containment Flooding SAG, Table 19
8. PPM 5.2.1 Primary Containment Control
9. FSAR Section 7.5.1.5.4
10. PPM 5.0.10 Flowchart Training Manual
11. PPM 4.814.J1 814.J1 Annunciator Panel Alarms, 2-2
12. PPM 4.814.J2 814.J2 Annunciator Panel Alarms, 2-2
13. PPM 5.3.1 Secondary Containment Control
14. NEI 99-01 CG1

ATTACHMENT 7.1: EAL Technical Bases

Category: C – Cold Shutdown / Refuel System Malfunction

Subcategory: 1 – RPV Level

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

EAL:

CG1.2 General Emergency

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

AND

Core uncovery is indicated by any of the following:

- UNPLANNED wetwell level rise GT 2 inches (PPM 5.2.1 entry condition)
- Valid indication of RB room flooding as identified by high level alarms (PPM 5.3.1 Table 25)
- Observation of UNISOLABLE RCS leakage outside primary containment of sufficient magnitude to indicate core uncover

AND

Any of the following indication of containment challenge:

- CONTAINMENT CLOSURE not established (Note 6)
- Explosive mixture inside PC (H_2 GE 6% and O_2 GE 5%)
- UNPLANNED rise in PC pressure
- RB area radiation GT any Maximum Safe Operating level (PPM 5.3.1 Table 24)

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

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

Mode Applicability:

			4	5	

Basis:

In this EAL, all RPV level indication is unavailable and the RPV inventory loss must be detected by the leakage indications provided. Level increases must be evaluated against other potential sources of leakage such as cooling water sources inside the drywell to ensure they are indicative of RPV leakage. Reactor Building equipment or floor drain sump level rise may be indicative of RPV inventory losses external to the primary containment from systems connected to the RPV (ref. 1, 2). With RHR System operating in the Shutdown Cooling mode, an unexplained rise in wetwell level could be indicative of RHR valve misalignment or leakage (ref. 3). If the make-up rate to the RPV unexplainably rises above the pre-established rate, a loss of RPV inventory may be occurring even if the source of the leakage cannot be immediately identified.

An UNPLANNED wetwell level increase to GT 2 inches or a VALID RB room high level alarm indicates a significant loss of RCS that could lead to core uncovery if not isolated (ref. 4, 5).

Visual observation of significant leakage from systems connected to the RCS in areas outside the primary containment that cannot be isolated could be indicative of a loss of RPV inventory sufficient to lead to core uncovery.

Four conditions are associated with a challenge to primary containment (PC) integrity:

- CONTAINMENT CLOSURE is not established.
- Explosive (deflagration) mixtures in the primary containment are assumed to be elevated concentrations of hydrogen and oxygen. BWR industry evaluation of hydrogen generation for development of EOPs/SAGs indicates that any hydrogen concentration above minimum detectable is not to be expected within the short term. Post-LOCA hydrogen generation primarily caused by radiolysis is a slowly evolving, long-term condition. Hydrogen concentrations that rapidly develop are most likely caused by metal-water reaction. A metal-water reaction is indicative of an accident more severe than accidents considered in the plant design basis and would be indicative, therefore, of a potential threat to primary containment integrity. Hydrogen concentration of approximately 6% is considered the global deflagration concentration limit (ref. 6).

  The specified values for this threshold are the minimum global deflagration concentration limits (6% hydrogen and 5% oxygen, ref. 6) and readily recognizable because 6% hydrogen is well above the EOP flowchart entry condition (ref. 8). The minimum global deflagration hydrogen/oxygen concentrations (6%/5%, respectively) require intentional primary containment venting, which is defined to be a loss of the primary containment barrier.

  Atmosphere samples from a minimum of two locations inside the primary containment and one location in the suppression chamber are sequentially monitored for hydrogen and oxygen percentage levels by each of two redundant analyzer systems. The analyzers are single range (0 to 30% hydrogen and 0 to 30% oxygen). Two redundant (divisional) recorders are provided in the Main Control Room CMS O2/H2R 1 (H13 P827) and CMS O2/H2R 2 (H13 P811). Hydrogen and oxygen concentrations can also be displayed on the plant computers (Ref. 9-12)

- Any unplanned rise in PC pressure in the Cold Shutdown or Refueling mode indicates Containment Closure cannot be assured and the primary containment cannot be relied upon as a barrier to fission product release.
- RB (Reactor Building) area radiation monitors should provide indication of increased release that may be indicative of a challenge to Containment Closure. The Maximum Safe Operating levels are indicative of problems in the secondary containment that are spreading. The locations into which the primary system discharge is of concern correspond to the areas addressed in Table 24 of the EOP flowcharts (ref.13).

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

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

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

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

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

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

The inability to monitor RPV level may be caused by instrumentation and/or power failures, or water level dropping below the range of available instrumentation. If water level cannot be monitored, operators may determine that an inventory loss is occurring by observing changes in sump and/or tank levels. Sump and/or tank level changes must be evaluated against other potential sources of water flow to ensure they are indicative of leakage from the 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 Operations at Commercial Nuclear Power Plants in the United States*; and NUMARC 91-06, *Guidelines for Industry Actions to Assess Shutdown Management*.

CGS Basis Reference(s):

1. SOP-EDR-OPS Equipment Drain System Operation
2. SOP-FDR-OPS Floor Drain System Operation
3. SOP-RHR-SDC RHR Shutdown Cooling
4. PPM 5.2.1 Primary Containment Control
5. PPM 5.3.1 Secondary Containment Control
6. BWROG EPG/SAG Revision 2, Sections PC/G
7. PPM 5.7.1 RPV and Primary Containment Flooding SAG, Table 19
8. PPM 5.2.1 Primary Containment Control
9. FSAR Section 7.5.1.5.4
10. PPM 5.0.10 Flowchart Training Manual
11. PPM 4.814.J1 814.J1 Annunciator Panel Alarms, 2-2
12. PPM 4.814.J2 814.J2 Annunciator Panel Alarms, 2-2
13. PPM 5.3.1 Secondary Containment Control
14. NEI 99-01 CG1

ATTACHMENT 7.1: EAL Technical Bases

Category: C – Cold Shutdown / Refuel 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 2, to emergency buses SM-7 and SM-8 reduced to a single power source for GE 15 min. (Note 1)

AND

Any additional single power source failure will result in a loss of all AC power to SAFETY SYSTEMS~~Any additional single power source failure will result in loss of all AC power to emergency buses SM-7 and SM-8~~

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

Table 2 AC Power Sources
Offsite • Startup Transformer TR-S • Backup Transformer TR-B • Backfeed 500 KV power through Main Transformers (if already aligned in modes 4, 5, def **only**)
Onsite • DG1 • DG2 • Main Generator via TR-N1/N2

Mode Applicability:

			4	5	def

Basis:

Table 2 provides the list of AC power sources available to power emergency buses (ref. 1, 2).

Station Startup 230KV power comes from the Ashe substation through Startup transformer TR-S. The startup transformer usually supplies station auxiliary loads when the main generator is not available. Station Backup 115KV power from the Benton Substation feeder can be supplied to emergency buses SM-7 and SM-8 (ref. 3, 4).

Credit is not taken in this EAL for SM-4/DG3 crosstie capability because establishing the crosstie to SM-7 or SM-8 is assumed to require more than 15 minutes (5). SM-4 is not a site specific emergency AC buss source since SM-4 does not provide core cooling or containment cooling.

It is possible to remove startup power from service and continue to supply the plant during shutdown conditions by backfeeding 500 KV power from Ashe Substation through the Main Transformers, the Normal Transformers and associated "N" breakers. This involves disconnecting the Main Generator from the Isolated Phase conductors (25 KV system) and overriding various interlocks. This action would take significantly longer than 15 minutes; therefore, backfeed must be in service to credit this source (ref. 7).

The second threshold statement in this EAL does not describe a separate condition, it is clarifying the first threshold statement.

This cold condition EAL is equivalent to the hot condition EAL MA1.1.

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, Refuel, 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 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 one division of emergency power sources (e.g., onsite diesel generators).
- 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 division of emergency buses being back-fed from an offsite power source.

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

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

CGS Basis Reference(s):

1. FSAR Figure 8.1-2.1 Main One-Line Diagram - Main Buses
2. FSAR Figure 8.1-2.2 Main One-Line Diagram - Emergency Buses
3. FSAR Section 8.2
4. OI-53 Offsite Power
5. FSAR Section 8.3
6. ABN-ELEC-LOOP Loss Of All Off-Site Electrical Power
7. SOP-ELECT-BACKFEED
8. NEI 99-01 CU2

Category: C – Cold Shutdown / Refuel 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 buses SM-7 and SM-8 for GE 15 min. (Note 1)

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

Mode Applicability:

			4	5	def

Basis:

~~Table 2 provides the list of AC power sources available to power emergency buses. (ref. 1, 2)~~

~~Station Startup 230KV power comes from the Ashe substation through Startup transformer TR-S. The startup transformer usually supplies station auxiliary loads when the main generator is not available. Station Backup 115KV power from the Benton Substation feeder can be supplied to emergency buses SM-7 and SM-8 (ref. 3, 4).~~

~~It is possible to remove startup power from service and continue to supply the plant during shutdown conditions by backfeeding 500 KV power from Ashe Substation through the Main Transformers, the Normal Transformers and associated "N" breakers. This involves disconnecting the Main Generator from the Isolated Phase conductors (25 KV system) and overriding various interlocks. This action would take significantly longer than 15 minutes; therefore, backfeed must be in service to credit this source (ref 7).~~

This cold condition EAL is equivalent to the hot condition EAL MS1.1.

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, Refuel, 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.

CGS Basis Reference(s):

1. FSAR Figure 8.1-2.1 Main One-Line Diagram - Main Buses
2. FSAR Figure 8.1-2.2 Main One-Line Diagram - Emergency Buses
3. FSAR Section 8.2

ATTACHMENT 7.1: EAL Technical Bases

4. OI-53 Offsite Power
5. FSAR Section 8.3
6. ABN-ELEC-LOOP Loss Of All Off-Site Electrical Power
7. SOP-ELECT-BACKFEED
8. NEI 99-01 CA2

ATTACHMENT 7.1: EAL Technical Bases

Category: C – Cold Shutdown / Refuel System Malfunction

Subcategory: 3 – RCS Temperature

Initiating Condition: UNPLANNED increase in RCS temperature

EAL:

CU3.1 Unusual Event

UNPLANNED increase in RCS temperature to GT 200°F

Mode Applicability:

			4	5	

Basis:

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

The Technical Specification cold shutdown temperature limit is 200°F (ref. 1).

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

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

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

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

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

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

CGS Basis Reference(s):

1. Technical Specifications Table 1.1-1
2. NEI 99-01 CU3

ATTACHMENT 7.1: EAL Technical Bases

Category: C – Cold Shutdown / Refuel System Malfunction

Subcategory: 3 – RCS Temperature

Initiating Condition: UNPLANNED increase in RCS temperature

EAL:

CU3.2 Unusual Event

Loss of all RCS temperature and RPV water level indication for GE 15 min. (Note 1)

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

Mode Applicability:

			4	5	

Basis:

Recirculation suction temperature, RRC-TR-650 pt 1(2), is the primary temperature measurement instrument when RPV pressure is less than 100 psig and the associated RRC pump is operating.

Monitoring of the RWCU bottom head drain temperature element, RWCU-TE-21, as read on RWCU-TI-607 pt 5 (H13 P602) or MS-TR-6 pt 316 (RB 522) is acceptable only if a RRC pump is operating for forced flow and RWCU flow of greater than 50 gpm exists. (ref. 4)

With flow through the RHR Heat Exchanger, the inlet temperature (TDAS pt. X045) is indicative of RRC system temperature. If adequate core flow cannot be provided, RPV metal temperature can be monitored on MS-TR-6. (ref. 5)

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

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

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.

CGS Basis Reference(s):

1. FSAR Table 7.5-1
2. FSAR Figure 7.7-1
3. FSAR Section 7.6.1.3
4. OSP-RCS-C102 RPV Non-Critical Cooldown Surveillance
5. SOP-RHR-SDC RHR Shutdown Cooling
6. NEI 99-01 CU3

Category: C – Cold Shutdown / Refuel System Malfunction

Subcategory: 3 – RCS Temperature

Initiating Condition: Inability to maintain the plant in cold shutdown

EAL:

CA3.1 Alert

UNPLANNED increase in RCS temperature to GT 200°F for GT Table 7 duration (Note 1)

OR

UNPLANNED RPV pressure increase GT 10 psig

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

Table 7 RCS Heat-up Duration Thresholds

RCS Status	**CONTAINMENT CLOSURE Status**	**Heat-up Duration**
Intact	N/A	60 min.*
Not intact	established	20 min.*
	not established	0 min.
* If an RCS heat removal system is in operation within this time frame and RCS temperature is being reduced, the EAL is **not** applicable.		

Mode Applicability:

			4	5	

Basis:

200°F is the Technical Specification cold shutdown temperature limit (ref. 1).

10 psi is one-half of the 20 psi minor division on the Wide Range RPV pressure instrument, RFW-PI-605, on Main Control Room Panel H13- P603 (ref. 2). This instrument has a range of 0 to 1200 psig. This RPV pressure indication is also displayed on plant computer point B016 (ref. 3).

Recirculation suction temperature, RRC TR 650 pt 1(2), is the primary temperature measurement instrument when RPV pressure is less than 100 psig and the associated RRC pump is operating.

Monitoring of the RWCU bottom head drain temperature element, RWCU TE 21, as read on RWCU TI 607 pt 5 (H13 P602) or MS TR 6 pt 316 (RB 522) is acceptable only if a RRC pump is operating for forced flow and RWCU flow of greater than 50 gpm exists. (ref. 4)

With flow through the RHR Heat Exchanger, the inlet temperature (TDAS pt. X045) is indicative of RRC system temperature. If adequate core flow cannot be provided, RPV metal temperature can be monitored on MS TR 6. (ref. 5)

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).

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.. 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 , 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.

CGS Basis Reference(s):

1. Technical Specifications Table 1.1-1
2. Instrument Master Datasheet for EPN RFW-PI-605
3. PPM 10.27.36 Reactor Pressure High Alarm – CC
4. OSP-RCS-C102 RPV Non-Critical Cooldown Surveillance
5. SOP-RHR-SDC RHR Shutdown Cooling
6. NEI 99-01 CA3

ATTACHMENT 7.1: EAL Technical Bases

Category: C – Cold Shutdown / Refuel 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 LT 108 VDC on required 125 VDC buses DP-S1-1 and DP-S1-2 for GE 15 min. (Note 1)

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

Mode Applicability:

			4	5	

Basis:

The 125 VDC Class 1E DC power system consists of three electrically independent and separate distribution systems (S1-1, S1-2, and S1-HPCS). S1-HPCS is not included in this EAL. Each DC distribution system has a battery and a battery charger that are normally connected to the bus such that these two sources of power are operating in parallel. The charger is normally supplying system electrical loads with the battery on a float charge. Each battery has the necessary amp-hour discharge capacity to sustain system loads for a minimum of two hours. This capacity is specifically for a loss of power to the charger coincident with a design basis accident. The batteries have capacity to carry design load at 60°F without decreasing battery voltage below 1.81 volts/cell (or 108 VDC, ref. 1) with loss of output from the battery chargers during the specified period. Battery capacity is sufficient to provide starting currents while operating at full load. (ref. 2)

This EAL is the cold condition equivalent of the hot condition loss of DC power EAL MS2.1.

This IC addresses a loss of essential DC power which compromises the ability to monitor and control operable SAFETY SYSTEMS when the plant is in the cold shutdown or Refuel 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 essential DC buses necessary to support operation of the in-service, or operable, train or trains of SAFETY SYSTEM equipment. For example, if Division I is out-of-service (inoperable) for scheduled outage maintenance work and Division II is in-service (operable), then a loss of essential DC power affecting Division II would require the declaration of an Unusual Event. A loss of essential DC power to Division I 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.

CGS Basis Reference(s):

1. Calculation No. 2.05.01 Battery Sizing, Voltage Drop, and Charger Studies for Div. 1 & 2 Systems
2. FSAR Section 8.3.2
3. NEI 99-01 CU4

ATTACHMENT 7.1: EAL Technical Bases

Category: C – Cold Shutdown / Refuel System Malfunction

Subcategory: 5 – Loss of Communications

Initiating Condition: Loss of all onsite or offsite communications capabilities

EAL:

CU5.1 Unusual Event

(1) Loss of all Table 4 onsite communication methods

OR

(2) Loss of all Table 4 ORO communication methods

OR

(3) Loss of all Table 4 NRC communication methods

Table 4 Communication Methods

System	Onsite	ORO	NRC
Plant Public Address (PA) System	X		
Plant Telephone System	X	X	
Plant Radio System Operations and Security Channels	X		
Offsite calling capability from the Control Room via direct telephone		X	X
Long distance calling capability on the commercial phone system		X	X

Mode Applicability:

			4	5	def

Basis:

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

Public Address (PA) System

The public address system provides a way of contacting personnel in the various buildings of the plant and locations of the site that might be inaccessible using other means of communication. The building-wide alarm system alerts (via the public address system speakers) operating personnel to fire hazards and other trouble conditions for which plant management finds it necessary to alert plant personnel.

Plant Telephone System

This system consists of interconnections to the public telephone network (and trunks to the PBX) with individual direct lines that provide inward and outward dialing access to most plant locations.

ATTACHMENT 7.1: EAL Technical Bases

Plant Radio System Operations and Security Channels

The radio communications system is used for communications with personnel involved in maintenance and security in and around the plant complex by means of hand-held portable radio units, mobile radio units, and paging receivers. The telephone link to BPA provides a direct communication link to the BPA Dittmer Control Center. The radio communications system provides a communications link for security and emergency communications to local law enforcement agencies and emergency control centers.

Offsite calling capability from the Control Room via direct telephone and fax lines

This communications method includes following dedicated phone networks that are available for emergency communications in addition to the normal Energy Northwest phone network:

- Energy Northwest Emergency Center Network
- Response Agency Network
- NRC Emergency Notification System

Various locations such as the Control Room, Technical Support Center, Emergency Operations Facility, Joint Information Center, Department of Energy-RL, Washington State Emergency Operations Center, Oregon State Emergency Coordination Center and the Benton and Franklin County Emergency Operations Centers have facsimile transceivers. The facsimile transceivers enable the transmission and receipt of printed material. The facsimile system which connects the Energy Northwest emergency centers with the county and state emergency centers uses dedicated phone lines.

Long distance calling capability on the commercial phone system

The Energy Northwest Richland phone system is a computer based, software controlled telephone exchange (Computerized Branch Exchange). It is equipped with redundant computerized processor units and is served by an uninterruptible power supply. The direct-dial private telephone system provides communication between the Energy Northwest facilities. The phone system is arranged such that plant telephones can reach other Energy Northwest facilities by direct-dialing and without the need of an operator.

This EAL is the cold condition equivalent of the hot condition EAL MU7.1.

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 Washington Stare, Benton County, Franklin County and DOE RL.

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

CGS Basis Reference(s):

1. Emergency Plan Section 6.6
2. FSAR Section 9.5.2
3. NEI 99-01 CU5

ATTACHMENT 7.1: EAL Technical Bases

Category: C – Cold Shutdown / Refuel 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 8 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

OR

The event has caused VISIBLE DAMAGE to a SAFETY SYSTEM component or structure, Table 5, needed for the current operating mode (Note 9)

Note 9: If the affected SAFETY SYSTEM (or component) was already inoperable or out of service before the event occurred, then **no** emergency classification is warranted as long as the damage was limited to the affected SAFETY SYSTEM (or component).

Table 8 Hazardous Events
• Seismic event • Internal or external FLOODING event • High winds • Tornado strike • FIRE • EXPLOSION • Volcanic ash fallout • Other events with similar hazard characteristics as determined by the Shift Manager

Table 5 Safe Shutdown Areas
• Vital portions of the Rad Waste/Control Building: - 467' elevation vital island - 487' elevation cable spreading room - Main Control Room and vertical cable chase - 525' elevation HVAC area • Reactor Building • Vital portions of the Turbine Building - DEH pressure switches - RPS switches on turbine throttle valves - Main steam line radiation monitors - Turbine Building ventilation radiation monitors - Main steam line piping up to MS-V-146 and the first stop valves • Standby Service Water Pump Houses • Diesel Generator Building

Mode Applicability:

			4	5	

Basis:

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

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

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

~~An emergency classification is required if a FIRE or EXPLOSION caused by an equipment failure damages safety system equipment that was otherwise functional or operable (i.e., equipment that was not the source/location of the failure). For example, if a FIRE or EXPLOSION resulting from the failure of a piece of safety system equipment causes damage to the other train of the affected safety system or another safety system, then an emergency declaration is required in accordance with this IC and EAL.~~

The significance of a seismic event is discussed under EAL HU2.1 (ref. 1).

Internal FLOODING may be caused by events such as component failures, equipment misalignment, or outage activity mishaps (ref. 2).

Seismic Category I structures are analyzed to withstand a sustained, design wind velocity of at least 100 mph (ref. 3).

Areas containing functions and systems required for safe shutdown of the plant are identified by Fire Areas in the fire response procedure (ref. 4).

The potential for volcanic eruption exists in the Pacific Northwest. Heavy ash fall, such as that experienced at certain locations following the eruption of Mt. St. Helens in 1980, could affect operation of plant equipment if precautionary measures are not taken. The design basis ash fall is projected for a twenty hour duration. (ref. 5)

Table 5 provides a list of CGS safety system areas (ref. 6).

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

CGS Basis Reference(s):

1. FSAR Section 3.7 Seismic Design
2. FSAR Section 3.4.1 Flood Protection
3. CGS Calculation CALC CE-02-93-16 Evaluate PMR/BDC 98-0131-0A change from 5 min. to 15 min. averaging of 33 ft. elev. met twr. wind speeds for UE and Alert declarations
4. ABN-FIRE Attachment 13.2, Fire Areas
5. ABN-ASH Ash Fall
6. FSAR Table 3.2-1 Equipment Classification
7. NEI 99-01 CA6

Category H – Hazards and Other Conditions Affecting Plant Safety

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

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

1. Security

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

2. Seismic Event

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

3. Natural or Technology Hazard

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

4. Fire

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

5. 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

   If the Control Room must be evacuated, additional support for monitoring and controlling plant functions is necessary through the emergency response facilities.

7. Emergency Director Judgment

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

ATTACHMENT 7.1: EAL Technical Bases

Category: H – Hazards

Subcategory: 1 – Security

Initiating Condition: Confirmed SECURITY CONDITION or threat

EAL:

HU1.1 Unusual Event

(1) A SECURITY CONDITION that does not involve a HOSTILE ACTION as reported by the Security Sergeant or Security Lieutenant

OR

(2) Notification of a credible security threat directed at the site

OR

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

Mode Applicability:

1	2	3	4	5	def

Basis:

The Security Shift Supervision is defined as either the Security Lieutenant or the Security Sergeant (ref. 1).

This EAL is based on the CGS Physical Security Plan (ref. 1).

The Safeguards Contingency Plan (Appendix C of CGS Physical Security Plan) defines the events that meet the criteria of a SECURITY CONDITION or HOSTILE ACTION (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 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]*.

Threshold #1 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.

Threshold #2 addresses the receipt of a credible security threat. The credibility of the threat is assessed in accordance with the CGS Physical Security Plan (ref. 1).

Threshold #3 addresses the threat from the impact of an aircraft on the plant. The NRC Headquarters Operations Officer (HOO) will communicate to the licensee if the threat involves an aircraft. The status and size of the plane may also be provided by NORAD through the NRC. Validation of the threat is performed in accordance with ABN-AIRBORNE-ATTACK (ref. 2).

Emergency plans and implementing procedures are public documents; therefore, EALs should not

incorporate Security-sensitive information. This includes information that may be advantageous to a potential adversary, such as the particulars concerning a specific threat or threat location. Security-sensitive information should be contained in non-public documents such as the CGS Physical Security Plan (ref. 1).

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

CGS Basis Reference(s):

1. CGS Physical Security Plan
2. ABN-AIRBORNE-ATTACK
2. NEI 99-01 HU1

ATTACHMENT 7.1: EAL Technical Bases

Category: H – Hazards

Subcategory: 1 – Security

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

EAL:

HA1.1 Alert

(1) A HOSTILE ACTION is occurring or has occurred within the OWNER CONTROLLED AREA as reported by the Security Sergeant or Security Lieutenant

OR

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

Mode Applicability:

1	2	3	4	5	def

Basis:

The Security Shift Supervision is defined as either the Security Lieutenant or the Security Sergeant (ref. 1).

Note that the ISFSI Protected Area is an area separate from the Protected Area surrounding the power block.

The Safeguards Contingency Plan (Appendix C of CGS Physical Security Plan) defines the events that meet the criteria of a SECURITY CONDITION or HOSTILE ACTION (ref. 1).

This IC addresses the occurrence of a HOSTILE ACTION within the OWNER CONTROLLED AREA or notification of an aircraft attack threat. This event will require rapid response and assistance due to the possibility of the attack progressing to the PROTECTED AREA, or the need to prepare the plant and staff for a potential aircraft impact.

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

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.

Threshold #1 is applicable for any HOSTILE ACTION occurring, or that has occurred, in the OWNER CONTROLLED AREA. This includes any action directed against the ISFSI which is located outside the plant PROTECTED AREA.

ATTACHMENT 7.1: EAL Technical Bases

Threshold #2 addresses the threat from the impact of an aircraft on the plant, and the anticipated arrival time is within 30 minutes. The intent of this EAL is to ensure that threat-related notifications are made in a timely manner so that plant personnel and OROs are in a heightened state of readiness. This EAL is met when the threat-related information has been validated in accordance with ABN-AIRBORNE-ATTACK (ref 2) s.

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

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

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

CGS Basis Reference(s):

1. CGS Physical Security Plan
2. ABN-AIRBORNE-ATTACK
2. NEI 99-01 HA1

ATTACHMENT 7.1: EAL Technical Bases

Category: H – Hazards

Subcategory: 1 – Security

Initiating Condition: HOSTILE ACTION within the Protected Area

EAL:

HS1.1 Site Area Emergency

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

Mode Applicability:

1	2	3	4	5	def

Basis:

The Security Shift Supervision is defined as either the Security Lieutenant or the Security Sergeant (ref. 1).

Note that the ISFSI Protected Area is an area separate from the Protected Area surrounding the power block.

The Safeguards Contingency Plan (Appendix C of CGS Physical Security Plan) defines the events that meet the criteria of a SECURITY CONDITION or HOSTILE ACTION (ref. 1).

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

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

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

As time and conditions allow, these events require a heightened state of readiness by the plant staff and implementation of onsite protective measures (e.g., evacuation, dispersal or sheltering). The 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 CGS Physical Security Plan (ref. 1).

CGS Basis Reference(s):

1. CGS Physical Security Plan

ATTACHMENT 7.1: EAL Technical Bases

2. NEI 99-01 HS1

ATTACHMENT 7.1: EAL Technical Bases

Category: H – Hazards and Other Conditions Affecting Plant Safety

Subcategory: 2 – Seismic Event

Initiating Condition: Seismic event GT OBE levels

EAL:

HU2.1 Unusual Event

Seismic event GT Operating Basis Earthquake (OBE) as indicated by H13.P851.S1.5-1 (OPERATING BASIS EARTHQUAKE EXCEEDED) activated

Mode Applicability:

1	2	3	4	5	def

Basis:

CGS seismic instrumentation consists of a Kinemetrics SMA-3 Strong Motion Accelerograph and associated sensors that are equipped with seismic triggers set to initiate recording at an acceleration equal to or exceeding 0.01 g (ref. 1, 2). This also annunciates the seismic activity alarm H13.P851.S1.2-5 Minimum Seismic Earthquake Exceeded (ref. 2, 3, 4).

A seismic switch unit that is similar to the seismic trigger unit is also provided. The trip point of the seismic switch unit is set at the maximum acceleration corresponding to the OBE, and it provides immediate Control Room annunciation that the OBE has been exceeded requiring declaration of an Unusual Event (ref. 1, 3, 4)

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

Event verification with external sources should not be necessary during or following an OBE. Earthquakes of this magnitude should be readily felt by on-site personnel and recognized as a seismic event (e.g., lateral accelerations in excess of 0.08g). The Shift Manager or Emergency 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 ~~SA8~~MA8.

CGS Basis Reference(s):

1. CGS FSAR Section 3.7.4 Seismic Instrumentation
2. ISP-SEIS-M201 Seismic Systems Channel Check
3. PPM 4.851.S1.2-5 Minimum Seismic Earthquake Exceeded
4. ABN-EARTHQUAKE Earthquake
5. NEI 99-01 HU2

ATTACHMENT 7.1: EAL Technical Bases

Category: H – Hazards and Other Conditions Affecting Plant Safety

Subcategory: 3 – Natural or Technology Hazard

Initiating Condition: Hazardous event

EAL:

HU3.1 Unusual Event

(1) A tornado strike within the PROTECTED AREA

OR

(2) Volcanic ash fallout requiring plant shutdown

Mode Applicability:

1	2	3	4	5	def

Basis:

If damage is confirmed visually or by other in-plant indications, the event may be escalated to an Alert under EAL CA6.1 or MA8.1.

Threshold #1

A tornado striking (touching down) within the PROTECTED AREA warrants declaration of an Unusual Event regardless of the measured wind speed at the meteorological tower. A tornado is defined as a violently rotating column of air in contact with the ground and extending from the base of a thunderstorm. A dust devil is not a tornado.

Threshold #2

The potential for volcanic eruption exists in the Pacific Northwest. Heavy ash fall, such as that experienced at certain locations following the eruption of Mt. St. Helens in 1980, could affect operation of plant equipment if precautionary measures are not taken. The design basis ash fall is projected for a 20 hour duration. Plant shutdown may be warranted, based on several individual criteria specified in ABN-ASH (ref. 1). This threshold is met when ABN-ASH requires plant shutdown.

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

Threshold #1 addresses a tornado striking (touching down) within the PROTECTED AREA.

Threshold #2 addresses a volcanic ash fallout event.

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

CGS Basis Reference(s):

1. ABN-ASH Ash Fall
2. NEI 99-01 HU3

ATTACHMENT 7.1: EAL Technical Bases

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:

1	2	3	4	5	def

Basis:

An uncontrolled flooding event may pose a direct threat to safety-related equipment. As such, the potential exists for substantial degradation of the level of safety of the plant. One indication of FLOODING is indicated by ECCS room level alarms on P601 (ref. 1, 2).

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, M or C.

CGS Basis Reference(s):

1. Calculation ME 02-02-02 Reactor Building Flooding
2. Calculation ME 02-02-46, RB/RW/TB/DG Corridor Flooding
3. 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

(1) Movement of personnel within the PROTECTED AREA is IMPEDED due to an offsite event involving hazardous materials (e.g., an offsite chemical spill, 618-11 event or toxic gas release)

OR

(2) 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:

1	2	3	4	5	def

Basis:

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

Threshold #1 includes an event at the 618-11 burial ground which would IMPEDE movement of personnel within the PROTECTED AREA.

Threshold #2 includes a range fire causing Hanford officials to limit vehicle access to the site. The origin of the hazardous event could be from on or off-site.

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

Threshold #1 addresses a hazardous materials event originating at an offsite location and of sufficient magnitude to impede the movement of personnel within the PROTECTED AREA.

Threshold #2 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, M or C.

CGS 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 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 5 area

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

Table 5 Safe Shutdown Areas
• Vital portions of the Rad Waste/Control Building: - 467' elevation vital island - 487' elevation cable spreading room - Main Control Room and vertical cable chase - 525' elevation HVAC area • Reactor Building • Vital portions of the Turbine Building - DEH pressure switches - RPS switches on turbine throttle valves - Main steam line radiation monitors - Turbine Building ventilation radiation monitors - Main steam line piping up to MS-V-146 and the first stop valves • Standby Service Water Pump Houses • Diesel Generator Building

Mode Applicability:

1	2	3	4	5	def

Basis:

A fire alarm can be confirmed by multiple/redundant indications such as additional alarms on FCP-1 or FCP-2, fire pumps starting, fire suppression system discharge, fire water header pressure fluctuations or by notification by plant personnel (ref. 1).

The Table 5 Safe Shutdown Areas include those structures/areas that contain any Class 1, 2 or 3 SSC. Table 5 includes those structures containing functions and systems required to achieve and maintain cold shutdown (including all auxiliary equipment such as AC/DC power, cooling water and instrumentation) (ref. 2).

The concept of this EAL is that a fire exists in a Table 5 area that is not extinguished within 15 minutes.

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 alarms, indications, or report was received, and not the time that a subsequent verification action was performed. Similarly, the fire duration clock also starts at the time of receipt of the initial alarms, indications 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 MA8.

CGS Basis Reference(s):

1. ABN-FIRE
2. FSAR Table 3.2-1 Equipment Classification
3. NEI 99-01 HU4

ATTACHMENT 7.1: EAL Technical Bases

Category: H – Hazards and Other Conditions Affecting Plant Safety

Subcategory: 4 – Fire

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

EAL:

HU4.2 Unusual Event

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

AND

The fire alarm is indicating a FIRE within any Table 5 area

AND

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

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

Table 5 Safe Shutdown Areas

- Vital portions of the Rad Waste/Control Building:
  - 467' elevation vital island
  - 487' elevation cable spreading room
  - Main Control Room and vertical cable chase
  - 525' elevation HVAC area
- Reactor Building
- Vital portions of the Turbine Building
  - DEH pressure switches
  - RPS switches on turbine throttle valves
  - Main steam line radiation monitors
  - Turbine Building ventilation radiation monitors
  - Main steam line piping up to MS-V-146 and the first stop valves
- Standby Service Water Pump Houses
- Diesel Generator Building

Mode Applicability:

1	2	3	4	5	def

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.

A single point fire alarm, with no other indications of a fire, may be more indicative of an instrumentation issue rather than a fire in the plant.

The concept of this EAL is that there is 30 minutes to determine if a fire exists when only one fire alarm is received.

The Table 5 Safe Shutdown Areas include those structures/areas that contain any Class 1, 2 or 3 SSC. Table 5 includes those structures containing functions and systems required to achieve and maintain cold shutdown (including all auxiliary equipment such as AC/DC power, cooling water and instrumentation) (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.

Basis-Related Requirements from Appendix R

Appendix R to 10 CFR 50, states in part:

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

maintain safe shutdown conditions is greater than the need to limit fire damage to those systems required to mitigate the consequences of design basis accidents.

In addition, Appendix R to 10 CFR 50, requires, among other considerations, the use of 1-hour fire barriers for the enclosure of cable and equipment and associated non-safety circuits of one redundant train (G.2.c). As used in this EAL, 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 MA8.

CGS Basis Reference(s):

1. FSAR Table 3.2-1 Equipment Classification
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

(1) A FIRE within the ISFSI or plant PROTECTED AREA not extinguished within 60 min. of the initial report, alarm or indication (Note 1)

OR

(2) A FIRE within the ISFSI or plant PROTECTED AREA that requires firefighting support by an offsite fire response agency to extinguish

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

Mode Applicability:

1	2	3	4	5	def

Basis:

These thresholds reflect the potential issues that can arise from a fire in other areas of the plant for greater than one-hour or a fire requiring offsite fire department to actively support firefighting efforts because the fire is beyond the capability of the Fire Brigade to extinguish.

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.

Threshold #1

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 PROTECTED AREA of an ISFSI located outside the plant PROTECTED AREA.

Threshold #2

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 Fire Brigade to extinguish. Declaration is not necessary if the agency resources are placed on stand-by, or supporting post-extinguishment recovery or investigation actions.

Depending upon the plant mode at the time of the event, escalation of the emergency classification level would be via IC CA6 or MA8.

CGS 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 9 rooms or areas

AND

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

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

Table 9 Safe Operation & Shutdown Areas	
Room/Area	**Mode Applicability**
RW 467' Radwaste Control Room (RHR flush to RW tanks)	3
RW 467' Vital Island (RHR-V-9 disconnect)	3
RB 422' B RHR Pump Rm (local pump temperatures)	3
RB 454' B RHR Pump Rm (operate RHR-V-85B)	3

Mode Applicability:

1	2	3	4	5	def

Basis:

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

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 Director's judgment that the gas concentration in the affected room/area is sufficient to preclude or

significantly impede procedurally required access. This judgment may be based on a variety of factors including an existing job hazard analysis, report of ill effects on personnel, advice from a subject matter expert or operating experience with the same or similar hazards. Access should be considered as impeded if extraordinary measures are necessary to facilitate entry of personnel into the affected room/area (e.g., requiring use of protective equipment, such as SCBAs, that is not routinely employed).

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

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

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

CGS Basis Reference(s):

1. Attachment 7.4 Safe Operation & Shutdown Areas Table 9 Bases
2. NEI 99-01 HA5

ATTACHMENT 7.1: EAL Technical Bases

Category: H – Hazards and Other Conditions Affecting Plant Safety

Subcategory: 6 – Control Room Evacuation

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

EAL:

HA6.1 Alert

An event has resulted in plant control being transferred from the Control Room to the Remote Shutdown Panel or Alternate Remote Shutdown Panel

Mode Applicability:

1	2	3	4	5	def

Basis:

The Shift Manager (SM) determines if the Control Room is inoperable and requires evacuation. This determination can depend on a number of factors, including Control Room habitability, loss of safe shutdown control circuity, or a Security event (ref. 1). For the purpose of this EAL the 15 minute classification clock starts when the last licensed operator leaves the Control Room.

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.

CGS Basis Reference(s):

1. ABN-CR-EVAC Control Room evacuation and Remote Cooldown
2. NEI 99-01 HA6

ATTACHMENT 7.1: EAL Technical Bases

Category: H – Hazards and Other Conditions Affecting Plant Safety

Subcategory: 6 – Control Room Evacuation

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

EAL:

HS6.1 Site Area Emergency

An event has resulted in plant control being transferred from the Control Room to the Remote Shutdown Panel or Alternate Remote Shutdown Panel

AND

Control of any of the following key safety functions is not reestablished within 15 min. (Note 1):

- Reactivity (Modes 1 and 2 **only**)
- RPV water level
- RCS heat removal

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

Mode Applicability:

1	2	3	4	5	

Basis:

The Shift Manager determines if the Control Room is inoperable and requires evacuation. This determination can depend on a number of factors, including Control Room habitability, loss of safe shutdown control circuity, or a Security event (ref. 1).

For the purpose of this EAL the 15 minute classification clock starts when the last licensed operator leaves the Control Room.

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

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

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

CGS Basis Reference(s):

1. ABN-CR-EVAC Control Room evacuation and Remote Cooldown
2. NEI 99-01 HS6

ATTACHMENT 7.1: EAL Technical Bases

Category: H – Hazards and Other Conditions Affecting Plant Safety

Subcategory: 7 – Emergency Director Judgment

Initiating Condition: Other conditions existing which in the judgment of the Emergency Director warrant declaration of a UE

EAL:

HU7.1 Unusual Event

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

Mode Applicability:

1	2	3	4	5	def

Basis:

The Emergency Director is the designated onsite individual having the responsibility and authority for implementing the Emergency Response Plan. The Shift Manager (SM) initially acts in the capacity of the Emergency Director and takes actions as outlined in the Emergency Plan implementing procedures. If required by the emergency classification or if deemed appropriate by the Emergency Director, 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 Director to fall under the emergency classification level description for an Unusual Event.

CGS Basis Reference(s):

1. NEI 99-01 HU7

ATTACHMENT 7.1: EAL Technical Bases

Category: H – Hazards and Other Conditions Affecting Plant Safety

Subcategory: 7 – Emergency Director Judgment

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

EAL:

HA7.1 Alert

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

Mode Applicability:

1	2	3	4	5	def

Basis:

The Emergency Director is the designated onsite individual having the responsibility and authority for implementing the Emergency Response Plan. The Shift Manager (SM) initially acts in the capacity of the Emergency Director and takes actions as outlined in the Emergency Plan implementing procedures. If required by the emergency classification or if deemed appropriate by the Emergency Director, 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 Director to fall under the emergency classification level description for an Alert.

CGS Basis Reference(s):

1. NEI 99-01 HA7

ATTACHMENT 7.1: EAL Technical Bases

Category: H – Hazards and Other Conditions Affecting Plant Safety

Subcategory: 7 – Emergency Director Judgment

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

EAL:

HS7.1 Site Area Emergency

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

Mode Applicability:

1	2	3	4	5	def

Basis:

The Emergency Director is the designated onsite individual having the responsibility and authority for implementing the Emergency Response Plan. The Shift Manager (SM) initially acts in the capacity of the Emergency Director and takes actions as outlined in the Emergency Plan implementing procedures. If required by the emergency classification or if deemed appropriate by the Emergency Director, 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 Director to fall under the emergency classification level description for a Site Area Emergency.

CGS Basis Reference(s):

1. NEI 99-01 HS7

ATTACHMENT 7.1: EAL Technical Bases

Category: H – Hazards and Other Conditions Affecting Plant Safety

Subcategory: 7 – Emergency Director Judgment

Initiating Condition: Other conditions exist which in the judgment of the Emergency Director warrant declaration of a General Emergency

EAL:

HG7.1 General Emergency

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

Mode Applicability:

1	2	3	4	5	def

Basis:

The Emergency Director is the designated onsite individual having the responsibility and authority for implementing the Emergency Response Plan. The Shift Manager(SM) initially acts in the capacity of the Emergency Director and takes actions as outlined in the Emergency Plan implementing procedures. If required by the emergency classification or if deemed appropriate by the Emergency Director, 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 Director to fall under the emergency classification level description for a General Emergency.

CGS Basis Reference(s):

1. NEI 99-01 HG7

ATTACHMENT 7.1: EAL Technical Bases

Category M – System Malfunction

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

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

The events of this category pertain to the following subcategories:

1. Loss of Emergency AC Power

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

2. Loss of vital DC Power

Loss of vital 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 vital 125 VDC buses.

3. Loss of Control Room Indications

Certain events that degrade plant operator ability to effectively assess plant conditions within the plant warrant emergency classification. Losses of indicators are in this subcategory.

4. RCS Activity

During normal operation, reactor coolant fission product activity is very low. Small concentrations of fission products in the coolant are primarily from the fission of tramp uranium in the fuel clad or minor perforations in the clad itself. Any significant 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 pressure vessel provides a volume for the coolant that covers the reactor core. The reactor pressure vessel and associated pressure piping (reactor coolant system) together provide a barrier to limit the release of radioactive material should the reactor fuel clad integrity fail. Excessive RCS leakage greater than Technical Specification limits indicates potential pipe cracks that may propagate to an extent threatening fuel clad, RCS and Containment integrity.

6. RPS Failure

This subcategory includes events related to failure of the Reactor Protection System (RPS) to initiate and complete reactor scrams. In the plant licensing basis, postulated failures of the RPS to complete a reactor scram comprise a specific set of analyzed events referred to as Anticipated Transient Without Scram (ATWS) events. For EAL classification, however, ATWS is intended to mean any 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. Hazardous Event Affecting Safety Systems

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

ATTACHMENT 7.1: EAL Technical Bases

Category: M – 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:

MU1.1 Unusual Event

Loss of all offsite AC power capability, Table 2, to emergency buses SM-7 and SM-8 for GE 15 min. (Note 1)

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

Table 2 AC Power Sources
Offsite • Startup Transformer TR-S • Backup Transformer TR-B • Backfeed 500 KV power through Main Transformers (if already aligned in modes 4, 5, def **only**)
Onsite • DG1 • DG2 • Main Generator via TR-N1/N2

Mode Applicability:

1	2	3			

Basis:

Table 2 provides the list of AC power sources available to power emergency buses (ref. 1, 2).

Station Startup 230KV power comes from the Ashe substation through Startup transformer TR-S. The startup transformer usually supplies station auxiliary loads when the main generator is not available. Station Backup 115KV power from the Benton Substation feeder can be supplied to emergency buses SM-7 and SM-8. (ref. 3, 4)

Credit is not taken in this EAL for SM-4/DG3 crosstie capability because establishing the crosstie to SM-7 or SM-8 is assumed to require more than 15 minutes (5). SM-4 is not a site specific emergency AC buss source since SM-4 does not provide core cooling or containment cooling.

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 MA1.

CGS Basis Reference(s):

1. FSAR Figure 8.1-2.1 Main One-Line Diagram - Main Buses
2. FSAR Figure 8.1-2.2 Main One-Line Diagram - Emergency Buses
3. FSAR Section 8.2
4. OI-53 Offsite Power
5. FSAR Section 8.3
6. NEI 99-01 SU1

ATTACHMENT 7.1: EAL Technical Bases

Category: M – 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:

MA1.1 Alert

AC power capability, Table 2, to emergency buses SM-7 and SM-8 reduced to a single power source for GE 15 min. (Note 1)

AND

Any additional single power source failure will result in a loss of all AC power to SAFETY SYSTEMS~~Any additional single power source failure will result in loss of all AC power to emergency buses SM-7 and SM-8~~

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

Table 2 AC Power Sources
Offsite
• Startup Transformer TR-S • Backup Transformer TR-B • Backfeed 500 KV power through Main Transformers (if already aligned in modes 4, 5, def **only**)
Onsite
• DG1 • DG2 • Main Generator via TR-N1/N2

Mode Applicability:

1	2	3			

Basis:

Table 2 provides the list of AC power sources available to power emergency buses (ref. 1, 2).

Station Startup 230KV power comes from the Ashe substation through Startup transformer TR-S. The startup transformer usually supplies station auxiliary loads when the main generator is not available. Station Backup 115KV power from the Benton Substation feeder can be supplied to emergency buses SM-7 and SM-8 (ref. 3, 4).

Credit is not taken in this EAL for SM-4/DG3 crosstie capability because establishing the crosstie to SM-7 or SM-8 is assumed to require more than 15 minutes (5). SM-4 is not a site specific emergency AC buss source since SM-4 does not provide core cooling or containment cooling.

The second threshold statement in this EAL does not describe a separate condition, it is clarifying the first threshold statement.

This hot condition EAL is equivalent to the cold condition EAL CU2.1.

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. This IC provides an escalation path from IC MU1.

An “AC power source” is a source recognized in AOPs, 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 back-fed from an offsite power source.

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

Escalation of the emergency classification level would be via IC MS1.

CGS Basis Reference(s):

1. FSAR Figure 8.1-2.1 Main One-Line Diagram - Main Buses
2. FSAR Figure 8.1-2.2 Main One-Line Diagram - Emergency Buses
3. FSAR Section 8.2
4. OI-53 Offsite Power
5. FSAR Section 8.3
6. ABN-ELEC-LOOP
7. NEI 99-01 SA1

ATTACHMENT 7.1: EAL Technical Bases

Category: M – 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:

MS1.1 Site Area Emergency

Loss of all offsite and all onsite AC power capability to emergency buses SM-7 and SM-8 for GE 15 min. (Note 1)

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

Mode Applicability:

1	2	3			

Basis:

This hot condition EAL is equivalent to the cold condition EAL CA2.1.

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.

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 MG1.

CGS Basis Reference(s):

1. PPM 5.6.1 Station Blackout (SBO)
2. NEI 99-01 SS1

Category: M –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:

MG1.1 General Emergency

Loss of all offsite AND all onsite AC power capability to emergency buses SM-7 and SM-8

AND EITHER:

Restoration of emergency bus SM-7 or SM-8 in LT 4 hours is not likely (Note 1)

OR

RPV level cannot be restored and maintained GT -186 in.

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

Mode Applicability:

1	2	3			

Basis:

Credit may be taken in this EAL for DG 3 crosstie capability provided a reasonable expectation exists that AC power can be restored to either SM-7 or SM-8 from DG3 and SM-4 within 4 hours. (ref. 1).

Four hours is the station blackout coping time (ref. 2).

Indication of continuing core cooling degradation is manifested by the inability to restore and maintain RPV water level above the Minimum Steam Cooling Reactor Water Level (-186 in.) (ref. 3). Core submergence is the most desirable means of core cooling, however when RPV level is below TAF, the uncovered portion of the core can be cooled by less reliable means (i.e., steam cooling or spray cooling).

This IC addresses a prolonged loss of all power sources to AC 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 essential 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.

CGS Basis Reference(s):

1. FSAR Section 8.2
2. PPM 5.6.1 Station Blackout (SBO)
3. PPM 5.1.1 RPV Control
4. NEI 99-01 SG1

Category: M –System Malfunction

Subcategory: 1 – Loss of Essential AC Power

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

EAL:

MG1.2 General Emergency

Loss of all offsite AND all onsite AC power capability to emergency buses SM-7 and SM-8 for GE 15 min. (Note 1)

AND

Indicated voltage is LT 108 VDC on both 125 VDC buses DP-S1-1 and DP-S1-2 for GE 15 min. (Note 1)

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

Mode Applicability:

1	2	3			

Basis:

This EAL addresses operating experience from the March 2011 accident at Fukushima Daiichi.

The 125 VDC Class 1E DC power system consists of three electrically independent and separate distribution systems (S1-1, S1-2, and S1-HPCS) (ref. 2). S1-HPCS is not included in this EAL. Each DC distribution system has a battery and a battery charger that are normally connected to the bus such that these two sources of power are operating in parallel. The charger is normally supplying system electrical loads with the battery on a float charge. Each battery has the necessary amp-hour discharge capacity to sustain system loads for a minimum of two hours. This capacity is specifically for a loss of power to the charger coincident with a design basis accident. The batteries have capacity to carry design load at 60°F without decreasing battery voltage below 1.81 volts/cell (or 108 VDC, ref. 3) with loss of output from the battery chargers during the specified period. Battery capacity is sufficient to provide starting currents while operating at full load. (ref. 1, 3).

~~PPM 5.6.1 Station Blackout, directs use of DG3, DG4 or DG5 to power vital DC battery chargers. If this is already performed, this EAL would not apply (ref. 4).~~

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.

CGS Basis Reference(s):

1. FSAR Section 8
2. E505 DC One Line Diagram
3. Calculation No. 2.05.01 Battery Sizing, Voltage Drop, and Charger Studies for Div. 1 & 2 Systems

4. PPM 5.6.1 Station Blackout (SBO)
5. NEI 99-01 SG8

Category: M – System Malfunction

Subcategory: 2 – Loss of Vital DC Power

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

EAL:

MS2.1 Site Area Emergency

Indicated voltage is LT 108 VDC on both 125 VDC buses DP-S1-1 and DP-S1-2 for GE 15 min. (Note 1)

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

Mode Applicability:

1	2	3			

Basis:

The 125 VDC Class 1E DC power system (ref. 1) consists of three electrically independent and separate distribution systems (S1-1, S1-2, and S1-HPCS) (ref. 2). S1-HPCS is not included in this EAL. Each DC distribution system has a battery and a battery charger that are normally connected to the bus such that these two sources of power are operating in parallel. The charger is normally supplying system electrical loads with the battery on a float charge. Each battery has the necessary amp-hour discharge capacity to sustain system loads for a minimum of two hours. This capacity is specifically for a loss of power to the charger coincident with a design basis accident. The batteries have capacity to carry design load at 60°F without decreasing battery voltage below 1.81 volts/cell (or 108 VDC, ref. 2) with loss of output from the battery chargers during the specified period. Battery capacity is sufficient to provide starting currents while operating at full load. (ref. 3).

This EAL is the hot condition equivalent of the cold condition loss of DC power EAL CU4.1.

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

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

Escalation of the emergency classification level would be via ICs RG1, FG1 or MG1.

CGS Basis Reference(s):

1. E505 DC One Line Diagram
2. Calculation No. 2.05.01 Battery Sizing, Voltage Drop, and Charger Studies for Div. 1 & 2 Systems
3. FSAR Section 8.3.2
4. NEI 99-01 SS8

Category: M – System Malfunction

Subcategory: 3 – Loss of Control Room Indications

Initiating Condition: UNPLANNED loss of Control Room indications for 15 minutes or longer

EAL:

MU3.1 Unusual Event

An UNPLANNED event results in the inability to monitor one or more Table 10 parameters from within the Control Room for GE 15 min. (Note 1)

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

Table 10 Safety System Parameters
• Reactor power • RPV level • RPV pressure • Primary containment pressure • Wetwell level • Wetwell temperature

Mode Applicability:

1	2	3			

Basis:

SAFETY SYSTEM parameters listed in Table 10 are monitored in the Control Room through a combination of hard control panel indicators as well as computer based information systems. The Plant Computers and Graphic Display System provide redundant parameter indications (ref. 1-4).

This IC addresses the difficulty associated with monitoring normal plant conditions without the ability to obtain SAFETY SYSTEM parameters from within the Control Room. This condition is a precursor to a more significant event and represents a potential degradation in the level of safety of the plant.

As used in this EAL, an "inability to monitor" means that values for one or more of the listed parameters cannot be determined from within the Control Room. This situation would require a loss of all of the Control Room sources for the given parameter(s). For example, the reactor power level cannot be determined from any analog, digital and recorder source within the Control Room.

An event involving a loss of plant indications, annunciators and/or display systems is evaluated in accordance with 10 CFR 50.72 (and associated guidance in NUREG-1022) to determine if an NRC event report is required. The event would be reported if it significantly impaired the capability to perform emergency assessments. In particular, emergency assessments necessary to implement abnormal operating procedures, emergency operating procedures, and emergency plan implementing procedures addressing emergency classification, accident assessment, or protective action decision-making.

This EAL is focused on a selected subset of plant parameters associated with the key safety functions of reactivity control, RPV level and RCS heat removal. The loss of the ability to determine one or more

of these parameters from within the Control Room is considered to be more significant than simply a reportable condition. In addition, if all indication sources for one or more of the listed parameters are lost, then the ability to determine the values of other SAFETY SYSTEM parameters may be impacted as well. For example, if the value for RPV water level cannot be determined from the indications and recorders on a main control board, the SPDS or the plant computer, the availability of other parameter values may be compromised as well.

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

Escalation of the emergency classification level would be via IC ~~SA3~~MA3.

CGS Basis Reference(s):

1. FSAR Section 7.7.1
2. ABN-COMPUTER
3. SOP-COMPUTER-OPS Plant Process Computer (PPC)
4. SOP-GDS-OPS Graphics Display System
5. NEI 99-01 SU2

Category: M – 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:

MA3.1 Alert

An UNPLANNED event results in the inability to monitor one or more Table 10 parameters from within the Control Room for GE 15 min. (Note 1)

AND

Any Table 11 transient event in progress

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

Table 10 Safety System Parameters
• Reactor power • RPV level • RPV pressure • Primary containment pressure • Wetwell level • Wetwell temperature

Table 11 Significant Transients
• Reactor scram • Runback GT 25% thermal reactor power • Electrical load rejection GT 25% full electrical load • ECCS injection • Thermal power oscillations GT 10%

Mode Applicability:

1	2	3			

Basis:

SAFETY SYSTEM parameters listed in Table 10 are monitored in the Control Room through a combination of hard control panel indicators as well as computer based information systems. The Plant Computers and Graphic Display System provide redundant parameter indications (ref. 1-4).

Significant transients are listed in Table 11 and include response to automatic or manually initiated functions such as scrams, runbacks involving greater than 25% thermal power change, electrical load rejections of greater than 25% full electrical load, ECCS injections, or thermal power oscillations of 10% or greater.

This IC addresses the difficulty associated with monitoring rapidly changing plant conditions during a transient without the ability to obtain SAFETY SYSTEM parameters from within the Control Room. During this condition, the margin to a potential fission product barrier challenge is reduced. It thus represents a potential substantial degradation in the level of safety of the plant.

As used in this EAL, an "inability to monitor" means that values for one or more of the listed parameters cannot be determined from within the Control Room. This situation would require a loss of all of the Control Room sources for the given parameter(s). For example, the reactor power level cannot be determined from any analog, digital and recorder source within the Control Room.

An event involving a loss of plant indications, annunciators and/or display systems is evaluated in accordance with 10 CFR 50.72 (and associated guidance in NUREG-1022) to determine if an NRC event report is required. The event would be reported if it significantly impaired the capability to perform emergency assessments. In particular, emergency assessments necessary to implement abnormal operating procedures, emergency operating procedures, and emergency plan implementing procedures addressing emergency classification, accident assessment, or protective action decision-making.

This EAL is focused on a selected subset of plant parameters associated with the key safety functions of reactivity control, RPV level and RCS heat removal. The loss of the ability to determine one or more of these parameters from within the Control Room is considered to be more significant than simply a reportable condition. In addition, if all indication sources for one or more of the listed parameters are lost, then the ability to determine the values of other SAFETY SYSTEM parameters may be impacted as well. For example, if the value for RPV water level cannot be determined from the indications and recorders on a main control board, the SPDS or the plant computer, the availability of other parameter values may be compromised as well.

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

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

CGS Basis Reference(s):

1. FSAR Section 7.7.1
2. ABN-COMPUTER
3. SOP-COMPUTER-OPS Plant Process Computer (PPC)
4. SOP-GDS-OPS Graphics Display System
5. NEI 99-01 SA2

Category: M – System Malfunction

Subcategory: 4 – RCS Activity

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

EAL:

MU4.1 Unusual Event

SJAE CONDSR OUTLET RAD HI-HI alarm (P602)

Mode Applicability:

1	2	3			

Basis:

The main condenser offgas gross gamma activity rate is an initial condition of the Main Condenser Offgas System failure event. The gross gamma activity rate is controlled to ensure that during the event, the calculated offsite doses will be well within the limits of 10 CFR 50.67 (ref. 1).

SJAE CONDSR OUTLET RAD HI HI monitor and alarm, OG-RIS-612 (GE 2300 mR/hr), senses the offgas effluent and, therefore, may be one of the first indicators of degrading fuel conditions. The alarm is confirmed by verification of greater than the current alarm setpoint on Recorder OG-RIS-612 on Panel P604 or high offgas pre-treatment air activity (determined by sample results) greater than limits specified in Technical Specification.

If OG-RIS-612 and OG-RR-604 are reading off-scale high, the alarm may be confirmed by a significant increase in the Main Steam Line radiation monitors (MS-RIS-610A-D) on H13-P606 and H13-P633 (ref. 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.

CGS Basis Reference(s):

1. Technical Specifications 3.7.5
2. PPM 4.602.A5 ANNUNCIATOR RESPONSE, P602 ANNUNCIATOR A5 3-3
3. NEI 99-01 SU3

Category: M – System Malfunction

Subcategory: 4 – RCS Activity

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

EAL:

MU4.2	**Unusual Event**
Coolant activity GT 0.2 µCi/gm dose equivalent I-131	

Mode Applicability:

1	2	3			

Basis:

The limits on the specific activity of the primary coolant ensure that the 2-hour thyroid and whole body doses at the SITE BOUNDARY, resulting from an Main Steam Line Break (MSLB) outside containment during steady state operation, will not exceed the dose guidelines of 10 CFR 50.67.

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.

CGS Basis Reference(s):

1. Technical Specifications 3.4.8
2. NEI 99-01 SU3

Category: M – System Malfunction

Subcategory: 5 – RCS Leakage

Initiating Condition: RCS leakage for 15 minutes or longer

EAL:

MU5.1 Unusual Event

(1) RCS unidentified or pressure boundary leakage GE 10 gpm for GE 15 min.

OR

(2) RCS identified leakage GT 25 gpm for GE 15 min.

OR

(3) Leakage from the RCS to a location outside containment GT 25 gpm for GE 15 min.

(Note 1)

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

Mode Applicability:

1	2	3			

Basis:

Pressure boundary leakage is defined to be leakage through a non-isolable fault in a RCS component body, pipe wall, or vessel wall.

This EAL does not apply to relief valves performing their normal design function.

Unidentified leakage is defined to be all leakage into the drywell that is not identified leakage.

Identified leakage is defined to be leakage into the drywell such as that from pump seals or valve packing, that is captured and conducted to a sump; or leakage into the drywell atmosphere from sources that are both specifically located and known either not to interfere with the operation of leakage detection systems or not to be pressure boundary leakage. (ref. 1)

The Leak Detection (LD) system is designed to monitor leakage from the reactor coolant pressure boundary and to isolate this leakage when limits are exceeded. Systems, or parts of systems, that are in direct communication with the reactor vessel (form part of the primary coolant pressure boundary) are provided with leakage detection systems. (ref. 2-8)

Drain flow from the drywell equipment and floor drain sumps is monitored and recorded (EDR-FRS-623) on P632. The flow rates for ~~identified and~~ unidentified leakage in the EAL ~~are~~ is equal to the full scale reading on EDR-FRS-623 Pen 1.

Leakage not explicitly identified by installed instrumentation requires analysis and declaration clock starts at completion of analysis. This includes use of alternate means.

As an alternate means, leaks within the drywell are detected by monitoring for abnormally high:

- Pressure or temperature inside the drywell
- Fill up rates of equipment and floor drain sumps
- Containment leak detection rad monitors (CMS-SR-20/21)

Outside Containment leakage may require analysis to quantify leak rate GT 25 gpm and declaration clock starts at completion of analysis.

Examples of outside Containment leakage include:

- GT 25 gpm RWCU differential flow (RWCU-FI-620) due to RCS leakage
- Instrument line break in the RX building with failure to isolate
- Rx Building sump fill timers due to RCS leakage

RFW and RCC are not considered part of RCS leakage for this EAL.

For classification under this EAL, RCS leakage includes a broken SRV tailpipe that is discharging into the drywell or wetwell airspace. Once the SRV is closed, however, this RCS leakage path is considered isolated.

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.

Threshold #1 and threshold #2 are focused on a loss of mass from the RCS due to “unidentified leakage", "pressure boundary leakage" or "identified leakage” (as these leakage types are defined in the plant Technical Specifications). Threshold #3 addresses an RCS mass loss caused by an UNISOLABLE leak through an interfacing system. These conditions thus apply to leakage into the containment, or a location outside of containment.

The leak rate values for each threshold 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). Threshold #1 uses a lower value that reflects the greater significance of unidentified or pressure boundary leakage.

A stuck-open Safety Relief Valve (SRV) or SRV leakage is not considered either identified or unidentified leakage by Technical Specifications and, therefore, is not applicable to this EAL.

The 15-minute threshold duration allows sufficient time for prompt operator actions to isolate the leakage, if possible.

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

CGS Basis Reference(s):

1. Technical Specification 1.1
2. Technical Specifications 3.4.7
3. FSAR Section 5.2.5
4. FSAR Section 7.6.1
5. ABN-LEAKAGE   Reactor Coolant Leakage
6. SOP-EDR-OPS Equipment Drain System Operation
7. SOP-FDR-OPS Floor Drain System Operation
8. PPM 10.27.35 Leakage Surveillance And Prevention Program
9. NEI 99-01 SU4

Category: M – System Malfunction

Subcategory: 6 – RPS Failure

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

EAL:

MU6.1 Unusual Event

An automatic OR manual scram did not shut down the reactor

AND

A subsequent automatic scram OR manual scram action taken at the reactor control console (mode switch in shutdown, manual push buttons or ARI) is successful in shutting down the reactor as indicated by reactor power LE 5% (APRM downscale) (Note 8)

Note 8: A manual scram action is any operator action, or set of actions, which causes the control rods to be rapidly inserted into the core, and does **not** include manually driving in control rods or implementation of boron injection strategies

Mode Applicability:

1	2				

Basis:

This EAL addresses a failure of an automatic or manually initiated scram and a subsequent automatic or manual scram is successful in shutting down the reactor (reactor power LE 5%) (ref.1).

A successful scram has occurred when there is sufficient rod insertion from the trip of RPS to bring the reactor power to or below the APRM downscale trip setpoint of 5%. For the purposes of this EAL, a successful automatic initiation of ARI that reduces reactor power to or below 5% is a not a successful automatic scram. (ref. 2, 3, 4, 5)

For the purposes of emergency classification at the Unusual Event level, successful manual scram actions are those which can be quickly performed from the reactor control console (i.e., mode switch, manual scram pushbuttons, and manual ARI actuation). Reactor shutdown achieved by use of the alternate control rod insertion methods of PPM 5.5.11 does not constitute a successful manual scram (ref. 6).

Following any automatic RPS scram signal plant procedures prescribe insertion of redundant manual scram signals to back up the automatic RPS scram function and ensure reactor shutdown is achieved. Even if the first subsequent manual scram signal inserts all control rods to the full-in position immediately after the initial failure of the automatic scram, the lowest level of classification that must be declared is an Unusual Event.

~~By procedure, operator actions include the initiation of an immediate manual scram following receipt of an automatic scram signal. If there are no clear indications that the automatic scram failed (such as a time delay following indications that a scram setpoint was exceeded), it may be difficult to determine if the reactor was shut down because of automatic scram or manual actions. If a subsequent review of the scram actuation indications reveals that the automatic scram did not cause the reactor to be shut down, then initiate a Transitory Event Notification per EPIP 13.4.1.~~

The operating mode change associated with movement of the Mode Switch, by itself, does not justify failure to declare an emergency for ATWS events.

If both subsequent automatic and subsequent manual reactor scram actions in the Control Room fail, the event escalates to an Alert under EAL MA6.1.

This IC addresses a failure of the RPS to initiate or complete an automatic or manual reactor scram that results in a reactor shutdown, and either a subsequent operator manual action taken at the reactor control consoles or an automatic scram is successful in shutting down the reactor. This event is a precursor to a more significant condition and thus represents a potential degradation of the level of safety of the plant.

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

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

A manual action at the reactor control consoles is any operator action, or set of actions, which causes the control rods to be rapidly inserted into the core (e.g., initiating a manual reactor scram). This action does not include manually driving in control rods or implementation of boron injection strategies. Actions taken at back-panels or other locations within the control room, or any location outside the control room, are not considered to be "at the reactor control consoles".

Taking the reactor mode switch to shutdown is considered to be a manual scram action.

The plant response to the failure of an automatic or manual reactor scram will vary based upon several factors including the reactor power level prior to the event, availability of the condenser, performance of mitigation equipment and actions, other concurrent plant conditions, etc. If subsequent operator manual actions taken at the reactor control consoles are also unsuccessful in shutting down the reactor, then the emergency classification level will escalate to an alert via IC SA6. Depending upon the plant response, escalation is also possible via IC FA1. Absent the plant conditions needed to meet either IC ~~SA6~~ MA6 or FA1, an unusual event declaration is appropriate for this event.

A reactor shutdown is determined in accordance with applicable emergency operating procedure criteria.

Should a reactor scram signal be generated as a result of plant work (e.g., RPS setpoint testing), the following classification guidance should be applied.

- If the signal causes a plant transient that should have included an automatic reactor scram 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 scram 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.

CGS Basis Reference(s):

1. Technical Specifications Table 3.3.1.1-1
2. FSAR Section 7.2
3. FSAR Section 7.4
4. PPM 5.1.1 RPV Control
5. PPM 5.1.2 RPV Control-ATWS
6. PPM 5.5.11 Alternate Control Rod Insertions
7. NEI 99-01 SU5

ATTACHMENT 7.1: EAL Technical Bases

Category: M – System Malfunction

Subcategory: 2 – RPS Failure

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

EAL:

MA6.1 Alert

An automatic OR manual scram fails to shut down the reactor

AND

Manual scram actions taken at the reactor control console (mode switch in shutdown, manual push buttons or ARI) are not successful in shutting down the reactor as indicated by reactor power GT 5% (Note 8)

Note 8: A manual scram action is any operator action, or set of actions, which causes the control rods to be rapidly inserted into the core, and does **not** include manually driving in control rods or implementation of boron injection strategies

Mode Applicability:

1	2				

Basis:

This EAL addresses any automatic or manual reactor scram signal that fails to shut down the reactor followed by a subsequent manual scram that fails to shut down the reactor to an extent the reactor is producing energy in excess of the heat load for which the safety systems were designed.

For the purposes of emergency classification, successful manual scram actions are those which can be quickly performed from the reactor control console (i.e., mode switch in shutdown, manual push buttons or ARI). Reactor shutdown achieved by use of the alternate control rod insertion methods of PPM 5.5.11 does not constitute a successful manual scram (ref. 1).

The APRM downscale trip setpoint (5%) is a minimum reading on the power range scale that indicates power production. It also approximates the decay heat which the shutdown systems were designed to remove and is indicative of a condition requiring immediate response to prevent subsequent core damage. Below the APRM downscale trip setpoint, plant response will be similar to that observed during a normal shutdown. Nuclear instrumentation (APRM) indications or other reactor parameters (steam flow, BPV position or continuous SRV operation) can be used to determine if reactor power is greater than 5% power (ref. 2).

Escalation of this event is via EAL MS6.1.

This IC addresses a failure of the RPS to initiate or complete an automatic or manual reactor scram that results in a reactor shutdown, and subsequent operator manual actions taken at the reactor control consoles to shutdown the reactor are also unsuccessful. This condition represents an actual or potential substantial degradation of the level of safety of the plant. An emergency declaration is required even if the reactor is subsequently shutdown by an action taken away from the reactor control consoles since this event entails a significant failure of the RPS.

A manual action at the reactor control console is any operator action, or set of actions, which causes the control rods to be rapidly inserted into the core (e.g., initiating a manual reactor scram). This action does not include manually driving in control rods or implementation of boron injection strategies. If this

action(s) is unsuccessful, operators would immediately pursue additional manual actions at locations away from the reactor control console (e.g., locally opening breakers). Actions taken at backpanels or other locations within the control room, or any location outside the control room, are not considered to be “at the reactor control console”.

Taking the reactor mode switch to shutdown is considered to be a manual scram action.

The plant response to the failure of an automatic or manual reactor scram will vary based upon several factors including the reactor power level prior to the event, availability of the condenser, performance of mitigation equipment and actions, other concurrent plant conditions, etc. If the failure to shut down the reactor is prolonged enough to cause a challenge to RPV water level or RCS heat removal safety functions, the emergency classification level will escalate to a Site Area Emergency via IC ~~SS6~~MS6. Depending upon plant responses and symptoms, escalation is also possible via IC FS1. Absent the plant conditions needed to meet either IC SS6 or FS1, an Alert declaration is appropriate for this event.

It is recognized that plant responses or symptoms may also require an Alert declaration in accordance with the Recognition Category F ICs; however, this IC and EAL are included to ensure a timely emergency declaration.

A reactor shutdown is determined in accordance with applicable Emergency Operating Procedure criteria.

CGS Basis Reference(s):

1. PPM 5.5.11 Alternate Control Rod Insertions
2. Technical Specifications Table 3.3.1.1-1
3. NEI 99-01 SA5

ATTACHMENT 7.1: EAL Technical Bases

Category: M – System Malfunction

Subcategory: 2 – RPS Failure

Initiating Condition: Inability to shut down the reactor causing a challenge to RPV water level or RCS heat removal

EAL:

MS6.1 Site Area Emergency

An automatic OR manual scram fails to shut down the reactor

AND

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

AND EITHER:

RPV level cannot be restored and maintained above -186 in. or cannot be determined

OR

WW temperature and RPV pressure cannot be maintained below the HCTL

Mode Applicability:

1	2				

Basis:

This EAL addresses the following:

- Any automatic reactor scram signal followed by a manual scram that fails to shut down the reactor to an extent the reactor is producing energy in excess of the heat load for which the safety systems were designed (EAL MA6.1), and
- Indications that either core cooling is extremely challenged or heat removal is extremely challenged.

Reactor shutdown achieved by use of control rod insertion methods in PPM 5.5.11 is also credited as a successful manual scram provided reactor power can be reduced below the APRM downscale trip setpoint before indications of an extreme challenge to either core cooling or heat removal exist. (ref. 1)

The APRM downscale trip setpoint (5%) is a minimum reading on the power range scale that indicates power production. It also approximates the decay heat which the shutdown systems were designed to remove and is indicative of a condition requiring immediate response to prevent subsequent core damage. Below the APRM downscale trip setpoint, plant response will be similar to that observed during a normal shutdown. Nuclear instrumentation (APRM) indications or other reactor parameters (steam flow, RPV pressure, torus temperature trend) can be used to determine if reactor power is greater than 5% power (ref. 2).

The combination of failure of both front line and backup protection systems to function in response to a plant transient, along with the continued production of heat, poses a direct threat to the Fuel Clad and RCS barriers.

Indication that core cooling is extremely challenged is manifested by inability to restore and maintain RPV water level above the Minimum Steam Cooling RPV Water Level (MSCRWL) (ref. 3). The MSCRWL is the lowest RPV level at which the covered portion of the reactor core will generate sufficient steam to prevent any clad temperature in the uncovered part of the core from exceeding

1500°F. This water level is utilized in the EOPs to preclude fuel damage when RPV level is below the top of active fuel. RPV level below the MSCRWL for an extended period of time without satisfactory core spray cooling could be a precursor of a core melt sequence.

The Heat Capacity Temperature Limit (HCTL) is the highest suppression pool water temperature from which Emergency RPV Depressurization will not raise suppression pool temperature above the maximum design suppression pool temperature.

The HCTL is a function of RPV pressure and wetwell level. It is utilized to preclude failure of the containment and equipment in the containment necessary for the safe shutdown of the plant (ref. 4).

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

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

A reactor shutdown is determined in accordance with applicable Emergency Operating Procedure criteria.

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

CGS Basis Reference(s):

1. PPM 5.5.11 Alternate Control Rod Insertions
2. Technical Specifications Table 3.3.1.1-1
3. PPM 5.1.2 RPV Control – ATWS
4. PPM 5.2.1 Primary Containment Control,
5. NEI 99-01 SS5

ATTACHMENT 7.1: EAL Technical Bases

Category: M – System Malfunction

Subcategory: 7 – Loss of Communications

Initiating Condition: Loss of all onsite or offsite communications capabilities

EAL:

MU7.1 Unusual Event

(1) Loss of all Table 4 onsite communication methods

OR

(2) Loss of all Table 4 ORO communication methods

OR

(3) Loss of all Table 4 NRC communication methods

Table 4 Communication Methods

System	Onsite	ORO	NRC
Plant Public Address (PA) System	X		
Plant Telephone System	X	X	
Plant Radio System Operations and Security Channels	X		
Offsite calling capability from the Control Room via direct telephone		X	X
Long distance calling capability on the commercial phone system		X	X

Mode Applicability:

1	2	3			

Basis:

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

Public Address (PA) System

The public address system provides a way of contacting personnel in the various buildings of the plant and locations of the site that might be inaccessible using other means of communication. The building-wide alarm system alerts (via the public address system speakers) operating personnel to fire hazards and other trouble conditions for which plant management finds it necessary to alert plant personnel.

Plant Telephone System

This system consists of interconnections to the public telephone network (and trunks to the PBX) with individual direct lines that provide inward and outward dialing access to most plant locations.

Plant Radio System Operations and Security Channels

The radio communications system is used for communications with personnel involved in maintenance and security in and around the plant complex by means of hand-held portable radio units, mobile radio units, and paging receivers. The telephone link to BPA provides a direct communication link to the BPA Dittmer Control Center. The radio communications system provides a communications link for security and emergency communications to local law enforcement agencies and emergency control centers.

Offsite calling capability from the Control Room via direct telephone and fax lines

This communications method includes following dedicated phone networks that are available for emergency communications in addition to the normal Energy Northwest phone network:

- Energy Northwest Emergency Center Network
- Response Agency Network
- NRC Emergency Notification System

Various locations such as the Control Room, Technical Support Center, Emergency Operations Facility, Joint Information Center, Department of Energy-RL, Washington State Emergency Operations Center, Oregon State Emergency Coordination Center and the Benton and Franklin County Emergency Operations Centers have facsimile transceivers. The facsimile transceivers enable the transmission and receipt of printed material. The facsimile system which connects the Energy Northwest emergency centers with the county and state emergency centers uses dedicated phone lines.

Long distance calling capability on the commercial phone system

The Energy Northwest Richland phone system is a computer based, software controlled telephone exchange (Computerized Branch Exchange). It is equipped with redundant computerized processor units and is served by an uninterruptible power supply. The direct-dial private telephone system provides communication between the Energy Northwest facilities. The phone system is arranged such that plant telephones can reach other Energy Northwest facilities by direct-dialing and without the need of an operator.

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.).

Threshold #1 addresses a total loss of the communications methods used in support of routine plant operations.

Threshold #2 addresses a total loss of the communications methods used to notify all OROs of an emergency declaration. The OROs referred to here are Washington Stare, Benton County, Franklin County and DOE RL.

Threshold #3 addresses a total loss of the communications methods used to notify the NRC of an emergency declaration.

CGS Basis Reference(s):

1. Emergency Plan Section 6.6
2. FSAR Section 9.5.2
3. NEI 99-01 SU6

ATTACHMENT 7.1: EAL Technical Bases

Category: M – System Malfunction

Subcategory: 8 – Hazardous Event Affecting Safety Systems

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

EAL:

MA8.1 Alert

The occurrence of any Table 8 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

OR

The event has caused VISIBLE DAMAGE to a SAFETY SYSTEM component or structure, Table 5, needed for the current operating mode (Note 9)

Note 9: If the affected SAFETY SYSTEM (or component) was already inoperable or out of service before the event occurred, then **no** emergency classification is warranted as long as the damage was limited to the affected SAFETY SYSTEM (or component).

Table 8 Hazardous Events
• Seismic event • Internal or external FLOODING event • High winds • Tornado strike • FIRE • EXPLOSION • Volcanic ash fallout • Other events with similar hazard characteristics as determined by the Shift Manager

Table 5 Safe Shutdown Areas
• Vital portions of the Rad Waste/Control Building: - 467' elevation vital island - 487' elevation cable spreading room - Main Control Room and vertical cable chase - 525' elevation HVAC area • Reactor Building • Vital portions of the Turbine Building - DEH pressure switches - RPS switches on turbine throttle valves - Main steam line radiation monitors - Turbine Building ventilation radiation monitors - Main steam line piping up to MS-V-146 and the first stop valves • Standby Service Water Pump Houses • Diesel Generator Building

Mode Applicability:

1	2	3			

Basis:

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

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

The second condition addresses damage to a SAFETY SYSTEM component that is not in service/operation or readily apparent through indications alone, or to a structure containing 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 a seismic event is discussed under EAL HU2.1 (ref. 1).

Internal FLOODING may be caused by events such as component failures, equipment misalignment, or outage activity mishaps (ref. 2).

Seismic Category I structures are analyzed to withstand a sustained, design wind velocity of at least 100 mph (ref. 3).

Areas containing functions and systems required for safe shutdown of the plant are identified by Fire Areas in the fire response procedure (ref. 4).

The potential for volcanic eruption exists in the Pacific Northwest. Heavy ash fall, such as that experienced at certain locations following the eruption of Mt. St. Helens in 1980, could affect operation of plant equipment if precautionary measures are not taken. The design basis ash fall is projected for a twenty hour duration (ref. 5).

Table 5 provides a list of CGS safety system structures/areas (ref. 6). Table 8 provides a list of hazardous events.

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

CGS Basis Reference(s):

1. FSAR Section 3.7 Seismic Design
2. FSAR Section 3.4.1 Flood Protection
3. CGS Calculation CALC CE-02-93-16 Evaluate PMR/BDC 98-0131-0A change from 5 min. to 15 min. averaging of 33 ft. elev. met twr. wind speeds for UE and Alert declarations
4. ABN-FIRE Attachment 13.2, Fire Areas
5. ABN-ASH Ash Fall
6. FSAR Table 3.2-1 Equipment Classification
7. NEI 99-01 SA9

Category E – Independent Spent Fuel Storage Installation (ISFSI)

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

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

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

A hostile security event that leads to a potential loss in the level of safety of the ISFSI is a classifiable event under Security category EAL HA1.1.

ATTACHMENT 7.1: EAL Technical Bases

Category: E - ISFSI

Sub-category: None

Initiating Condition: Damage to a loaded cask CONFINEMENT BOUNDARY

EAL:

EU1.1 Unusual Event

Damage to a loaded canister (MPC) CONFINEMENT BOUNDARY as indicated by measured dose rates on a loaded overpack GT EITHER:

- 20 mrem/hr (gamma + neutron) on the top of the overpack
- 100 mrem/hr (gamma + neutron) on the side of the overpack, excluding inlet and outlet ducts

Mode Applicability:

Storage Operations

Basis:

The Independent Spent Fuel Storage Installation utilizes the HOLTEC International (HOLTEC) HI-STORM 100 Spent Fuel Dry Storage (SFDS) system. HI-STORM overpack or storage overpack means the cask that receives and contains the sealed multi-purpose canisters containing spent nuclear fuel. It provides the gamma and neutron shielding, ventilation passages, missile protection, and protection against natural phenomena and accidents for the MPC. (ref. 1, 2)

The EAL threshold values represent two-times the limits specified in the ISFSI Certificate of Compliance Technical Specification Section 3.2, Radiation Protection Program (ref. 2).

CGS has casks loaded to various amendments to the Certificate of Compliance (COC) Technical Specifications~~ with a proposed amendment coming in 2017~~. The numbers above reflect the most limiting Technical Specification (TS) values (Amendment 1)~~ and can be updated using 10 CFR 50.54(q) process, if CGS adopts a common TS amendment~~.

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

The existence of “damage” is determined by radiological survey. The technical specification multiple of “2 times”, which is also used in Recognition Category R IC RU1, is used here to distinguish between non-emergency and emergency conditions. The emphasis for this classification is the degradation in the level of safety of the spent fuel cask and not the magnitude of the associated dose or dose rate. It is recognized that in the case of extreme damage to a loaded cask, the fact that the “on-contact” dose rate limit is exceeded may be determined based on measurement of a dose rate at some distance from the cask.

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

CGS Basis Reference(s):

1. ABN-ISFSI, ISFSI Abnormal Conditions
2. ISFSI Certificate of Compliance No. 1014 Amendment 1, Appendix A, Technical Specifications for the HI-STORM 100 Cask System, Section 3.2 Radiation Protection Program
3. NEI 99-01 E-HU1

Category F – Fission Product Barrier Degradation

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

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

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

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

C. Containment (PC): The drywell, the wetwell, their respective interconnecting paths, and other connections up to and including the outermost containment isolation valves comprise the PC barrier. Containment Barrier thresholds are used as criteria for escalation of the ECL from Alert to either 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 CGS design and operating characteristics.

ATTACHMENT 7.1: EAL Technical Bases

- As used in this category, the term RCS leakage encompasses not just those types defined in Technical Specifications but also includes the loss of RCS mass to any location– inside the containment, an interfacing system, or outside of the containment. The release of liquid or steam mass from the RCS due to the as-designed/expected operation of a relief valve is not considered RCS leakage.
- At the Site Area Emergency level, EAL users should maintain cognizance of how far present conditions are from meeting a threshold that would require a General Emergency declaration. For example, if the Fuel Clad and RCS fission product barriers were both lost, there should be frequent assessments of containment radioactive inventory and integrity. Alternatively, if both the Fuel Clad and RCS fission product barriers were potentially lost, the Emergency Director would have more assurance that there was no immediate need to escalate to a General Emergency.

Category: Fission Product Barrier Degradation

Subcategory: N/A

Initiating Condition: Any loss or any potential loss of either Fuel Clad or RCS

EAL:

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

Mode Applicability:

1	2	3			

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

CGS 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	2	3			

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 Director would have greater assurance that escalation to a General Emergency is less imminent.

CGS Basis Reference(s):

1. NEI 99-01 FS1

ATTACHMENT 7.1: EAL Technical Bases

Category: Fission Product Barrier Degradation

Subcategory: N/A

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

EAL:

FG1.1 General Emergency

Loss of any two barriers

AND

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

Mode Applicability:

1	2	3			

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

CGS Basis Reference(s):

1. NEI 99-01 FG1

ATTACHMENT 7.2: Fission Product Barrier Matrix and Bases

Introduction

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

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

A. RPV Water Level

B. RCS Leak Rate

B. PC Conditions

C. PC Radiation / RCS Activity

D. PC Integrity or Bypass

E. Emergency Director Judgment

Each category occupies a row in Table F-1 thus forming a matrix defined by the categories. The intersection of each row with each Loss/Potential Loss column forms a cell in which one or more fission product barrier thresholds appear. If NEI 99-01 does not define a threshold for a barrier Loss/Potential Loss, the word “None” is entered in the cell.

Thresholds are assigned sequential numbers within each Loss and Potential Loss column beginning with number one. In this manner, a threshold can be identified by its category title and number. For example, the first Fuel Clad barrier Loss in Category A would be assigned “FC Loss A.1,” the third Containment barrier Potential Loss would be assigned “PC P-Loss B.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

ATTACHMENT 7.2: Fission Product Barrier Matrix and Bases

Loss of the primary containment barrier can occur. Barrier Losses and Potential Losses are then applied to the algorithms given in EALs FG1.1, FS1.1, and FA1.1 to determine the appropriate emergency classification.

In the remainder of this Attachment, the Fuel Clad barrier threshold bases appear first, followed by the RCS barrier and finally the Containment barrier threshold bases. In each barrier, the bases are given according category Loss followed by category Potential Loss beginning with Category A, then B,…, F.

ATTACHMENT 7.2: Fission Product Barrier Matrix and Bases

Table F-1 Fission Product Barrier Threshold Matrix

FC - Fuel Clad Barrier		RCS - Reactor Coolant System Barrier		PC - Containment Barrier	
Loss	**Potential Loss**	**Loss**	**Potential Loss**	**Loss**	**Potentia**
G entry required	RPV level cannot be restored and maintained GT -161 in. or cannot be determined	RPV level cannot be restored and maintained GT -161 in. or cannot be determined	None	None	SAG entry required
None	None	UNISOLABLE break in any of the following: • Main steam lines • RCIC steam Line • RWCU • Feedwater OR Emergency RPV Depressurization is required	UNISOLABLE primary system leakage that results in exceeding EITHER: RB area temperature alarm level (PPM 5.3.1 Table 23) OR RB area radiation alarm level (PPM 5.3.1 Table 24)	UNISOLABLE primary system leakage that results in exceeding EITHER: RB area maximum safe operating temperature (PPM 5.3.1 Table 23) OR RB area maximum safe operating radiation (PPM 5.3.1 Table 24)	None
None	None	PC pressure GT 1.68 psig due to RCS leakage	None	UNPLANNED rapid drop in PC pressure following PC pressure rise OR PC pressure response not consistent with LOCA conditions	PC pressure GT 45 ps OR Explosive mixture exis GE 6% and O_2 GE 5% OR WW temperature and cannot be maintained
ntainment Radiation Monitor IS-RIS-27E or CMS-RIS-27F ding GT 3,600 R/hr R mary coolant activity GT 300 i/gm dose equivalent I-131	None	Containment Radiation Monitor CMS-RIS-27E or CMS-RIS-27F reading GT 70 R/hr	None	None	Containment Radiatio RIS-27E or CMS-RIS- GT 14,000 R/hr
None	None	None	None	UNISOLABLE direct downstream pathway to the environment exists after PC isolation signal OR Intentional PC venting per EOPs	Nor
y condition in the opinion of the ergency Director that indicates s of the fuel clad barrier	Any condition in the opinion of the Emergency Director that indicates potential loss of the fuel clad barrier	Any condition in the opinion of the Emergency Director that indicates loss of the RCS barrier	Any condition in the opinion of the Emergency Director that indicates potential loss of the RCS barrier	Any condition in the opinion of the Emergency Director that indicates loss of the Containment barrier	Any condition in the o Emergency Director t potential loss of the C barrier

ATTACHMENT 7.2: Fission Product Barrier Matrix and Bases

Barrier: Fuel Clad

Category: A. RPV Level

Degradation Threat: Loss

Threshold:

SAG entry required

Basis:

EOP flowcharts provide instructions to assure adequate core cooling by restoring and maintaining RPV water level above prescribed limits, operate sufficient RPV injection sources to assure adequate core cooling, and assess the possibility of core damage when RPV level cannot be determined. The Fuel Clad Loss threshold conditions are the EOP flowchart conditions that signal a loss of adequate core cooling and a requirement to exit all EOPs and enter the SAGs (ref. 1-6).

This threshold is also a Loss of the RCS barrier (RCS Loss A) and a Potential Loss of the Containment barrier (PC P-Loss A), and therefore represents a Loss of two barriers and a Potential Loss of a third, which requires a General Emergency classification.

The Loss threshold represents the EOP requirement for entry to the Severe Accident Guidelines (SAGs).

CGS Basis Reference(s):

1. PPM 5.1.1 RPV Control
2. PPM 5.1.2 RPV Control – ATWS
3. Calculation NE-02-03-06 Attachment 10 RPV Variables
4. PPM 5.0.10 Flowchart Training Manual
5. PPM 5.1.4 RPV Flooding
6. PPM 5.1.6 RPV Flooding – ATWS
7. NEI 99-01 RPV Water Level Fuel Clad Loss 2.A

ATTACHMENT 7.2: Fission Product Barrier Matrix and Bases

Barrier: Fuel Clad

Category: A. RPV Level

Degradation Threat: Potential Loss

Threshold:

RPV level cannot be restored and maintained GT -161 in. or cannot be determined

Basis:

An RPV water level instrument reading of -161 in. indicates RPV level is at the top of active fuel (TAF) (ref. 1, 2). When RPV level is at or above the TAF, the core is completely submerged. Core submergence is the most desirable means of core cooling. When RPV level is below TAF, the uncovered portion of the core must be cooled by less reliable means (i.e., steam cooling or spray cooling). If core uncovery is threatened, the EOPs specify alternate, more extreme, RPV water level control measures in order to restore and maintain adequate core cooling. Since core uncovery begins if RPV level drops to TAF, the level is indicative of a challenge to core cooling and the Fuel Clad barrier.

When RPV level cannot be determined, EOPs require RPV flooding strategies. RPV water level indication provides the primary means of knowing if adequate core cooling is being maintained. When all means of determining RPV level are unavailable, the fuel clad barrier is threatened and reliance on alternate means of assuring adequate core cooling must be attempted. The instructions in PPM 5.1.4 and PPM 5.1.6 specify these means, which include emergency depressurization of the RPV and injection into the RPV at a rate needed to flood to the elevation of the main steam lines or hold RPV pressure above the Minimum Steam Cooling Pressure (in ATWS events) (ref. 3, 4). If RPV level cannot be determined with respect to the top of active fuel, a potential loss of the fuel clad barrier exists.

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

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

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

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

In high-power ATWS/failure to scram events, EOPs may direct the operator to deliberately lower RPV water level to the top of active fuel in order to reduce reactor power. RPV water level is then controlled between the top of active fuel and the Minimum Steam Cooling RPV Water Level (MSCRWL). Although such action is a challenge to core cooling and the Fuel Clad barrier, the immediate need to reduce reactor power is the higher priority. For such events, ICs MA6 or MS6 will dictate the need for emergency classification.

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

CGS Basis Reference(s):

1. Calculation NE-02-03-05 Attachment 3 Note 8
2. PPM 5.1.1 RPV Control
3. PPM 5.1.4 RPV Flooding
4. PPM 5.1.6 RPV Flooding – ATWS
5. PPM 5.1.2 RPV Control – ATWS
6. NEI 99-01 RPV Water Level Fuel Clad Potential Loss 2.A

ATTACHMENT 7.2: Fission Product Barrier Matrix and Bases

Barrier: Fuel Clad

Category: B. RCS Leak Rate

Degradation Threat: Loss

Threshold:

None

Barrier: Fuel Clad

Category: B. RCS Leak Rate

Degradation Threat: Potential Loss

Threshold:

None

ATTACHMENT 7.2: Fission Product Barrier Matrix and Bases

Barrier: Fuel Clad

Category: C. PC Conditions

Degradation Threat: Loss

Threshold:

None

ATTACHMENT 7.2: Fission Product Barrier Matrix and Bases

Barrier: Fuel Clad

Category: C. PC Conditions

Degradation Threat: Potential Loss

Threshold:

None

ATTACHMENT 7.2: Fission Product Barrier Matrix and Bases

Barrier: Fuel Clad

Category: D. PC Radiation / RCS Activity

Degradation Threat: Loss

Threshold:

Containment Radiation Monitor CMS-RIS-27E or CMS-RIS-27F reading GT 3,600 R/hr

Basis:

Four high range area radiation detectors (CMS-RE-27A, B, E and F) are installed to monitor the drywell. CMS-RE-27A and -27B are located in the bioshield wall at elevations 522′ and 525′, azimuth 60° and 297°, respectively. CMS-RE-27E and -27F are located inside containment at elevation 515′, azimuth 290° and 51.5°, respectively. The companion containment radiation monitors (CMS-RIS-27A, B, E and F) are located on RAD Boards 22 and 23 in the Main Control Room. (ref. 1)

The threshold value was calculated assuming the instantaneous release and dispersal of the reactor coolant noble gas and iodine inventory associated with a concentration of 300 µCi/gm dose equivalent I-131 (or approximately 5% clad failure) into the drywell atmosphere. Evaluation of detector location, geometry and anticipated response suggests CMS-RIS-27E or F will provide the desired response to a given radiation source in the drywell and are, therefore, identified as the preferred monitors for evaluating this Fuel Clad Loss threshold. (ref. 2)

The radiation monitor reading corresponds to an instantaneous release of all reactor coolant mass into the primary 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 D since it indicates a loss of both the Fuel Clad Barrier and the RCS Barrier. Note that a combination of the two monitor readings appropriately escalates the emergency classification level to a Site Area Emergency.

There is no Potential Loss threshold associated with primary containment radiation.

CGS Basis Reference(s):

1. TM-2117 TSG – Core Thermal Engineer, Attachment 4.2
2. Calculation NE-02-94-57
2. NEI 99-01 Primary Containment Radiation Fuel Clad Loss 4.A

ATTACHMENT 7.2: Fission Product Barrier Matrix and Bases

Barrier: Fuel Clad

Category: D. PC Radiation / RCS Activity

Degradation Threat: Loss

Threshold:

Primary coolant activity GT 300 µCi/gm dose equivalent I-131

Basis:

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

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

There is no Potential Loss threshold associated with RCS Activity.

CGS Basis Reference(s):

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

ATTACHMENT 7.2: Fission Product Barrier Matrix and Bases

Barrier: Fuel Clad

Category: D. PC Radiation / RCS Activity

Degradation Threat: Potential Loss

Threshold:

None

ATTACHMENT 7.2: Fission Product Barrier Matrix and Bases

Barrier: Fuel Clad

Category: E. PC Integrity or Bypass

Degradation Threat: Loss

Threshold:

None

ATTACHMENT 7.2: Fission Product Barrier Matrix and Bases

Barrier: Fuel Clad

Category: E. PC Integrity or Bypass

Degradation Threat: Potential Loss

Threshold:

None

ATTACHMENT 7.2: Fission Product Barrier Matrix and Bases

Barrier: Fuel Clad

Category: F. Emergency Director Judgment

Degradation Threat: Loss

Threshold:

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

Basis:

The Emergency Director judgment threshold addresses any other factors relevant to determining if the Fuel Clad barrier is lost. Such a determination should include imminent barrier degradation, barrier monitoring capability and dominant accident sequences.

- Imminent barrier degradation exists if the degradation will likely occur within two hours based on a projection of current safety system performance. The term “imminent” refers to recognition of the inability to reach safety acceptance criteria before completion of all checks.
- Barrier monitoring capability is decreased if there is a loss or lack of reliable indicators. This assessment should include instrumentation operability concerns, readings from portable instrumentation and consideration of offsite monitoring results.
- Dominant accident sequences lead to degradation of all fission product barriers and likely entry to the EOPs. The Emergency Director 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 Director in determining whether the Fuel Clad barrier is lost.

CGS Basis Reference(s):

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

ATTACHMENT 7.2: Fission Product Barrier Matrix and Bases

Barrier: Fuel Clad

Category: F. Emergency Director Judgment

Degradation Threat: Potential Loss

Threshold:

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

Basis:

The Emergency Director judgment threshold addresses any other factors relevant to determining if the Fuel Clad barrier is potentially lost. Such a determination should include imminent barrier degradation, barrier monitoring capability and dominant accident sequences.

- Imminent barrier degradation exists if the degradation will likely occur within two hours based on a projection of current safety system performance. The term “imminent” refers to recognition of the inability to reach safety acceptance criteria before completion of all checks.
- Barrier monitoring capability is decreased if there is a loss or lack of reliable indicators. This assessment should include instrumentation operability concerns, readings from portable instrumentation and consideration of offsite monitoring results.
- Dominant accident sequences lead to degradation of all fission product barriers and likely entry to the EOPs. The Emergency Director 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 Director in determining whether the Fuel Clad barrier is potentially lost. The Emergency Director should also consider whether or not to declare the barrier potentially lost in the event that barrier status cannot be monitored.

CGS Basis Reference(s):

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

ATTACHMENT 7.2: Fission Product Barrier Matrix and Bases

Barrier: Reactor Coolant System

Category: A. RPV Water Level

Degradation Threat: Loss

Threshold:

RPV level cannot be restored and maintained GT -161 in. or cannot be determined

Basis:

An RPV water level instrument reading of -161 in. indicates level is at the top of active fuel (TAF) (ref. 1, 2). TAF is significantly lower than the normal operating RPV level control band. To reach this level, RPV inventory loss would have previously required isolation of the RCS and PC barriers, and initiation of all ECCS. If RPV water level cannot be maintained above TAF, ECCS and other sources of RPV injection have been ineffective or incapable of reversing the decreasing level trend. The cause of the loss of RPV inventory is therefore assumed to be a LOCA. By definition, a LOCA event is a Loss of the RCS barrier.

When RPV water level cannot be determined, EOPs require RPV flooding strategies. RPV water level indication provides the primary means of knowing if adequate core cooling is being maintained. The instructions in PPM 5.1.4 and PPM 5.1.6 specify emergency depressurization of the RPV, which is defined to be a Loss of the RCS barrier (RCS Loss B threshold #2). (ref. 3, 4)

The conditions of this threshold are also a Potential Loss of the Fuel Clad barrier (FC P-Loss A). A Loss of the RCS barrier and Potential Loss of the Fuel Clad barrier requires a Site Area Emergency classification.

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

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

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

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

In high-power ATWS/failure to scram events, EOPs may direct the operator to deliberately lower RPV water level to the top of active fuel in order to reduce reactor power. RPV water level is then controlled between the top of active fuel and the Minimum Steam Cooling RPV Water Level (MSCRWL). Although such action is a challenge to core cooling and the Fuel Clad barrier, the immediate need to reduce reactor power is the higher priority. For such events, ICs MA6 or MS6 will dictate the need for emergency classification.

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

CGS Basis Reference(s):

1. Calculation NE-02-03-05 Attachment 3 Note 8
2. PPM 5.1.1 RPV Control
3. PPM 5.1.4 RPV Flooding
4. PPM 5.1.6 RPV Flooding – ATWS
5. PPM 5.1.2 RPV Control – ATWS
6. NEI 99-01 RPV Water Level RCS Loss 2.A

ATTACHMENT 7.2: Fission Product Barrier Matrix and Bases

Barrier: Reactor Coolant System

Category: A. RPV Water Level

Degradation Threat: Potential Loss

Threshold:

None

ATTACHMENT 7.2: Fission Product Barrier Matrix and Bases

Barrier: Reactor Coolant System

Category: B. RCS Leak Rate

Degradation Threat: Loss

Threshold:

UNISOLABLE break in any of the following:

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

Basis:

The conditions of this threshold include required containment isolation failures allowing a flow path to the environment. A release pathway outside primary containment exists when flow is not prevented by downstream isolations. In the case of a failure of both isolation valves to close but in which no downstream flowpath exists, emergency declaration under this threshold would not be required. Similarly, if the emergency response requires the normal process flow of a system outside containment (e.g., EOP requirement to bypass MSIV low RPV water level interlocks and maintain the main condenser as a heat sink using main turbine bypass valves), the threshold is not met. The combination of these threshold conditions represent the loss of both the RCS and Containment (see PC Loss E Threshold #1) barriers and justifies declaration of a Site Area Emergency (i.e., Loss or Potential Loss of any two barriers). (ref. 1-4)

Even though RWCU and Feedwater systems do not contain steam, they are included in the list because an unisolable break could result in the high-pressure discharge of fluid that is flashed to steam from relatively large volume systems directly connected to the RCS. (ref. 1)

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

CGS Basis Reference(s):

1. FSAR Section 5.4.5
2. FSAR Section 5.4.6
3. FSAR Section 5.4.8
4. FSAR Section 10.3
5. NEI 99-01 RCS Leak Rate RCS Loss 3.A

ATTACHMENT 7.2: Fission Product Barrier Matrix and Bases

Barrier: Reactor Coolant System

Category: B. RCS Leak Rate

Degradation Threat: Loss

Threshold:

Emergency RPV Depressurization is required

Basis:

Plant symptoms requiring Emergency RPV Depressurization per the EOPs are indicative of a loss of the RCS barrier. Emergency RPV Depressurization is specified in the EOP flowcharts when symbols containing the phrase “EMERG DEPRESS REQ’D” are reached (ref. 1-7). If Emergency RPV Depressurization is required, the plant operators are directed to open safety relief valves (SRVs) and keep them open as needed to maintain adequate core cooling with available injection sources (ref. 8, 9). Even though the RCS is being vented into the suppression pool, a loss of the RCS exists due to the diminished effectiveness of the RCS pressure barrier to a release of fission products beyond its boundary.

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

CGS Basis Reference(s):

1. PPM 5.1.1 RPV Control
2. PPM 5.1.2 RPV Control – ATWS
3. PPM 5.1.4 RPV Flooding
4. PPM 5.1.6 RPV Flooding – ATWS
5. PPM 5.2.1 Primary Containment Control
6. PPM 5.3.1 Secondary Containment Control
7. PPM 5.4.1 Radioactivity Release Control
8. PPM 5.1.3 Emergency RPV Depressurization
9. PPM 5.1.5 Emergency RPV Depressurization – ATWS
10. NEI 99-01 RCS Leak Rate RCS Loss 3.B

ATTACHMENT 7.2: Fission Product Barrier Matrix and Bases

Barrier: Reactor Coolant System

Category: B. RCS Leak Rate

Degradation Threat: Potential Loss

Threshold:

UNISOLABLE primary system leakage that results in exceeding EITHER:

RB area temperature alarm level (PPM 5.3.1 Table 23)

OR

RB area radiation alarm level (PPM 5.3.1 Table 24)

Basis:

The presence of elevated general area temperatures or radiation levels in the secondary containment may be indicative of unisolable primary system leakage outside the primary containment. The PPM 5.3.1 Table 23 and Table 24 alarm levels define this RCS threshold because they are the maximum normal operating values and signify the onset of abnormal system operation. When parameters reach this level, equipment failure or misoperation may be occurring. Elevated parameters may also adversely affect the ability to gain access to or operate equipment within the affected area. The locations into which the primary system discharge is of concern correspond to the areas addressed in PPM 5.3.1 Tables 23 and 24 (ref. 1).

Area temperature alarms are provided by the leak detection and reactor building recirculation air (RRA) systems (ref. 2)

The ARM alarm setpoints listed in Table 24 vary due to plant operating mode and Health Physics radiation surveys. A program is established to maintain the current setpoint values in PPM 4.602.A5 for annunciator window 3-1; thus, reference is made to the annunciator response procedure in Table 24. (ref. 2)

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

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

A Max Normal Operating value is the highest value of the identified parameter expected to occur during normal plant operating conditions with all directly associated support and control systems functioning properly.

ATTACHMENT 7.2: Fission Product Barrier Matrix and Bases

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

An UNISOLABLE leak which is indicated by Max Normal Operating values escalates to a Site Area Emergency when combined with Containment Barrier Loss threshold 3.A (after a containment isolation) and a General Emergency when the Fuel Clad Barrier criteria is also exceeded.

CGS Basis Reference(s):

1. PPM 5.3.1 Secondary Containment Control
2. PPM 5.0.10 Flowchart Training Manual
3. NEI 99-01 RCS Leak Rate RCS Potential Loss 3.A

ATTACHMENT 7.2: Fission Product Barrier Matrix and Bases

Barrier: Reactor Coolant System

Category: C. PC Conditions

Degradation Threat: Loss

Threshold:

PC pressure GT 1.68 psig due to RCS leakage

Basis:

The drywell high pressure scram setpoint is an entry condition to the EOP flowcharts: PPM 5.1.1, RPV Control, and PPM 5.2.1, Primary Containment Control (ref. 1, 2, 3). Normal primary containment (PC) pressure control functions such as operation of drywell cooling and venting through SGT are specified in PPM 5.2.1 in advance of less desirable but more effective functions such as operation of drywell or wetwell sprays.

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

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

1.68 psig is the drywell high pressure setpoint which indicates a LOCA by automatically initiating the ECCS or equivalent makeup system.

There is no Potential Loss threshold associated with drywell pressure.

CGS Basis Reference(s):

1. Technical Specifications Table 3.3.5.1-1
2. PPM 5.1.1 RPV Control
3. PPM 5.2.1 Primary Containment Control
4. FSAR Section 6
5. NEI 99-01 Primary Containment Pressure RCS Loss 1.A

ATTACHMENT 7.2: Fission Product Barrier Matrix and Bases

Barrier: Reactor Coolant System

Category: C. PC Conditions

Degradation Threat: Potential Loss

Threshold:

None

ATTACHMENT 7.2: Fission Product Barrier Matrix and Bases

Barrier: Reactor Coolant System

Category: D. PC Radiation / RCS Activity

Degradation Threat: Loss

Threshold:

Containment Radiation Monitor CMS-RIS-27E or CMS-RIS-27F reading GT 70 R/hr

Basis:

Four high range area radiation detectors (CMS-RE-27A, B, E and F) are installed in the drywell. CMS-RE-27A and -27B are located in the bioshield wall at elevations 522′ and 525′, azimuth 60° and 297°, respectively. CMS-RE-27E and -27F are located inside containment at elevation 515′, azimuth 290° and 51.5°, respectively. The companion containment radiation monitors (CMS-RIS-27A, B, E and F) are located on RAD Boards 22 and 23 in the Main Control Room. (ref. 1)

The threshold value was calculated assuming 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 drywell atmosphere. Evaluation of detector location, geometry and anticipated response suggests CMS-RIS-27E or F will provide the desired response to a given radiation source in the drywell and are, therefore, identified as the preferred monitors for evaluating this RCS Loss threshold. (ref. 2)

The radiation monitor reading corresponds to an instantaneous release of all reactor coolant mass into the primary containment, assuming that reactor coolant activity equals Technical Specification allowable limits. This value is lower than that specified for Fuel Clad Barrier Loss threshold D.1 since it indicates a loss of the RCS Barrier only.

There is no Potential Loss threshold associated with primary containment radiation.

CGS Basis Reference(s):

1. TM-2117 TSG – Core Thermal Engineer, Attachment 4.2
2. Calculation NE-02-94-57
3. NEI 99-01 Primary Containment Radiation RCS Loss 4.A

ATTACHMENT 7.2: Fission Product Barrier Matrix and Bases

Barrier: Reactor Coolant System

Category: D. PC Radiation / RCS Activity

Degradation Threat: Potential Loss

Threshold:

None

ATTACHMENT 7.2: Fission Product Barrier Matrix and Bases

Barrier: Reactor Coolant System

Category: E. PC Integrity or Bypass

Degradation Threat: Loss

Threshold:

None

Barrier: Reactor Coolant System

Category: E. PC Integrity or Bypass

Degradation Threat: Potential Loss

Threshold:

None

ATTACHMENT 7.2: Fission Product Barrier Matrix and Bases

Barrier: Reactor Coolant System

Category: F. Emergency Director Judgment

Degradation Threat: Loss

Threshold:

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

Basis:

The Emergency Director judgment threshold addresses any other factors relevant to determining if the RCS barrier is lost. Such a determination should include imminent barrier degradation, barrier monitoring capability and dominant accident sequences.

- Imminent barrier degradation exists if the degradation will likely occur within two hours based on a projection of current safety system performance. The term "imminent" refers to the recognition of the inability to reach safety acceptance criteria before completion of all checks.
- Barrier monitoring capability is decreased if there is a loss or lack of reliable indicators. This assessment should include instrumentation operability concerns, readings from portable instrumentation and consideration of offsite monitoring results.
- Dominant accident sequences lead to degradation of all fission product barriers and likely entry to the EOPs. The Emergency Director 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 Director in determining whether the RCS Barrier is lost.

CGS Basis Reference(s):

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

ATTACHMENT 7.2: Fission Product Barrier Matrix and Bases

Barrier: Reactor Coolant System

Category: F. Emergency Director Judgment

Degradation Threat: Potential Loss

Threshold:

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

Basis:

The Emergency Director judgment threshold addresses any other factors relevant to determining if the RCS barrier is potentially lost. Such a determination should include imminent barrier degradation, barrier monitoring capability and dominant accident sequences.

- Imminent barrier degradation exists if the degradation will likely occur within two hours based on a projection of current safety system performance. The term “imminent” refers to the inability to reach final safety acceptance criteria before completing all checks.
- Barrier monitoring capability is decreased if there is a loss or lack of reliable indicators. This assessment should include instrumentation operability concerns, readings from portable instrumentation and consideration of offsite monitoring results.
- Dominant accident sequences lead to degradation of all fission product barriers and likely entry to the EOPs. The Emergency Director 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 Director in determining whether the RCS Barrier is potentially lost. The Emergency Director should also consider whether or not to declare the barrier potentially lost in the event that barrier status cannot be monitored.

CGS Basis Reference(s):

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

ATTACHMENT 7.2: Fission Product Barrier Matrix and Bases

Barrier: Containment

Category: A. RPV Water Level

Degradation Threat: Loss

Threshold:

None

ATTACHMENT 7.2: Fission Product Barrier Matrix and Bases

Barrier: Containment

Category: A. RPV Water Level

Degradation Threat: Potential Loss

Threshold:

SAG entry required

Basis:

EOP flowcharts provide instructions to assure adequate core cooling by restoring and maintaining RPV water level above prescribed limits, operate sufficient RPV injection sources to assure adequate core cooling, and assess the possibility of core damage when RPV level cannot be determined. The Fuel Clad Loss threshold conditions are the EOP flowchart conditions that signal a loss of adequate core cooling and a requirement to exit all EOPs and enter the SAGs (ref. 1-6).

This threshold is also a Loss of the RCS barrier (RCS Loss A) and a Loss of the Fuel Clad barrier (FC Loss A), and therefore represents a Loss of two barriers and a Potential Loss of a third, which requires a General Emergency classification.

The Potential Loss threshold is identical to the Fuel Clad Loss RPV Water Level threshold. The Potential Loss requirement for SAG entry indicates adequate core cooling cannot be restored and maintained and that core damage is possible. BWR EPGs/SAGs specify the conditions that require SAG entry. When SAG entry is required, the EPGs are exited and SAGs are entered. Entry into SAGs is a logical escalation in response to the inability to restore and maintain adequate core cooling.

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

CGS Basis Reference(s):

1. PPM 5.1.1 RPV Control
2. PPM 5.1.2 RPV Control – ATWS
3. Calculation NE-02-03-06 Attachment 10 RPV Variables
4. PPM 5.0.10 Flowchart Training Manual
5. PPM 5.1.4 RPV Flooding
6. PPM 5.1.6 RPV Flooding – ATWS
7. NEI 99-01 RPV Water Level PC Potential Loss 2.A

ATTACHMENT 7.2: Fission Product Barrier Matrix and Bases

Barrier: Containment

Category: B. RCS Leak Rate

Degradation Threat: Loss

Threshold:

UNISOLABLE primary system leakage that results in exceeding EITHER:

RB area maximum safe operating temperature (PPM 5.3.1 Table 23)

OR

RB area maximum safe operating radiation (PPM 5.3.1 Table 24)

Basis:

The presence of elevated general area temperatures or radiation levels in the Reactor Building (RB) may be indicative of unisolable primary system leakage outside the primary containment. The maximum safe operating values define this Containment barrier threshold because they are indicative of problems in the secondary containment that are spreading and pose a threat to achieving a safe plant shutdown. This threshold addresses problematic discharges outside primary containment that may not originate from a high-energy line break. The locations into which the primary system discharge is of concern correspond to the areas addressed in PPM 5.3.1 Tables 23 and 24 (ref. 1).

RB maximum safe operating temperatures are conservatively defined by the qualification temperature of safety related equipment in the area. The equipment qualification program has proven that safety related equipment will perform satisfactorily to at least this temperature. In an area with multiple components and different qualification temperatures, the maximum safe operating temperature assigned to that area is generally the lowest of the individual temperatures. (ref. 2)

The maximum safe operating radiation value is defined to be 10,000 mR/hr in areas other than the refueling floor. This is the maximum indication on all but the high level instruments. This value is high enough to be indicative of substantial and immediate problems yet low enough to allow time for shutdown or isolation of a leak without exceeding the total integrated dose allowable for even the most sensitive safety related equipment. No area radiation levels are defined for the refueling floor because no primary systems are routed there. (ref. 2)

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

The Max Safe Operating Temperature and the Max Safe Operating Radiation Level are each the highest value of these parameters at which neither: (1) equipment necessary for the safe shutdown of the plant will fail, nor (2) personnel access necessary for the safe shutdown of the plant will be

precluded. EOPs utilize these temperatures and radiation levels to establish conditions under which RPV depressurization is required.

The temperatures and radiation levels should be confirmed to be caused by RCS leakage from a primary system. A primary system is defined to be the pipes, valves, and other equipment which connect directly to the RPV such that a reduction in RPV pressure will effect a decrease in the steam or water being discharged through an unisolated break in the system.

In combination with the RCS Potential Loss RCS Leak Rate threshold this threshold would result in a Site Area Emergency.

There is no Potential Loss threshold associated with primary containment isolation failure.

CGS Basis Reference(s):

1. PPM 5.3.1 Secondary Containment Control
2. PPM 5.0.10 Flowchart Training Manual
3. NEI 99-01 RCS Leak Rate PC Loss 3.C

ATTACHMENT 7.2: Fission Product Barrier Matrix and Bases

Barrier: Containment

Category: B. RCS Leak Rate

Degradation Threat: Potential Loss

Threshold:

None

ATTACHMENT 7.2: Fission Product Barrier Matrix and Bases

Barrier: Containment

Category: C. PC Conditions

Degradation Threat: Loss

Threshold:

UNPLANNED rapid drop in PC pressure following PC pressure rise

Basis:

Rapid UNPLANNED loss of primary containment pressure (i.e., not attributable to containment spray or condensation effects) following an initial pressure increase indicates a loss of primary containment integrity.

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

CGS Basis Reference(s):

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

ATTACHMENT 7.2: Fission Product Barrier Matrix and Bases

Barrier: Containment

Category: C. PC Conditions

Degradation Threat: Loss

Threshold:

PC pressure response not consistent with LOCA conditions

Basis:

This indicator is considered to be a loss of both the RCS and PC barriers.

Normal LOCA conditions are drywell pressure rising with wetwell pressure following. Primary containment or drywell pressure responses not consistent with LOCA conditions indicate a loss of the primary containment barrier. This may be noticed as a decrease in drywell pressure when no operator action (e.g., starting drywell cooling fans) has been taken. It would also include a failure of the drywell pressure to increase as expected during a LOCA. Also, a loss of suppression function in conjunction with a LOCA would indicate a loss of the primary containment barrier. Exceeding Pressure Suppression Pressure (PSP) is an indication of loss of pressure suppression function.

Primary containment pressure should increase as a result of mass and energy release into the primary containment from a LOCA. Thus, primary containment pressure not increasing under these conditions indicates a loss of primary containment integrity.

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

CGS Basis Reference(s):

1. FSAR Section 6.2.1.1.3.3
2. FSAR Figure 6.2-3
3. FSAR Table 6.2-5
4. FSAR Table 6.2-1
5. NEI 99-01 Primary Containment Conditions PC Loss 1.B

ATTACHMENT 7.2: Fission Product Barrier Matrix and Bases

Barrier: Containment

Category: C. PC Conditions

Degradation Threat: Potential Loss

Threshold:

PC pressure GT 45 psig

Basis:

If this threshold is exceeded, a challenge to the primary containment structure has occurred because assumptions used in the accident analysis are no longer valid and an unanalyzed condition exists (ref. 1, 2). This constitutes a Potential Loss of the Containment barrier even if a containment breach has not occurred.

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

CGS Basis Reference(s):

1. FSAR Table 6.2-1
2. FSAR Section 6.2
3. NEI 99-01 Primary Containment Conditions PC Potential Loss 1.A

ATTACHMENT 7.2: Fission Product Barrier Matrix and Bases

Barrier: Containment

Category: C. PC Conditions

Degradation Threat: Potential Loss

Threshold:

Explosive mixture exists inside PC (H_2 GE 6% and O_2 GE 5%)

Basis:

Explosive (deflagration) mixtures in the primary containment are assumed to be elevated concentrations of hydrogen and oxygen. BWR industry evaluation of hydrogen generation for development of EOPs/SAGs indicates that any hydrogen concentration above minimum detectable is not to be expected within the short term. Post-LOCA hydrogen generation primarily caused by radiolysis is a slowly evolving, long-term condition. Hydrogen concentrations that rapidly develop are most likely caused by metal-water reaction. A metal-water reaction is indicative of an accident more severe than accidents considered in the plant design basis and would be indicative, therefore, of a potential threat to primary containment integrity. Hydrogen concentration of approximately 6% is considered the global deflagration concentration limit (ref. 1).

Except for brief periods during plant startup and shutdown, oxygen concentration in the primary containment is maintained at insignificant levels by nitrogen inerting. The specified values for this Potential Loss threshold are the minimum global deflagration concentration limits (6% hydrogen and 5% oxygen, ref. 2) and readily recognizable because 6% hydrogen is well above the EOP flowchart entry condition (ref. 3). The minimum global deflagration hydrogen/oxygen concentrations (6%/5%, respectively) require intentional primary containment venting, which is defined to be a Loss of Containment (PC Integrity or Bypass).

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

CGS Basis Reference(s):

1. BWROG EPG/SAG Revision 2, Sections PC/G
2. PPM 5.7.1 RPV and Primary Containment Flooding SAG, Table 19
3. PPM 5.2.1 Primary Containment Control
4. FSAR Section 7.5.1.5.4
5. PPM 5.0.10 Flowchart Training Manual
6. PPM 4.814.J1 814.J1 Annunciator Panel Alarms, 2-2
7. PPM 4.814.J2 814.J2 Annunciator Panel Alarms, 2-2
8. NEI 99-01 Primary Containment Conditions PC Potential Loss 1.B

ATTACHMENT 7.2: Fission Product Barrier Matrix and Bases

Barrier: Containment

Category: C. PC Conditions

Degradation Threat: Potential Loss

Threshold:

WW temperature and RPV pressure cannot be maintained below the HCTL

Basis:

The HCTL is given in EOP flowchart Figure C (ref. 1). This is the only instance in which the threshold could be met.

Heat Capacity Temperature Limit (HCTL) is the highest Wetwell temperature from which emergency RPV depressurization will not exceed:

- Capability of the Wetwell, and equipment within the Wetwell which may be required to operate, when the RPV is pressurized
- Pressure Limit (PCPL), while the rate of energy transfer from the RPV to the Containment is GT the capacity of the Containment vent

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

CGS Basis Reference(s):

1. PPM 5.2.1 Primary Containment Control
2. NEI 99-01 Primary Containment Conditions PC Potential Loss 1.C

ATTACHMENT 7.2: Fission Product Barrier Matrix and Bases

Barrier: Containment

Category: D. PC Radiation / RCS Activity

Degradation Threat: Loss

Threshold:

None

ATTACHMENT 7.2: Fission Product Barrier Matrix and Bases

Barrier: Containment

Category: D. PC Radiation / RCS Activity

Degradation Threat: Potential Loss

Threshold:

Containment Radiation Monitor CMS-RIS-27E or CMS-RIS-27F reading GT 14,000 R/hr

Basis:

Four high range area radiation detectors (CMS-RE-27A, B, E and F) are installed in the drywell. CMS-RE-27A and -27B are located in the bioshield wall at elevations 522′ and 525′, azimuth 60° and 297°, respectively. CMS-RE-27E and -27F are located inside containment at elevation 515′, azimuth 290° and 51.5°, respectively. The companion containment radiation monitors (CMS-RIS-27A, B, E and F) are located on RAD Boards 22 and 23 in the Main Control Room. (ref. 1)

The threshold value was calculated assuming the instantaneous release and dispersal of the reactor coolant noble gas and iodine inventory associated with 20% fuel clad damage into the drywell atmosphere. Evaluation of detector location, geometry and anticipated response suggests CMS-RIS-27E or F will provide the desired response to a given radiation source in the drywell and are, therefore, identified as the preferred monitors for evaluating this Containment barrier Potential Loss threshold. (ref. 2)

The radiation monitor reading corresponds to an instantaneous release of all reactor coolant mass into the primary containment, assuming that 20% of the fuel cladding has failed. This level of fuel clad failure is well above that used to determine the analogous Fuel Clad Barrier Loss and RCS Barrier Loss thresholds.

NUREG-1228, *Source Estimations During Incident Response to Severe Nuclear Power Plant Accidents*, indicates the fuel clad failure must be greater than approximately 20% in order for there to be a major release of radioactivity requiring offsite protective actions. For this condition to exist, there must already have been a loss of the RCS Barrier and the Fuel Clad Barrier. It is therefore prudent to treat this condition as a potential loss of containment which would then escalate the emergency classification level to a General Emergency.

CGS Basis Reference(s):

1. TM-2117 TSG – Core Thermal Engineer, Attachment 4.2
2. Calculation NE-02-94-57
2. NEI 99-01 Primary Containment Radiation Fuel Clad Potential Loss 1.D

Barrier: Containment

Category: E. PC Integrity or Bypass

Degradation Threat: Loss

Threshold:

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

Basis:

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

Leakage into a closed system is to be considered only if the closed system is breached and thereby creates a significant pathway to the environment. Examples include unisolable main steam line or RCIC steam line breaks, unisolable RWCU system breaks, and unisolable PC vent paths. ~~If the main condenser is available with an unisolable main steam line, there may be releases through the steam jet air ejectors and gland seal exhausters. These pathways are monitored, however, and do not meet the intent of a nonisolable release path to the environment. These minor releases are assessed using the Category R, Abnormal Rad Release / Rad Effluent, EALs.~~

PPM 5.2.1, Primary Containment Control, may specify primary containment venting and intentional bypassing of the containment isolation valve logic, even if offsite radioactivity release rate limits are exceeded (ref. 1). Under these conditions with a valid containment isolation signal, the Containment barrier should be considered lost.

The use of the modifier “direct” in defining the release path discriminates against release paths through interfacing liquid systems or minor release pathways, such as instrument lines, not protected by the Primary Containment Isolation System (PCIS).

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.

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

CGS Basis Reference(s):

1. PPM 5.2.1 Primary Containment Control
2. NEI 99-01 Primary Containment Isolation Failure PC Loss 3.A

ATTACHMENT 7.2: Fission Product Barrier Matrix and Bases

Barrier: Containment

Category: E. PC Integrity or Bypass

Degradation Threat: Loss

Threshold:

Intentional PC venting per EOPs

Basis:

EOP flowcharts (PPM 5.2.1, Primary Containment Control) may specify primary containment venting and intentional bypassing of the containment isolation valve logic, even if offsite radioactivity release rate limits are exceeded (ref. 1). The threshold is met when the operator begins venting the primary containment in accordance with EOP Support Procedures (PPM 5.5.14 or PPM 5.5.15) or ABN-CONT-VENT, not when actions are taken to bypass interlocks prior to opening the vent valves (ref. 2, 3, 4). Purge and vent actions specified in PPM 5.2.1 to control primary containment pressure below the drywell high pressure scram setpoint or to lower hydrogen concentration does not meet this threshold because such action is only permitted if offsite radioactivity release rates will remain below the ODCM RFO limits (ref. 1).

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

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

CGS Basis Reference(s):

1. PPM 5.2.1 Primary Containment Control
2. PPM 5.5.14 Emergency Wetwell Venting
3. PPM 5.5.15 Emergency Drywell Venting
4. ABN-CONT-VENT
5. NEI 99-01 Primary Containment Isolation Failure PC Loss 3.B

ATTACHMENT 7.2: Fission Product Barrier Matrix and Bases

Barrier: Containment

Category: E. PC Integrity or Bypass

Degradation Threat: Potential Loss

Threshold:

None

ATTACHMENT 7.2: Fission Product Barrier Matrix and Bases

Barrier: Containment

Category: F. Emergency Director Judgment

Degradation Threat: Loss

Threshold:

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

Basis:

The Emergency Director judgment threshold addresses any other factors relevant to determining if the Containment barrier is lost. Such a determination should include imminent barrier degradation, barrier monitoring capability and dominant accident sequences.

- Imminent barrier degradation exists if the degradation will likely occur within two hours based on a projection of current safety system performance. The term “imminent” refers to recognition of the inability to reach safety acceptance criteria before completion of all checks.
- Barrier monitoring capability is decreased if there is a loss or lack of reliable indicators. This assessment should include instrumentation operability concerns, readings from portable instrumentation and consideration of offsite monitoring results.
- Dominant accident sequences lead to degradation of all fission product barriers and likely entry to the EOPs. The Emergency Director 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 Director in determining whether the Containment Barrier is lost.

CGS Basis Reference(s):

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

ATTACHMENT 7.2: Fission Product Barrier Matrix and Bases

Barrier: Containment

Category: F. Emergency Director Judgment

Degradation Threat: Potential Loss

Threshold:

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

Basis:

The Emergency Director judgment threshold addresses any other factors relevant to determining if the Containment barrier is potentially lost. Such a determination should include imminent barrier degradation, barrier monitoring capability and dominant accident sequences.

- Imminent barrier degradation exists if the degradation will likely occur within two hours based on a projection of current safety system performance. The term “imminent” refers to recognition of the inability to reach safety acceptance criteria before completion of all checks.
- Barrier monitoring capability is decreased if there is a loss or lack of reliable indicators. This assessment should include instrumentation operability concerns, readings from portable instrumentation and consideration of offsite monitoring results.
- Dominant accident sequences lead to degradation of all fission product barriers and likely entry to the EOPs. The Emergency Director 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 Director in determining whether the Containment Barrier is lost.

CGS Basis Reference(s):

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

ATTACHMENT 7.3: NOTES AND TABLES

Table 1 Sumps/Pool
• Any valid Hi-Hi level alarm on R-1 through R-5 sumps • EDR GE 25 GPM • FDR GE 10 GPM • Wetwell level rise • Observation of UNISOLABLE RCS leakage

Table 2 AC Power Sources
Offsite • Startup Transformer TR-S • Backup Transformer TR-B • Backfeed 500 KV power through Main Transformers (if already aligned in modes 4, 5, def **only**)
Onsite • DG1 • DG2 • Main Generator via TR-N1/N2

ATTACHMENT 7.3: NOTES AND TABLES

Table 3 Effluent Monitor Classification Thresholds

	Release Point	Monitor	~~GE~~General	SAE	Alert	UE
Gaseous	Reactor Building Exhaust	PRM-RE-1B (I)	----	----	----	6.00E+03 cps
		PRM-RE-1C (H)	2.00E+04 cps	2.00E+03 cps	4.00E+02 cps	----
	Turbine Building Exhaust	TEA-RIS-13	8.35E-02 µCi/cc	8.35E-03 µCi/cc	8.35E-04 µCi/cc	1.02E-04 µCi/cc
	Radwaste Building Exhaust	WEA-RIS-14	3.45E-01 µCi/cc	3.45E-02 µCi/cc	3.45E-03 µCi/cc	1.98E-03 µCi/cc
Liquid	Radwaste Effluent	FDR-RIS-606	----	----	----	2 X HI-HI alarm
	TSW Effluent	TSW-RIS-5	----	----	----	3.00E-05 µCi/cc
	Service Water Process A Service Water Process B	SW-RIS-604 SW-RIS-605	----	----	----	1.00E+02 cps 1.00E+02 cps

Table 4 Communication Methods

System	Onsite	ORO	NRC
Plant Public Address (PA) System	X		
Plant Telephone System	X	X	
Plant Radio System Operations and Security Channels	X		
Offsite calling capability from the Control Room via direct telephone		X	X
Long distance calling capability on the commercial phone system		X	X

ATTACHMENT 7.3: NOTES AND TABLES

Table 5 Safe Shutdown Areas

- Vital portions of the Rad Waste/Control Building:
  - 467' elevation vital island
  - 487' elevation cable spreading room
  - Main Control Room and vertical cable chase
  - 525' elevation HVAC area
- Reactor Building
- Vital portions of the Turbine Building
  - DEH pressure switches
  - RPS switches on turbine throttle valves
  - Main steam line radiation monitors
  - Turbine Building ventilation radiation monitors
  - Main steam line piping up to MS-V-146 and the first stop valves
- Standby Service Water Pump Houses
- Diesel Generator Building

Table 7 RCS Heat-up Duration Thresholds

RCS Status	CONTAINMENT CLOSURE Status	Heat-up Duration
Intact	N/A	60 min.*
Not intact	established	20 min.*
	not established	0 min.

* If an RCS heat removal system is in operation within this time frame and RCS temperature is being reduced, the EAL is **not** applicable.

ATTACHMENT 7.3: NOTES AND TABLES

Table 8 Hazardous Events
• Seismic event • Internal or external FLOODING event • High winds • Tornado strike • FIRE • EXPLOSION • Volcanic ash fallout • Other events with similar hazard characteristics as determined by the Shift Manager

Table 9 Safe Operation & Shutdown Areas	
Room/Area	**Mode Applicability**
RW 467’ Radwaste Control Room (RHR flush to RW tanks)	3
RW 467’ Vital Island (RHR-V-9 disconnect)	3
RB 422’ B RHR Pump Rm (local pump temperatures)	3
RB 454’ B RHR Pump Rm (operate RHR-V-85B)	3

Table 10 Safety System Parameters
• Reactor power • RPV level • RPV pressure • Primary containment pressure • Wetwell level • Wetwell temperature

ATTACHMENT 7.3: NOTES AND TABLES

Table 11 Significant Transients
• Reactor scram • Runback GT 25% thermal reactor power • Electrical load rejection GT 25% full electrical load • ECCS injection • Thermal power oscillations GT 10%

ATTACHMENT 7.3: NOTES AND TABLES

Table 12 Notes
Note 1: The Emergency Director should declare the event promptly upon determining that time limit has been exceeded, or will likely be exceeded
Note 2: If an ongoing release is detected and the release start time is unknown, assume that the release duration has exceeded the specified time limit
Note 3: If the effluent flow past an effluent monitor is known to have stopped, indicating that the release path is isolated, the effluent monitor reading is no longer VALID for classification purposes
Note 4: The pre-calculated effluent monitor values presented in EALs RA1.1, RS1.1 and RG1.1 should be used for emergency classification assessments until the results from a dose assessment using actual meteorology are available
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 scram action is any operator action, or set of actions, which causes the control rods to be rapidly inserted into the core, and does **not** include manually driving in control rods or implementation of boron injection strategies
Note 9: If the affected SAFETY SYSTEM (or component) was already inoperable or out of service before the event occurred, then **no** emergency classification is warranted as long as the damage was limited to the affected SAFETY SYSTEM (or component)

ATTACHMENT 7.4: Safe Operation & Shutdown Areas Table 9 Bases

Background

NEI 99-01 Revision 6 ICs AA3 and HA5 prescribe declaration of an Alert based on impeded access to rooms or areas (due to either area radiation levels or hazardous gas concentrations) where equipment necessary for normal plant operations, cooldown or shutdown is located. These areas are intended to be plant operating mode dependent. Specifically the Developers Notes For AA3 and HA5 states:

> *The "site-specific list of plant rooms or areas with entry-related mode applicability identified" should specify those rooms or areas that contain equipment which require a manual/local action as specified in operating procedures used for normal plant operation, cooldown and shutdown. Do not include rooms or areas in which actions of a contingent or emergency nature would be performed (e.g., an action to address an off-normal or emergency condition such as emergency repairs, corrective measures or emergency operations). In addition, the list should specify the plant mode(s) during which entry would be required for each room or area.*
>
> *The list should not include rooms or areas for which entry is required solely to perform actions of an administrative or record keeping nature (e.g., normal rounds or routine inspections).*

Further, as specified in IC HA5:

> *The list need not include the Control Room if adequate engineered safety/design features are in place to preclude a Control Room evacuation due to the release of a hazardous gas. Such features may include, but are not limited to, capability to draw air from multiple air intakes at different and separate locations, inner and outer atmospheric boundaries, or the capability to acquire and maintain positive pressure within the Control Room envelope.*

ATTACHMENT 7.4: Safe Operation & Shutdown Areas Table 9 Bases

Table 9 Bases

The following table lists the locations into which an operator may be dispatched in order to safely shut down the reactor and reach cold shutdown conditions in accordance with plant procedures. The reason for these in-plant actions has been evaluated and a determination made whether or not the actions, if not performed, would prevent achieving cold shutdown. The minimum set of in-plant actions, associated locations, and operating modes to shut down and cool down the reactor are identified as "yes". These comprise the rooms/areas to be included in EAL Table 9.

Building	Elevation	Room	Modes	Reason	If not performed, prevents cooldown/shutdown?
TG	441	Booster pump area	1,3,4	Condensate Booster Pump S/D per SOP-COND-SHUTDOWN	No
		RFT Area	1,3	RFT S/D per SOP-RFT-SHUTDOWN	No
		IR-9 Area	1	Verify Desuperheater pressure per SOP-MT-SHUTDOWN	No
		Mech Vacuum Pmp Rm	3	Mech Vacuum Pmp Start per SOP-AR-SHUTDOWN	No, can break vacuum and cool down with SRVs
		Mech Vacuum Pmp Rm	3	Mech Vacuum Pmp Stop per SOP-AR-START	No
		OG Preheater Rm	3	OG System S/D per SOP-OG-SHUTDOWN	No
		Gland Exh Condenser Area	3	OG System S/D per SOP-OG-SHUTDOWN	No
		H2 valve station	1,3,4	H2 makeup to Mn Generator per SOP-H2/CO2-OPS	No
	501	MT Turning Gear Area	1	Place MT on Turning Gear per SOP-MT-START	No
CW Pump House	n/a	CW Pmp Area	1	CW Pmp S/D per SOP-CW-SHUTDOWN	No
		Towers and CW Basin	1	Monitor water level per SOP-CW-SHUTDOWN	No

ATTACHMENT 7.4: Safe Operation & Shutdown Areas Table 9 Bases

Building	Elevation	Room	Modes	Reason	If not performed, prevents cooldown/shutdown?
RW	467	Radwaste Control Room	1,3	Remove CFDs from service per SOP-CFD-SHUTDOWN	No
			3	Align RW tanks to receive RHR water per SOP-RHR-SDC	**Yes, RWCR operator will need to align Radwaste tanks to accept RHR SDC flush water**.
		Vital Island	3	Close disc for RHR-V-9 per SOP-RHR-SDC	**Yes, Disconnect for RHR-V-9 is normally left open during power operations**.
	525	Communication Rm	4	Check Oscillograph per PPM 3.2.1	No
TMU	n/a	TMU Pump Area	1	TMU Pmp Shutdown per SOP-TMU-SHUTDOWN	No
Switchyard	n/a	500KV MODs	1	Open MODs per SOP-MT-SHUTDOWN	No
Rx Bldg	422	B RHR Pump Rm	3	RHR Pump local temperature reading per SOP-RHR-SDC	**Yes, local readings of RHR pump taken prior to and during flush to ensure minimal delta-T is established**
	441	Railroad Bay	1	CIA N2 Bottle Change out per SOP-CIA-OPS	No, Many installed bottles, infrequent task
	454	B RHR Pump Rm	3	Cycle RHR-V-85B for flush per SOP-RHR-SDC	**Yes, valve must be cycled to perform RHR SDC line flush**
	501	HCU Area	1	HCU Charging per SOP-CRD-HCU	No, infrequent task
	548	B RHR Valve Rm	3	Vent RHR system post flush per SOP-RHR-SDC	No, vent not necessary to enter SDC
	572	B RHR HX Rm	3	Vent RHR system post flush per SOP-RHR-SDC	No, vent not necessary to enter SDC

ATTACHMENT 7.4: Safe Operation & Shutdown Areas Table 9 Bases

Table 9 Results

Table 9 Safe Operation & Shutdown Areas	
Room/Area	**Mode Applicability**
RW 467’ Radwaste Control Room (RHR flush to RW tanks)	3
RW 467’ Vital Island (RHR-V-9 disconnect)	3
RB 422’ B RHR Pump Rm (local pump temperatures)	3
RB 454’ B RHR Pump Rm (operate RHR-V-85B)	3

Plant Operating Procedures Reviewed

1. PPM 3.2.1 NORMAL PLANT SHUTDOWN
2. SOP-FWH-SHUTDOWN
3. SOP-MSR-OPS
4. SOP-CW-SHUTDOWN
5. SOP-COND-SHUTDOWN
6. SOP-CFD-SHUTDOWN
7. SOP-TMU-SHUTDOWN
8. SOP-AS-START
9. SOP-SS-OPS
10. SOP-RFT-SHUTDOWN
11. SOP-RFT-OPS
12. SOP-AR-SHUTDOWN
13. SOP-MT-SHUTDOWN
14. SOP-CW-OPS
15. SOP-OG-SHUTDOWN
16. SOP-AR-START
17. SOP-MT-START
18. OSP-RHR-M102
19. SOP-RHR-SDC
20. SOP-RCIC-SHUTDOWN
21. SOP-SS-SHUTDOWN
22. SOP-H2/CO2-OPS
23. SOP-CIA-OPS
24. SOP-CRD-HCU

ATTACHMENT 7.5: Columbia Generating Station Emergency Classification Chart Distribution

NOTE: The Emergency Classification Chart is provided in a separate, controlled distribution to the following locations:

Location	No. Of Copies
Control Room (MCR)	2 half size
Control Room Simulator	2 half size
Technical Support Center (TSC)	2 half size, 1 full size
Alternate TSC	2 half size, 1 full size
Emergency Operations Facility (EOF)	2 half size, 2 full size
Alternate EOF	2 half size
Joint Information Center (JIC)	1 half size
Remote Shutdown Room	1 half size
Simulator Remote S/D Room	1 half size

NOTE: Information Only charts should be provided to the following locations:

Benton County EOC	1 half size
Franklin County EOC	1 half size
Washington State EOC	1 half size
Grant County EOC	1 half size
Adams County EOC	1 half size
Yakima County EOC	1 half size