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

Oyster Creek Generating Station
Facility Operating License No. DPR-16
NRC Docket No. 50-219

Subject: Exelon/AmerGen Supplement to Submittal to Revise Oyster Creek Emergency Action Levels Developed from NEI 99-01

References:
(1) Letter from Pamela B. Cowan (Exelon Generation Company, LLC) to U. S. Nuclear Regulatory Commission, dated August 14, 2006

(2) Letter from J. Honcharik (U. S. Nuclear Regulatory Commission) to Christopher M. Crane (Exelon Generation Company, LLC), dated November 2, 2006

(3) Letter from Pamela B. Cowan (Exelon Generation Company, LLC) to U. S. Nuclear Regulatory Commission, dated December 1, 2006

In Reference (1), Exelon Generation Company, LLC (Exelon) and AmerGen Energy Company, LLC (AmerGen) submitted a request for a change to the Exelon and AmerGen Emergency Action Levels (EALs) for the Exelon/AmerGen plants including Oyster Creek Generating Station. The proposed revision supports a change to Oyster Creek's EAL schemes currently based on NUMARC/NESP-007, Revision 2 "Methodology for Development of Emergency Action Levels" to a scheme based on NEI 99-01, "Methodology for Development of Emergency Action Levels" (Revision 4, January 2003).

In Reference (2), the U. S. Nuclear Regulatory Commission (NRC) requested additional information regarding the EAL revision request. Exelon/AmerGen provided a response to the request in the Reference (3) letter. Per a teleconference on December 13, 2006, the NRC identified a typographical error in the referenced Applicability Mode in Oyster Creek's "typed" version of EAL MU8. The error has been corrected and is provided as Attachment 1 to this letter. Per NRC request, Exelon/AmerGen is re-submitting a complete response to the

AX45

reference (2) Request for Additional Information for Oyster Creek (Attachment 7 of Reference 3). The proposed revision contained in this submittal does not reduce the capability to meet the standards in 10CFR 50.47.

Should you have any questions concerning this letter, please contact Mr. Doug Walker at (610) 765-5726.

Very truly yours,

Pamela B. Cowan

Pamela B. Cowan
Director – Licensing & Regulatory Affairs
AmerGen Energy Company, LLC

Attachments: 1) Re-submittal of Enclosure 7 Oyster Creek RAI Response

cc: Regional Administrator – NRC Region I
NRC Senior Resident Inspector – Oyster Creek
NRC Project Manager, NRR – Oyster Creek (w/o Attachments)
R. R. Janati, Commonwealth of Pennsylvania
File No. 06042

ATTACHMENT 1

EXELON/AMERGEN

SUPPLEMENTAL INFORMATION REGARDING THE

OYSTER CREEK GENERATING STATION EAL REVISION

"Oyster Creek RAI Response"

ENCLOSURE 7A

OYSTER CREEK GENERATING STATION ANNEX

EP-OC-1010

"Response to the Request for Additional Information"

RESPONSE TO THE REQUEST FOR ADDITIONAL INFORMATION
Oyster Creek Generating Station

1. Several inconsistencies exist related to accurately reflecting the correct Operating Modes applicable to EALs. For example, for Oyster Creek EAL MG3 and MS3, the Operating Mode applicability is Mode 1 only. However, for all other, it is Modes 1 and 2. Perform a review for inconsistencies in the submittal and correct these discrepancies. (NEI EAL: Various / Exelon EAL: Various)

   Response:

   Oyster Creek's EALs were reviewed and there was one typographical error identified in the submitted Basis for MU8 that has been corrected. However, it should be noted that Oyster Creek, per Technical Specifications, has 4 Modes plus Defuel. Oyster Creek's Mode 1 Power Operations is the same as NEI's Mode 1 Power Operations and Mode 2 Startup. Please note comparison below:

Oyster Creek	NEI
Mode 1 – Power Operations	Mode 1 and Mode 2
Mode 2 – Hot Shutdown	Mode 3
Mode 3 – Cold Shutdown	Mode 4
Mode 4 – Refuel	Mode 5
Mode D – Defueled	Defuel

2. The containment barrier EAL "Other site-specific indications" stated in NEI 99-01 is not addressed. The bases for this EAL in NEI 99-01 states that the EAL is intended to cover other site-specific indications. Revise the proposed EALs and their bases to provide the other site-specific indications identified in NEI 99-01 bases or provide further justification for having none available. (NEI EAL: FPB / Exelon EAL: FPB)

   Response:

   A review of existing SER EAL thresholds was performed to ensure the EAL Thresholds should not be considered in the "Other" category. Justifications were provided for Oyster Creek in the submittal package (see justification 2.2.7 and 2.2.9 of the submittal package for OC). Oyster Creek did include one EAL which would fall within the 'Other Category." This EAL was a Loss of RCS threshold which reads ' UNISOLABLE Isolation Condenser Tube rupture or line break.' The inclusion of this EAL is appropriate based on the design of Oyster Creek.

   In addition, a review was performed of NEI 99-01 submittals for similar design type plants to determine if additional means or indications were being utilized in determining Barrier status. Based on this review no other EAL Thresholds were identified for inclusion in determining Barrier status. Exelon's proposed Fission Product Barrier EALs were found to be consistent with other previously approved NEI 99-01 EALs for similarly designed plants.

3. A threshold value "and drywell pressure rise due to RCS leakage" was added to the EAL RCS2.c. Provide justification for adding this threshold to the EAL. (NEI EAL: FB-BWR-RCS-L1 / Exelon EAL: FPB)

   Response:

   NEI 99-01 EAL for Drywell Pressure states:

   > *The (site-specific) drywell pressure is based on the drywell high pressure set point which indicates a LOCA by automatically initiating the ECCS or equivalent makeup system.*

   The use of the qualifier "*and drywell pressure rise due to RCS leakage*" ensures the EAL is classified based on a pressure that would be the result of a LOCA (RCS leakage) as specified in NEI. Other events could potentially result in the Drywell pressure setpoint being reached (e.g., loss of drywell cooling, improper containment venting/purging). These events would not be an indication of a RCS breach and therefore a classification for a loss of RCS would be inappropriate. Using the qualifier ensures the classification is properly declared for a loss of RCS. Use of this qualifier was found to be consistent with other previously approved NEI 99-01 EALs for similarly designed plants.

4. Explain why "response" in regards to drywell pressure was changed to "rise"? (NEI EAL: FPB Cont / Exelon EAL: FPB Cont 3.C)

   Response:

   EAL was revised to replace the word "rise" with "response" in accordance with NEI 99-01 wording. See mark-up of EAL basis provided in Enclosure C.

5. The threshold value for a Loss (RPV Level) was changed from "<-30" to "<-20." Provide justification for this change. (NEI EAL: FPB Clad #2 / Exelon EAL: FPB Clad 1.a)

   Response:

   This RPV level value is based on the corresponding RPV level associated with Minimum Steam Cooling RPV Water Level (MSCWL). Oyster Creek revised the stations Emergency Operating Procedures the revise this value from –30 inches to -20 inches based on an analysis performed by General Electric. The EAL value was revised under 50.54q to maintain alignment with the stations Emergency Operating Procedures.

6. Clarify whether the proposed wind speed setpoint is within the calibrated range of the instrument used for determining wind speed. (NEI EAL: HA1, HU1 / Exelon EAL: HA5, HU5)

Response:

Oyster Creek revised the wind speed setpoint for HU5 and HA5 to greater than 99 mph such that the setpoint is within the calibrated range of the instrument and is readable.

7. Provide additional justification for the inclusion of this seismic alarm EAL threshold (i.e., why instrumentation alone is not sufficient to declare the seismic EAL). In addition, does this annunciate in the control room, and is it reliable for indicating the EAL? If it does not, use the guidance in NEI 99-01. In addition, explain why there are limits, e.g. 0.01, referenced in some EALs and not in others.

   Response:

   Oyster Creek does not have seismic monitors. See RAI #10.

8. Not Applicable for Oyster Creek.

9. Not Applicable for Oyster Creek.

10. How is a seismic event determined in ABN-38? Referencing procedures in the EAL does not allow for timely decision making in declaring an EAL. Provide the EAL threshold in a clear, concise manner in the body of the EAL in order to ensure timely decision making in declaring an EAL. (NEI EAL: <u>HA1</u> / Exelon EAL: <u>HA5.1</u>)

    Response:

    Since Oyster Creek does not have installed Seismic instrumentation, an alternate means for determining the Alert threshold must be used. ABN-38 provides the direction for plant shutdown based on the affects of the seismic event. In addition, to ensure a timely classification is determined, the basis provides the following guidance:

    A reactor scram is required by procedure ABN-38, Station Seismic Event, if:

    - *The seismic event affects safe plant operation by jeopardizing the availability of safety systems, systems required to complete safe shutdown, or causing spurious actuation of equipment, or*
    - *The Shift Manager determines it necessary to scram the Reactor to protect public safety.*

    Guidance from ABN-38 has been incorporated in the HA5.1 threshold. See EAL Basis in Enclosure C.

11. The word "may" has been removed from EAL1. Provide justification for this change. For consistency, review EAL Frequently Asked Questions (FAQ) 2006-024

Language specific to asphyxiates, and include this in the EAL. (NEI EAL: HA3 / Exelon EAL: HA7)

Response:

The word "may" is not included in the EAL threshold because the use of "may" is inconsistent with the NEI 99-01 basis which states "EAL #1 is met if measurement of toxic gas concentration results in a atmosphere that is IDLH". Additionally the use of "may" could result in inconsistent interpretation of the threshold. Exelon added the following guidance to the EAL bases to capture the intent of the word "may":

> *Declaration should not be delayed for confirmation from atmospheric testing if it is reasonable to conclude that the IDLH concentrations have been met (e.g. documented analysis, indication of personal ill effects from exposure, or operating experience with the hazards).*

EAL HA7 was revised to include asphyxiant gases in the EAL Threshold and provided a description of asphyxiant gases in the Basis section. See EAL Basis markup provided in Enclosure C and revised justification 2.1.8 provided in Enclosure B.

12. Provide additional justification for the inclusion of this seismic alarm EAL threshold (i.e., why instrumentation alone is not sufficient to declare the seismic EAL). In addition, does this annunciate in the control room, and is it reliable for indicating the EAL? If it does not, use the guidance in NEI 99-01. In addition, explain why there are limits, e.g. 0.01, referenced in some EALs and not in others.

Response:

Oyster Creek does not have seismic monitors. See RAI #10.

13. Not Applicable for Oyster Creek.

14. Explain the difference between HU5 (EAL Threshold #3 concerning physical damage) and HA5 (EAL Threshold #3 concerning visible damage). Since these EALs are similar (only differing in one word), explain how errors in declaration will be avoided, or follow NEI 99-01 guidance. (NEI EAL: HU1 / Exelon EAL: HU5.3)

Response:

Oyster Creek removed reference to Physical Damage and reworded proposed EALs to align with NEI 99-01 wording. See EAL Basis mark-up provided in Enclosure C.

15. EAL Threshold values specify "physical damage", while the EAL Threshold basis specify "visible damage." Explain why there is a change in terminology. (NEI EAL: HU2 / Exelon EAL: HU6.3)

Response:

Oyster Creek removed reference to Physical Damage and reworded proposed EALs to align with NEI 99-01 wording. See EAL Basis mark-up provided in Enclosure C and revised justification 2.1.7 provided in Enclosure B.

16. The EAL includes initiating conditions and EAL Threshold values when manual scram has not been successful, but provides no guidance on what constitutes a successful manual scram. Therefore, provide additional guidance in the EAL, or EAL Basis Document, explaining what constitutes successful manual scram actions. (NEI EAL: SA2, SS2, SG2 / Exelon EAL: MA3, MS3, MG3)

    Response:

    Oyster Creek revised MG3 and MS3 Threshold and Basis to define what constitutes a successful Manual Scram. In addition the EAL Threshold was revised to specify that actions must be taken from Reactor Console. See EAL Basis mark-up provided in Enclosure C and justification 2.2.13 provided in Enclosure B.

    For MA3, the threshold is independent of Manual Scram actions and additional guidance for explaining what constitutes successful manual scram actions is not needed.

17. What instrument is used to measure EAL #2 and does the scale of that instrument cover all conditions where the EAL might be declared. (NEI EAL: CA4 / Exelon EAL: MA5)

    Response:

    Oyster Creek validated the Threshold value for pressure change is within the range of Control Room digital pressure instrumentation and is readable.

18. Could this EAL (3.b) be declared if RPV level is unknown? (NEI EAL: SG1 / Exelon EAL: MG1)

    Response:

    Oyster Creek revised EAL Threshold 3.b to now read *'RPV level cannot be determined to be above TAF (0")'*. This revision takes into account 'RPV level unknown." See EAL Basis mark-up provided in Enclosure C.

19. Not Applicable for Oyster Creek.

20. For consistency, review EAL Frequently Asked Questions (FAQ) 2006-016 wording and implement this in the EAL. (NEI EAL: SG1 / Exelon EAL: MG1)

Response:

Oyster Creek reviewed FAQ 2006-016 for applicability. Oyster Creek uses the phrase 'Essential Buses' in the IC in accordance with NEI 99-01 and FAQ 2006-016. Oyster Creek also used site-specific terminology for Buses in the Threshold as allowed by FAQ 2006-016. The EAL, as written, is consistent with FAQ 2006-016 and no additional changes are required.

21. Not Applicable for Oyster Creek.

22. Proposed EAL MS8 2.b. eliminates one of the two conditions that indicate core uncovery per the NEI 99-01 scheme without proposing an alternative. Provide the either the conditions as identified in NEI 99-01, or provide an alternative with justification, or justify why these conditions are not needed. (NEI EAL: CS1 / Exelon EAL: MS8)

    Response:

    Oyster Creek revised their EAL to align with NEI 99-01. See EAL Basis mark-up provided in Enclosure C and revised justification 2.1.3 provided in Enclosure B.

23. Proposed EAL FAQ 2006-005 describes the 30-minute threshold as not applicable to this EAL. For consistency, review this EAL FAQ for implementation. NEI 99-01 1.b uses core uncovery, however, the Exelon submittal uses loss of RPV inventory. Clarify how loss of RPV inventory meets the intent of the NEI 99-01 EAL. (NEI EAL: CS2 / Exelon EAL: MS9)

    Response:

    Oyster Creek revised their EAL to remove the 30-minute time period contained in NEI 99-01 Bases and revised the EAL to be consistent with NEI 99-01 threshold and FAQ 2006-005. See revised justification 2.1.4 provided in Enclosure B and EAL Basis mark-up provided in Enclosure C.

24. For consistency, review EAL Frequently Asked Questions (FAQ) 2006-011 wording and implement this in the EAL. (NEI EAL: CS2 / Exelon EAL: MS9)

    Response:

    The Oyster Creek EAL submittal had previously incorporated this FAQ and submitted this as a deviation. There are no additional changes required as a result of this FAQ.

25. For consistency, review EAL Frequently Asked Questions (FAQ) 2006-014 wording and implement this in the EAL. (NEI EAL: CU1 / Exelon EAL: MU8)

Response:

Oyster Creek revised MU8 to align with the wording in FAQ 2006-014. See EAL Basis mark-up provided in Enclosure C and revised justification 2.1.1 provided in Enclosure B.

26. Describe the basis for the EAL Threshold values in EAL #2. (NEI EAL: CU1 / Exelon EAL: MU8)

    Response:

    Oyster Creek revised the Basis for MU8 to state the threshold value corresponds to the RPS low-level scram setpoint. See EAL Basis mark-up provided in Enclosure C and justification 2.1.1 provided in Enclosure B.

27. RCS Barrier EAL 4. - EAL basis information appears to apply for a PWR only. Therefore, provide the EAL basis for a BWR, which is applicable to Clinton.

    Response:

    The justification for RCS Barrier EAL 4 has been corrected to include BWR description. See revised justification 2.1.6 provided in Enclosure B.

28. Provide additional justification for the elimination of this EAL, or propose an alternative method. (NEI EAL: RCS#4 / Exelon EAL: None)

    Response:

    Oyster Creek revised their proposed NEI 99-01 EALs to include this EAL. See EAL Basis mark-up provided in Enclosure C and revised justification 2.1.6 provided in Enclosure B.

29. Are there any other effluent monitors? If there are other monitors, include them in the EAL, or follow the guidance in NEI 99-01. (NEI EAL: AA1 / Exelon EAL: RA1.1)

    Response:

    Oyster Creek revised their EALs to include monitors specified in the Discharge Permit. See EAL Basis mark-up provided in Enclosure C.

30. Not Applicable for Oyster Creek.

31. For consistency, review EAL Frequently Asked Questions (FAQ) 2006-025 wording and implement this in the EAL. (NEI EAL: AA3.2 / Exelon EAL: RA3.2)

Response:

Oyster Creek revised their EAL setpoint to align with the guidance provided in FAQ 2006-025. See EAL Basis mark-up provided in Enclosure C.

32. Are there any other effluent monitors? If there are other monitors, include them in the EAL, or follow the guidance in NEI 99-01. (NEI EAL: AU1 / Exelon EAL: RU1.1)

    Response:

    Oyster Creek revised their EALs to include monitors specified in the Discharge Permit. See EAL Basis mark-up provided in Enclosure C.

33. Exelon EAL: did not include the fuel transfer canal as indicated in NEI 99-01. Include the fuel transfer canal in the EAL or provide justification for not including it in the EAL. (NEI EAL: AU2 / Exelon EAL: RU2)

    Response:

    Oyster Creek revised their EALs to include fuel transfer canal.' See EAL Basis mark-up provided in Enclosure C.

34. Not Applicable for Oyster Creek.

35. Not Applicable for Oyster Creek.

ENCLOSURE 7B

OYSTER CREEK GENERATING STATION ANNEX

EP-AA-1010

"Revised EAL Technical Justification"

2.0 EVALUATION

The revision to the Oyster Creek EALs incorporates NEI 99-01, Revision 4, with noted deviations.

The operating modes for Oyster Creek are defined as:

Mode 1 – Power Operations

Mode 2 – Hot Shutdown

Mode 3 – Cold Shutdown

Mode 4 – Refuel

Mode D – Defueled

2.1 Summary of Deviations from Proposed EALs to NEI 99-01 Rev 4

The following identifies deviations from the NEI 99-01, Rev 4 EALs which have been identified as requiring prior NRC approval before implementing at Oyster Creek Generating Station. A detailed description of the changes and justifications for the changes are contained in the following section.

2.1.1 Deviation 1 – Reactor Coolant System Leakage in Cold Shutdown

NEI EAL: CU1 **Oyster Creek EAL:** MU8

Operational Modes: 3

Description of the Deviation

NEI 99-01 uses Technical Specification Leakage limits that are based on crack propagation which are not applicable in Cold Shutdown. Leakage cannot be accurately measured to the limits specified in CSD; therefore, use Unable to maintain RCS Vessel level above <site-specific value>.

NEI 99-01 Example EAL:

1. Unidentified or pressure boundary leakage greater than 10 gpm.
2. Identified leakage greater than 25 gpm.

Proposed Oyster Creek MU8 EAL

~~1. UNPLANNED loss of RPV Inventory per Table M5 indications.~~

~~**AND**~~

~~2.~~1. RPV level **cannot** be restored and maintained **> 139 inches TAF**.

~~**Table M5 – Indications of RCS Leakage**~~

~~Unexplained Identified or unidentified leakage rise~~
~~Unexplained Torus rise~~
~~Unexplained vessel make-up rise~~
~~Observation of leakage or Inventory loss~~

Technical Justification

System alignment changes and variations in instrumentation availability during the Cold Shutdown conditions, would result in the Operators being significantly challenged to accurately monitor and quantify RCS leakage to the limits as shown in NEI 99-01 CU1 RCS Leakage. Instrumentation normally available during power operations may not be available in Cold Shutdown due to maintenance activities. Alternate methods to quantify leakage (RCS makeup rate, sump levels, sump pump run times, etc.) are required.

While the example EAL is applicable in Cold Shutdown, it uses hot mode Technical Specification Leakage limits which not applicable in Cold Shutdown. These limits are based on crack propagation with RCS pressure as the major contributor, and in Cold Shutdown the reactor is shut down and depressurized so leak propagation is not the concern. This EAL is further explained in NEI 99-01 Appendix C. Appendix C states:

> *For BWRs, the shutdown EALS are intended to address concerns raised by NRC Office for Analysis and Evaluation of Operational Data (AEOD) Report AEOD/EG09, "BWR Operating Experience Involving Inadvertent Draining of the Reactor Vessel," dated August 8, 1986. This report states: "In broadest terms, the dominant causes of inadvertent reactor vessel draining are related to the operational and design problems associated with the residual heat removal system when it is entering into or exiting from the shutdown cooling mode.*

As an alternative method of detecting leakage in the Cold Shutdown condition, Oyster Creek has chosen to use ~~a combination of the general (non-quantitative) indicators of leakage mentioned above, in addition to~~ indicated vessel level reaching a point of inventory loss (RPS lo level scram setpoint) indicative of being outside of the normally expected operational range during Cold Shutdown conditions which would indicate a possible inadvertent reactor vessel draining event in progress or RPV leak. The use of RPV level is consistent with the Alert, Site Area Emergency and General Emergency escalation path EALs for RCS Leakage in Cold Shutdown or Refuel Modes.

This ~~combination~~ use of this indication~~s~~ meets the intent of the NEI 99-01 example EAL by providing clear ~~indicators~~ indications that inventory changes are occurring, and that the inventory changes have resulted in reaching a point in vessel level that is considered to be a potential degradation of the level of safety of the plant.

Supporting Information

Enclosure 8B contains the proposed EALs and EAL Bases, as well as the corresponding EAL ICs in the applicable IC logic grouping (i.e., the related UE, Alert, SAE, and GE).

MU8, MA8, MS8, MG8

Enclosure 8C contains a comparison table which highlights the proposed changes to the EALs. The table compares the current approved SER version of the EAL, the NEI 99-01 EAL, and the proposed deviation to the EAL.

Enclosure 8D contains the supporting documents referenced by the technical evaluation (i.e., calculations, simplified plant system drawings, technical Specification references, etc). The supporting document(s) referenced by this deviation include:

- Technical Specification 3.3.D Reactor Coolant System Leakage

2.1.3 **Deviation 3** – Loss of Inventory affecting Core DHR Capability

NEI EAL: CS1 **Oyster Creek EAL:** MS8

Operational Modes: 3

Description of the Deviation

Based on RAI, Oyster Creek revised this EAL to align with the wording provided in NEI 99-01.~~NEI EAL CS1 threshold #2.b specifies the use of rising sump level or erratic Source Range Monitor indication when vessel level is unknown. Proposed EAL MS8 threshold #2.b will use only erratic Source Range Neutron Monitor indication when vessel level is unknown.~~

Technical Justification

~~NEI EAL CS1~~

> ~~*2.b. RPV level cannot be monitored for > 30 minutes with a loss of RPV inventory as indicated by either:*~~
>
> - ~~*Unexplained {site-specific} sump and tank level increase*~~
> - ~~*Erratic Source Range Monitor Indication*~~

~~Proposed EAL MS8~~

> ~~*2.b. RPV level unknown for > 30 minutes with indication of core uncovery as evidenced by erratic Source Range Neutron Monitor indication.*~~

~~NEI EAL CS-1 Thresholds #1 and #2.a both appropriately establish conditions for a SAE with the loss or potential loss of two fission product barriers.~~

~~NEI EAL CS1 threshold #2.b however, could require a SAE based on only one fission product barrier challenged. Specifically, a sump level rise due to reactor coolant leakage concurrent with unknown RPV level could be indicative of loss of RCS only. This could result in an unintended escalation from the Alert (NEI EAL CA1 / Exelon EAL MA8) prior to meeting the conditions warranting a SAE.~~

~~The proposed EAL MS8 threshold #2.b has been written to maintain consistent application of a loss or potential loss of two fission product barriers for the SAE when containment is intact. By establishing a challenge to reactor coolant and fuel barriers (indication of core uncovery as evidenced by erratic Source Range Neutron Monitor indication) a consistent SAE threshold is sustained throughout the EAL series.~~

~~Continuity of the EAL string and escalation pathway is maintained from the Alert to the SAE to the GE. The Alert (MA8) is declared when vessel level cannot be~~

~~monitored for > 15 minutes and sump levels (or other inventory loss indications) are observed. When containment is intact, the SAE threshold would then occur when there are indications of core uncovery. The GE would then take place when any of the multiple indications of containment challenges are met.~~

~~This change provides a clear and appropriate escalation path and does not alter the intent of the EAL.~~

Supporting Information

~~Enclosure 8B contains the proposed EALs and EAL Bases, as well as the corresponding EAL ICs in the applicable IC logic grouping (i.e., the related UE, Alert, SAE, and GE).~~

~~MU8, MA8, MS8, MG8,~~

~~Enclosure 8C contains a comparison table which highlights the proposed changes to the EALs. The table compares the current approved SER version of the EAL, the NEI 99-01 EAL, and the proposed deviation to the EAL.~~

2.1.4 **Deviation 4** – Loss of RPV Level, Refueling

NEI EAL: CS2 **Oyster Creek EAL:** MS9

Operational Modes: 4

Description of the Deviation

CS2 as defined by NEI 99-01 was a significant revision to NESP-007 SS5 and Oyster Creeks MS7 EALs. NEI 99-01 states *'Recognition Category C completely replaces Recognition Category S when in Cold Shutdown and Refueling modes'*. To implement this EAL, Oyster Creek will delete NESP-007 MS7 EAL and replace it with the proposed MS9 EAL, which implements NEI CS2.

OC proposed EAL revises ~~both~~ NEI CS2 example EAL~~s~~ 2.b. ~~Each EAL change will be discussed.~~

~~Deviation a: CS2 1.b (OC MS9 1.b and 2.b) added a 30 minute time limit. Additionally, for CS2 1.b, OC removed reference core uncovery indication and replaced with indication of RCS leakage. This revision is being performed to provide the prescribed escalation path from CA2 to CS2 to CG1 as defined by the NEI 99-01 guidance.~~

~~Deviation b:~~ CS2 2.b (OC MS9 2.b) provides an alternate method to determine core uncovery.

Technical Justification

~~Deviation a:~~

~~NEI CS2 Example EAL 1.b reads:~~

> ~~*b. RPV level cannot be monitored with Indication of core uncovery as evidenced by one or more of the following:*~~
>
> - ~~*Containment High Range Radiation Monitor reading > {site-specific} setpoint*~~
> - ~~*Erratic Source Range Monitor Indication*~~
> - ~~*Other {site-specific} indications*~~

~~Oyster Creek proposes to revise EAL 1.b to read:~~

> ~~*b. RPV level unknown for > 30 minutes with a loss of RPV inventory per Table M5 indications.*~~

~~**Table M5 – Indications of RCS Leakage**~~
~~Unexplained Identified or unidentified leakage rise~~
~~Unexplained Torus rise~~
~~Unexplained vessel make-up rise~~
~~Observation of leakage or Inventory loss~~

~~This revision is being performed to provide the prescribed escalation path from CA2 to CS2 to CG1 as defined by the NEI 99-01 guidance. This escalation path is established by RPV level being unknown for > 15 minutes in CA2 to core uncovery for > 30 minutes in CG1. The basis for both CA2 and CS2 implies that the 30 minute time interval be included in CS2.~~

~~NEI CA2 EAL #2 reads:~~

> ~~2. a. Loss of RPV inventory as indicated by unexplained {site-specific} sump and tank level increase~~
>
> ~~AND~~
>
> ~~*b. RPV level cannot be monitored for > 15 minutes*~~

~~The basis for the Alert CA2 states:~~

> ~~*The 15-minute duration for the loss of level indication was chosen because it is half of the CS2 Site Area Emergency EAL duration. The 15-minute duration allows CA2 to be an effective precursor to CS2. Significant fuel damage is not expected to occur until the core has been uncovered for greater than 1 hour per the analysis referenced in the CS2 basis. Therefore this EAL meets the definition for an Alert.*~~

~~The basis for the Site Area Emergency CS2 states in part:~~

> ~~*Analysis in the above references indicates that core damage may occur within an hour following continued core uncovery therefore, conservatively, 30 minutes was chosen.*~~

~~As stated in the basis sections for CA2 and CS2 the 30-minute time interval is required and was added to CS2 EAL. This addition now establishes the escalation path from CA2 to CS2 based on time associated with the RPV level transient.~~

~~To establish the escalation path from CS2 to CG1, CS2 threshold was revised to remove the indications of core uncovery.~~

~~NEI General Emergency CG1 reads as follows:~~

> ~~*2. RPV Level:*~~
>
> ~~*a. less than TOAF for > 30 minutes*~~
>
> ~~***OR***~~
>
> ~~*b. cannot be monitored with Indication of core uncovery for > 30 minutes as evidenced by one or more of the following:*~~
>
> - ~~*Containment High Range Radiation Monitor reading > {site-specific} setpoint*~~
> - ~~*Erratic Source Range Monitor Indication*~~
> - ~~*Other {site-specific} indications*~~

~~To establish the escalation path from Site Area Emergency to General Emergency (CS2 to CG1), CS2 threshold was revised to remove the indications of core uncovery and replaced it with indications of leakage.~~

~~Since General Emergency CG1 implies that core uncovery has occurred following the RPV level status for > 30 minutes, it is appropriate to remove the core uncovery indications from CS2 to allow this as the escalation path to CG1. If core uncovery remained in CS2, the General Emergency would be declared concurrent with the Site Area Emergency.~~

~~Deviation b:~~

~~The following changes were made to NEI CS2 Example EAL #2. The first change added the 30-minute time interval for the same reasons as described above. The second change involves use of an alternate method to detect core uncovery than prescribed for in the NEI 99-01 guidance.~~

Oyster Creek proposes to revise EAL 2.b to read

2. **With** Secondary CONTAINMENT CLOSURE established:

   b. RPV level unknown ~~for **> 30 minutes**~~ with indication of core uncovery as evidenced by one or more of the following:

      - Refuel Floor Area Radiation Monitor C-10, North Wall reading **> 3 R/hr**.
      - Erratic Source Range Neutron Monitor indication..

NEI 99-01 basis for Site Area Emergency CS2 states:

> *As water level in the RPV lowers, the dose rate above the core will increase. The dose rate due to this core shine should result in up-scaled Containment High Range Monitor indication and possible alarm. EAL 2.b calculations should be performed to conservatively estimate a site-specific dose rate setpoint indicative of core uncovery (ie., level at TOAF). Additionally, post-TMI studies indicated that the installed nuclear instrumentation will operate erratically when the core is uncovered and that this should be used as a tool for making such determinations.*

NEI 99-01 specifies use of containment high range radiation monitor readings corresponding to core uncovery while in refueling mode as an alternative RPV level indication when RPV level instrumentation is unavailable. The criteria: "Fuel Floor ARM indicates >3 R/hr" has been used in lieu of "Containment High Range Radiation Monitor reading > {Site-Specific} setpoint." This is acceptable because the calculations performed to estimate the radiation dose rates indicated that levels would be below the lower range of the Containment High Range monitors, and the shielding between the anticipated "shine" from the top of the fuel and these monitors would make them ineffective. The radiation monitors selected have a better "view" of the area above the refueling vessel and a lower range that is well within the calculated range of the fuel uncovery event.

Calculation EP-AEL-0501, Estimation of Radiation Monitor Readings Indicating Core Uncovery During Refuel contains the assumptions and bases for electing to use the Refuel Floor ARMs instead of the Containment High Range Radiation Monitors.

The 30 minute time period specified in NEI Bases for this EAL was not incorporated.

Both changes described still maintain the intent of the EAL while providing proper escalation paths and reliable indications of core uncovery using an alternate but more appropriate method based on availability and limitations of installed instrumentation.

Supporting Information

Enclosure 8B contains the proposed EALs and EAL Bases, as well as the corresponding EAL ICs in the applicable IC logic grouping (i.e., the related UE, Alert, SAE, and GE).

MU9, MS9, MA8, MG8

Enclosure 8C contains a comparison table which highlights the proposed changes to the EALs. The table compares the current approved SER version of the EAL, the NEI 99-01 EAL, and the proposed deviation to the EAL.
Enclosure 8D contains the supporting documents referenced by the technical evaluation (i.e., calculations, simplified plant system drawings, technical Specification references, etc).

- EP-AEL-0501, "Estimation of Radiation Monitor Readings Indicating Core Uncovery During Refuel"

2.1.6 Deviation 6 – Fission Product Barrier - Loss of Reactor Coolant System (RCS) Containment / Drywell Radiation Levels

NEI EAL: FPB/Loss of RCS #4 **Oyster Creek EAL:** None

Operational Modes: 1, 2

Description of the Deviation

NEI 99-01 requires use of RCS activity at Technical Specification levels to determine RCS leakage by using Drywell Radiation Monitor readings.~~that won't register above background for Exelon plants. This EAL will not be implemented.~~

NEI 99-01 FPB Loss of RCS #4

4. Drywell Radiation Monitoring

Drywell Radiation monitor reading GREATER THAN (site-specific) R/hr.

Proposed EAL

1. Containment Hi Range Radiation Monitoring System (CHRRMS) **> 100 R/hr**.

AND

2. Indications of RCS leakage into the Drywell.

Technical Justification

For Loss of the RCS Fission Product Barrier as indicated by Containment Radiation Monitoring NEI 99-01, Rev 4 provides the following guidance:

The (site-specific) reading is a value which indicates the release of reactor coolant to the drywell. The reading should be calculated assuming the instantaneous release and dispersal of the reactor coolant noble gas and iodine inventory associated with normal operating concentrations (i.e., within T/S) into the drywell atmosphere. This reading will be less than that specified for Fuel Clad Barrier EAL #3. Thus, this EAL would be indicative of a RCS leak only. If the radiation monitor reading increased to that value specified by Fuel Clad Barrier EAL #3, fuel damage would also be indicated.

However, if the site specific physical location of the drywell radiation monitor is such that radiation from a cloud of released RCS gases could not be distinguished from radiation from adjacent piping and components containing elevated reactor coolant activity, this EAL should be omitted and other site specific indications of RCS leakage substituted.

> ~~*"The (site-specific) reading is a value which indicates the release of reactor coolant to the containment. The reading should be calculated assuming the instantaneous release and dispersal of the reactor coolant noble gas and iodine inventory associated with normal operating concentrations (i.e., within T/S) into the containment atmosphere. This reading will be less than that specified for Fuel Clad Barrier EAL #5. Thus, this EAL would be indicative of a RCS leak only. If the radiation monitor reading increased to that specified by Fuel Clad Barrier EAL #5, fuel damage would also be indicated."*~~
>
> ~~*"However, if the site specific physical location of the containment radiation monitor is such that radiation from a cloud of released RCS gases could not be distinguished from radiation from nearby piping and components containing elevated reactor coolant activity, this EAL should be omitted and other site specific indications of RCS leakage substituted."*~~

Calculations were performed to estimate the containment radiation reading resulting from a total release of the RCS inventory to the containment atmosphere with coolant activity at the Technical Specification limit. See Attachment 1 of EP-EAL-0611, Criteria for Choosing Containment Radiation Monitor Reading Indicative of Loss of RCS Barrier, for assumptions and results. These calculations use conservative assumptions (i.e., assumed value will produce higher dose than would actually be expected) for shielding, isotopic mix and exact monitor locations.

Based on calculation EP-EAL-0611, a conservative value of 100 R/Hr was selected as the threshold value. In addition the qualifier *"Indications of RCS leakage into the Drywell"* was added to ensure classifications were not made based on shine. This qualifier is required to enable the application of this EAL to be used as an indicator of RCS leakage, which should be classified at the Alert level.

~~Under expected conditions early in an event, the use of Containment Monitors to determine Loss of RCS is not reliable based on coolant activity at the Technical Specification limit for RCS isotopic inventory, because the normal dose rates in the drywell would mask the values resulting from release of RCS inventory. Normal drywell radiation readings during operation are about 2 R/hr compared to about 750 mR/hr calculated dose from the coolant release.~~

~~The NEI 99-01 FPB matrix already establishes a Potential Loss of the RCS based on leakage greater than 50 gpm. To establish a Loss of RCS EAL due to leakage, the leak must be greater than the 50 gpm for the Potential Loss of RCS and provide a drywell radiation monitor reading greater than the normally expected during plant operations.~~

~~Leaks of this magnitude would result in high-pressure conditions within the drywell due to the relatively small drywell volume. The high drywell pressure provides an alternate means to detect the barrier failure. A Loss of RCS is declared when drywell pressure exceeds the ECCS setpoint with indications of drywell leakage. Since the use of Drywell Radiation monitor readings to signify a RCS leak is~~

~~bounded by the Drywell High Pressure threshold, the threshold for RCS Loss is redundant and is not required.~~

~~This satisfies the NEI 99-01 statement that if the released RCS gases cannot be distinguished, then the "EAL should be omitted and other site specific indications of RCS leakage substituted." The use of Drywell pressure in this context is appropriate in meeting the intent of this EAL by providing a Loss of RCS criterion prior to reaching a Loss of Fuel Clad per the NEI 99-01 guidance.~~

Supporting Information

Enclosure 8B contains the proposed EALs and EAL Bases, as well as the corresponding EAL ICs in the applicable IC logic grouping (i.e., the related UE, Alert, SAE, and GE).

- Fission Product Barrier Matrix

Enclosure 8C contains a comparison table which highlights the proposed changes to the EALs. The table compares the current approved SER version of the EAL, the NEI 99-01 EAL, and the proposed deviation to the EAL.

Enclosure 8D contains the supporting documents referenced by the technical evaluation (i.e., calculations, simplified plant system drawings, technical Specification references, etc). The supporting document(s) referenced by this deviation include:

- Calculation EP-EAL-0611, Criteria for Choosing Containment Radiation Monitor Reading Indicative of Loss of RCS Barrier

2.1.7 Deviation 7 – Fire / Explosion

NEI EAL: HU2 **Oyster Creek EAL:** HU6

Operational Modes: 1, 2, 3, 4, D

Description of the Deviation

NEI 99-01 EAL H2 was revised to remove the word "contiguous" from threshold and to add another threshold value to clarify that the fire should have the potential to damage safety systems. This more accurately reflects intent of EAL and avoids unnecessary classifications for fires in areas that may be considered contiguous but there is no threat to any vital equipment or plant safety.

NEI 99-01 HU2 EAL:

FIRE in buildings or areas contiguous to any of the following (site-specific) areas not extinguished within 15 minutes of control room notification or verification of a control room alarm:

(Site-specific) list

Oyster Creek proposed HU6 EAL:

1. FIRE in any Table H2 area not extinguished within 15 minutes of Control Room notification or verification of a Control Room alarm.

OR

2. FIRE outside any Table H2 area with the potential to damage safety systems in any Table H2 area not extinguished within 15 minutes of Control Room notification or verification of a Control Room alarm.

OR

~~3.~~3. Report by plant personnel of an unanticipated EXPLOSION within PROTECTED AREA boundary resulting in ~~PHYSICAL~~ VISIBLE DAMAGE to permanent structure or equipment.

Table H2 – Vital Areas
Reactor Bldg
Control Room Complex
Main Transformer/Condensate Transfer Pad
Intake Structure
#1 EDG Vault
#2 EDG Vault
EDG Fuel Oil Storage Tank

Technical Justification

This deviation is appropriate because it ensures that the correct criteria are considered for the classification. Due to the design of the Exelon Nuclear Stations, there are many large areas (or "buildings") that are connected (or contiguous) to areas that contain vital equipment. The example EAL wording implies that a fire anywhere within these large areas warrants an emergency declaration whether it has any affect on equipment in, or personnel access to, the vital area. The IC and basis wording of the example EAL are clear that unless these conditions are met, an emergency declaration is not required.

An example of the inappropriateness of the use of "contiguous" would be a small isolated fire in a far corner of the Turbine Building that is a far distance from the Main Control Room. Because the Turbine Building is "contiguous" to the Main Control Room, once the fire has been identified then an emergency declaration would be required when no risk to the equipment in the Main Control Room exists, and no impediment to access to the Main Control Room has occurred. This wording clarification will eliminate this type of inappropriate event declaration.

Note, the Turbine Building is no longer included in "Table H2 -Vital Areas" since the Turbine Building does not contain vital areas and any effects on Safety Systems would still be classified under the existing EAL thresholds.

The wording clarifications brings the intent of the EAL as explained in the basis into the EAL threshold value to enable the Operators to make the correct declaration in a more timely manner and eliminate the ambiguity introduced by the term "contiguous."

Note that Threshold #3 originates from NEI HU1 and has been included in Exelon EAL HU6 since the escalation path is HA6.

Supporting Information

Enclosure 8B contains the proposed EALs and EAL Bases, as well as the corresponding EAL ICs in the applicable IC logic grouping (i.e., the related UE, Alert, SAE, and GE).

HU6, HA6

Enclosure 8C contains a comparison table which highlights the proposed changes to the EALs. The table compares the current approved SER version of the EAL, the NEI 99-01 EAL, and the proposed deviation to the EAL.

2.1.8 **Deviation 8** – Toxic or Flammable Gas

NEI EAL: HA3 **Oyster Creek EAL:** HA7

Operational Modes: 1, 2, 3, 4, D

Description of the Deviation

NEI 99-01 HA3 EAL was revised to change the word "Contiguous" to "Restricting Access" and "(or area that restricts access to listed areas)". This more accurately reflects intent of EAL and avoids unnecessary classifications for events in areas that may be considered contiguous but there is no threat to any vital equipment or plant safety.

NEI 99-01 HA3 EAL

1. *Report or detection of toxic gases within or contiguous to a VITAL AREA in concentrations that may result in an atmosphere IMMEDIATELY DANGEROUS TO LIFE AND HEALTH (IDLH).*

2. *Report or detection of gases in concentration greater than the LOWER FLAMMABILITY LIMIT within or contiguous to a VITAL AREA.*

Oyster Creek HA7 EAL

1. Report or detection of toxic or asphyxiant gases within a Table H2 area (or area that restricts access to a Table H2 area) in concentrations that result in an atmosphere IMMEDIATELY DANGEROUS TO LIFE AND HEALTH (IDLH).

   OR

2. Report or detection of flammable gases within a Table H2 area (or area that restricts access to a Table H2 area) in concentration greater than the LOWER FLAMMABILITY LIMIT (LFL).

Table H2 – Vital Areas
Reactor Bldg
Control Room Complex
Main Transformer/Condensate Transfer Pad
Intake Structure
#1 EDG Vault
#2 EDG Vault
EDG Fuel Oil Storage Tank

Technical Justification

This deviation is appropriate because it ensures that the correct criteria are considered for the classification. Due to the design of the Exelon Nuclear Stations, there are many large areas (or "buildings") that are connected (or contiguous) to areas that contain vital equipment. The example EAL wording implies that a hazardous atmosphere anywhere within these large areas warrants an emergency declaration whether or not it has any affect on equipment in, or personnel access to, the vital area. The IC and basis wording of the example EAL are clear that unless these conditions are met, an emergency declaration is not required.

An example of the inappropriateness of the use of "contiguous" would be an isolated discharge of a toxic gas in a far corner of the Turbine Building that is a far distance from the Main Control Room. Because the Turbine Building is "contiguous" to the Main Control Room, once the area of gas discharge is considered to be IDLH, and then an emergency declaration would be required when no risk to the operation of equipment in the Main Control Room has occurred, and no impediment to access to the Main Control Room has occurred. This wording clarification will eliminate this type of inappropriate event declaration.

Note, the Turbine Building is no longer included in "Table H2 -Vital Areas" since the Turbine Building does not contain vital areas and any effects on Safety Systems would still be classified under the existing EAL thresholds.

The wording clarifications bring the intent of the EAL as explained in the basis into the EAL threshold value to enable the Operators to make the correct declaration in a more timely manner and eliminate the ambiguity introduced by the term "contiguous."

Supporting Information

Enclosure 8B contains the proposed EALs and EAL Bases, as well as the corresponding EAL ICs in the applicable IC logic grouping (i.e., the related UE, Alert, SAE, and GE).

HU7, HA7

Enclosure 8C contains a comparison table which highlights the proposed changes to the EALs. The table compares the current approved SER version of the EAL, the NEI 99-01 EAL, and the proposed deviation to the EAL.

2.2 EAL Deviations Related To Last Approved SER

The following are identified as deviations when comparing the proposed new EAL revision to the previous EALs as they existed at the time of the last NRC SER under NUMARC/NESP-007. The changes from NESP-007 SER when compared to the proposed version might be perceived as a Deviation and therefore are being submitted for prior NRC approval. A detailed description of the changes and justifications for the changes are provided below.

Note: For the purpose of this evaluation, the term "SER EAL" refers to an EAL as it existed at the time of the last NRC approved SER.

2.2.5 Deviation 13 – Fuel Clad Degradation

NEI EAL: SA4 **Oyster Creek EAL:** RA3

Operational Modes: 1,2,3,4, D

Description of the Deviation

Oyster Creek's NESP-007 SER Approved RA3 EAL value for Threshold #2 is being raised from 1 R/hr to ~~3000~~ 2000 mr/hr to align with NEI 99-01 Rev 4 requirements to base the threshold on 10CFR20.

Technical Justification

NEI 99-01 Basis for AA3 (Oyster Creek RA3) states the following for determining the site-specific threshold value for this EAL:

> *"For areas requiring infrequent access, the site-specific value(s) should be based on radiation levels which result in exposure control measures intended to maintain doses within normal occupational exposure guidelines and limits (i.e., 10 CFR 20), and in doing so, will impede necessary access. As used here, impede, includes hindering or interfering provided that the interference or delay is sufficient to significantly threaten the safe operation of the plant."*

This threshold addresses increased radiation levels in areas that require infrequent access in order to maintain safe plant operation or perform a safe plant shutdown. Typically areas requiring infrequent access to maintain plant safety functions will include plant vital areas. Area radiation levels at or above 3,000 mR/hr are indicative of radiation fields that limit personnel access to equipment, the operation of which may be needed to assure adequate core cooling or shutdown the reactor.

The ~~3,~~2000 mR/hr) dose rate threshold selected is based ~~on not exceeding 10CFR20 limits (5,000 mR/yr), assuming:~~

- ~~Emergency response personnel are already at the Oyster Creek TEDE annual administrative dose limit of 2 Rem (2,000 mRem) per year. This means that any emergency response worker could receive up to an additional 3,000 mRem without be required to implement emergency exposure guidelines.~~

- ~~A one hour stay time (one hour is the maximum time an individual would be expected to remain in a vital area during emergency conditions).~~

- ~~3,000 mRem in 1 hour = 3,000 mR/hour~~site administrative limits

NEI 99-01 also states:

> *"It is this impaired ability to operate the plant that results in the actual or potential substantial degradation of the level of safety of the plant. The cause and/or magnitude of the increase in radiation levels is not a concern of this IC."*

This change does not alter the intent of the EAL but ensures that it is properly applied for its designed application and would not result in an overly conservative classification. The proposed ~~3,~~2000 mR/hr threshold is more aligned with the NEI 99-01 Rev 4 guidance than the current SER approved threshold value of 1R/hr.

Supporting Information

Enclosure 8B contains the proposed EALs and EAL Bases, as well as the corresponding EAL ICs in the applicable IC logic grouping (i.e., the related UE, Alert, SAE, and GE).

RA3

Enclosure 8C contains a comparison table which highlights the proposed changes to the EALs. The table compares the current approved SER version of the EAL, the NEI 99-01 EAL, and the proposed deviation to the EAL.

2.2.12 Deviation 20 – Loss of AC Power

NEI EAL: SG1 **Oyster Creek EAL:** MG1

Operational Modes: 1, 2

Description of the Deviation

Oyster Creek is removing an EAL threshold from the NESP-007 SER approved EAL for MG1, Loss of AC Power. This threshold deletion is being proposed to align Oyster Creek, Loss of AC Power EAL MG1 with the guidance provided in NEI 99-01 for SG1.

NEI 99-01 provides the following Example EALs for SG1:

1. Loss of power to (site-specific) transformers.

   AND

   Failure of (site-specific) emergency diesel generators to supply power to emergency busses.

   AND

   Either of the following: (a or b)

   a. Restoration of at least one emergency bus within (site-specific) hours is not likely

      OR

   b. (Site-Specific) Indication of continuing degradation of core cooling based on Fission Product Barrier monitoring.

Oyster Creek had added the following thresholds to MG1 to satisfy the above Example EAL (b):

OC's current MG 1:

1. BOTH 4160V Busses 1C and 1D de-energized for > 15 min.

   AND

   ANY of the following:

   - Restoration of at least one emergency bus within 1 hour is not likely
   - RPV level CANNOT be maintained > 0" TAF OR CANNOT be determined
   - Torus water temperature and RPV pressure exceeds the Heat Capacity Temperature Limit (Figure F, EMG-3200.02)

OC is revising the second bulleted criteria to state " RPV level **cannot** be determined to be **> 0 inches TAF**.~~RPV level < 0 inches TAF~~." OC is also removing "*Torus water temperature and RPV pressure exceeds the Heat Capacity Temperature Limit (Figure F, EMG-3200.02)*" from the third MG1 bulleted threshold.

Technical Justification

NEI 99-01 provides the following guidance:

> *b. (Site-Specific) Indication of continuing degradation of core cooling based on Fission Product Barrier monitoring":*
>
> *In addition, under these conditions, fission product barrier monitoring capability may be degraded. Although it may be difficult to predict when power can be restored, it is necessary to give the Emergency Director a reasonable idea of how quickly (s)he may need to declare a General Emergency based on two major considerations:*
>
> 1. *Are there any present indications that core cooling is already degraded to the point that Loss or Potential Loss of Fission Product Barriers is imminent?*
>
> 2. *If there are no present indications of such core cooling degradation, how likely is it that power can be restored in time to assure that a loss of two barriers with a potential loss of the third barrier can be prevented?*
>
> *Thus, indication of continuing core cooling degradation must be based on Fission Product Barrier monitoring with particular emphasis on Emergency Director judgment as it relates to imminent Loss or Potential Loss of fission product barriers and degraded ability to monitor fission product barriers.*

NEI 99-01 guidance is clear that this EAL is to be based on site-specific indications of continuing degradation of core cooling based on Fission Product Barrier monitoring. The use of Torus temperature does not meet the intent of this requirement. Plant procedures would direct a plant depressurization prior to exceeding this curve. Exceeding the temperature on the Heat Capacity Temperature Limit (HCTL) curve does not provide an indication of continuing core cooling degradation based on the Fission Product Barrier Matrix.

NEI 99-01 further states that:

> *Although this IC may be viewed as redundant to the Fission Product Barrier Degradation IC, its inclusion is necessary to better assure timely recognition and emergency response.*

Based on the guidance provided in NEI 99-01 it would be appropriate to use " RPV level **cannot** be determined to be **> 0 inches TAF**.~~RPV level < 0 inches TAF~~" as the threshold since this is the parameter from the Fission Product Barrier Matrix that represents core-cooling degradation. Removal of the threshold concerning the HCTL curve and utilizing Top of Active Fuel does not alter or change the intent of the EAL as defined by NEI 99-01.

Supporting Information

Enclosure 8B contains the proposed EALs and EAL Bases, as well as the corresponding EAL ICs in the applicable IC logic grouping (i.e., the related UE, Alert, SAE, and GE).

MU1, MS1, MA1, MG1

Enclosure 8C contains a comparison table which highlights the proposed changes to the EALs. The table compares the current approved SER version of the EAL, the NEI 99-01 EAL, and the proposed deviation to the EAL.

2.2.13 Deviation 21 – RPS Failure

NEI EAL: SG2 **Oyster Creek EAL:** MG3

Operational Modes: 1

Description of the Deviation

Oyster Creek Generating Station (OCGS) is revising NESP-007 General Emergency EAL associated with 'Failure of Reactor Protection System" to incorporate the enhancements provided in NEI 99-01 SG2 EAL basis and align threshold values with the Station's Emergency Operating Procedure wording. Oyster Creek's Failure to Scram, MG3 EAL (NEI SG2), includes the following qualifier when evaluating RPV level:

> *RPV level CANNOT be restored and maintained > -30" TAF OR CANNOT be determined*

The phrase "OR CANNOT be determined" is being removed from Oyster Creek's MG3 EAL. Note that the proposed –20" TAF level was revised from -30" TAF under 10CFR50.54(q) in 2005.

NESP-007 EAL

1. *RPS setpoint for an automatic SCRAM exceeded*

   AND

   *Failure of automatic RPS, ARI and manual SCRAM to reduce reactor power **< 2%***

   AND EITHER:

   - *RPV level **CANNOT** be restored and maintained **> -30" TAF OR CANNOT** be determined*

     OR

   - *Torus water temperature and RPV pressure exceeds the Heat Capacity Temperature Limit (Figure F, EMG-3200.02)*

Proposed EAL

1. *Automatic scram, Manual scram and ARI were not successful from Reactor Console as indicated by EITHER:~~Automatic and Manual SCRAM were **not** successful as indicated by **EITHER**:~~*

   a. *Reactor Power remains **≥ 2%.***

   OR

   b. *Torus temperature **> 110°F AND** boron injection required for reactivity control.*

AND

2. a. *RPV level **cannot** be restored and maintained **> -20 inches TAF.***

OR

b. Heat Capacity Temperature Limit (EMG-3200.02 Fig. F) exceeded.

Technical Justification

The phrase "OR CANNOT be determined" was added as an approved deviation to the NESP-007 EAL scheme when Oyster Creek converted their EALs from NUREG-0654. This phrase was not consistently applied throughout the EALs. NEI 99-01, Rev 4 does not include the phrase 'OR CANNOT be determined' when specifying RPV levels and is being deleted to be consistent with the equivalent NEI 99-01, Rev 4 EAL.

NEI SG2 states the threshold should be:

Indication(s) exists that the core cooling is extremely challenged.

NEI SG2 Bases explains the above threshold as follows:

For BWRs, the extreme challenge to the ability to cool the core is intended to mean that the reactor vessel water level cannot be restored and maintained above Minimum Steam Cooling RPV Water Level as described in the EOP bases.

Additionally, an indeterminate RPV level from installed instrumentation does not necessarily mean that the RPV level is low and a declaration is warranted. Decisions on actual RPV level during conditions when indications are lost are based on last known level, trends, and transients that are in progress. Emergency Operating Procedures (EOPs) provide guidance regarding indeterminate RPV level conditions. This guidance directs the utilization of alternate means for level determination and prescribes appropriate courses of action. The availability of this guidance negates the need for the indeterminate RPV level IC.

An indeterminate RPV level may lead to a premature declaration if actual level remains above the top of active fuel (TAF). This situation would result in a premature declaration of a General Emergency.

To maintain consistency with NEI 99-01 and remove possible situations where a premature declaration could occur, the phrase '**OR CANNOT** be determined' will be deleted from this EAL. This will not change or alter the intent of the EAL as defined by NEI 99-01.

Supporting Information

Enclosure 8B contains the proposed EALs and EAL Bases, as well as the corresponding EAL ICs in the applicable IC logic grouping (i.e., the related UE, Alert, SAE, and GE).

MA3, MS3, MG3

Enclosure 8C contains a comparison table which highlights the proposed changes to the EALs. The table compares the current approved SER version of the EAL, the NEI 99-01 EAL, and the proposed deviation to the EAL.

ENCLOSURE 7C

OYSTER CREEK GENERATING STATION ANNEX

EP-AA-1010

"Revised Markup of EAL Bases"

Table D-2: OCGS EAL Technical Basis

RECOGNITION CATEGORY
FISSION PRODUCT BARRIER DEGRADATION

RA1

Initiating Condition

Any UNPLANNED Release of Gaseous or Liquid Radioactivity to the Environment that Exceeds 200 Times the Radiological Effluent Technical Specifications for 15 Minutes or Longer.

Operating Mode Applicability:

1, 2, 3, 4, D

EAL Threshold Values

1. VALID reading on any of the following effluent monitors~~the Radwaste Discharge effluent monitor~~ **> 200 times alarm setpoint** established by a current radioactivity discharge permit for **≥ 15 minutes**:
   - Radwaste Overboard Discharge effluent monitor
   - Discharge Permit specified monitor

OR

2. VALID reading on one or more of the following radiation monitors that exceeds the Table R1 values for **≥ 15 minutes**:

Table R1 – Effluent Monitor Thresholds	
	Alert
Main Stack RAGEMS	1.93 E+00 μCi/cc HRM
Turbine Building RAGEMS	8.11E+04 cpm LRM

HRM = High Range Monitor LRM = Low Range Monitor

OR

3. Confirmed sample analyses for gaseous or liquid releases indicate concentrations or release rates **> 200 times ODCM Limit** with a release duration of **≥ 15 minutes**.

Table D-2: OCGS EAL Technical Basis

RECOGNITION CATEGORY
FISSION PRODUCT BARRIER DEGRADATION

RA1 (cont)

Basis

UNPLANNED, as used in this context, includes any release for which a radioactivity discharge permit was not prepared, or a release that exceeds the conditions (e.g., minimum dilution flow, maximum discharge flow, alarm setpoints, etc.) on the applicable permit.

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.

The site design incorporates features intended to control the release of radioactive effluents to the environment. Further, there are administrative controls established to prevent unintentional releases, or control and monitor intentional releases. These controls are located in the Offsite Dose Calculation Manual (ODCM). The occurrence of extended, uncontrolled radioactive releases to the environment is indicative of a degradation in these features and/or controls.

The Emergency Director should not wait until 15 minutes has elapsed, but should declare the event as soon as it is determined that the release duration has or will likely exceed 15 minutes. Also, if an ongoing release is detected and the starting time for that release is unknown, the Emergency Director should, in the absence of data to the contrary, assume that the release has exceeded 15 minutes.

Threshold #1 Basis:

The threshold addresses radioactivity releases that cause effluent radiation monitor readings that exceed two hundred times the alarm setpoint established by the radioactivity discharge permit. This alarm setpoint may be associated with a planned batch release, or a continuous release path. In either case, the setpoint is established by the ODCM to warn of a release that is not in compliance with the RETS. Indexing the EAL threshold to the ODCM setpoints in this manner ensures that the EAL threshold will never be less than the setpoint established by a specific discharge permit.

Should 200 times the HI-HI Radiation alarm result in an offscale high meter reading, then the EAL threshold would be considered met when the meter goes offscale high for ≥ 15 minutes, provided there are no other direct or indirect means available to determine actual value.

An elevated monitor reading while the effluent flow path is isolated is not considered a VALID reading.

The effluent monitors listed are those normally used for planned discharges. If a discharge is performed using a different flowpath or effluent monitor other than those listed (e.g., a portable or temporary effluent monitor), then the declaration criteria will be based on the monitor specified in the Discharge Permit.

~~**Threshold #2 Basis:**~~

~~This EAL addresses a potential or actual drop in the level of safety of the plant as indicated by a radiological release that exceeds, by a factor of 200, regulatory commitments for an extended period of time. The Alert gaseous effluent value was determined using formulas, isotopic dose factors and meteorology data as specified by the ODCM. The Table R1 values~~

~~were determined in the units of a station-generated normal operating mixture for the no clad damage condition.~~

Table D-2: OCGS EAL Technical Basis

RECOGNITION CATEGORY
FISSION PRODUCT BARRIER DEGRADATION

RA1 (cont)

Basis (cont):

Threshold #2 Basis:

This EAL addresses a potential or actual drop in the level of safety of the plant as indicated by a radiological release that exceeds, by a factor of 200, regulatory commitments for an extended period of time. The Alert gaseous effluent value was determined using formulas, isotopic dose factors and meteorology data as specified by the ODCM. The Table R1 values were determined in the units of a station-generated normal operating mixture for the no clad damage condition.

Threshold #3 Basis:

Confirmed sample analyses in excess of two hundred times the site Offsite Dose Calculation Manual (ODCM) limits that continue for 15 minutes or longer represent an uncontrolled situation and hence, a potential degradation in the level of safety. This event escalates from the Unusual Event by increasing the magnitude of the release by a factor of 100 over the Unusual Event level (i.e., 200 times ODCM). Prorating the 500 mR/yr basis of the 10CFR20 non-occupational MPC limits for both time (8766 hr/yr) and the 200 multiplier, the associated site boundary dose rate would be approximately 10 mR/hr. The required release duration was reduced to 15 minutes in recognition of the raised severity.

The 'site boundary' is defined by an approximately 400-meter (1/4-mile) radius around the plant. This is the nearest distance from potential release points at which Protective Actions would be required for members of the public.

Grab samples are used to determine release concentrations or rates to confirm meter readings or when the effluent monitors are not in service. The ODCM uses 10 CFR 20 Appendix B Table 2 data to generate maximum instantaneous release rate limits. These are indicated on Release Packages, which are approved.

Basis: (References)

1. CY-OC-170-301, Offsite Dose Calculation Manual for Oyster Creek
2. EP-EAL-0610, Criteria for Choosing Radiological Gaseous Effluent EAL Threshold Values, Oyster Creek Generating Station
3. NEI 99-01, Rev. 4 AA1

Table D-2: OCGS EAL Technical Basis

RECOGNITION CATEGORY
FISSION PRODUCT BARRIER DEGRADATION

RU1

Initiating Condition

Any UNPLANNED Release of Gaseous or Liquid Radioactivity to the Environment that Exceeds Two Times the Radiological Effluent Technical Specifications for 60 Minutes or Longer.

Operating Mode Applicability:

1, 2, 3, 4, D

EAL Threshold Value:

1. VALID reading on any of the following effluent monitors~~the Radwaste Discharge effluent monitor~~ **> 2 times alarm setpoint** established by a current radioactivity discharge permit for **≥ 60 minutes**:
   - Radwaste Overboard Discharge effluent monitor
   - Discharge Permit specified monitor

OR

2. VALID reading on one or more of the following radiation monitors that exceeds the Table R1 values for **≥ 60 minutes**.

Table R1 – Effluent Monitor Thresholds	
	Unusual Event
Main Stack RAGEMS	7.92 E+03 cps LRM
Turbine Building RAGEMS	8.11E+02 cpm LRM

HRM = High Range Monitor LRM = Low Range Monitor

OR

3. Confirmed sample analyses for gaseous or liquid releases indicate concentrations or release rates in **> 2 times ODCM Limit** with a release duration of **≥ 60 minutes**.

Table D-2: OCGS EAL Technical Basis

RECOGNITION CATEGORY
FISSION PRODUCT BARRIER DEGRADATION

RU1 (cont)

Basis

UNPLANNED, as used in this context, includes any release for which a radioactivity discharge permit was not prepared, or a release that exceeds the conditions (e.g., minimum dilution flow, maximum discharge flow, alarm setpoints, etc.) on the applicable permit.

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.

Since the assumptions used in calculating the radiation monitor threshold values and alarm setpoints with respect to ODCM release rate limits may not exactly match the conditions present when the classification is considered, results of available sample analyses override the monitor readings listed.

The Emergency Director should not wait until 60 minutes has elapsed, but should declare the event as soon as it is determined that the release duration has or will likely exceed 60 minutes. Also, if an ongoing release is detected and the starting time for that release is unknown, the Emergency Director should, in the absence of data to the contrary, assume that the release has exceeded 60 minutes.

Threshold #1 Basis:

The site design incorporates features intended to control the release of radioactive effluents to the environment. Further, there are administrative controls established to prevent unintentional releases, or control and monitor intentional releases. These controls are located in the Offsite Dose Calculation Manual (ODCM). The occurrence of extended, uncontrolled radioactive releases to the environment is indicative of a degradation in these features and/or controls.

An elevated monitor reading while the effluent flow path is isolated is not considered to be a VALID reading.

The effluent monitors listed are those normally used for planned discharges. If a discharge is performed using a different flowpath or effluent monitor other than those listed (e.g., a portable or temporary effluent monitor), then the declaration criteria will be based on the monitor specified in the Discharge Permit.

Threshold #2 Basis:

This EAL addresses a potential drop in the level of safety of the plant as indicated by a radiological release that exceeds, by a factor of 2, regulatory commitments for an extended period of time. The Unusual Event gaseous effluent value was determined using formulas, isotopic dose factors and meteorology data as specified by the ODCM.

The release rate was determined in the units of a station-generated normal operating mixture for the no clad damage condition.

Table D-2: OCGS EAL Technical Basis

RECOGNITION CATEGORY
FISSION PRODUCT BARRIER DEGRADATION

RU1 (cont)

Basis (cont):

Threshold #3 Basis:

Confirmed sample analyses in excess of two times the site Offsite Dose Calculation Manual (ODCM) limits that continue for 60 minutes or longer represent an uncontrolled situation and hence, a potential degradation in the level of safety. The final integrated dose (which is very low in the Unusual Event emergency class) is not the primary concern here; it is the degradation in plant control implied by the fact that the release was not isolated within 60 minutes. Therefore, it is not intended that the release be averaged over 60 minutes. For example, a release of 4 times ODCM for 30 minutes does not exceed this EAL. Grab samples are used to determine release concentrations or rates to confirm meter readings or when the effluent monitors are not in service. The ODCM uses 10 CFR 20 Appendix B Table 2 data to generate maximum instantaneous release rate limits. These are indicated on Release Packages, which are approved.

Basis: (References)

1. CY-OC-170-301, Offsite Dose Calculation Manual for Oyster Creek
2. EP-EAL-0610, Criteria for Choosing Radiological Gaseous Effluent EAL Threshold Values, Oyster Creek Generating Station
3. NEI 99-01, Rev. 4 AU1

Table D-2: OCGS EAL Technical Basis

RECOGNITION CATEGORY
FISSION PRODUCT BARRIER DEGRADATION

RU2

Initiating Condition

Unexpected Rise in Plant Radiation.

Operating Mode Applicability

1, 2, 3, 4, D

EAL Threshold Value

1. a. VALID indication of uncontrolled drop in water level in the Reactor Cavity, ~~or~~ Spent Fuel Pool or Fuel Transfer Canal with all irradiated fuel assemblies remaining covered by water as indicated by:

    - Reactor Cavity water level **< 583 inches**. (GEMAC Wide Range, floodup calibration)

    OR

    - Report of visual observation of an uncontrolled drop in water level in the Reactor Cavity or Spent Fuel Pool.

   AND

   b. UNPLANNED VALID Area Radiation Monitor reading rise on one or more radiation monitor in Table R2.

Table R2 – Refuel Floor ARMs
C-5, Crit Mon
C-10, North Wall
C-9, North Wall
B-9, Open Floor

OR

2. UNPLANNED VALID Area Radiation Monitor readings rise by a factor of **1000** over NORMAL LEVELS or VALID upscale reading.

Basis

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.

UNPLANNED: a parameter change or an event that is not the result of an intended evolution and requires corrective or mitigative actions.

Table D-2: OCGS EAL Technical Basis

RECOGNITION CATEGORY
FISSION PRODUCT BARRIER DEGRADATION

RU2 (cont)

Basis (cont)

NORMAL LEVELS: Normal levels can be considered as the highest reading in the past twenty-four hours excluding the current peak value.

Threshold #1 Basis:

This EAL addresses unplanned increases in radiation levels inside the plant. These radiation levels represent a degradation in the control of radioactive material and a potential degradation in the level of safety of the plant.

In light of Reactor Cavity Seal failure incidents at two different PWRs and loss of water in the Spent Fuel Pit/Fuel Transfer Canal at a BWR, explicit coverage of these types of events in the EALs is appropriate given their potential for increased doses to plant workers. Classification as an Unusual Event is warranted as a precursor to a more serious event.

Since no remote indication of Spent Fuel Pool water level exists, decreases in Spent Fuel Pool water level can normally be detected only through visual observation or decrease in skimmer surge tank level.

During refueling when the RPV head is removed, the GEMAC Wide Range instrumentation is calibrated to indicate water level to the elevation of the refuel floor. With the refueling cavity in communication with the Spent Fuel Pool through the fuel transfer canal, uncontrolled inventory loss can be remotely monitored.

The Refueling Cavity includes the fuel transfer canal. When the Refueling Cavity is flooded to normal level, water level is approximately one foot below the refuel floor. ~~Technical Specifications require~~ Reactor Cavity water level is~~be~~ maintained at least 23 ft above the top of the RPV flange (307 in. + 276 in.) or 583 inches when irradiated fuel or control rods are being handled within the RPV. During refueling when the RPV head is removed, plant procedures (e.g., 205.95.0 Reactor Flood-up / Drain-down, etc.) provide alternate level monitoring capabilities when the normal level instrumentation is unavailable for the desired level range or the head vent piping is removed. When calibrated for refueling operations, the GEMAC Wide Range instrument indicates from 100 in. above top of active fuel in the RPV to the maximum refuel floor water level. In addition, visual observation of level from the refueling floor can be used to monitor water level when the RPV head is removed.

Table D-2: OCGS EAL Technical Basis

RECOGNITION CATEGORY
FISSION PRODUCT BARRIER DEGRADATION

RU2 (cont)

Basis (cont)

Threshold #2 Basis:
Valid elevated area radiation levels usually have long lead times relative to the potential for radiological release beyond the site boundary, thus impact to public health and safety is very low.

This EAL addresses unplanned rise in radiation levels inside the plant

Basis (References)

1. RP-AA-203 Exposure Control and Authorization
2. RAP-G-7-a, SKM SRG TNK LVL LO-LO
3. ~~Technical Specifications~~

~~4~~3. 205.94.0 RPV Floodup Using Core Spray

~~5~~4. 205.95.0 Reactor Flood-up / Drain-down

~~6~~5. FSAR Figure 7.6-3

~~7~~6. NEI 99-01, Rev. 4 AU2

Table D-2: OCGS EAL Technical Basis

RECOGNITION CATEGORY
FISSION PRODUCT BARRIER DEGRADATION

RA3

Initiating Condition

Release of Radioactive Material or Rise in Radiation Levels Within the Facility That Impedes Operation of Systems Required to Maintain Safe Operations or to Establish or Maintain Cold Shutdown

Operating Mode Applicability:

1, 2, 3, 4, D

EAL Threshold Values

1. VALID radiation monitor or survey reading **>15 mR/hr** in any of the following areas requiring continuous occupancy to maintain plant safety functions.
   - Main Control Room
   - Central Alarm Station (by survey)

OR

2. VALID radiation monitor or survey reading **> ~~3000~~ 2000 mR/hr** in areas requiring infrequent access (Table R3) which will impede necessary access and threaten safe operation of the plant.

Table R3 – Areas Requiring Infrequent Access
Reactor Bldg
Turbine Bldg
Control Room Complex
Main Transformer/Condensate Transfer Pad
Intake Structure
#1 EDG Vault
#2 EDG Vault
EDG Fuel Oil Storage Tank

Table D-2: OCGS EAL Technical Basis

RECOGNITION CATEGORY
FISSION PRODUCT BARRIER DEGRADATION

RA3 (cont)

Basis

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.

Threshold #1 Basis:

This threshold addresses increased radiation levels that impede necessary access to operating stations requiring continuous occupancy to maintain safe plant operation or perform a safe plant shutdown. Areas requiring continuous occupancy include the Main Control Room and the Central Alarm Station (CAS). The security alarm station is included in this threshold because of its importance to permitting access to areas required to assure safe plant operations.

The value of 15 mR/hr is derived from the GDC 19 value of 5 rem in 30 days with adjustment for expected occupancy times. Although Section III.D.3 of NUREG-0737, "Clarification of TMI Action Plan Requirements", provides that the 15 mR/hr value can be averaged over the 30 days, the value is used here without averaging. A 30 day duration implies an event potentially more significant than an Alert.

It is the impaired ability to operate the plant that results in the actual or potential degradation of the level of safety of the plant. The cause or magnitude of the increase in radiation levels is not a concern of this threshold. The Emergency Director must consider the source or cause of the increased radiation levels and determine if any other EALs may be involved. For example, a dose rate of 15 mR/hr in the Main Control Room may be a problem in itself. However, the increase may also be indicative of high dose rates in the containment due to a LOCA. In this latter case, a Site Area Emergency or a General Emergency may be indicated by other EAL categories.

This threshold is not intended to apply to anticipated temporary radiation increases due to planned events (e.g., radwaste container movement, depleted resin transfers, etc.).

Threshold #2 Basis:

This threshold addresses increased radiation levels in areas requiring infrequent access in order to maintain safe plant operation or perform a safe plant shutdown. Typically areas requiring infrequent access to maintain plant safety functions include plant vital areas. Area radiation levels at or above ~~3,~~2000 mR/hr are indicative of radiation fields, which may limit personnel access to equipment, the operation of which may be needed to assure adequate core cooling or shutdown the reactor.

The Control Room Complex consists of the Main Office Building, Upper and Lower Cable Spread Rooms.

Table D-2: OCGS EAL Technical Basis

RECOGNITION CATEGORY
FISSION PRODUCT BARRIER DEGRADATION

RA3 (cont)

Basis: (cont)

The dose rate threshold selected is based on site administrative limits.~~not exceeding 10CFR20 limits (5,000 mR/yr), assuming:~~

- ~~Emergency response personnel are at their Oyster Creek TEDE annual administrative dose limit of 2 Rem (2,000 mRem) per year. This means that any emergency worker could receive up to an additional 3,000 mRem without requiring to implement emergency exposure guidelines.~~
- ~~A one hour stay time (one hour is the maximum time an individual would be expected to remain in a vital area during emergency conditions).~~
- ~~3,000 mRem in 1 hour = 3,000 mR/hour~~

It is the impaired ability to operate the plant that results in the actual or potential degradation of the level of safety of the plant. The cause or magnitude of the increase in radiation levels is not a concern of this threshold. The Emergency Director must consider the source or cause of the increased radiation levels and determine if any other EAL may be involved. For example, a dose rate of ~~3~~ 2 R/hr may be a problem in itself. However, the increase may also be indicative of high dose rates in the containment due to a LOCA. In this latter case, a Site Area Emergency or a General Emergency may be indicated by other EAL categories.

This threshold is not intended to apply to anticipated temporary radiation rise due to planned events (e.g., radwaste container movement, depleted resin transfers, etc.) or pre-existing radiation areas for which radiological controls already exist. The concern of this threshold is the unanticipated rise in radiation levels that results in unplanned restrictions to areas requiring infrequent access in order to maintain safe plant operation or perform a safe plant shutdown.~~It is not intended to apply to anticipated temporary radiation increases due to planned events (e.g., radwaste container movement, depleted resin transfers, etc.).~~

Basis (References):

1. ABN-29, Plant Fires
2. EMG-3200.11, Secondary Containment Control Safe Shutdown Area
3. NEI 99-01, Rev. 4 AA3

Table D-2: OCGS EAL Technical Basis

RECOGNITION CATEGORY
FISSION PRODUCT BARRIER DEGRADATION

RCS 2.b

Initiating Condition

Drywell (DW) High Range Rad Monitor

Operating Mode Applicability

1, 2

EAL Threshold Value

LOSS:

1. Containment Hi Range Radiation Monitoring System (CHRRMS) > **100 R/hr**.

AND

2. Indications of RCS leakage into the Drywell.

POTENTIAL LOSS:

NONE

Basis:

The radiation monitor reading is a value that indicates a significant release of reactor coolant to the containment. A reading was calculated assuming the instantaneous release and dispersal of the reactor coolant iodine inventory associated with normal operating concentrations (i.e., within Technical Specifications) into the drywell atmosphere. This provides a reading that is observable and indicative of significant release of reactor coolant to the drywell. The reading is less than that specified for Fuel Cladding barrier Loss because no damage to the fuel cladding is assumed. Only leakage from the RCS is assumed for this barrier loss threshold. The value is high enough to preclude erroneous classification of barrier loss due to normal plant operations.

Indication of a RCS leak into the drywell is added to qualify the radiation monitor indication to avoid declaring the loss of RCS barrier for situations where the radiation rise is not due to primary a system leak. For situations that involve failure of the Fuel Clad barrier alone, radiation monitor readings would rise due to shine and potentially giving a false indication of a loss of the RCS barrier. Therefore this EAL contains a qualifier to preclude over classification of the event if only fuel clad barrier failed.

Basis Reference(s):

1. EP-EAL-0611, Criteria for Choosing Containment Radiation Monitor Reading Indicative of Loss of RCS Barrier
2. NEI 99-01, Rev. 4 Table 5-F-2

RECOGNITION CATEGORY
FISSION PRODUCT BARRIER DEGRADATION

CONTAINMENT 3.c

Initiating Condition

Drywell (DW) Pressure / Hydrogen Concentration

Operating Mode Applicability

1, 2

EAL Threshold Value

LOSS:

1. Rapid unexplained drop in Drywell pressure following an initial pressure rise.

OR

2. Drywell pressure ~~rise~~ response **not** consistent with LOCA conditions.

POTENTIAL LOSS:

3. Drywell pressure **> 44 psig** and rising.

OR

4. a. Drywell or Torus Hydrogen concentration **≥ 6%**

AND

b. Drywell or Torus Oxygen concentration **≥ 5%**

Basis

LOSS – [Threshold Value #1]

A rapid unexplained drop in Drywell pressure not due to use of containment sprays or condensation effects following an initial pressure rise indicates a loss of containment integrity.

LOSS – [Threshold Value #2]

Drywell pressure should rise as a result of mass and energy release into the containment from a Loss of Coolant Accident (LOCA). Thus, Drywell pressure ~~NOT rising under these~~response not consistent with LOCA conditions indicates a loss of containment integrity. This indicator relies on operator recognition of an unexpected response for the condition and therefore does not include a specific pressure value or trend. Due to conservatisms in LOCA analyses, actual pressure response is expected to be less than the analyzed response. For example, blowdown mass flowrate may be only 60-80% of the analyzed rate.

The unexpected response of Drywell pressure is important because it is the indicator for a containment downcomer to torus bypass condition. Under these conditions the pressure suppression function of the Primary Containment is reduced and Drywell pressure will be significantly higher for any LOCA. A large bypass of the pressure suppression function could result in failure of the containment on high pressure.

Table D-2: OCGS EAL Technical Basis

RECOGNITION CATEGORY
FISSION PRODUCT BARRIER DEGRADATION

CONTAINMENT 3.c (cont)

Basis: (cont)

POTENTIAL LOSS - [Threshold Value #3]

When the Primary Containment design pressure is challenged, primary containment venting may be required even if offsite radioactivity release rate limits will be exceeded. This condition if compounded by further plant degradation may challenge primary containment integrity and is, therefore, an appropriate threshold for potential loss of the Primary Containment barrier.

Drywell pressure of 44 psig is based on the containment/drywell design pressure. If the containment design pressure is exceeded this represents a challenge to the containment structure because assumptions used in the accident analysis are no longer valid and an unanalyzed condition exists. This constitutes a potential loss of the containment barrier even if a breach has NOT occurred.

POTENTIAL LOSS - [Threshold Value #4]

Explosive 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/SAMGs 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. The specified values for this potential loss threshold are the minimum global deflagration concentration limits (6% hydrogen and 5% oxygen concentration).

Except for brief periods during plant startup and shutdown, oxygen concentration in the primary containment is maintained at insignificant levels by nitrogen inertion. The specified values for this potential loss threshold are readily recognizable because 6% hydrogen is above the hydrogen monitor alarm setpoint (minimum detectable concentration and the Primary Containment Control EOP entry condition). The minimum global deflagration hydrogen/oxygen concentrations require prior entry to the SAMGs and intentional primary containment venting. These conditions represent a potential barrier loss under Primary Containment 3.a. and a barrier loss under Primary Containment 3.d.2.

Basis Reference(s):

1. EMG-3200.02 Primary Containment Control
2. FSAR Update 6.2.1.1.3
3. Technical Specifications 5.2 Basis
4. NEI 99-01, Rev. 4 Table 5-F-2

Table D-2: OCGS EAL Technical Basis

RECOGNITION CATEGORY
HAZARDS AND OTHER CONITIONS AFFECTING PLANT SAFETY

MG1

Initiating Condition

Prolonged Loss of All Offsite Power and Prolonged Loss of All Onsite AC Power to Essential Busses.

Operating Mode Applicability:

1, 2

EAL Threshold Values:

1. Loss of power to Startup Transformers SA and SB.

AND

2. Failure of EDG-1 and EDG-2 Emergency Diesel Generators to supply power to 4160V Buses 1C and 1D.

AND

3. a. Restoration of at least one 4160V Bus (1C or 1D) within **1 hour** is <u>**not**</u> likely.

   OR

   b. RPV level <u>cannot</u> be determined to be **>< 0 inches TAF**.

Basis:

Loss of all AC power to the 4160V emergency buses compromises the availability of all plant safety systems. Prolonged loss of all AC power may lead to loss of Fuel Cladding, RCS and Primary Containment barriers. The one-hour interval to restore AC power to either emergency 4160V bus is based on the blackout coping analysis performed in conformance with 10 CFR 50.63 and Regulatory Guide 1.155, "Station Blackout."

The likelihood of restoring at least one emergency 4160V bus should be based on a realistic appraisal of the situation since a delay in an upgrade decision based on only a chance of mitigating the event could result in a loss of valuable time in preparing and implementing public protective actions. Emergency buses 1C and 1D can be powered from non-emergency buses 1A and 1B as well as from Emergency Diesel Generators EDG-1 and EDG-2. Buses 1A and 1B can be powered from Startup Transformers SA and SB. In addition, the SBO Transformer can also power bus 1B.

An additional source of offsite power is available when the main generator is off-line by backfeeding through the main power transformer and Auxiliary transformer. The backfeed operation must be manually performed and involves removal of flexible link connections between the main generator and the main power and auxiliary transformers. (Due to the time required to affect the backfeed, this source is likely only to be available when previously configured.)

Table D-2: OCGS EAL Technical Basis

RECOGNITION CATEGORY
HAZARDS AND OTHER CONITIONS AFFECTING PLANT SAFETY

MG1 (cont)

Basis (cont)

Under these conditions, fission product barrier monitoring capability may be degraded. Although it may be difficult to predict when power can be restored, it is necessary to give the Emergency Director a reasonable idea of how quickly to declare a General Emergency based on two major considerations:

1. Are there any present indications that core cooling is already degraded to the point that loss or potential loss of Fission Product Barriers is imminent?
2. If there are no present indications of such core cooling degradation, how likely is it that power can be restored in time to assure that a loss of two barriers with a potential loss of the third barrier can be prevented?

Thus, indication of continuing core cooling degradation must be based on Fission Product Barrier monitoring with particular emphasis on Emergency Director judgment as it relates to imminent loss or potential loss of fission product barriers and degraded ability to monitor fission product barriers.

When RPV water level is above the Top of Active Fuel (0 inches TAF), the core is completely submerged. Core submergence is the most desirable means of core cooling. When RPV water 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 water level drops below TAF, the level is indicative of a challenge to core cooling and the Fuel Cladding barrier.

Basis References:

1. OCNGS Drawing BR 3000
2. ABN-36, Loss of Off-Site Power
3. ABN-37, Station Blackout
4. ABN-60, Grid Emergency
5. Regulatory Guide 1.155, Station Blackout
6. TDR-1099, "Station Blackout Evaluation Report"
7. 2000-BAS-3200.02, EOP User's Guide
8. 2000-GLN-3200.01, Plant Specific Technical Guideline
9. NEI 99-01, Rev. 4 SG1

Table D-2: OCGS EAL Technical Basis

RECOGNITION CATEGORY
HAZARDS AND OTHER CONITIONS AFFECTING PLANT SAFETY

MG3

Initiating Condition

Failure of the Reactor Protection System to Complete an Automatic Scram and Manual Scram was NOT Successful and There is Indication of an Extreme Challenge to the Ability to Cool the Core.

Operating Mode Applicability:

1

EAL Threshold Values:

1. Automatic scram,~~and manual~~ Manual scram and ARI were **not** successful from Reactor Console as indicated by **EITHER**:

   a. Reactor Power remains **≥ 2%.**

   OR

   b. Torus temperature **> 110°F AND** boron injection required for reactivity control.

AND

2. a. RPV level **cannot** be restored and maintained **> -20 inches TAF**.

   OR

   b. Heat Capacity Temperature Limit (EMG-3200.02 Fig. F) exceeded.

Basis:

Automatic scram, manual scram and ARI are not considered successful if action away from the reactor control console was required to scram the reactor (i.e., actions from the console include mode switch to shutdown, using the manual scram pushbuttons, or manual ARI initiation).

Taking the mode switch to shutdown is a manual scram action.

This EAL encompasses events in which the automatic and manual scrams were not successful and the reactor is producing more heat than the maximum decay heat load for which the safety systems are designed. The reactor power threshold (2%) is approximately equal to the the APRM downscale setpoint and the maximum decay heat generation rate that should exist shortly after shutdown. Below the APRM downscale 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 2% power.

The torus water temperature criterion (110°F) is the Boron Injection Initiation Temperature (BIIT). The BIIT ensures that the Standby Liquid Control (SLC) system will inject the Hot Shutdown Boron Weight (HSBW) into the RPV before the total amount of energy rejected to the torus heats the suppression pool to the Heat Capacity Temperature Limit (HCTL). If suppression pool temperature exceeds the BIIT, reactor power is heating the suppression pool and the suppression pool cooling may be inadequate or incapable of performing its design function.

Table D-2: OCGS EAL Technical Basis

RECOGNITION CATEGORY
HAZARDS AND OTHER CONITIONS AFFECTING PLANT SAFETY

MG3 (cont)

Basis (cont)

The second condition of this EAL indicates either:

- An extreme challenge to the ability to cool the core as indicated when RPV water level cannot be maintained above -20 inches TAF. The specified water level is the Minimum Steam Cooling RPV Water Level (MSCRWL). The MSCRWL is the lowest RPV water 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 water level is below the top of active fuel. RPV water level below the MSCRWL for an extended period of time without satisfactory core spray cooling could be precursors of a core melt sequence.
- An extreme challenge to the primary containment as indicated when heat cannot be removed from the primary containment resulting in elevated torus water temperature. The Heat Capacity Temperature Limit is the highest torus water temperature from which a blowdown will not raise torus pressure above the Primary Containment Pressure Limit (PCPL) before the rate of energy transfer from the RPV to the primary containment is within the capacity of the primary containment vent. (When the PCPL is challenged, primary containment venting may be required even if offsite radioactivity release rate limits will be exceeded.) The HCTL is a function of RPV pressure and torus water temperature and is a measure of the maximum heat load, which the primary containment can withstand. Plant parameters in excess of the HCTL could be a precursor of primary containment failure. The Heat Capacity Limit is given in Figure F of EMG-3200.02, Primary Containment Control.

Basis Reference(s):

1. EMG-3200.01A, RPV Control – no ATWS
2. EMG-3200.01B, RPV Control – with ATWS
3. EMG-3200.02, Primary Containment Control
4. 2000-BAS-3200.02, EOP User's Guide
5. 2000-GLN-3200.01, Plant Specific Technical Guideline
6. NEI 99-01, Rev. 4 SG2

Table D-2: OCGS EAL Technical Basis

RECOGNITION CATEGORY
HAZARDS AND OTHER CONITIONS AFFECTING PLANT SAFETY

MS3

Initiating Condition

Failure of the Reactor Protection System Instrumentation to Complete or Initiate an Automatic Reactor Scram Once a Reactor Protection System Setpoint Has Been Exceeded and Manual Scram was NOT Successful.

Operating Mode Applicability:

1

EAL Threshold Values:

1. Automatic scram,~~and manual~~ Manual scram and ARI were **not** successful from Reactor Console as indicated by **EITHER**:

   a. Reactor Power remains **≥ 2%.**

   OR

   b. Torus temperature **> 110°F AND** boron injection required for reactivity control.

Basis:

Automatic scram, manual scram and ARI are not considered successful if action away from the reactor control console was required to scram the reactor (i.e., actions from the console include mode switch to shutdown, using the manual scram pushbuttons, or manual ARI initiation).

Taking the mode switch to shutdown is a manual scram action.

This EAL encompasses events in which the automatic and manual scrams were not successful and the reactor is producing more heat than the maximum decay heat load for which the safety systems are designed The reactor power threshold (2%) is approximately equal to the APRM downscale setpoint and the maximum decay heat generation rate that should exist shortly after shutdown. Below the APRM downscale 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~~≥ 2% power. Classification at the Site Area Emergency level is appropriate because conditions exist that can lead to imminent loss or potential loss of both the Fuel Cladding and RCS barriers.

The torus water temperature criterion (110°F) is the Boron Injection Initiation Temperature (BIIT). The BIIT ensures that the Standby Liquid Control (SLC) system will inject the Hot Shutdown Boron Weight (HSBW) into the RPV before the total amount of energy rejected to the torus heats the suppression pool to the Heat Capacity Temperature Limit (HCTL). If torus temperature exceeds the BIIT, reactor power is heating the torus and the suppression pool cooling may be inadequate or incapable of performing its design function.

Basis Reference(s):

1. EMG-3200.01A, RPV Control – no ATWS
2. EMG-3200.01B, RPV Control – with ATWS
3. 2000-BAS-3200.02, EOP User's Guide

Table D-2: OCGS EAL Technical Basis

RECOGNITION CATEGORY
HAZARDS AND OTHER CONITIONS AFFECTING PLANT SAFETY

4. 2000-GLN-3200.01, Plant Specific Technical Guideline
5. NEI 99-01, Rev. 4 SS2

Table D-2: OCGS EAL Technical Basis

RECOGNITION CATEGORY
HAZARDS AND OTHER CONITIONS AFFECTING PLANT SAFETY

MA3

Initiating Condition

Failure of the Reactor Protection System Instrumentation to Complete or Initiate an Automatic Reactor Scram Once a Reactor Protection System Setpoint Has Been Exceeded.

Operating Mode Applicability:

1

EAL Threshold Values:

1. A Reactor Protection System setpoint was exceeded.

AND

2. Automatic SCRAM did **not** reduce Reactor Power to subcritical with power below the Heating Range.

Basis:

This condition indicates a failure of the automatic Reactor Protection System (RPS) to successfully scram the reactor. This condition is more than a potential degradation of a safety system in that a front line automatic protection system did not function in response to a plant transient and thus the plant safety has been compromised, and design limits of the fuel may have been exceeded. Site-specific indication of reactor shutdown is included as the criteria of whether the scram was successful when required. An Alert is indicated because conditions exist that lead to potential loss of fuel clad or RCS. Reactor Protection System setpoint being exceeded, rather than limiting safety system setpoint being exceeded, is specified here because failure of the automatic protection system is the issue.

Taking the mode switch to shutdown is a manual scram action. ~~When the mode switch is taken out of the Run position, however, the nuclear instrumentation scram setpoint is lowered. If reactor power remains above the lowered setpoint, an automatic scram is initiated. If the RPS fails to achieve reactor shutdown, an automatic RPS setpoint has been exceeded and the automatic scram was not successful.~~

The first condition of this EAL identifies the need to cease critical reactor operations by actuation of the automatic Reactor Protection System (RPS) scram function.

The second condition of this EAL indicates a failure of the automatic RPS scram function to rapidly insert a sufficient number of control rods to achieve reactor shutdown. The CRD system backup scram valves and the Alternate Rod Insertion (ARI) system provide automatic, alternate methods of completing the scram function. These backups, however, insert control rods at a much slower rate than the automatic RPS scram function. For the purpose of emergency classification at the Alert level, reactor shutdown achieved by automatic backup scram valve operation and ARI initiation does not constitute a successful RPS automatic scram. The significance of the second condition, therefore, is that a potential degradation of a safety system exists because a front line automatic protection system did not function in response to a plant transient. Thus, plant safety has been compromised.

RECOGNITION CATEGORY
HAZARDS AND OTHER CONITIONS AFFECTING PLANT SAFETY

MA3 (cont)

Basis (cont):

Following any automatic RPS scram signal EMG-3200.01B RPV Control – with ATWS, prescribes 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 Alert.

This threshold indicates failure of all manual scram capability. While failure of all manual SCRAM capability does not challenge fuel design limits, it is indicative of a condition in which rapid reactor shutdown cannot be established prior to the fuel being challenged should an RPS setpoint subsequently be exceeded.

A manual scram is any set of actions by the reactor operator(s) at the reactor control console, which causes control rods to be rapidly inserted into the core, and brings the reactor subcritical, including manual scram buttons, Mode Switch and actuation of ARI.

If by procedure, operator actions include the initiation of an immediate manual scram following receipt of an automatic scram signal and there are no clear indications that the automatic scram failed (such as a time delay following indications that a scram setpoint was exceeded, or first-out annunciators), 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 consideration should be given to evaluating the fuel for potential damage, and the reporting requirements of 50.72 should be considered for the transient event.

Basis Reference(s):

1. EMG-3200.01A, RPV Control – no ATWS
2. EMG-3200.01B, RPV Control – with ATWS
3. 2000-BAS-3200.02, EOP User's Guide
4. 2000-GLN-3200.01, Plant Specific Technical Guideline
5. NEI 99-01, Rev. 4 SA2

Table D-2: OCGS EAL Technical Basis

RECOGNITION CATEGORY
HAZARDS AND OTHER CONITIONS AFFECTING PLANT SAFETY

MS8

Initiating Condition

Loss of RPV Inventory Affecting Core Decay Heat Removal Capability

Operating Mode Applicability:

3

EAL Threshold Values:

1. **Without** Primary or Secondary CONTAINMENT CLOSURE established:
   a. RPV level **< 84 inches TAF**.

   OR

   b. RPV level unknown for **> 30 minutes** with a loss of RPV inventory per Table M5 indications.

OR

2. **With** Primary or Secondary CONTAINMENT CLOSURE established:
   a. RPV level **< 0 inches TAF**.

   OR

   b. RPV level unknown for **> 30 minutes** with a loss of RPV inventory as evidenced by either of the following:~~indication of core uncovery as evidenced by Erratic Source Range Monitor indication.~~
      - Per Table M5 indications.
      - Erratic Source Range Monitor indication.

Table M5 – Indications of RCS Leakage
Unexplained Identified or Unidentified leakage rise
Unexplained Torus rise
Unexplained vessel make-up rise
Observation of leakage or Inventory loss

Basis:

CONTAINMENT CLOSURE: Containment Closure is considered to be Containment as required by Technical Specifications.

Table D-2: OCGS EAL Technical Basis

RECOGNITION CATEGORY
HAZARDS AND OTHER CONITIONS AFFECTING PLANT SAFETY

MS8 (cont)

Basis (cont)

Under the conditions specified by this EAL, continued decrease in RPV water level is indicative of a loss of inventory control. Inventory loss may be due to an RPV breach, RCS pressure boundary leakage or continued boiling in the RPV. If a low-pressure boundary to fission product release does not exist (i.e., containment closure is not established), the RPV water level associated with this threshold is six inches below the Core Spray ECCS actuation setpoint (i.e., 90 in. - 6 in. = 84 inches.). If containment closure is established, a low-pressure boundary to fission product release exists and RPV water level can decrease to the top of active fuel, 0 in. (TAF), before a Site Area Emergency declaration is required. 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.

In the refueling mode, when RPV water level indication is unavailable, the inventory loss must be detected by drywell floor and equipment drain sump pumpout rates or erratic Source Range Monitor indication. Sump pumpout rate rises must be evaluated against other potential sources of leakage such as cooling water sources inside the primary containment to ensure they are indicative of RCS leakage. Post-TMI studies indicate that the installed nuclear instrumentation will operate erratically when the core is uncovered and that Source Range Monitors can be used as a tool for making such determinations. As water level in the Reactor Vessel lowers, the dose rate above the core will rise. The dose rate due to this core shine, scattering and radiation bounce off of the solid surfaces in the area will result in readings on the Refuel floor ARMs indicating ≥ 3 R/hr. This threshold radiation value is based on calculations documented in EP-AEL-0501.

This threshold is based on concerns raised by Generic Letter 88-17, Loss of Decay Heat Removal, SECY 91-283, Evaluation of Shutdown and Low Power Risk Issues, NUREG-1449, Shutdown and Low-Power Operation at Commercial Nuclear Power Plants in the United States, and, NUMARC 91-06, Guidelines for Industry Actions to Assess Shutdown Management. A number of variables (e.g., decay heat removal system design, etc.) can have a significant impact on heat removal capability challenging the Fuel Cladding barrier. Analysis in the above references indicates that core damage may occur within an hour following continued core uncovery, therefore, the thirty-minute interval was conservatively chosen.

The 30 minute interval allows sufficient time for actions to be performed to recover needed cooling equipment.

Basis Reference(s):

1. EMG-3200.01A, RPV Control – No ATWS
2. 2000-GLN-3200.03, Plant Specific Technical Guidelines for EOPs
3. NEI 99-01, Rev. 4 CS1
4. EP-AEL-0501,, Estimation of Radiation Monitor Readings Indicating Core Uncovery

Table D-2: OCGS EAL Technical Basis

RECOGNITION CATEGORY
HAZARDS AND OTHER CONITIONS AFFECTING PLANT SAFETY

MU8

Initiating Condition

RCS Leakage.

Operating Mode Applicability:

3

EAL Threshold Values:

~~1. UNPLANNED loss of RPV Inventory per Table M5 indications.~~

~~**AND**~~

~~2~~1. RPV level **cannot** be restored and maintained **> 139 inches TAF**.

~~**Table M5 – Indications of RCS Leakage**~~
~~Unexplained Identified or Unidentified leakage rise~~ ~~Unexplained Torus rise~~ ~~Unexplained vessel make-up rise~~ ~~Observation of leakage or Inventory loss~~

Basis:

~~UNPLANNED: a parameter change or an event that is not the result of an intended evolution and requires corrective or mitigative actions.~~

The inability to restore and maintain level after reaching the RPS low level scram~~this~~ setpoint infers a degradation of the level of safety of the plant.

Basis Reference(s):

1. NEI 99-01, Rev. 4 CU1

Table D-2: OCGS EAL Technical Basis

RECOGNITION CATEGORY
HAZARDS AND OTHER CONITIONS AFFECTING PLANT SAFETY

MS9

Initiating Condition

Loss of RPV Inventory Affecting Core Decay Heat Removal Capability with Irradiated Fuel in the RPV.

Operating Mode Applicability:

4

EAL Threshold Values:

1. **Without** Secondary CONTAINMENT CLOSURE established:

   a. RPV level **< 84 inches TAF**.

   OR

   b. **RPV level unknown** with indication of core uncovery as evidenced by one or more of the following:~~for **> 30 minutes** with a loss of RPV inventory per Table M5 indications.~~

   - Refuel Floor Area Radiation Monitor C-10, North Wall reading **> 3 R/hr**.
   - Erratic Source Range Monitor indication.

OR

2. **With** Secondary CONTAINMENT CLOSURE established:

   a. RPV level **< 0 inches TAF**.

   OR

   b. RPV level unknown ~~for **> 30 minutes**~~ with indication of core uncovery as evidenced by one or more of the following:

   - Refuel Floor Area Radiation Monitor C-10, North Wall reading **> 3 R/hr**.
   - Erratic Source Range Monitor indication.

~~Table M5 – Indications of RCS Leakage~~
~~Unexplained Identified or Unidentified leakage rise~~ ~~Unexplained Torus rise~~ ~~Unexplained vessel make-up rise~~ ~~Observation of leakage or Inventory loss~~

Table D-2: OCGS EAL Technical Basis

RECOGNITION CATEGORY
HAZARDS AND OTHER CONITIONS AFFECTING PLANT SAFETY

MS9 (cont)

Basis (cont)

CONTAINMENT CLOSURE: Containment Closure is considered to be Containment as required by Technical Specifications.

Under the conditions specified by this EAL, continued decrease in RPV water level is indicative of a loss of inventory control. Inventory loss may be due to an RPV breach, RCS pressure boundary leakage or continued boiling in the RPV. If a low-pressure boundary to fission product release does not exist (i.e., containment closure is not established), the RPV water level associated with this threshold is six inches below the Core Spray ECCS actuation setpoint (i.e., 90 in. - 6 in. = 84 inches). If containment closure is established, a low-pressure boundary to fission product release exists and RPV water level can decrease to the top of active fuel, 0 inches (TAF), before a Site Area Emergency declaration is required. 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 RPV water level after reaching this setpoint infers a failure of the RCS barrier and potential loss of the Fuel Cladding barrier. If it is determined that RPV water level cannot be monitored, the need for declaration of the Site Area Emergency is evaluated.

Under the Refueling conditions specified in this EAL, loss of the ability to monitor RPV water level in conjunction with indirect indication of possible core uncovery infer a continued lowering in RPV water level and loss of inventory control. Inventory loss may be due to an RPV breach, RCS pressure boundary leakage or continued boiling in the RPV.

In the Refueling mode, when RPV water level indication is unavailable, the inventory loss must be detected by erratic Source Range Monitor indication or elevated refuel floor radiation.

Post-TMI studies indicate that the installed nuclear instrumentation will operate erratically when the core is uncovered and that Source Range Neutron Monitors can be used as a tool for making such determinations. The Refuel Floor ARMs reading > 3R/hr is based on calculation EP-AEL-0501, Estimation of Radiation Monitor Readings Indicating Core Uncovery During Refuel.

~~This EAL is based on concerns raised by Generic Letter 88-17, Loss of Decay Heat Removal, SECY 91-283, Evaluation of Shutdown and Low Power Risk Issues, NUREG-1449, Shutdown and Low-Power Operation at Commercial Nuclear Power Plants in the United States, and, NUMARC 91-06, Guidelines for Industry Actions to Assess Shutdown Management. A number of variables (e.g., decay heat removal system design, etc.) can have a significant impact on heat removal capability challenging the Fuel Cladding barrier. Analysis in the above references indicates that core damage may occur within an hour following continued core uncovery therefore, the thirty-minute interval was conservatively chosen.~~

Table D-2: OCGS EAL Technical Basis

RECOGNITION CATEGORY
HAZARDS AND OTHER CONITIONS AFFECTING PLANT SAFETY

MS9 (cont)

Basis Reference(s):

1. EMG-3200.01A, RPV Control – No ATWS
2. 2000-GLN-3200.03, Plant Specific Technical Guidelines for Severe Accident Guidelines
3. EP-AEL-0501, Estimation of Radiation Monitor Readings Indicating Core Uncovery
4. NEI 99-01, Rev. 4 CS2

Table D-2: OCGS EAL Technical Basis

RECOGNITION CATEGORY
HAZARDS AND OTHER CONITIONS AFFECTING PLANT SAFETY

HA5

Initiating Condition

Natural and Destructive Phenomena Affecting the Plant VITAL AREA.

Operating Mode Applicability:

1, 2, 3, 4, D

EAL Threshold Values:

1. A ~~C~~confirmed **Seismic event** that affects safety systems or systems required for safe shutdown **requiring reactor scram**~~ in accordance with ABN-38, Station Seismic Event~~.

OR

2. Tornado or high winds **> ~~100~~ 99 mph** within PROTECTED AREA boundary and resulting in VISIBLE DAMAGE to any plant structures or equipment contained in any Table H2 area or Control Room indication of degraded performance of those systems.

Table H2 – Vital Areas
Reactor Bldg
Control Room Complex
Main Transformer/Condensate Transfer Pad
Intake Structure
#1 EDG Vault
#2 EDG Vault
EDG Fuel Oil Storage Tank

OR

3. Vehicle crash within PROTECTED AREA boundary and resulting in VISIBLE DAMAGE to any plant structures or equipment contained in any Table H2 area or Control Room indication of degraded performance of those systems.

OR

4. Turbine failure-generated missiles result in any VISIBLE DAMAGE or penetration of any Table H2 area.

Table D-2: OCGS EAL Technical Basis

RECOGNITION CATEGORY
HAZARDS AND OTHER CONITIONS AFFECTING PLANT SAFETY

HA5 (cont)

EAL Threshold Value (cont)

OR

5. Uncontrolled flooding that results in **EITHER**:
   a. Degraded safety system performance in any Table H3 area as indicated in the Control Room.

   OR

   b. Industrial safety hazards (e.g., electric shock) that preclude access necessary to operate or monitor safety equipment.

Table H3 – Internal Flooding Areas
Reactor Building NE Corner Room
Reactor Building SE Corner Room
Reactor Building SW Corner Room (RBEDT Rm)
Reactor Building NW Corner Room (CRD Pp Rm)

OR

6. Abnormal Intake Structure level, as indicated by **EITHER**:
   - **> 6.0 ft. MSL** (> 4.92 psig on PI-533-1172 and PI-533-1173 or > 6.0 ft MSL on CR-423-11 pt 24 and pt 23).

   OR

   - **≤ -4.0 ft. MSL** (≤ 0.50 psig on PI-533-1172 and PI-533-1173 or ≤ -4.0 ft MSL on CR-423-11 pt 24 and pt 23).

   MSL = Mean Sea Level

Basis:

VITAL AREA: is any area, normally within the PROTECTED AREA, which contains equipment, systems, components, or material, the failure, destruction, or release of which could directly or indirectly endanger the public health and safety by exposure to radiation.

PROTECTED AREA: is an area, which normally encompasses all controlled areas within the security protected area fence

VISIBLE DAMAGE: is damage to equipment or structure that is readily observable without measurements, testing, or analysis.~~PHYSICAL DAMAGE~~ Damage is sufficient to cause concerns regarding the continued operability or reliability of affected safety structures, systems, or components. Example damage includes: deformation due to heat or impact, denting, penetration, rupture, cracking, paint blistering. Surface blemishes (e.g., paint chipping, scratches) should not be included.

~~PHYSICAL DAMAGE: is damage to equipment or structure that is readily observable without measurements, testing, or analysis.~~

Table D-2: OCGS EAL Technical Basis

RECOGNITION CATEGORY
HAZARDS AND OTHER CONITIONS AFFECTING PLANT SAFETY

HA5 (cont)

Basis (cont)

Threshold #1 Basis:

This threshold addresses events that may have resulted in a VITAL AREA (Table H2) being subjected to forces beyond design limits and thus damage may be assumed to have occurred to plant safety systems. The initial report should not be interpreted as mandating a lengthy damage assessment prior to classification. No attempt is made in this EAL to assess the actual magnitude of the damage.

A reactor scram is required by procedure ABN-38, Station Seismic Event, if:

- The seismic event affects safe plant operation by jeopardizing the availability of safety systems, systems required to complete safe shutdown, or causing spurious actuation of equipment, or
- The Shift Manager determines it necessary to scram the Reactor to protect public safety.

Threshold #2 Basis:

This threshold addresses events that may have resulted in a Vital Area being subjected to forces beyond design limits and thus damage may be assumed to have occurred to plant safety systems. Vital Areas are areas that house equipment the operation of which may be needed to ensure the reactor safely reaches and is maintained in its lowest energy state. Vital Areas include structures that are in contact with or immediately adjacent to the areas that actually contains the equipment of concern. The Alert classification is appropriate if relevant plant parameters indicate that the performance of safety systems in the affected Vital Areas has been degraded. No attempt should be made to fully inventory the actual magnitude of the damage or quantify the degradation of safety system performance prior to declaration of an Alert under this threshold. The declaration of an Alert and the activation of the TSC provide the Emergency Director with the resources needed to perform detailed damage assessments.

The wind speed threshold is the FSAR design basis wind speed. Sustained winds present a more severe loading on the buildings than a gust.

Table D-2: OCGS EAL Technical Basis

RECOGNITION CATEGORY
HAZARDS AND OTHER CONITIONS AFFECTING PLANT SAFETY

HA5 (cont)

Basis (cont)

Threshold #3 Basis:

This threshold addresses events such as plane, helicopter, train, barge, car or truck crashes, or impact of projectiles into a plant Vital Area. This threshold addresses vehicle crashes that challenge the operability of systems necessary for safe shutdown of the plant. Vital Areas (Table H2) include Class 1 structures and those Class 2 structures that contain Class 1 Systems and components.

The Alert classification is appropriate if relevant plant parameters indicate that the performance of safety systems in the affected Vital Areas has been degraded. No attempt should be made to fully inventory the actual magnitude of the damage or quantify the degradation of safety system performance prior to declaration of an Alert under this threshold. The declaration of an Alert and the activation of the TSC provide the Emergency Director with the resources needed to perform detailed damage assessments. The Emergency Director also needs to consider the security ramifications of such crashes.

Threshold #4 Basis:

This EAL is intended to address the threat to safety related equipment impacted by missiles generated by main turbine rotating component failures. Table H2 includes all areas containing safety related equipment, their controls, and their power supplies.

Threshold #5 Basis:

This Threshold addresses the effect of internal flooding that has resulted in degraded performance of systems affected by the flooding, or has created industrial safety hazards (e.g., electrical shock) that preclude necessary access to operate or monitor safety equipment. The inability to operate or monitor safety equipment represents a potential for substantial degradation of the level of safety of the plant. This flooding may have been caused by internal events such as component failures, equipment misalignment, and fire suppression system actuation or outage activity mishaps. The Internal Flooding Areas listed in Table H3 include areas containing systems that are:

- Required for safe shutdown of the plant
- Not designed to be wetted or submerged
- Susceptible to internal flooding events

Table D-2: OCGS EAL Technical Basis

RECOGNITION CATEGORY
HAZARDS AND OTHER CONITIONS AFFECTING PLANT SAFETY

HA5 (cont)

Basis (cont)

Threshold #6 Basis:

This threshold covers high and low water level conditions as well as internal flooding events that may have resulted in a plant Vital Area being subjected to levels beyond design limits, and thus damage may be assumed to have occurred to plant safety systems.

Both pressure gauges listed (PI-533-1172 and 1173) are located inside each of the intake structure bays; they should both be considered to determine the overall effect on plant operations due to water level changes, since one structure could be isolated or have its associated screens clogged resulting in a low intake level condition on one side of the intake structure and therefore not provide positive indication of actual water level trend.

High Intake Structure level, > 6.0 feet MSL (> 4.92 psig on PI-533-1172 [CR-423-11 pt 24] and PI-533-1173 [CR-423-11 pt 23]) is capable of causing flooding that can affect Plant Vital Structures. At levels > 6.5 ft. above MSL, Circulating Water Pumps may become flooded. At levels > 8.0 ft. above MSL, Service Water pumps may become flooded. No attempt should be made to determine the magnitude of flooding. This is a long lead-time event but this level is at the intake structure lower deck so classification as an Alert is appropriate. The evidence of flooding is sufficient for declaration. PI-533-1172 and PI-533-1173 are local pressure indicators that provide a reading in psig that corresponds to Mean Sea Level (MSL) in feet. CR-423-11 points 23 and 24 are Main Control Room indications of intake bay levels displayed directly in feet MSL. CR-423-11 points 23 and 24 receive the same instrument signals and provide the same related indications as their associated local indicators discussed above.

Low Intake Structure level ≤ -4.0 feet MSL (< 0.50 psig on PI-533-1172 [CR-423-11 pt 24] and PI-533-1173 [CR-423-11 pt 23]) indicates the possible loss of Emergency Service Water pumps. PI-533-1172 and PI-533-1173 are local pressure indicators that provide a reading in psig that corresponds to Mean Sea Level (MSL) in feet. CR-423-11 points 23 and 24 are Main Control Room indications of intake bay levels displayed directly in feet MSL. CR-423-11 points 23 and 24 receive the same instrument signal and provide the same related indications as their associated local indicators discussed above.

Basis Reference(s):

1. ABN-38, Station Seismic Event
2. FSAR Update Section 3.3.7 (Seismic)
3. FSAR Update Section 3.3.1 (High winds)
4. ABN-31, High Winds
5. ABN-32, Abnormal Intake Level
6. ABN-29, Plant Fires
7. LES Calculation No. 72-01-01, Turbine Missile Analysis for New Monoblock Rotor and Blades," October 1996, Revision 3
8. NEI 99-01, Rev. 4 HA1

Table D-2: OCGS EAL Technical Basis

RECOGNITION CATEGORY
HAZARDS AND OTHER CONITIONS AFFECTING PLANT SAFETY

HU5

Initiating Condition

Natural and Destructive Phenomena Affecting the PROTECTED AREA.

Operating Mode Applicability:

1, 2, 3, 4, D

EAL Threshold Values:

1. a. Seismic event felt in plant.

   AND

   b. Confirmed by National Earthquake Center.

OR

2. Report by plant personnel of tornado striking or high winds **> ~~100~~ 99 mph** within PROTECTED AREA boundary.

OR

3. Vehicle crash into plant structures or systems within PROTECTED AREA boundary affecting Table H2 area ~~causing PHYSICAL DAMAGE~~.

Table H2 – Vital Areas
Reactor Bldg
Control Room Complex
Main Transformer/Condensate Transfer Pad
Intake Structure
#1 EDG Vault
#2 EDG Vault
EDG Fuel Oil Storage Tank

OR

4. Report of turbine failure resulting in casing penetration or damage to turbine or generator seals.

RECOGNITION CATEGORY
HAZARDS AND OTHER CONITIONS AFFECTING PLANT SAFETY

HU5 (cont.)

EAL Threshold Value (cont)

OR

5. Uncontrolled flooding in any Table H3 area that has the potential to affect safety related equipment needed for the current operating mode.

Table H3 Internal Flooding Areas
Reactor Building NE Corner Room
Reactor Building SE Corner Room
Reactor Building SW Corner Room (RBEDT Rm)
Reactor Building NW Corner Room (CRD Pp Rm)

OR

6. Abnormal Intake Structure level, as indicated by **EITHER**:
   - **> 4.5 ft. MSL** (> 4.26 psig on PI-533-1172 and PI-533-1173 or > 4.5 ft MSL on CR-423-11 pt 24 and pt 23).

   OR

   - **≤ -3.0 ft. MSL** (≤ 0.94 psig on PI-533-1172 and PI-533-1173 or ≤ -3.0 ft MSL on CR-423-11 pt 24 and pt 23).

   MSL = Mean Sea Level

Basis:

PROTECTED AREA: is an area, which normally encompasses all controlled areas within the security protected area fence.

~~PHYSICAL DAMAGE: is damage to equipment or structure that is readily observable without measurements, testing, or analysis.~~

~~VISIBLE DAMAGE: is PHYSICAL DAMAGE sufficient to cause concerns regarding the continued operability or reliability of affected safety structures, systems, or components. Example damage includes: deformation due to heat or impact, denting, penetration, rupture, cracking, paint blistering. Surface blemishes (e.g., paint chipping, scratches) should not be included.~~

Table D-2: OCGS EAL Technical Basis

RECOGNITION CATEGORY
HAZARDS AND OTHER CONITIONS AFFECTING PLANT SAFETY

HU5 (cont)

Basis: (cont)

Threshold #1 Basis:

Confirmation from the National Earthquake center shall not delay declaration in the presence of other reliable confirming indications.

A felt earthquake is an earthquake of sufficient intensity such that the vibratory ground motion is felt at the nuclear plant site. An earthquake of this magnitude may be sufficient to cause minor damage to plant structures or equipment within the Protected Area. Damage is considered to be minor, as it would not affect physical or structural integrity. This event is not expected to affect the capabilities of plant safety functions.

The method of earthquake detection relies on the agreement of the shift operators on duty in the Control Room that the suspected ground motion is a "felt earthquake" followed by one or more reports confirming seismic activity near the station. Consensus of the Control Room operators with respect to ground motion helps avoid unnecessary classification if the motion were not due to seismic motion. As defined in the EPRI-sponsored "Guidelines for Nuclear Plant Response to an Earthquake", dated October 1989, a "felt earthquake" is:

"An earthquake of sufficient intensity such that: (a) the inventory ground motion is felt at the nuclear plant site and recognized as an earthquake based on a consensus of Control Room operators on duty at the time, and (b) for plants with operable seismic instrumentation, the seismic switches of the plant are activated. For most plants with seismic instrumentation, the seismic switches are set at an acceleration of about 0.01 g."

Threshold #2 Basis:

A tornado touching down within the Protected Area or sustained wind speeds > ~~100~~ 99 mph within the Owner Controlled Area are of sufficient velocity to have the potential to cause damage to ~~Plant~~ Vital ~~Structures~~Areas. These criteria are indicative of unstable weather conditions and represent a potential degradation in the level of safety of the plant. Verification of a tornado will be by direct observation and reporting by station personnel. Verification of wind speeds > ~~100~~ 99 mph will be via meteorological data in the control room. This event will be escalated to an Alert if the tornado or high wind speeds result in damage to ~~Plant~~ Vital ~~Structures~~Areas.

Threshold #3 Basis:

In this context, a "vehicle crash" is intended to address crashes of vehicle types large enough to cause significant damage to plant structures containing functions and systems required for safe shutdown of the plant.~~This threshold addresses events such as plane, helicopter, train, barge, car or truck crashes that may potentially damage plant structures containing functions and systems required for safe shutdown of the plant.~~

Table D-2: OCGS EAL Technical Basis

RECOGNITION CATEGORY
HAZARDS AND OTHER CONITIONS AFFECTING PLANT SAFETY

HU5 (cont)

Basis: (cont)

Threshold #4 Basis:

This Threshold addresses main turbine rotating component failures of sufficient magnitude to cause observable damage to the turbine casing or to the seals of the turbine generator. Of major concern is the potential for significant leakage of combustible fluids (lubricating oils) and gases (hydrogen cooling) to the plant environs. It is not the intent of this EAL to classify minor operational leakage.

Threshold #5 Basis:

This threshold addresses the effect of flooding caused by internal events such as component failures, equipment misalignment, fire suppression system actuation or outage activity mishaps. The Internal Flooding Areas of concern for the Unusual Event declaration are those Table H3 areas that have the potential to affect safety related equipment needed for the current operating mode.

Threshold #6 Basis:

Both pressure gauges listed (PI-533-1172 and 1173) are located inside each of the intake structure bays; they should both be considered to determine the overall effect on plant operations due to water level changes, since one structure could be isolated or have its associated screens clogged resulting in a low intake level condition on one side of the intake structure and therefore not provide positive indication of actual water level trend.

High Intake Structure level, > 4.5 feet MSL (> 4.26 psig on PI-533-1172 [CR-423-11 pt 24] and PI-533-1173 [CR-423-11 pt 23]) is sufficiently high to require plant shutdown per ABN-32, Abnormal Intake Level. This event will be escalated to an Alert classification based on water level reaching the elevation of the Intake Structure lower deck. PI-533-1172 and PI-533-1173 are local pressure indicators that provide a reading in psig that corresponds to Mean Sea Level (MSL) in feet. CR-423-11 points 23 and 24 are Main Control Room indications of intake bay levels displayed directly in feet MSL. CR-423-11 points 23 and 24 receive the same instrument signals and provide the same related indications as their associated local indicators discussed above.

Low Intake Structure level ≤ -3.0 feet MSL (< 0.94 psig on PI-533-1172 [CR-423-11 pt 24] and PI-533-1173 [CR-423-11 pt 23]) indicates the possible loss of Radwaste Service Water pumps and is approaching levels, which may result in a loss of vital cooling equipment. This event will be escalated to an Alert based upon water level dropping to ≤ -4.0 feet MSL. PI-533-1172 and PI-533-1173 are local pressure indicators that provide a reading in psig that corresponds to Mean Sea Level (MSL) in feet. CR-423-11 points 23 and 24 are Main Control Room indications of intake bay levels displayed directly in feet MSL. CR-423-11 points 23 and 24 receive the same instrument signal and provide the same related indications as their associated local indicators discussed above.

Table D-2: OCGS EAL Technical Basis

RECOGNITION CATEGORY
HAZARDS AND OTHER CONITIONS AFFECTING PLANT SAFETY

HU5 (cont)

Basis Reference(s):

1. ABN-38, Station Seismic Event
2. FSAR Update 3.3.7 (Seismic)
3. FSAR Update 3.3.1 (High winds)
4. ABN-31, High Winds
5. ABN-32, Abnormal Intake Level
6. ABN-29, Plant Fire
7. NEI 99-01, Rev. 4 HU1

Table D-2: OCGS EAL Technical Basis

RECOGNITION CATEGORY
HAZARDS AND OTHER CONITIONS AFFECTING PLANT SAFETY

HA6

Initiating Condition

FIRE or EXPLOSION Affecting the Operability of Plant Safety Systems Required to Establish or Maintain Safe Shutdown.

Operating Mode Applicability:

1, 2, 3, 4, D

EAL Threshold Values:

1. FIRE or EXPLOSION in any Table H2 area.

Table H2 – Vital Areas
Reactor Bldg
Control Room Complex
Main Transformer/Condensate Transfer Pad
Intake Structure
#1 EDG Vault
#2 EDG Vault
EDG Fuel Oil Storage Tank

AND

2. a. Affected safety system parameter indications show degraded performance.

   OR

   b. Plant personnel report VISIBLE DAMAGE to permanent structures or safety system equipment within the specified area.

Basis:

FIRE: is 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.

EXPLOSION: is a rapid, violent, unconfined combustion, or catastrophic failure of pressurized equipment that imparts energy of sufficient force to potentially damage permanent structures, systems, or components.

Table D-2: OCGS EAL Technical Basis

RECOGNITION CATEGORY
HAZARDS AND OTHER CONITIONS AFFECTING PLANT SAFETY

HA6 (cont.)

Bases: (cont)

VISIBLE DAMAGE: is damage to equipment or structure that is readily observable without measurements, testing, or analysis.~~PHYSICAL DAMAGE~~ Damage is sufficient to cause concerns regarding the continued operability or reliability of affected safety structures, systems, or components. Example damage includes: deformation due to heat or impact, denting, penetration, rupture, cracking, paint blistering. Surface blemishes (e.g., paint chipping, scratches) should not be included.

~~PHYSICAL DAMAGE: is damage to equipment or structure that is readily observable without measurements, testing, or analysis.~~

The areas listed in Table H2 are VITAL AREAs that house equipment the operation of which may be needed to ensure the reactor safely reaches and is maintained in its lowest energy state. Personnel access to these areas may be an important factor in monitoring and controlling equipment operability. This EAL addresses fires and explosions that challenge the operability of systems necessary for safe shutdown of the plant.

The only fires and explosions that should be considered are those of sufficient force to visibly damage permanent structures or equipment required for safe shutdown. Visual observation of damage infers the ability to approach or enter the affected Vital Areas. Lacking the ability to adequately inspect the area for damage, the Alert classification is appropriate if relevant plant parameters indicate that the performance of safety systems in the affected Vital Areas has been degraded. No attempt should be made to fully inventory the actual magnitude of the damage or quantify the degradation of safety system performance prior to declaration of an Alert under this EAL. The declaration of an Alert and the activation of the TSC provide the Emergency Director with the resources needed to perform detailed damage assessments.

A steam line break or steam explosion that damages permanent structures or equipment in a Vital Area would be classified under this EAL. The method of damage is not as important as the degradation of plant structures or equipment.

Basis Reference(s):

1. ABN-29, Plant Fires
2. NEI 99-01, Rev. 4 HA2

Table D-2: OCGS EAL Technical Basis

RECOGNITION CATEGORY
HAZARDS AND OTHER CONITIONS AFFECTING PLANT SAFETY

HU6

Initiating Condition

FIRE Not Extinguished Within 15 Minutes of Detection, or EXPLOSION, within PROTECTED AREA Boundary.

Operating Mode Applicability:

1, 2, 3, 4, D

EAL Threshold Values:

1. FIRE in any Table H2 area <u>**not**</u> extinguished within **15 minutes** of Control Room notification or verification of a Control Room alarm.

OR

2. FIRE outside any Table H2 area with the potential to damage safety systems in any Table H2 area <u>**not**</u> extinguished within **15 minutes** of Control Room notification or verification of a Control Room alarm.

Table H2 – Vital Areas
Reactor Bldg
Control Room Complex
Main Transformer/Condensate Transfer Pad
Intake Structure
#1 EDG Vault
#2 EDG Vault
EDG Fuel Oil Storage Tank

OR

3. Report by plant personnel of an unanticipated EXPLOSION within PROTECTED AREA boundary resulting in ~~PHYSICAL~~ VISIBLE DAMAGE to permanent structure or equipment.

Basis:

<u>FIRE</u>: is 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.

<u>PROTECTED AREA</u>: is an area, which normally encompasses all controlled areas within the security protected area fence.

<u>EXPLOSION</u>: is a rapid, violent, unconfined combustion, or catastrophic failure of pressurized equipment that imparts energy of sufficient force to potentially damage permanent structures, systems, or components.

Table D-2: OCGS EAL Technical Basis

RECOGNITION CATEGORY
HAZARDS AND OTHER CONITIONS AFFECTING PLANT SAFETY

VISIBLE DAMAGE: is damage to equipment or structure that is readily observable without measurements, testing, or analysis. Damage is sufficient to cause concern regarding the continued operability or reliability of affected safety structure, system, or component. Example damage includes: deformation due to heat or impact, denting, penetration, rupture, cracking, paint blistering. Surface blemishes (e.g., paint chipping, scratches) should not be included.~~PHYSICAL DAMAGE: is damage to equipment or structure that is readily observable without measurements, testing, or analysis.~~

Table D-2: OCGS EAL Technical Basis

RECOGNITION CATEGORY
HAZARDS AND OTHER CONITIONS AFFECTING PLANT SAFETY

HU6 (cont)

Basis: (cont)

Threshold #1 and #2 Basis

For the purposes of declaring an emergency event, the term "extinguished" means no visible flames.

The purpose of this threshold is to address the magnitude and extent of fires that may be potentially significant precursors to damage to safety systems. As used here, notification is visual observation and report by plant personnel or sensor alarm indication. The 15-minute period begins with a credible notification that a fire is occurring or indication of a valid fire detection system alarm. A verified alarm is assumed to be an indication of a fire unless personnel dispatched to the scene disprove the alarm within the 15-minute period. The report, however, shall not be required to verify the alarm.

The intent of the 15-minute period is to size the fire and discriminate against small fires that are readily extinguished (e.g., smoldering waste paper basket, etc.). Such fires are excluded from consideration in this EAL since they have no safety consequence.

Areas directly associated with a fire that may be considered to have a hazardous environment (due to smoke or administrative controls awaiting ventilation and/or testing) do not warrant declaration of an Alert under EAL HA7, Release of Toxic or Flammable Gases. However, an IDLH atmosphere resulting from the discharge of a fire-extinguishing agent (Cardox or Halon) should be evaluated under EAL HA7.

Threshold #3 Basis:

The only EXPLOSIONS that should be considered are those of sufficient force to visibly damage permanent structures or equipment in the PROTECTED AREA.

A steam line break or steam explosion that damages permanent structures or equipment in a PROTECTED AREA would be classified under this EAL. The method of damage is not as important as the degradation of plant structures or equipment.

Basis Reference(s):

1. ABN-29, Plant Fires
2. Station Security Plan – Appendix C
3. NEI 99-01, Rev. 4 HU2

Table D-2: OCGS EAL Technical Basis

RECOGNITION CATEGORY
HAZARDS AND OTHER CONITIONS AFFECTING PLANT SAFETY

HA7

Initiating Condition

Release of Toxic or Flammable Gases Within or Contiguous to a VITAL AREA Which Jeopardizes Operation of Systems Required to Maintain Safe Operations or Establish or Maintain Safe Shutdown.

Operating Mode Applicability:

1, 2, 3, 4, D

EAL Threshold Values:

1. Report or detection of toxic or asphyxiant gases within a Table H2 area (or area that restricts access to a Table H2 area) in concentrations that result in an atmosphere IMMEDIATELY DANGEROUS TO LIFE AND HEALTH (IDLH).

OR

2. Report or detection of flammable gases within a Table H2 area (or area that restricts access to a Table H2 area) in concentration greater than the LOWER FLAMMABILITY LIMIT (LFL).

Table H2 – Vital Areas
Reactor Bldg
Control Room Complex
Main Transformer/Condensate Transfer Pad
Intake Structure
#1 EDG Vault
#2 EDG Vault
EDG Fuel Oil Storage Tank

Basis:

VITAL AREA: is any area, normally within the PROTECTED AREA, which contains equipment, systems, components, or material, the failure, destruction, or release of which could directly or indirectly endanger the public health and safety by exposure to radiation.

IMMEDIATELY DANGEROUS TO LIFE AND HEALTH (IDLH): A condition that either poses an immediate threat to life and health or an immediate threat of severe exposure to contaminants which are likely to have adverse delayed effects on health.

LOWER FLAMMABILITY LIMIT (LFL): The minimum concentration of a combustible substance that is capable of propagating a flame through a homogenous mixture of the combustible and a gaseous oxidizer.

Bases: (cont)

This EAL is based on toxic, asphyxiant or flammable gases that have entered a plant structure in concentrations that are unsafe for plant personnel and, therefore, preclude access to equipment necessary for the safe operation of the plant. Toxic or flammable gases detected outside of these areas need not be considered for this EAL unless there is a spread of the gases into one of these areas.

Concentrations above life-threatening or flammable concentrations that result from planned maintenance or repair activities on-site, where planned contingency measures are identified to monitor and control gas(es), do not require classification.

Threshold #1:

Declaration should not be delayed for conformation from atmospheric testing if it is reasonable to conclude that the IDLH concentrations have been met (e.g. documented analysis, indication of personal ill effects from exposure, or operating experience with the hazards).~~An atmosphere that is IDLH may be determined by:~~

~~Direct measurement, or~~

~~Other indication of personal ill effects from exposure, or~~

~~A judgment that respirators must be worn for entry to the area."~~

~~Should a situation arise where there is no direct measurement of the gas concentration, the fact that the building needs to be evacuated due to the release of the gas would satisfy the "Other indication of personal ill effects from exposure" condition and the determination should be made that an IDLH atmosphere exists.~~

Areas directly associated with a fire that may be considered to have a hazardous environment (due to smoke or administrative controls awaiting ventilation and/or testing) do not warrant declaration of an Alert under this EAL. However, an IDLH atmosphere resulting from the discharge of a fire-extinguishing agent (Cardox or Halon) should be evaluated under this EAL.

The first condition is met if measurement of toxic gas concentration results in an atmosphere that is immediately dangerous to life and health (IDLH) within a Table H2 area. Non-Toxic Gases which displace oxygen (Halon or Nitrogen) to a life threatening level due to asphyxiation (oxygen deprivation) should also be considered for this EAL.

An asphyxiant is a material 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.

Threshold #2:

The second condition is met when the flammable gas concentration in a Table H2 area exceeds the lower flammability limit. Flammable gases such as hydrogen and acetylene are routinely used to maintain plant systems (hydrogen – main generator cooling, reactor coolant chemistry control) or repair equipment/components (acetylene - welding). This condition

Table D-2: OCGS EAL Technical Basis

RECOGNITION CATEGORY
HAZARDS AND OTHER CONITIONS AFFECTING PLANT SAFETY

addresses concentrations at which gases can ignite or support combustion. An uncontrolled release of flammable gases within a Table H2 area has the potential to affect safe operation of the plant by limiting either operator or equipment operations due to the potential for ignition and resulting equipment damage or personnel injury. Once it has been determined that an uncontrolled release of flammable gas is occurring, sampling must be done to determine if the gas concentration exceeds the lower flammability limit.

Table D-2: OCGS EAL Technical Basis

RECOGNITION CATEGORY
HAZARDS AND OTHER CONITIONS AFFECTING PLANT SAFETY

HA7 (cont.)

Basis Reference(s):

1. ABN-33, Toxic or Flammable Gas Release
2. NEI 99-01, Rev. 4 HA3

Table D-2: OCGS EAL Technical Basis

RECOGNITION CATEGORY
HAZARDS AND OTHER CONITIONS AFFECTING PLANT SAFETY

HU7

Initiating Condition

Release of Toxic or Flammable Gases Deemed Detrimental to Normal Operation of the Plant.

Operating Mode Applicability:

1, 2, 3, 4, D

EAL Threshold Values:

1. Report or detection of toxic, asphyxiant or flammable gases that has or could enter the site area boundary in amounts that can affect NORMAL PLANT OPERATIONS.

OR

2. Report by Local, County or State Officials for evacuation or sheltering of site personnel based on an offsite event.

Basis:

NORMAL PLANT OPERATIONS: activities at the plant site associated with routine testing, maintenance, or equipment operations, in accordance with normal operating or administrative procedures. Entry into abnormal or emergency operating procedures, or deviation from normal security or radiological controls posture, is a departure from NORMAL PLANT OPERATIONs.

This EAL is based on the existence of uncontrolled releases of toxic, asphyxiant or flammable gas affecting safe plant operations or the health of plant personnel. The release may have originated within the Protected Area boundary, or it may have originated offsite and subsequently drifted inside the Protected Area boundary. Offsite events (e.g., tanker truck accident releasing toxic gases, etc.) resulting in the plant being within the evacuation area should also be considered in this EAL because of the adverse affect on normal plant operations.

It is intended that releases of toxic, asphyxiant or flammable gases are of sufficient quantity and the release point of such gases is such that safe plant operations would be affected. This would preclude small or incidental releases, or releases that do not impact structures needed for safe plant operation. The EAL is not intended to require significant assessment or quantification. The EAL assumes an uncontrolled process that has the potential to affect safe plant operations or plant personnel safety.

An Asphyxiant is a material 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.

Basis Reference(s):

1. ABN-33, Toxic or Flammable Gas Release
2. NEI 99-01, Rev. 4 HU3

ENCLOSURE 7D

OYSTER CREEK GENERATING STATION ANNEX

EP-AA-1010

"EAL Bases for Approval"

RG1	FPB RCS 2.a	MS8
RS1	FPB RCS 2.b	MU8
RA1	FPB Cont 3.b	MS9
RU1	FPB Cont 3.d	HA5
RA2	MG1	HU5
RU2	MG3	HA6
RA3	MS3	HU6
RU3	MA3	HA7
FPB FC1.a	MA5	HU7
FPB FC1.b	MG8	

Table D-2: OCGS EAL Technical Basis

RECOGNITION CATEGORY
ABNORMAL RAD LEVELS / RADIOLOGICAL EFFLUENT

RG1

Initiating Condition

Offsite Dose Resulting from an Actual or Imminent Release of Gaseous Radioactivity Exceeds 1000 mRem TEDE or 5000 mRem Thyroid CDE for the Actual or Projected Duration of the Release Using Actual Meteorology.

Operating Mode Applicability:

1, 2, 3, 4, D

EAL Threshold Values

Note: If dose assessment results are available at the time of declaration, the classification should be based on EAL Threshold #2 instead of EAL Threshold #1. Do **not** delay declaration awaiting dose assessment results.

1. VALID reading on one or more of the following radiation monitors that exceeds or is expected to exceed the reading shown (Table R1) for **≥ 15 minutes**:

Table R1 – Effluent Monitor Thresholds	
	General Emergency
Main Stack RAGEMS	1.27 E+02 μCi/cc HRM
Turbine Building RAGEMS	1.40 E+00 μCi/cc HRM

HRM = High Range Monitor LRM = Low Range Monitor

OR

2. Dose assessment using actual meteorology indicates doses at or beyond the Site Boundary of **EITHER**:

   a. **> 1000 mRem TEDE**

   OR

   b. **> 5000 mRem CDE Thyroid**

OR

3. Field survey results at or beyond Site Boundary indicate **EITHER**:

   a. Gamma (closed window) dose rates **> 1000 mR/hr** are expected to continue for more than one hour.

   OR

   b. Analyses of field survey samples indicate **> 5000 mRem CDE Thyroid** for one hour of inhalation.

Table D-2: OCGS EAL Technical Basis

RECOGNITION CATEGORY
ABNORMAL RAD LEVELS / RADIOLOGICAL EFFLUENT

RG1 (cont)

Basis:

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.

Since dose assessment is based on actual meteorology and the EAL monitor readings are based on annual average meteorology, the results of dose assessments may indicate that the classification threshold has not been reached. For this reason, emergency implementing procedures call for the timely performance of dose assessments using actual meteorology and release information. If the results of dose assessments are available when the classification is made (e.g., initiated at a lower classification level), the dose assessment results override the monitor reading EALs.

Threshold #1 Basis:

The Emergency Director should not wait until 15 minutes has elapsed, but should declare the event as soon as it is determined that the release duration has or will likely exceed 15 minutes.

This EAL addresses radioactivity releases that result in doses at or beyond the site boundary that exceed the EPA Protective Action Guides (PAGs). Public protective actions will be necessary. Releases of this magnitude are associated with the failure of plant systems needed for the protection of the public and likely involve fuel damage. While these failures are addressed by other EALs, this EAL provides appropriate diversity and addresses events, which may not be able to be classified on the basis of plant status alone. It is important to note that, for the more severe accidents, the release may be unmonitored or there may be large uncertainties associated with the source term and/or meteorology.

Threshold #2 Basis:

The TEDE (1000 mRem) and the CDE Thyroid (5000 mRem) doses are set at the EPA PAG Limits.

The 'Site Boundary' is defined by an approximately 400-meter (1/4-mile) radius around the plant. This is the nearest distance from potential release points at which Protective Actions would be required for members of the public.

Table D-2: OCGS EAL Technical Basis

RECOGNITION CATEGORY
ABNORMAL RAD LEVELS / RADIOLOGICAL EFFLUENT

RG1 (cont)

Basis (cont):

Threshold #3 Basis:

The values are for surveys or iodine air samples taken at or beyond the site boundary and are the most accurate indicator of the condition. Field data are independent of release elevation and meteorology. Expected post accident source terms would be dominated by noble gases providing the dose rate value. Direct reading iodine monitors are not available. Sampling of radioiodine by adsorption on industry standard sample media followed by field analysis are used for determining the iodine (CDE) thyroid value.

The term "expected to continue for more than one hour" would not apply if:

- The release has been stopped and was less than one hour.

OR

- It is known it will be stopped with a release duration of less than one hour.

In all other cases it should be considered to last more than one hour.

Basis: (References)

1. EP-OC-110-200, Dose Assessment
2. NEI 99-01, Rev. 4 AG1
3 EP-EAL-0610, Criteria for Choosing Radiological Gaseous Effluent EAL Threshold Values, Oyster Creek Generating Station

Table D-2: OCGS EAL Technical Basis

RECOGNITION CATEGORY
ABNORMAL RAD LEVELS / RADIOLOGICAL EFFLUENT

RS1

Initiating Condition

Offsite Dose Resulting from an Actual or Imminent Release of Gaseous Radioactivity Exceeds 100 mRem TEDE or 500 mRem Thyroid CDE for the Actual or Projected Duration of the Release Using Actual Meteorology.

Operating Mode Applicability:

1, 2, 3, 4, D

EAL Threshold Values

Note: If dose assessment results are available at the time of declaration, the classification should be based on EAL Threshold #2 instead of EAL Threshold #1. Do **not** delay declaration awaiting dose assessment results.

1. VALID reading on one or more of the following radiation monitors that exceeds or is expected to exceed the reading shown (Table R1) for **≥ 15 minutes**:

Table R1 – Effluent Monitor Thresholds

	Site Area Emergency
Main Stack RAGEMS	1.69 E+01 μCi/cc HRM
Turbine Building RAGEMS	1.40 E-01 μCi/cc HRM

HRM = High Range Monitor LRM = Low Range Monitor

OR

2. Dose assessment using actual meteorology indicates doses at or beyond the Site Boundary of **EITHER**:

   a. **> 100 mRem TEDE**

   OR

   b. **> 500 mRem CDE Thyroid**

OR

3. Field survey results at or beyond Site Boundary indicate **EITHER**:

   a. Gamma (closed window) dose rates **> 100 mR/hr** are expected to continue for more than one hour.

   OR

   b. Analyses of field survey samples indicate **> 500 mRem CDE Thyroid** for one hour of inhalation.

Table D-2: OCGS EAL Technical Basis

RECOGNITION CATEGORY
ABNORMAL RAD LEVELS / RADIOLOGICAL EFFLUENT

RS1 (cont)

Basis

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.

Since dose assessment is based on actual meteorology and the EAL monitor readings are based on annual average meteorology, the results of dose assessments may indicate that the classification threshold has not been reached. For this reason, emergency implementing procedures call for the timely performance of dose assessments using actual meteorology and release information. If the results of dose assessments are available when the classification is made (e.g., initiated at a lower classification level), the dose assessment results override the monitor reading EALs

Threshold #1 Basis:

The Emergency Director should not wait until 15 minutes has elapsed, but should declare the event as soon as it is determined that the release duration has or will likely exceed 15 minutes.

This EAL addresses radioactivity releases that result in doses at or beyond the site boundary that exceed 10% of the EPA Protective Action Guides (PAGs). Releases of this magnitude are associated with the failure of plant systems needed for the protection of the public. While these failures are addressed by other EALs, this EAL provides appropriate diversity and addresses events, which may not be able to be classified on the basis of plant status alone. It is important to note that, for the more severe accidents, the release may be unmonitored or there may be large uncertainties associated with the source term and/or meteorology.

Threshold #2 Basis:

The TEDE (100 mRem) and the CDE Thyroid (500 mRem) doses are set at 10% of the EPA PAG Limits.

The 'Site Boundary' is defined by an approximately 400-meter (1/4-mile) radius around the plant. This is the nearest distance from potential release points at which Protective Actions would be required for members of the public.

Table D-2: OCGS EAL Technical Basis

RECOGNITION CATEGORY
ABNORMAL RAD LEVELS / RADIOLOGICAL EFFLUENT

RS1 (cont)

Basis (cont):

Threshold #3 Basis:

The values are for surveys or iodine air samples taken at or beyond the site boundary and are the most accurate indicator of the condition. Field data are independent of release elevation and meteorology. Expected post accident source terms would be dominated by noble gases providing the dose rate value. Direct reading iodine monitors are not available. Sampling of radioiodine by adsorption on industry standard sample media followed by field analysis are used for determining the iodine (CDE) thyroid value.

The term "expected to continue for more than one hour" would not apply if:

- The release has been stopped and was less than one hour.

OR

- It is known it will be stopped with a release duration of less than one hour.

In all other cases it should be considered to last more than one hour.

Basis: (References)

1. EP-OC-110-200, Dose Assessment
2. NEI 99-01, Rev. 4 AS1
3. EP-EAL-0610, Criteria for Choosing Radiological Gaseous Effluent EAL Threshold Values, Oyster Creek Generating Station

Table D-2: OCGS EAL Technical Basis

RECOGNITION CATEGORY
ABNORMAL RAD LEVELS / RADIOLOGICAL EFFLUENT

RA1

Initiating Condition

Any UNPLANNED Release of Gaseous or Liquid Radioactivity to the Environment that Exceeds 200 Times the Radiological Effluent Technical Specifications for 15 Minutes or Longer.

Operating Mode Applicability:

1, 2, 3, 4, D

EAL Threshold Values

1. VALID reading on any of the following effluent monitors **> 200 times alarm setpoint** established by a current radioactivity discharge permit for **≥ 15 minutes**:
   - Radwaste Overboard Discharge effluent monitor
   - Discharge Permit specified monitor

OR

2. VALID reading on one or more of the following radiation monitors that exceeds the Table R1 values for **≥ 15 minutes**:

Table R1 – Effluent Monitor Thresholds	
	Alert
Main Stack RAGEMS	1.93 E+00 μCi/cc HRM
Turbine Building RAGEMS	8.11E+04 cpm LRM

HRM = High Range Monitor LRM = Low Range Monitor

OR

3. Confirmed sample analyses for gaseous or liquid releases indicate concentrations or release rates **> 200 times ODCM Limit** with a release duration of **≥ 15 minutes.**

RA1 (cont)

Basis

UNPLANNED, as used in this context, includes any release for which a radioactivity discharge permit was not prepared, or a release that exceeds the conditions (e.g., minimum dilution flow, maximum discharge flow, alarm setpoints, etc.) on the applicable permit.

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.

The site design incorporates features intended to control the release of radioactive effluents to the environment. Further, there are administrative controls established to prevent unintentional releases, or control and monitor intentional releases. These controls are located in the Offsite Dose Calculation Manual (ODCM). The occurrence of extended, uncontrolled radioactive releases to the environment is indicative of a degradation in these features and/or controls.

The Emergency Director should not wait until 15 minutes has elapsed, but should declare the event as soon as it is determined that the release duration has or will likely exceed 15 minutes. Also, if an ongoing release is detected and the starting time for that release is unknown, the Emergency Director should, in the absence of data to the contrary, assume that the release has exceeded 15 minutes.

Threshold #1 Basis:

The threshold addresses radioactivity releases that cause effluent radiation monitor readings that exceed two hundred times the alarm setpoint established by the radioactivity discharge permit. This alarm setpoint may be associated with a planned batch release, or a continuous release path. In either case, the setpoint is established by the ODCM to warn of a release that is not in compliance with the RETS. Indexing the EAL threshold to the ODCM setpoints in this manner ensures that the EAL threshold will never be less than the setpoint established by a specific discharge permit.

Should 200 times the HI-HI Radiation alarm result in an offscale high meter reading, then the EAL threshold would be considered met when the meter goes offscale high for ≥ 15 minutes, provided there are no other direct or indirect means available to determine actual value.

An elevated monitor reading while the effluent flow path is isolated is not considered a VALID reading.

The effluent monitors listed are those normally used for planned discharges. If a discharge is performed using a different flowpath or effluent monitor other than those listed (e.g., a portable or temporary effluent monitor), then the declaration criteria will be based on the monitor specified in the Discharge Permit.

RECOGNITION CATEGORY
ABNORMAL RAD LEVELS / RADIOLOGICAL EFFLUENT

RA1 (cont)

Basis (cont):

Threshold #2 Basis:

This EAL addresses a potential or actual drop in the level of safety of the plant as indicated by a radiological release that exceeds, by a factor of 200, regulatory commitments for an extended period of time. The Alert gaseous effluent value was determined using formulas, isotopic dose factors and meteorology data as specified by the ODCM. The Table R1 values were determined in the units of a station-generated normal operating mixture for the no clad damage condition.

Threshold #3 Basis:

Confirmed sample analyses in excess of two hundred times the site Offsite Dose Calculation Manual (ODCM) limits that continue for 15 minutes or longer represent an uncontrolled situation and hence, a potential degradation in the level of safety. This event escalates from the Unusual Event by increasing the magnitude of the release by a factor of 100 over the Unusual Event level (i.e., 200 times ODCM). Prorating the 500 mR/yr basis of the 10CFR20 non-occupational MPC limits for both time (8766 hr/yr) and the 200 multiplier, the associated site boundary dose rate would be approximately 10 mR/hr. The required release duration was reduced to 15 minutes in recognition of the raised severity.

The 'site boundary' is defined by an approximately 400-meter (1/4-mile) radius around the plant. This is the nearest distance from potential release points at which Protective Actions would be required for members of the public.

Grab samples are used to determine release concentrations or rates to confirm meter readings or when the effluent monitors are not in service. The ODCM uses 10 CFR 20 Appendix B Table 2 data to generate maximum instantaneous release rate limits. These are indicated on Release Packages, which are approved.

Basis: (References)

1. CY-OC-170-301, Offsite Dose Calculation Manual for Oyster Creek
2. EP-EAL-0610, Criteria for Choosing Radiological Gaseous Effluent EAL Threshold Values, Oyster Creek Generating Station
3. NEI 99-01, Rev. 4 AA1

Table D-2: OCGS EAL Technical Basis

RECOGNITION CATEGORY
ABNORMAL RAD LEVELS / RADIOLOGICAL EFFLUENT

RU1

Initiating Condition

Any UNPLANNED Release of Gaseous or Liquid Radioactivity to the Environment that Exceeds Two Times the Radiological Effluent Technical Specifications for 60 Minutes or Longer.

Operating Mode Applicability:

1, 2, 3, 4, D

EAL Threshold Value

1. VALID reading on any of the following effluent monitors **> 2 times alarm setpoint** established by a current radioactivity discharge permit for **≥ 60 minutes**:
   - Radwaste Overboard Discharge effluent monitor
   - Discharge Permit specified monitor

OR

2. VALID reading on one or more of the following radiation monitors that exceeds the Table R1 values for **≥ 60 minutes**.

Table R1 – Effluent Monitor Thresholds	
	Unusual Event
Main Stack RAGEMS	7.92 E+03 cps LRM
Turbine Building RAGEMS	8.11E+02 cpm LRM

HRM = High Range Monitor LRM = Low Range Monitor

OR

3. Confirmed sample analyses for gaseous or liquid releases indicate concentrations or release rates in **> 2 times ODCM Limit** with a release duration of **≥ 60 minutes**.

Table D-2: OCGS EAL Technical Basis

RECOGNITION CATEGORY
ABNORMAL RAD LEVELS / RADIOLOGICAL EFFLUENT

RU1 (cont)

Basis

UNPLANNED, as used in this context, includes any release for which a radioactivity discharge permit was not prepared, or a release that exceeds the conditions (e.g., minimum dilution flow, maximum discharge flow, alarm setpoints, etc.) on the applicable permit.

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.

Since the assumptions used in calculating the radiation monitor threshold values and alarm setpoints with respect to ODCM release rate limits may not exactly match the conditions present when the classification is considered, results of available sample analyses override the monitor readings listed.

The Emergency Director should not wait until 60 minutes has elapsed, but should declare the event as soon as it is determined that the release duration has or will likely exceed 60 minutes. Also, if an ongoing release is detected and the starting time for that release is unknown, the Emergency Director should, in the absence of data to the contrary, assume that the release has exceeded 60 minutes.

Threshold #1 Basis:

The site design incorporates features intended to control the release of radioactive effluents to the environment. Further, there are administrative controls established to prevent unintentional releases, or control and monitor intentional releases. These controls are located in the Offsite Dose Calculation Manual (ODCM). The occurrence of extended, uncontrolled radioactive releases to the environment is indicative of a degradation in these features and/or controls.

An elevated monitor reading while the effluent flow path is isolated is not considered to be a VALID reading.

The effluent monitors listed are those normally used for planned discharges. If a discharge is performed using a different flowpath or effluent monitor other than those listed (e.g., a portable or temporary effluent monitor), then the declaration criteria will be based on the monitor specified in the Discharge Permit.

Threshold #2 Basis:

This EAL addresses a potential drop in the level of safety of the plant as indicated by a radiological release that exceeds, by a factor of 2, regulatory commitments for an extended period of time. The Unusual Event gaseous effluent value was determined using formulas, isotopic dose factors and meteorology data as specified by the ODCM.

The release rate was determined in the units of a station-generated normal operating mixture for the no clad damage condition.

Table D-2: OCGS EAL Technical Basis

RECOGNITION CATEGORY
ABNORMAL RAD LEVELS / RADIOLOGICAL EFFLUENT

RU1 (cont)

Basis (cont):

Threshold #3 Basis:

Confirmed sample analyses in excess of two times the site Offsite Dose Calculation Manual (ODCM) limits that continue for 60 minutes or longer represent an uncontrolled situation and hence, a potential degradation in the level of safety. The final integrated dose (which is very low in the Unusual Event emergency class) is not the primary concern here; it is the degradation in plant control implied by the fact that the release was not isolated within 60 minutes. Therefore, it is not intended that the release be averaged over 60 minutes. For example, a release of 4 times ODCM for 30 minutes does not exceed this EAL. Grab samples are used to determine release concentrations or rates to confirm meter readings or when the effluent monitors are not in service. The ODCM uses 10 CFR 20 Appendix B Table 2 data to generate maximum instantaneous release rate limits. These are indicated on Release Packages, which are approved.

Basis: (References)

1. CY-OC-170-301, Offsite Dose Calculation Manual for Oyster Creek
2. EP-EAL-0610, Criteria for Choosing Radiological Gaseous Effluent EAL Threshold Values, Oyster Creek Generating Station
3. NEI 99-01, Rev. 4 AU1

Table D-2: OCGS EAL Technical Basis

RECOGNITION CATEGORY
ABNORMAL RAD LEVELS / RADIOLOGICAL EFFLUENT

RA2

Initiating Condition

Damage to Irradiated Fuel or Loss of Water Level that Has or Will Result in the Uncovering of Irradiated Fuel Outside the Reactor Vessel.

Operating Mode Applicability:

1, 2, 3, 4, D

EAL Threshold Values

1. VALID reading of **> 1000 mR/hr** or upscale reading on one or more radiation monitors in Table R2.

Table R2 – Refuel Floor ARMs
C-5, Crit Mon
C-10, North Wall
C-9, North Wall
B-9, Open Floor

OR

2. Water level drop in the Reactor Cavity or Spent Fuel Pool that will result in irradiated fuel becoming uncovered.

Basis

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.

Threshold #1 Basis:

Uncovering irradiated fuel represents a substantial degradation of the level of safety of the plant and warrants an Alert classification. Time is available to take corrective actions. NUREG/CR-4982, "Severe Accident in Spent Fuel Pools in Support of Generic Safety Issue 82," (July, 1987) indicates that even if corrective actions are not taken, no prompt fatalities are predicted and the risk of injury is low. Visual observation of irradiated fuel uncovery represents a major ALARA concern in that radiation levels could exceed 10,000 R/hr on the refuel bridge when fuel uncovery begins. The value of a valid reading > 1000 mR/hr was conservatively chosen for classification purposes and consistent with the design range of the ARMs.

Dropping heavy loads onto the irradiated fuel can cause significant damage to the irradiated fuel and an Alert is also warranted under these conditions provided that the above radiation monitor threshold readings are reached.

Basis (cont)

Threshold #2 Basis:

When the RPV head is removed and the GEMAC Wide Range instrumentation is calibrated to indicate levels as high as the refuel floor elevation, remote indication of Refueling Cavity water level is available in the Control Room. Spent Fuel Pool water level decreases can be directly monitored only by visual observation. If the Spent Fuel Pool is in communication with the Refueling Cavity, however, remote indication of Spent Fuel Pool water level from the bottom of the Fuel Pool gate to the refueling floor is provided by the GEMAC Wide Range instrumentation. Even so, uncovery of irradiated fuel seated in the Spent Fuel Pool storage racks cannot be monitored remotely because the bottom of the Fuel Pool gate is above the elevation of the top of the storage racks. Any fuel that becomes uncovered while suspended from the refuel grapple may be indicated on the GEMAC Wide Range but, without report of the vertical position of the grapple, fuel uncovery cannot be determined. Visual observation, therefore, provides the only viable mechanism of determining if irradiated fuel in the fuel pool or Refueling Cavity will be uncovered.

This EAL applies to irradiated fuel requiring water coverage and is not intended to address spent fuel which is licensed for dry storage.

Basis Reference(s):

1. RAP G-7-a, SKM SRG TNK LVL LO-LO
2. NEI 99-01, Rev. 4 AA2
3. RAP-10F-1-m, Crit Mon C5 Hi
4. RAP-10F-3-m, North Wall C9 Hi Vent Trip
5. RAP-10F-2-m, North Wall C10 Hi
6. RAP-10F-4-m, North Wall B9 Hi Vent Trip

Table D-2: OCGS EAL Technical Basis

RECOGNITION CATEGORY
ABNORMAL RAD LEVELS / RADIOLOGICAL EFFLUENT

RU2

Initiating Condition

Unexpected Rise in Plant Radiation.

Operating Mode Applicability

1, 2, 3, 4, D

EAL Threshold Value

1. a. VALID indication of uncontrolled drop in water level in the Reactor Cavity, Spent Fuel Pool or Fuel Transfer Canal with all irradiated fuel assemblies remaining covered by water as indicated by:

   - Reactor Cavity water level **< 583 inches**. (GEMAC Wide Range, floodup calibration)

   OR

   - Report of visual observation of an uncontrolled drop in water level in the Reactor Cavity or Spent Fuel Pool.

   AND

   b. UNPLANNED VALID Area Radiation Monitor reading rise on one or more radiation monitor in Table R2.

Table R2 – Refuel Floor ARMs
C-5, Crit Mon
C-10, North Wall
C-9, North Wall
B-9, Open Floor

OR

2. UNPLANNED VALID Area Radiation Monitor readings rise by a factor of **1000** over NORMAL LEVELS or VALID upscale reading.

Basis

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.

UNPLANNED: a parameter change or an event that is not the result of an intended evolution and requires corrective or mitigative actions.

RU2 (cont)

Basis (cont)

NORMAL LEVELS: Normal levels can be considered as the highest reading in the past twenty-four hours excluding the current peak value.

Threshold #1 Basis:

This EAL addresses unplanned increases in radiation levels inside the plant. These radiation levels represent a degradation in the control of radioactive material and a potential degradation in the level of safety of the plant.

In light of Reactor Cavity Seal failure incidents at two different PWRs and loss of water in the Spent Fuel Pit/Fuel Transfer Canal at a BWR, explicit coverage of these types of events in the EALs is appropriate given their potential for increased doses to plant workers. Classification as an Unusual Event is warranted as a precursor to a more serious event.

Since no remote indication of Spent Fuel Pool water level exists, decreases in Spent Fuel Pool water level can normally be detected only through visual observation or decrease in skimmer surge tank level.

During refueling when the RPV head is removed, the GEMAC Wide Range instrumentation is calibrated to indicate water level to the elevation of the refuel floor. With the refueling cavity in communication with the Spent Fuel Pool through the fuel transfer canal, uncontrolled inventory loss can be remotely monitored.

The Refueling Cavity includes the fuel transfer canal. When the Refueling Cavity is flooded to normal level, water level is approximately one foot below the refuel floor. Reactor Cavity water level is maintained at least 23 ft above the top of the RPV flange (307 in. + 276 in.) or 583 inches when irradiated fuel or control rods are being handled within the RPV. During refueling when the RPV head is removed, plant procedures (e.g., 205.95.0 Reactor Flood-up / Drain-down, etc.) provide alternate level monitoring capabilities when the normal level instrumentation is unavailable for the desired level range or the head vent piping is removed. When calibrated for refueling operations, the GEMAC Wide Range instrument indicates from 100 in. above top of active fuel in the RPV to the maximum refuel floor water level. In addition, visual observation of level from the refueling floor can be used to monitor water level when the RPV head is removed.

Table D-2: OCGS EAL Technical Basis

RECOGNITION CATEGORY
ABNORMAL RAD LEVELS / RADIOLOGICAL EFFLUENT

RU2 (cont)

Basis (cont)

Threshold #2 Basis:
Valid elevated area radiation levels usually have long lead times relative to the potential for radiological release beyond the site boundary, thus impact to public health and safety is very low.

This EAL addresses unplanned rise in radiation levels inside the plant

Basis (References)

1. RP-AA-203 Exposure Control and Authorization
2. RAP-G-7-a, SKM SRG TNK LVL LO-LO
3. 205.94.0 RPV Floodup Using Core Spray
4. 205.95.0 Reactor Flood-up / Drain-down
5. FSAR Figure 7.6-3
6. NEI 99-01, Rev. 4 AU2

Table D-2: OCGS EAL Technical Basis

RECOGNITION CATEGORY
ABNORMAL RAD LEVELS / RADIOLOGICAL EFFLUENT

RA3

Initiating Condition

Release of Radioactive Material or Rise in Radiation Levels Within the Facility That Impedes Operation of Systems Required to Maintain Safe Operations or to Establish or Maintain Cold Shutdown

Operating Mode Applicability:

1, 2, 3, 4, D

EAL Threshold Values

1. VALID radiation monitor or survey reading **>15 mR/hr** in any of the following areas requiring continuous occupancy to maintain plant safety functions.
   - Main Control Room
   - Central Alarm Station (by survey)

OR

2. VALID radiation monitor or survey reading **> 2000 mR/hr** in areas requiring infrequent access (Table R3) which will impede necessary access and threaten safe operation of the plant.

Table R3 – Areas Requiring Infrequent Access
Reactor Bldg
Turbine Bldg
Control Room Complex
Main Transformer/Condensate Transfer Pad
Intake Structure
#1 EDG Vault
#2 EDG Vault
EDG Fuel Oil Storage Tank

RA3 (cont)

Basis

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.

Threshold #1 Basis:

This threshold addresses increased radiation levels that impede necessary access to operating stations requiring continuous occupancy to maintain safe plant operation or perform a safe plant shutdown. Areas requiring continuous occupancy include the Main Control Room and the Central Alarm Station (CAS). The security alarm station is included in this threshold because of its importance to permitting access to areas required to assure safe plant operations.

The value of 15 mR/hr is derived from the GDC 19 value of 5 rem in 30 days with adjustment for expected occupancy times. Although Section III.D.3 of NUREG-0737, "Clarification of TMI Action Plan Requirements", provides that the 15 mR/hr value can be averaged over the 30 days, the value is used here without averaging. A 30 day duration implies an event potentially more significant than an Alert.

It is the impaired ability to operate the plant that results in the actual or potential degradation of the level of safety of the plant. The cause or magnitude of the increase in radiation levels is not a concern of this threshold. The Emergency Director must consider the source or cause of the increased radiation levels and determine if any other EALs may be involved. For example, a dose rate of 15 mR/hr in the Main Control Room may be a problem in itself. However, the increase may also be indicative of high dose rates in the containment due to a LOCA. In this latter case, a Site Area Emergency or a General Emergency may be indicated by other EAL categories.

This threshold is not intended to apply to anticipated temporary radiation increases due to planned events (e.g., radwaste container movement, depleted resin transfers, etc.).

Threshold #2 Basis:

This threshold addresses increased radiation levels in areas requiring infrequent access in order to maintain safe plant operation or perform a safe plant shutdown. Typically areas requiring infrequent access to maintain plant safety functions include plant vital areas. Area radiation levels at or above 2000 mR/hr are indicative of radiation fields, which may limit personnel access to equipment, the operation of which may be needed to assure adequate core cooling or shutdown the reactor.

The Control Room Complex consists of the Main Office Building, Upper and Lower Cable Spread Rooms.

Table D-2: OCGS EAL Technical Basis

RECOGNITION CATEGORY
ABNORMAL RAD LEVELS / RADIOLOGICAL EFFLUENT

RA3 (cont)

Basis: (cont)

The dose rate threshold selected is based on site administrative limits.

It is the impaired ability to operate the plant that results in the actual or potential degradation of the level of safety of the plant. The cause or magnitude of the increase in radiation levels is not a concern of this threshold. The Emergency Director must consider the source or cause of the increased radiation levels and determine if any other EAL may be involved. For example, a dose rate of 2 R/hr may be a problem in itself. However, the increase may also be indicative of high dose rates in the containment due to a LOCA. In this latter case, a Site Area Emergency or a General Emergency may be indicated by other EAL categories.

This threshold is not intended to apply to anticipated temporary radiation rise due to planned events (e.g., radwaste container movement, depleted resin transfers, etc.) or pre-existing radiation areas for which radiological controls already exist. The concern of this threshold is the unanticipated rise in radiation levels that results in unplanned restrictions to areas requiring infrequent access in order to maintain safe plant operation or perform a safe plant shutdown.

Basis (References):

1. ABN-29, Plant Fires
2. EMG-3200.11, Secondary Containment Control Safe Shutdown Area
3. NEI 99-01, Rev. 4 AA3

Table D-2: OCGS EAL Technical Basis

RECOGNITION CATEGORY
ABNORMAL RAD LEVELS / RADIOLOGICAL EFFLUENT

RU3

Initiating Condition

Fuel Clad Degradation.

Operating Mode Applicability:

1, 2

EAL Threshold Values

1. Offgas system isolation required due to VALID Offgas radiation monitor signal.

OR

2 Coolant activity **> 4.0 µCi/gm** Dose Equivalent I-131.

Basis:

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.

Threshold #1 Basis:

During unit operation, the steam jet air ejectors (SJAEs) remove all non-condensable gases from the main condenser including air in-leakage and disassociated products originating in the reactor and exhausts them to the offgas holdup volume. A rise in offgas activity could therefore indicate damage to the fuel cladding, a potential degradation in the level of safety of the plant and a potential precursor of more serious problems. The gas from the main condenser normally includes relatively low levels of radioactivity. If radioactivity of the gas reaches the Off Gas Rad Monitor Hi-Hi alarm setpoint and the Offgas isolation timer is not reset, the Offgas system isolates (i.e., closure of V-7-31, V-7-29 and OG-AOV-001A/B) after a fifteen-minute time delay. The fifteen-minute time delay is allotted for operator action to reduce the offgas radiation levels and exclude transient conditions.

The modifier "valid" is appropriate because there are several conditions that may cause the monitor to alarm that are not related to fuel clad degradation and therefore should not result in the declaration of an Unusual Event.

Threshold #2 Basis:

Elevated reactor coolant activity represents a potential degradation in the level of safety of the plant and a potential precursor of more serious problems. This EAL addresses reactor coolant samples exceeding coolant Technical Specifications for iodine spiking. The specific iodine activity ensures the source term assumed in the safety analysis for the Main Steam Line Break (MSLB) accident is not exceeded, so any release of radioactivity to the environment during an MSLB is less than a small fraction of the 10 CFR 100 limits.

However, fuel clad damage should be assumed to be the cause of elevated Reactor Coolant activity unless another cause is known, e.g., Reactor Coolant System chemical decontamination evolution (during shutdown) is ongoing with resulting high activity levels.

Table D-2: OCGS EAL Technical Basis

RECOGNITION CATEGORY
ABNORMAL RAD LEVELS / RADIOLOGICAL EFFLUENT

RU3 (cont)

Basis (References)

1. ABN-26, Increase in Off Gas Activity
2. RAP10F-1-c, Offgas HI-HI
3. Technical Specifications 3.6.A
4. NEI 99-01, Rev. 4 SU4

Table D-2: OCGS EAL Technical Basis

RECOGNITION CATEGORY
HAZARDS AND OTHER CONITIONS AFFECTING PLANT SAFETY

FUEL CLAD 1.a

Initiating Condition

Reactor Pressure Vessel (RPV) Water Level

Operating Mode Applicability:

1, 2

EAL Threshold Value

LOSS:

1. RPV Level **< -20 inches TAF**

POTENTIAL LOSS:

2. RPV Level **< 0 inches TAF**

Basis:

LOSS - [Threshold Value #1]

The specified RPV water level is the Minimum Steam Cooling RPV Water Level (MSCRWL) and is used in EOPs to indicate challenge to core cooling. The MSCRWL is the lowest RPV water level at which the submerged portion of the reactor core will generate sufficient steam to prevent any clad in the uncovered portion of the core from heating above 1500°F; the threshold temperature of fuel clad perforation. This water level is utilized to preclude fuel damage when RPV water level is below the top of active fuel (TAF).

The MSCRWL appears in the RPV CONTROL - WITH ATWS procedure when RPV water level is intentionally lowered to reduce reactor power. When RPV water level is deliberately lowered, power instabilities may produce noticeable oscillations in RPV water level and make it difficult to maintain water level. This level is also used in the RPV CONTROL – NO ATWS procedure when all attempts to restore and maintain RPV water level above TAF have failed.

RPV water level instrumentation is referenced to the Top of Active Fuel. 0 inches TAF equates to water level at TAF. –20 inchesTAF therefore means that RPV water level is 20 inches below TAF.

POTENTIAL LOSS – [Threshold Value #2]

Core submergence is the mechanism of core cooling whereby each fuel element is completely covered with water. Indicated RPV water level at or above the Top of Active Fuel (0 inches TAF) provides direct confirmation that adequate core cooling exists. Assurance of continued adequate core cooling through core submergence is achieved when RPV water level can be maintained at or above TAF.

Table D-2: OCGS EAL Technical Basis

RECOGNITION CATEGORY
HAZARDS AND OTHER CONITIONS AFFECTING PLANT SAFETY

FUEL CLAD 1.a (cont)

Basis Reference(s):

1. EMG-3200.01A, RPV Control – No ATWS
2. EMG-3200.01B, RPV Control – With ATWS
3. EMG-3200.08A, RPV Flooding – No ATWS
4. EMG-3200.08B, RPV Flooding – With ATWS
5. EMG-3200.02, Primary Containment Control
6. NEI 99-01, Rev. 4 Table 5-F-2

Table D-2: OCGS EAL Technical Basis

RECOGNITION CATEGORY
HAZARDS AND OTHER CONITIONS AFFECTING PLANT SAFETY

FUEL CLAD 1.b

Initiating Condition

Drywell (DW) High Range Rad Monitor

Operating Mode Applicability:

1, 2

EAL Threshold value

LOSS:

1. Containment Hi Range Radiation Monitoring System (CHRRMS) > Fuel Cladding Loss Threshold, Table F1

POTENTIAL LOSS:

NONE

Basis:

The drywell radiation monitor readings specified in Table F1 provide values that indicate the release of reactor coolant into the drywell with elevated activity indicative of fuel damage. The values are a function of time after shutdown (TAS) and were derived using CDAM with 2% clad damage, no drywell sprays in operation and a LOCA depressurized system. The reading is calculated assuming the instantaneous release and dispersal of the above reactor coolant noble gas and iodine inventory into the drywell atmosphere. Reactor coolant concentrations of this magnitude are several times larger than the maximum concentrations allowed within Technical Specifications (including iodine spiking) and are therefore indicative of fuel damage (approximately 2% - 5% cladding failure).

During at power (including ATWS) conditions the value listed for the "< 2 hours after shutdown" is used as an indication of fuel damage.

Table F1 – Drywell Radiation Thresholds

Time After Shutdown (hrs)	Fuel Cladding Loss (R/hr)
≤ 2	530
>2 – 4	465
>4 – 8	400
>8 – 16	335
>16 - 23	300
> 23	295

Basis Reference(s):

1. Core Damage Assessment Methodology (CDAM)
2. NEI 99-01, Rev. 4 Table 5-F-2

Table D-2: OCGS EAL Technical Basis

RECOGNITION CATEGORY
HAZARDS AND OTHER CONITIONS AFFECTING PLANT SAFETY

RCS 2.a

Initiating Condition

Reactor Pressure Vessel (RPV) Water Level

Operating Mode Applicability

1, 2

EAL Threshold Value

LOSS:

1. RPV level **< 0 inches TAF**

POTENTIAL LOSS:

NONE

Basis:

RPV water level instrument reading of < 0 inches TAF indicates RPV water level is below the Top of Active Fuel (TAF). TAF is significantly lower than the normal operating RPV water level control band. To reach this level, RPV inventory loss would have previously required isolation of the RCS and Primary Containment 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 any unplanned loss of RPV inventory is therefore assumed to be a LOCA. By definition, a LOCA event is a loss of the RCS barrier.

Basis Reference(s):

1. 2000-GLN-3200.01, Plant Specific Technical Guideline
2. 2000-BAS-3200.02, EOP Users Guide
3. NEI 99-01, Rev. 4 Table 5-F-2

Table D-2: OCGS EAL Technical Basis

RECOGNITION CATEGORY
HAZARDS AND OTHER CONITIONS AFFECTING PLANT SAFETY

RCS 2.b

Initiating Condition

Drywell (DW) High Range Rad Monitor

Operating Mode Applicability

1, 2

EAL Threshold Value

LOSS:

1. Containment Hi Range Radiation Monitoring System (CHRRMS) **> 100 R/hr**.

AND

2. Indications of RCS leakage into the Drywell.

POTENTIAL LOSS:

NONE

Basis:

The radiation monitor reading is a value that indicates a significant release of reactor coolant to the containment. A reading was calculated assuming the instantaneous release and dispersal of the reactor coolant iodine inventory associated with normal operating concentrations (i.e., within Technical Specifications) into the drywell atmosphere. This provides a reading that is observable and indicative of significant release of reactor coolant to the drywell. The reading is less than that specified for Fuel Cladding barrier Loss because no damage to the fuel cladding is assumed. Only leakage from the RCS is assumed for this barrier loss threshold. The value is high enough to preclude erroneous classification of barrier loss due to normal plant operations.

Indication of a RCS leak into the drywell is added to qualify the radiation monitor indication to avoid declaring the loss of RCS barrier for situations where the radiation rise is not due to primary a system leak. For situations that involve failure of the Fuel Clad barrier alone, radiation monitor readings would rise due to shine and potentially giving a false indication of a loss of the RCS barrier. Therefore this EAL contains a qualifier to preclude over classification of the event if only fuel clad barrier failed.

Basis Reference(s):

1. EP-EAL-0611, Criteria for Choosing Containment Radiation Monitor Reading Indicative of Loss of RCS Barrier
2. NEI 99-01, Rev. 4 Table 5-F-2

Table D-2: OCGS EAL Technical Basis

RECOGNITION CATEGORY
HAZARDS AND OTHER CONITIONS AFFECTING PLANT SAFETY

CONTAINMENT 3.b

Initiating Condition

Drywell (DW) High Range Rad Monitor

Operating Mode Applicability

1, 2

EAL Threshold Value

LOSS:

NONE

POTENTIAL LOSS:

1. Containment Hi Range Radiation Monitor System (CHRRMS) > Primary Containment Potential Loss Threshold, Table F2.

Basis

The drywell radiation monitor reading is a value that indicates significant fuel damage well in excess of that required for loss of the Fuel Cladding barrier. NUREG-1228 “Source Estimations During Incident Response to Severe Nuclear Power Plant Accidents” states that such readings do not exist when the amount of cladding damage is less than 20%.

The values are a function of time after shutdown (TAS) and were derived using CDAM assuming 20% clad damage, no drywell sprays in operation and a LOCA depressurized system. A major release of radioactivity requiring offsite protective actions from core damage is not possible unless a significant failure into the reactor coolant has occurred. Regardless of whether the Primary Containment barrier itself is challenged, this amount of activity in containment could have severe consequences if released. It is, therefore, prudent to treat this as a potential loss of the Primary Containment barrier.

Table F2 – Drywell Radiation Thresholds	
Time After Shutdown (hrs)	PC Potential Loss (R/hr)
≤ 2	1210
>2 – 4	1060
>4 – 8	910
>8 – 16	765
>16 - 23	685
> 23	680

Basis Reference(s):

1. Core Damage Assessment Methodology
2. NEI 99-01, Rev. 4 Table 5-F-2

Table D-2: OCGS EAL Technical Basis

RECOGNITION CATEGORY
HAZARDS AND OTHER CONITIONS AFFECTING PLANT SAFETY

CONTAINMENT 3.d

Initiating Condition

Breached / Bypassed

Operating Mode Applicability

1, 2

EAL Threshold Value

LOSS:

1. a. Failure of all isolation valves in any one line to close.

   AND

   b. Downstream pathway to the environment exists.

OR

2. Intentional venting/purging of Primary Containment per EOPs or SAMGs due to accident conditions.

OR

3. UNISOLABLE primary system leakage outside of drywell as indicated by Secondary Containment area temperatures or radiation levels **> EMG-3200.11 Max Safe Operating**.

POTENTIAL LOSS:

NONE

Basis:

UNISOLABLE – is a breach or leak that cannot be isolated from the Control Room. When evaluating this EAL for unisolable primary system leakage, it is appropriate to attempt isolation from the Control Room prior to classification.

LOSS – [Threshold #1]

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 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 primary containment integrity.

Failure of containment isolation valves to isolate with a downstream pathway to the environment is only a concern during an accident. If this condition exists during normal power operations, a Technical Specification Action Statement will address it. However, during accident conditions, this will represent a breach of Primary Containment.

Table D-2: OCGS EAL Technical Basis

RECOGNITION CATEGORY
HAZARDS AND OTHER CONITIONS AFFECTING PLANT SAFETY

CONTAINMENT 3.d (cont)

Basis: (cont)

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, steamline breaks, RWCU system breaks, Isolation Condenser Tube ruptures and containment atmosphere vent paths. Minor release paths such as instrument and sample lines are not considered under this threshold.

Examples of 'downstream pathway to the Environment" could be through Turbine/Condenser, or direct release to Turbine or Reactor Building.

The breach is NOT isolable from the Control Room or an attempt for isolation from the Control Room has been made and was unsuccessful. An attempt for isolation from the Control Room should be made prior to the accident classification. If Operator actions from the Control Room are successful, then this IC is not applicable. Credit is NOT given for Operator actions taken in-plant (outside the Control Room) to isolate the leak.

This EAL is intended to cover containment isolation failures allowing a direct flow path to the environment such as failure of both MSIVs to close with open valves downstream to the turbine or to the condenser, even if these systems are not breached.

LOSS – [Threshold #2]

Intentional venting of the Primary Containment per EOP or SAMGs due to accident conditions procedures to the secondary containment and/or the environment is considered to be a breach of the primary containment for the purposes of accident classification.

LOSS – [Threshold #3]

The presence of elevated general area temperatures and radiation levels in the secondary containment may be indicative of an unisolable primary system leakage outside the primary containment. Temperatures or radiation levels beyond their Maximum Safe Operating limits in one or more areas are indicative of problems in the secondary containment that are spreading and pose a threat to achieving a safe plant shutdown. These conditions represent a loss of the Primary Containment barrier and a potential loss of the RCS barrier. High-energy line breaks are primarily addressed by RCS Loss 2.c and PC Loss 3.d. This EAL threshold addresses other problematic discharges outside primary containment that may not originate from a high-energy line break.

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 Reactor Building 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 Reactor Building, an unexpected rise in Feedwater flowrate, or unexpected Main Turbine Control Valve closure) may indicate that a primary system is discharging into the Reactor Building.

RECOGNITION CATEGORY
HAZARDS AND OTHER CONITIONS AFFECTING PLANT SAFETY

CONTAINMENT 3.d (cont)

Basis References:

1. EMG-3200.11, Secondary Containment Control
2. 2000-GLN-3200.01, Plant Specific Technical Guideline
3. EMG-3200.02, Primary Containment Control
4. Support Procedures -32, -34, -41, -44
5. 2000-GLN-3200.03, OCNGS Plant Specific Technical Guidelines for Severe Accident Guidelines
6. NEI 99-01, Rev. 4 Table 5-F-2

Table D-2: OCGS EAL Technical Basis

RECOGNITION CATEGORY
HAZARDS AND OTHER CONITIONS AFFECTING PLANT SAFETY

MG1

Initiating Condition

Prolonged Loss of All Offsite Power and Prolonged Loss of All Onsite AC Power to Essential Busses.

Operating Mode Applicability:

1, 2

EAL Threshold Values:

1. Loss of power to Startup Transformers SA and SB.

AND

2. Failure of EDG-1 and EDG-2 Emergency Diesel Generators to supply power to 4160V Buses 1C and 1D.

AND

3. a. Restoration of at least one 4160V Bus (1C or 1D) within **1 hour** is **not** likely.

   OR

   b. RPV level **cannot** be determined to be **> 0 inches TAF**.

Basis:

Loss of all AC power to the 4160V emergency buses compromises the availability of all plant safety systems. Prolonged loss of all AC power may lead to loss of Fuel Cladding, RCS and Primary Containment barriers. The one-hour interval to restore AC power to either emergency 4160V bus is based on the blackout coping analysis performed in conformance with 10 CFR 50.63 and Regulatory Guide 1.155, "Station Blackout."

The likelihood of restoring at least one emergency 4160V bus should be based on a realistic appraisal of the situation since a delay in an upgrade decision based on only a chance of mitigating the event could result in a loss of valuable time in preparing and implementing public protective actions. Emergency buses 1C and 1D can be powered from non-emergency buses 1A and 1B as well as from Emergency Diesel Generators EDG-1 and EDG-2. Buses 1A and 1B can be powered from Startup Transformers SA and SB. In addition, the SBO Transformer can also power bus 1B.

An additional source of offsite power is available when the main generator is off-line by backfeeding through the main power transformer and Auxiliary transformer. The backfeed operation must be manually performed and involves removal of flexible link connections between the main generator and the main power and auxiliary transformers. (Due to the time required to affect the backfeed, this source is likely only to be available when previously configured.)

Table D-2: OCGS EAL Technical Basis

RECOGNITION CATEGORY
HAZARDS AND OTHER CONITIONS AFFECTING PLANT SAFETY

MG1 (cont)

Basis (cont)

Under these conditions, fission product barrier monitoring capability may be degraded. Although it may be difficult to predict when power can be restored, it is necessary to give the Emergency Director a reasonable idea of how quickly to declare a General Emergency based on two major considerations:

1. Are there any present indications that core cooling is already degraded to the point that loss or potential loss of Fission Product Barriers is imminent?
2. If there are no present indications of such core cooling degradation, how likely is it that power can be restored in time to assure that a loss of two barriers with a potential loss of the third barrier can be prevented?

Thus, indication of continuing core cooling degradation must be based on Fission Product Barrier monitoring with particular emphasis on Emergency Director judgment as it relates to imminent loss or potential loss of fission product barriers and degraded ability to monitor fission product barriers.

When RPV water level is above the Top of Active Fuel (0 inches TAF), the core is completely submerged. Core submergence is the most desirable means of core cooling. When RPV water 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 water level drops below TAF, the level is indicative of a challenge to core cooling and the Fuel Cladding barrier.

Basis References:

1. OCNGS Drawing BR 3000
2. ABN-36, Loss of Off-Site Power
3. ABN-37, Station Blackout
4. ABN-60, Grid Emergency
5. Regulatory Guide 1.155, Station Blackout
6. TDR-1099, "Station Blackout Evaluation Report"
7. 2000-BAS-3200.02, EOP User's Guide
8. 2000-GLN-3200.01, Plant Specific Technical Guideline
9. NEI 99-01, Rev. 4 SG1

Table D-2: OCGS EAL Technical Basis

RECOGNITION CATEGORY
HAZARDS AND OTHER CONITIONS AFFECTING PLANT SAFETY

MG3

Initiating Condition

Failure of the Reactor Protection System to Complete an Automatic Scram and Manual Scram was NOT Successful and There is Indication of an Extreme Challenge to the Ability to Cool the Core.

Operating Mode Applicability:

1

EAL Threshold Values:

1. Automatic scram, Manual scram and ARI were **not** successful from Reactor Console as indicated by **EITHER**:

   a. Reactor Power remains $\geq$ **2%.**

   OR

   b. Torus temperature **> 110°F AND** boron injection required for reactivity control.

AND

2. a. RPV level **cannot** be restored and maintained **> -20 inches TAF**.

   OR

   b. Heat Capacity Temperature Limit (EMG-3200.02 Fig. F) exceeded.

Basis:

Automatic scram, manual scram and ARI are not considered successful if action away from the reactor control console was required to scram the reactor (i.e., actions from the console include mode switch to shutdown, using the manual scram pushbuttons, or manual ARI initiation).

Taking the mode switch to shutdown is a manual scram action.

This EAL encompasses events in which the automatic and manual scrams were not successful and the reactor is producing more heat than the maximum decay heat load for which the safety systems are designed. The reactor power threshold (2%) is approximately equal to the the APRM downscale setpoint and the maximum decay heat generation rate that should exist shortly after shutdown. Below the APRM downscale 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 2% power.

The torus water temperature criterion (110°F) is the Boron Injection Initiation Temperature (BIIT). The BIIT ensures that the Standby Liquid Control (SLC) system will inject the Hot Shutdown Boron Weight (HSBW) into the RPV before the total amount of energy rejected to the torus heats the suppression pool to the Heat Capacity Temperature Limit (HCTL). If suppression pool temperature exceeds the BIIT, reactor power is heating the suppression pool and the suppression pool cooling may be inadequate or incapable of performing its design function.

Table D-2: OCGS EAL Technical Basis

RECOGNITION CATEGORY
HAZARDS AND OTHER CONITIONS AFFECTING PLANT SAFETY

MG3 (cont)

Basis (cont)

The second condition of this EAL indicates either:

- An extreme challenge to the ability to cool the core as indicated when RPV water level cannot be maintained above -20 inches TAF. The specified water level is the Minimum Steam Cooling RPV Water Level (MSCRWL). The MSCRWL is the lowest RPV water 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 water level is below the top of active fuel. RPV water level below the MSCRWL for an extended period of time without satisfactory core spray cooling could be precursors of a core melt sequence.
- An extreme challenge to the primary containment as indicated when heat cannot be removed from the primary containment resulting in elevated torus water temperature. The Heat Capacity Temperature Limit is the highest torus water temperature from which a blowdown will not raise torus pressure above the Primary Containment Pressure Limit (PCPL) before the rate of energy transfer from the RPV to the primary containment is within the capacity of the primary containment vent. (When the PCPL is challenged, primary containment venting may be required even if offsite radioactivity release rate limits will be exceeded.) The HCTL is a function of RPV pressure and torus water temperature and is a measure of the maximum heat load, which the primary containment can withstand. Plant parameters in excess of the HCTL could be a precursor of primary containment failure. The Heat Capacity Limit is given in Figure F of EMG-3200.02, Primary Containment Control.

Basis Reference(s):

1. EMG-3200.01A, RPV Control – no ATWS
2. EMG-3200.01B, RPV Control – with ATWS
3. EMG-3200.02, Primary Containment Control
4. 2000-BAS-3200.02, EOP User's Guide
5. 2000-GLN-3200.01, Plant Specific Technical Guideline
6. NEI 99-01, Rev. 4 SG2

Table D-2: OCGS EAL Technical Basis

RECOGNITION CATEGORY
HAZARDS AND OTHER CONITIONS AFFECTING PLANT SAFETY

MS3

Initiating Condition

Failure of the Reactor Protection System Instrumentation to Complete or Initiate an Automatic Reactor Scram Once a Reactor Protection System Setpoint Has Been Exceeded and Manual Scram was NOT Successful.

Operating Mode Applicability:

1

EAL Threshold Values:

1. Automatic scram, Manual scram and ARI were **not** successful from Reactor Console as indicated by **EITHER**:

   a. Reactor Power remains **≥ 2%.**

   OR

   b. Torus temperature **> 110°F AND** boron injection required for reactivity control.

Basis:

Automatic scram, manual scram and ARI are not considered successful if action away from the reactor control console was required to scram the reactor (i.e., actions from the console include mode switch to shutdown, using the manual scram pushbuttons, or manual ARI initiation).

Taking the mode switch to shutdown is a manual scram action.

This EAL encompasses events in which the automatic and manual scrams were not successful and the reactor is producing more heat than the maximum decay heat load for which the safety systems are designed The reactor power threshold (2%) is approximately equal to the APRM downscale setpoint and the maximum decay heat generation rate that should exist shortly after shutdown. Below the APRM downscale 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 ≥ 2% power. Classification at the Site Area Emergency level is appropriate because conditions exist that can lead to imminent loss or potential loss of both the Fuel Cladding and RCS barriers.

The torus water temperature criterion (110°F) is the Boron Injection Initiation Temperature (BIIT). The BIIT ensures that the Standby Liquid Control (SLC) system will inject the Hot Shutdown Boron Weight (HSBW) into the RPV before the total amount of energy rejected to the torus heats the suppression pool to the Heat Capacity Temperature Limit (HCTL). If torus temperature exceeds the BIIT, reactor power is heating the torus and the suppression pool cooling may be inadequate or incapable of performing its design function.

Table D-2: OCGS EAL Technical Basis

RECOGNITION CATEGORY
HAZARDS AND OTHER CONITIONS AFFECTING PLANT SAFETY

MS3 (cont)

Basis Reference(s):

1. EMG-3200.01A, RPV Control – no ATWS
2. EMG-3200.01B, RPV Control – with ATWS
3. 2000-BAS-3200.02, EOP User's Guide
4. 2000-GLN-3200.01, Plant Specific Technical Guideline
5. NEI 99-01, Rev. 4 SS2

RECOGNITION CATEGORY
HAZARDS AND OTHER CONITIONS AFFECTING PLANT SAFETY

MA3

Initiating Condition

Failure of the Reactor Protection System Instrumentation to Complete or Initiate an Automatic Reactor Scram Once a Reactor Protection System Setpoint Has Been Exceeded.

Operating Mode Applicability:

1

EAL Threshold Values:

1. A Reactor Protection System setpoint was exceeded.

AND

2. Automatic SCRAM did **not** reduce Reactor Power to subcritical with power below the Heating Range.

Basis:

This condition indicates a failure of the automatic Reactor Protection System (RPS) to successfully scram the reactor. This condition is more than a potential degradation of a safety system in that a front line automatic protection system did not function in response to a plant transient and thus the plant safety has been compromised, and design limits of the fuel may have been exceeded. Site-specific indication of reactor shutdown is included as the criteria of whether the scram was successful when required. An Alert is indicated because conditions exist that lead to potential loss of fuel clad or RCS. Reactor Protection System setpoint being exceeded, rather than limiting safety system setpoint being exceeded, is specified here because failure of the automatic protection system is the issue.

Taking the mode switch to shutdown is a manual scram action.

The first condition of this EAL identifies the need to cease critical reactor operations by actuation of the automatic Reactor Protection System (RPS) scram function.

The second condition of this EAL indicates a failure of the automatic RPS scram function to rapidly insert a sufficient number of control rods to achieve reactor shutdown. The CRD system backup scram valves and the Alternate Rod Insertion (ARI) system provide automatic, alternate methods of completing the scram function. These backups, however, insert control rods at a much slower rate than the automatic RPS scram function. For the purpose of emergency classification at the Alert level, reactor shutdown achieved by automatic backup scram valve operation and ARI initiation does not constitute a successful RPS automatic scram. The significance of the second condition, therefore, is that a potential degradation of a safety system exists because a front line automatic protection system did not function in response to a plant transient. Thus, plant safety has been compromised.

Table D-2: OCGS EAL Technical Basis

RECOGNITION CATEGORY
HAZARDS AND OTHER CONITIONS AFFECTING PLANT SAFETY

MA3 (cont)

Basis (cont)

Following any automatic RPS scram signal EMG-3200.01B RPV Control – with ATWS, prescribes 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 Alert.

This threshold indicates failure of all manual scram capability. While failure of all manual SCRAM capability does not challenge fuel design limits, it is indicative of a condition in which rapid reactor shutdown cannot be established prior to the fuel being challenged should an RPS setpoint subsequently be exceeded.

A manual scram is any set of actions by the reactor operator(s) at the reactor control console, which causes control rods to be rapidly inserted into the core, and brings the reactor subcritical, including manual scram buttons, Mode Switch and actuation of ARI.

If by procedure, operator actions include the initiation of an immediate manual scram following receipt of an automatic scram signal and there are no clear indications that the automatic scram failed (such as a time delay following indications that a scram setpoint was exceeded, or first-out annunciators), 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 consideration should be given to evaluating the fuel for potential damage, and the reporting requirements of 50.72 should be considered for the transient event.

Basis Reference(s):

1. EMG-3200.01A, RPV Control – no ATWS
2. EMG-3200.01B, RPV Control – with ATWS
3. 2000-BAS-3200.02, EOP User's Guide
4. 2000-GLN-3200.01, Plant Specific Technical Guideline
5. NEI 99-01, Rev. 4 SA2

Table D-2: OCGS EAL Technical Basis

RECOGNITION CATEGORY
HAZARDS AND OTHER CONITIONS AFFECTING PLANT SAFETY

MA5

Initiating Condition

Inability to Maintain Plant in Cold Shutdown with Irradiated Fuel in the RPV.

Operating Mode Applicability:

3, 4

EAL Threshold Values:

1. UNPLANNED loss of decay heat removal capability results in RCS temperature **> 212°F** for > Table M1 duration.

Table M1 – RCS Reheat Duration Thresholds		
RCS	**Secondary Containment Closure**	**Duration**
Intact	N/A	60 minutes*
Not Intact	Established	20 minutes*
	Not Established	0 minutes
*If an RCS heat removal system is in operation within this time frame and RCS temperature is being reduced, then EAL Threshold #1 is **not** applicable.		

OR

2. UNPLANNED RPV pressure rise **> 10 psig** as a result of temperature rise due to loss of decay heat removal.

Basis:

CONTAINMENT CLOSURE - Containment Closure is considered to be Containment as required by Technical Specifications.

UNPLANNED - a parameter change or an event that is not the result of an intended evolution and requires corrective or mitigative actions.

RCS intact - When the RCS pressure boundary is in its normal condition for the Cold Shutdown mode of operation (e.g., no freeze seals or main steam line nozzle plugs, etc.)

Table D-2: OCGS EAL Technical Basis

RECOGNITION CATEGORY
HAZARDS AND OTHER CONITIONS AFFECTING PLANT SAFETY

MA5 (cont)

BASIS (cont)

Threshold #1 Basis:

This threshold is based on concerns raised by Generic Letter 88-17, "Loss of Decay Heat Removal." A number of phenomena such as decay heat removal system design and RPV water level instrumentation problems can lead to conditions in which decay heat removal is lost and core uncovery can occur. NRC analyses show that sequences that can cause core uncovery in 15 to 20 minutes and severe core damage within an hour after decay heat removal is lost.

Several instruments are capable of providing indication of RCS temperature with respect to the Technical Specification cold shutdown temperature limit (212°F), such as:

- Reactor Vessel Metal temperatures
- Recirculation loop suction temperatures (representative of the most restrictive beltline region metal temperatures.)

The first condition in Table M1 addresses complete loss of functions required for core cooling for greater than sixty minutes during Refueling and Cold Shutdown modes when RCS integrity is established. RCS integrity should be considered to be in place when the RCS pressure boundary is in its normal condition for the Cold Shutdown mode of operation (e.g., no freeze seals or main steam line nozzle plugs, etc.). With containment closure established, a low-pressure barrier to fission product release exists. In this condition, containment status is of less importance than the status of RCS integrity because the RCS is intact and providing a high-pressure barrier to fission product release. The sixty-minute interval should allow sufficient time to restore cooling without a substantial degradation in plant safety. The asterisk highlights the note at the bottom of the table. The note indicates that the first threshold is not applicable if actions are successful in restoring an RCS heat removal system to operation and RPV temperature is being reduced within the sixty-minute interval.

The second condition in Table M1 addresses the complete loss of functions required for core cooling for greater than twenty minutes during Refueling and Cold Shutdown modes when secondary containment closure is established but RCS integrity is not established or RPV inventory is reduced. RCS integrity should be assumed to be in place when the RCS pressure boundary is in its normal condition for the Cold Shutdown mode of operation (e.g., no freeze seals or main steam line nozzle plugs, etc.). The allowed twenty-minute interval is included to allow operator action to restore the heat removal function, if possible.

Table D-2: OCGS EAL Technical Basis

RECOGNITION CATEGORY
HAZARDS AND OTHER CONITIONS AFFECTING PLANT SAFETY

MA5 (cont)

BASIS (cont)

The allowed time frame is consistent with the guidance provided by Generic Letter 88-17, "Loss of Decay Heat Removal" (discussed earlier in this basis) and is believed to be conservative given that a low-pressure barrier to fission product release is established (i.e., secondary containment closure). The asterisk highlights the note at the bottom of the table. The note indicates that the second threshold is not applicable if actions are successful in restoring an RCS heat removal system to operation and RPV temperature is being reduced within the twenty-minute interval.

The third condition in Table M1 addresses complete loss of functions required for core cooling during Refueling and Cold Shutdown modes when primary containment closure, secondary containment closure, and RCS integrity are not established. RCS integrity is in place when the RCS pressure boundary is in its normal condition for the Cold Shutdown mode of operation (e.g., no freeze seals or main steam line nozzle plugs, etc.). No delay time is allowed for this threshold because the evaporated reactor coolant that may be released into the containment during this heatup condition could also be directly released to the environment.

A loss of Technical Specification components alone is not intended to constitute an Alert. The same is true of a momentary unplanned excursion above 212°F when the heat removal function is available.

Threshold #2 Basis:

The 10 psig pressure rise infers an uncontrolled RPV temperature rise in excess of the Technical Specification cold shutdown limit (212° F) for which MA5 Threshold #1 would permit up to sixty minutes to restore RCS cooling before declaration of an Alert. This EAL therefore covers situations of high decay heat loads, in which the event should be declared without delay.

This EAL is based on concerns raised by Generic Letter 88-17, "Loss of Decay Heat Removal." A number of phenomena such as decay heat removal system design and RPV water level instrumentation problems can lead to conditions in which decay heat removal is lost and core uncovery can occur. NRC analyses show that sequences that can cause core uncovery in 15 to 20 minutes and severe core damage within an hour after decay heat removal is lost.

Basis reference(s):

1. Technical Specifications 1.7
2. OU-AA-103, Shutdown Safety
3. NEI 99-01, Rev. 4 CA4

Table D-2: OCGS EAL Technical Basis

RECOGNITION CATEGORY
HAZARDS AND OTHER CONITIONS AFFECTING PLANT SAFETY

MG8

Initiating Condition

Loss of RPV Inventory Affecting Fuel Clad Integrity with Containment Challenged with Irradiated Fuel in the RPV.

Operating Mode Applicability:

3, 4

EAL Threshold Values:

1. Loss of RPV inventory per Table M5 indications.

AND

2. a. RPV level **< 0 inches TAF** for **> 30 minutes**.

   OR

   b. RPV Level unknown with Indication of core uncovery for **> 30 minutes** as evidenced by one or more of the following:

   - Refuel Floor Area Radiation Monitor C-10, North Wall, reading **> 3 R/hr**.
   - Erratic Source Range Monitor Indication.

AND

3. Containment is challenged as indicated by one or more of the following:
   - Primary Containment Hydrogen concentration **≥ 6%** and Oxygen **≥ 5%**.
   - Drywell pressure **> 44 psig**.
   - Primary and Secondary CONTAINMENT CLOSURE **not** established.
   - Any Secondary Containment radiation monitor reading **> EMG-3200.11 Maximum Safe Operating**.

Table M5 – Indications of RCS Leakage
Unexplained Identified or Unidentified leakage rise Unexplained Torus rise Unexplained vessel make-up rise Observation of leakage or Inventory loss

Basis:

CONTAINMENT CLOSURE: Containment Closure is considered to be Containment as required by Technical Specifications.

Table D-2: OCGS EAL Technical Basis

RECOGNITION CATEGORY
HAZARDS AND OTHER CONITIONS AFFECTING PLANT SAFETY

MG8 (cont.)

Basis: (cont)

This EAL represents the inability to restore and maintain RPV water level to above the Top of Active Fuel, 0 inches. Fuel damage is probable if core uncovery is prolonged and submergence cannot be restored and maintained. Available decay heat will cause boiling and further decrease RPV water level.

This EAL is based on concerns raised by Generic Letter 88-17, Loss of Decay Heat Removal, SECY 91-283, Evaluation of Shutdown and Low Power Risk Issues, NUREG-1449, Shutdown and Low-Power Operation at Commercial Nuclear Power Plants in the United States, and, NUMARC 91-06, Guidelines for Industry Actions to Assess Shutdown Management. A number of variables (e.g., decay heat removal system design, etc.) can have a significant affect on heat removal capability challenging the Fuel Cladding barrier. Analysis in the above references indicates that core damage may occur within an hour following continued core uncovery, therefore, the thirty-minute interval was conservatively chosen.

When RPV water level indication is unavailable, the inventory loss must be detected by erratic Source Range Monitor indication, elevated drywell radiation or unexplained rise in drywell floor or equipment drain sump pumpout rate. Post-TMI studies indicate that the installed nuclear instrumentation will operate erratically when the core is uncovered and that Source Range Monitors (SRM CH 21, CH 22, CH 23, or CH 24) can be used as a tool for making such determinations.

The Refuel Floor ARMs reading > 3R/hr is based on calculation EP-AEL-0501, Estimation of Radiation Monitor Readings Indicating Core Uncovery.

Sump pumpout rate rises must be evaluated against other potential sources of leakage such as cooling water sources inside the primary containment to ensure they are indicative of RCS leakage.

Four conditions are associated with the challenge to containment integrity:

- With Primary or Secondary containment closure not established with prolonged core uncovery, the health and safety of the public may be threatened.
- When hydrogen and oxygen concentrations in primary containment reach or exceed the deflagration limits, imminent loss of the Primary Containment barrier exists. To generate such levels of combustible gas, loss of the Fuel Cladding and RCS barriers must also have occurred.
- The secondary containment area radiation level is the EOP Maximum Safe Operating level. The Maximum Safe Operating radiation level is based on the highest radiation level at which neither equipment necessary for the safe shutdown of the plant will fail nor personnel access necessary for the safe shutdown of the plant will be precluded.
- The primary containment design pressure (44 psig at 292°F) is well in excess of that expected from the design basis loss of coolant accident. The threshold is indicative of a loss of both RCS and Fuel Cladding barriers in that it is not possible to reach this condition without severe core degradation.

Table D-2: OCGS EAL Technical Basis

RECOGNITION CATEGORY
HAZARDS AND OTHER CONITIONS AFFECTING PLANT SAFETY

MG8 (cont)

Basis Reference(s):

1. EMG-3200.01A, RPV Control – No ATWS
2. Technical Specifications 5.2 Basis
3. 2000-GLN-3200.03, Plant Specific Technical Guidelines for Severe Accident Guidelines
4. EMG-3200.11, Secondary Containment Control
5. FSAR Update 6.2.1.1.3
6. EMG-3200.02, Primary Containment Control
7. EP-AEL-0501,, Estimation of Radiation Monitor Readings Indicating Core Uncovery
8. NEI 99-01, Rev. 4 CG1

Table D-2: OCGS EAL Technical Basis

RECOGNITION CATEGORY
HAZARDS AND OTHER CONITIONS AFFECTING PLANT SAFETY

MS8

Initiating Condition

Loss of RPV Inventory Affecting Core Decay Heat Removal Capability

Operating Mode Applicability:

3

EAL Threshold Values:

1. **Without** Primary or Secondary CONTAINMENT CLOSURE established:

   a. RPV level **< 84 inches TAF**.

   OR

   b. RPV level unknown for **> 30 minutes** with a loss of RPV inventory per Table M5 indications.

OR

2. **With** Primary or Secondary CONTAINMENT CLOSURE established:

   a. RPV level **< 0 inches TAF**.

   OR

   b. RPV level unknown for **> 30 minutes** with a loss of RPV inventory as evidenced by either of the following:

   - Per Table M5 indications.
   - Erratic Source Range Monitor indication.

Table M5 – Indications of RCS Leakage
Unexplained Identified or Unidentified leakage rise Unexplained Torus rise Unexplained vessel make-up rise Observation of leakage or Inventory loss

Basis:

CONTAINMENT CLOSURE: Containment Closure is considered to be Containment as required by Technical Specifications.

MS8 (cont)

Basis (cont)

Under the conditions specified by this EAL, continued decrease in RPV water level is indicative of a loss of inventory control. Inventory loss may be due to an RPV breach, RCS pressure boundary leakage or continued boiling in the RPV. If a low-pressure boundary to fission product release does not exist (i.e., containment closure is not established), the RPV water level associated with this threshold is six inches below the Core Spray ECCS actuation setpoint (i.e., 90 in. - 6 in. = 84 inches.). If containment closure is established, a low-pressure boundary to fission product release exists and RPV water level can decrease to the top of active fuel, 0 in. (TAF), before a Site Area Emergency declaration is required. 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.

In the refueling mode, when RPV water level indication is unavailable, the inventory loss must be detected by drywell floor and equipment drain sump pumpout rates or erratic Source Range Monitor indication. Sump pumpout rate rises must be evaluated against other potential sources of leakage such as cooling water sources inside the primary containment to ensure they are indicative of RCS leakage. Post-TMI studies indicate that the installed nuclear instrumentation will operate erratically when the core is uncovered and that Source Range Monitors can be used as a tool for making such determinations. As water level in the Reactor Vessel lowers, the dose rate above the core will rise. The dose rate due to this core shine, scattering and radiation bounce off of the solid surfaces in the area will result in readings on the Refuel floor ARMs indicating ≥ 3 R/hr. This threshold radiation value is based on calculations documented in EP-AEL-0501.

This threshold is based on concerns raised by Generic Letter 88-17, Loss of Decay Heat Removal, SECY 91-283, Evaluation of Shutdown and Low Power Risk Issues, NUREG-1449, Shutdown and Low-Power Operation at Commercial Nuclear Power Plants in the United States, and, NUMARC 91-06, Guidelines for Industry Actions to Assess Shutdown Management. A number of variables (e.g., decay heat removal system design, etc.) can have a significant impact on heat removal capability challenging the Fuel Cladding barrier. Analysis in the above references indicates that core damage may occur within an hour following continued core uncovery, therefore, the thirty-minute interval was conservatively chosen.

The 30 minute interval allows sufficient time for actions to be performed to recover needed cooling equipment.

Basis Reference(s):

1. EMG-3200.01A, RPV Control – No ATWS
2. 2000-GLN-3200.03, Plant Specific Technical Guidelines for EOPs
3. NEI 99-01, Rev. 4 CS1
4. EP-AEL-0501,, Estimation of Radiation Monitor Readings Indicating Core Uncovery

Table D-2: OCGS EAL Technical Basis

RECOGNITION CATEGORY
HAZARDS AND OTHER CONITIONS AFFECTING PLANT SAFETY

MU8

Initiating Condition

RCS Leakage.

Operating Mode Applicability:

3

EAL Threshold Values:

1. RPV level **cannot** be restored and maintained **> 139 inches TAF**.

Basis:

The inability to restore and maintain level after reaching the RPS low level scram setpoint infers a degradation of the level of safety of the plant.

Basis Reference(s):

1. NEI 99-01, Rev. 4 CU1

Table D-2: OCGS EAL Technical Basis

RECOGNITION CATEGORY
HAZARDS AND OTHER CONITIONS AFFECTING PLANT SAFETY

MS9

Initiating Condition

Loss of RPV Inventory Affecting Core Decay Heat Removal Capability with Irradiated Fuel in the RPV.

Operating Mode Applicability:

4

EAL Threshold Values:

1. **Without** Secondary CONTAINMENT CLOSURE established:
   a. RPV level **< 84 inches TAF**.
   OR
   b. RPV level unknown with indication of core uncovery as evidenced by one or more of the following:
      - Refuel Floor Area Radiation Monitor C-10, North Wall reading **> 3 R/hr**.
      - Erratic Source Range Monitor indication.

OR

2. **With** Secondary CONTAINMENT CLOSURE established:
   a. RPV level **< 0 inches TAF**.
   OR
   b. RPV level unknown with indication of core uncovery as evidenced by one or more of the following:
      - Refuel Floor Area Radiation Monitor C-10, North Wall reading **> 3 R/hr**.
      - Erratic Source Range Monitor indication.

Table D-2: OCGS EAL Technical Basis

RECOGNITION CATEGORY
HAZARDS AND OTHER CONITIONS AFFECTING PLANT SAFETY

MS9 (cont)

Basis (cont)

CONTAINMENT CLOSURE: Containment Closure is considered to be Containment as required by Technical Specifications.

Under the conditions specified by this EAL, continued decrease in RPV water level is indicative of a loss of inventory control. Inventory loss may be due to an RPV breach, RCS pressure boundary leakage or continued boiling in the RPV. If a low-pressure boundary to fission product release does not exist (i.e., containment closure is not established), the RPV water level associated with this threshold is six inches below the Core Spray ECCS actuation setpoint (i.e., 90 in. - 6 in. = 84 inches). If containment closure is established, a low-pressure boundary to fission product release exists and RPV water level can decrease to the top of active fuel, 0 inches (TAF), before a Site Area Emergency declaration is required. 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 RPV water level after reaching this setpoint infers a failure of the RCS barrier and potential loss of the Fuel Cladding barrier. If it is determined that RPV water level cannot be monitored, the need for declaration of the Site Area Emergency is evaluated.

Under the Refueling conditions specified in this EAL, loss of the ability to monitor RPV water level in conjunction with indirect indication of possible core uncovery infer a continued lowering in RPV water level and loss of inventory control. Inventory loss may be due to an RPV breach, RCS pressure boundary leakage or continued boiling in the RPV.

In the Refueling mode, when RPV water level indication is unavailable, the inventory loss must be detected by erratic Source Range Monitor indication or elevated refuel floor radiation.

Post-TMI studies indicate that the installed nuclear instrumentation will operate erratically when the core is uncovered and that Source Range Neutron Monitors can be used as a tool for making such determinations. The Refuel Floor ARMs reading > 3R/hr is based on calculation EP-AEL-0501, Estimation of Radiation Monitor Readings Indicating Core Uncovery During Refuel.

Table D-2: OCGS EAL Technical Basis

RECOGNITION CATEGORY
HAZARDS AND OTHER CONITIONS AFFECTING PLANT SAFETY

MS9 (cont)

Basis Reference(s):

1. EMG-3200.01A, RPV Control – No ATWS
2. 2000-GLN-3200.03, Plant Specific Technical Guidelines for Severe Accident Guidelines
3. EP-AEL-0501, Estimation of Radiation Monitor Readings Indicating Core Uncovery
4. NEI 99-01, Rev. 4 CS2

Table D-2: OCGS EAL Technical Basis

RECOGNITION CATEGORY
HAZARDS AND OTHER CONITIONS AFFECTING PLANT SAFETY

HA5

Initiating Condition

Natural and Destructive Phenomena Affecting the Plant VITAL AREA.

Operating Mode Applicability:

1, 2, 3, 4, D

EAL Threshold Values:

1. A confirmed Seismic event that affects safety systems or systems required for safe shutdown requiring reactor scram.

OR

2. Tornado or high winds **> 99 mph** within PROTECTED AREA boundary and resulting in VISIBLE DAMAGE to any plant structures or equipment contained in any Table H2 area or Control Room indication of degraded performance of those systems.

Table H2 – Vital Areas
Reactor Bldg
Control Room Complex
Main Transformer/Condensate Transfer Pad
Intake Structure
#1 EDG Vault
#2 EDG Vault
EDG Fuel Oil Storage Tank

OR

3. Vehicle crash within PROTECTED AREA boundary and resulting in VISIBLE DAMAGE to any plant structures or equipment contained in any Table H2 area or Control Room indication of degraded performance of those systems.

OR

4. Turbine failure-generated missiles result in any VISIBLE DAMAGE or penetration of any Table H2 area.

HA5 (cont)

EAL Threshold Value (cont)

OR

5. Uncontrolled flooding that results in **EITHER**:

   a. Degraded safety system performance in any Table H3 area as indicated in the Control Room.

   OR

   b. Industrial safety hazards (e.g., electric shock) that preclude access necessary to operate or monitor safety equipment.

Table H3 – Internal Flooding Areas
Reactor Building NE Corner Room
Reactor Building SE Corner Room
Reactor Building SW Corner Room (RBEDT Rm)
Reactor Building NW Corner Room (CRD Pp Rm)

OR

6. Abnormal Intake Structure level, as indicated by **EITHER**:

   - **> 6.0 ft. MSL** (> 4.92 psig on PI-533-1172 and PI-533-1173 or > 6.0 ft MSL on CR-423-11 pt 24 and pt 23).

   OR

   - **≤ -4.0 ft. MSL** (≤ 0.50 psig on PI-533-1172 and PI-533-1173 or ≤ -4.0 ft MSL on CR-423-11 pt 24 and pt 23).

     MSL = Mean Sea Level

Basis:

VITAL AREA: is any area, normally within the PROTECTED AREA, which contains equipment, systems, components, or material, the failure, destruction, or release of which could directly or indirectly endanger the public health and safety by exposure to radiation.

PROTECTED AREA: is an area, which normally encompasses all controlled areas within the security protected area fence

VISIBLE DAMAGE: is damage to equipment or structure that is readily observable without measurements, testing, or analysis. Damage is sufficient to cause concerns regarding the continued operability or reliability of affected safety structures, systems, or components. Example damage includes: deformation due to heat or impact, denting, penetration, rupture, cracking, paint blistering. Surface blemishes (e.g., paint chipping, scratches) should not be included.

Table D-2: OCGS EAL Technical Basis

RECOGNITION CATEGORY
HAZARDS AND OTHER CONITIONS AFFECTING PLANT SAFETY

HA5 (cont)

Basis (cont)

Threshold #1 Basis:

This threshold addresses events that may have resulted in a VITAL AREA (Table H2) being subjected to forces beyond design limits and thus damage may be assumed to have occurred to plant safety systems. The initial report should not be interpreted as mandating a lengthy damage assessment prior to classification. No attempt is made in this EAL to assess the actual magnitude of the damage.

A reactor scram is required by procedure ABN-38, Station Seismic Event, if:

- The seismic event affects safe plant operation by jeopardizing the availability of safety systems, systems required to complete safe shutdown, or causing spurious actuation of equipment, or
- The Shift Manager determines it necessary to scram the Reactor to protect public safety.

Threshold #2 Basis:

This threshold addresses events that may have resulted in a Vital Area being subjected to forces beyond design limits and thus damage may be assumed to have occurred to plant safety systems. Vital Areas are areas that house equipment the operation of which may be needed to ensure the reactor safely reaches and is maintained in its lowest energy state. Vital Areas include structures that are in contact with or immediately adjacent to the areas that actually contains the equipment of concern. The Alert classification is appropriate if relevant plant parameters indicate that the performance of safety systems in the affected Vital Areas has been degraded. No attempt should be made to fully inventory the actual magnitude of the damage or quantify the degradation of safety system performance prior to declaration of an Alert under this threshold. The declaration of an Alert and the activation of the TSC provide the Emergency Director with the resources needed to perform detailed damage assessments.

The wind speed threshold is the FSAR design basis wind speed. Sustained winds present a more severe loading on the buildings than a gust.

Table D-2: OCGS EAL Technical Basis

RECOGNITION CATEGORY
HAZARDS AND OTHER CONITIONS AFFECTING PLANT SAFETY

HA5 (cont)

Basis (cont)

Threshold #3 Basis:

This threshold addresses events such as plane, helicopter, train, barge, car or truck crashes, or impact of projectiles into a plant Vital Area. This threshold addresses vehicle crashes that challenge the operability of systems necessary for safe shutdown of the plant. Vital Areas (Table H2) include Class 1 structures and those Class 2 structures that contain Class 1 Systems and components.

The Alert classification is appropriate if relevant plant parameters indicate that the performance of safety systems in the affected Vital Areas has been degraded. No attempt should be made to fully inventory the actual magnitude of the damage or quantify the degradation of safety system performance prior to declaration of an Alert under this threshold. The declaration of an Alert and the activation of the TSC provide the Emergency Director with the resources needed to perform detailed damage assessments. The Emergency Director also needs to consider the security ramifications of such crashes.

Threshold #4 Basis:

This EAL is intended to address the threat to safety related equipment impacted by missiles generated by main turbine rotating component failures. Table H2 includes all areas containing safety related equipment, their controls, and their power supplies.

Threshold #5 Basis:

This Threshold addresses the effect of internal flooding that has resulted in degraded performance of systems affected by the flooding, or has created industrial safety hazards (e.g., electrical shock) that preclude necessary access to operate or monitor safety equipment. The inability to operate or monitor safety equipment represents a potential for substantial degradation of the level of safety of the plant. This flooding may have been caused by internal events such as component failures, equipment misalignment, and fire suppression system actuation or outage activity mishaps. The Internal Flooding Areas listed in Table H3 include areas containing systems that are:

- Required for safe shutdown of the plant
- Not designed to be wetted or submerged
- Susceptible to internal flooding events

Table D-2: OCGS EAL Technical Basis

RECOGNITION CATEGORY
HAZARDS AND OTHER CONITIONS AFFECTING PLANT SAFETY

HA5 (cont)

Basis (cont)

Threshold #6 Basis:

This threshold covers high and low water level conditions as well as internal flooding events that may have resulted in a plant Vital Area being subjected to levels beyond design limits, and thus damage may be assumed to have occurred to plant safety systems.

Both pressure gauges listed (PI-533-1172 and 1173) are located inside each of the intake structure bays; they should both be considered to determine the overall effect on plant operations due to water level changes, since one structure could be isolated or have its associated screens clogged resulting in a low intake level condition on one side of the intake structure and therefore not provide positive indication of actual water level trend.

High Intake Structure level, > 6.0 feet MSL (> 4.92 psig on PI-533-1172 [CR-423-11 pt 24] and PI-533-1173 [CR-423-11 pt 23]) is capable of causing flooding that can affect Plant Vital Structures. At levels > 6.5 ft. above MSL, Circulating Water Pumps may become flooded. At levels > 8.0 ft. above MSL, Service Water pumps may become flooded. No attempt should be made to determine the magnitude of flooding. This is a long lead-time event but this level is at the intake structure lower deck so classification as an Alert is appropriate. The evidence of flooding is sufficient for declaration. PI-533-1172 and PI-533-1173 are local pressure indicators that provide a reading in psig that corresponds to Mean Sea Level (MSL) in feet. CR-423-11 points 23 and 24 are Main Control Room indications of intake bay levels displayed directly in feet MSL. CR-423-11 points 23 and 24 receive the same instrument signals and provide the same related indications as their associated local indicators discussed above.

Low Intake Structure level ≤ -4.0 feet MSL (< 0.50 psig on PI-533-1172 [CR-423-11 pt 24] and PI-533-1173 [CR-423-11 pt 23]) indicates the possible loss of Emergency Service Water pumps. PI-533-1172 and PI-533-1173 are local pressure indicators that provide a reading in psig that corresponds to Mean Sea Level (MSL) in feet. CR-423-11 points 23 and 24 are Main Control Room indications of intake bay levels displayed directly in feet MSL. CR-423-11 points 23 and 24 receive the same instrument signal and provide the same related indications as their associated local indicators discussed above.

Basis Reference(s):

1. ABN-38, Station Seismic Event
2. FSAR Update Section 3.3.7 (Seismic)
3. FSAR Update Section 3.3.1 (High winds)
4. ABN-31, High Winds
5. ABN-32, Abnormal Intake Level
6. ABN-29, Plant Fires
7. LES Calculation No. 72-01-01, Turbine Missile Analysis for New Monoblock Rotor and Blades," October 1996, Revision 3
8. NEI 99-01, Rev. 4 HA1

Table D-2: OCGS EAL Technical Basis

RECOGNITION CATEGORY
HAZARDS AND OTHER CONITIONS AFFECTING PLANT SAFETY

HU5

Initiating Condition

Natural and Destructive Phenomena Affecting the PROTECTED AREA.

Operating Mode Applicability:

1, 2, 3, 4, D

EAL Threshold Values:

1. a. Seismic event felt in plant.

   AND

   b. Confirmed by National Earthquake Center.

OR

2. Report by plant personnel of tornado striking or high winds **> 99 mph** within PROTECTED AREA boundary.

OR

3. Vehicle crash into plant structures or systems within PROTECTED AREA boundary affecting Table H2 area.

Table H2 – Vital Areas
Reactor Bldg
Control Room Complex
Main Transformer/Condensate Transfer Pad
Intake Structure
#1 EDG Vault
#2 EDG Vault
EDG Fuel Oil Storage Tank

OR

4. Report of turbine failure resulting in casing penetration or damage to turbine or generator seals.

HU5 (cont.)

EAL Threshold Value (cont)

OR

5. Uncontrolled flooding in any Table H3 area that has the potential to affect safety related equipment needed for the current operating mode.

Table H3 Internal Flooding Areas
Reactor Building NE Corner Room
Reactor Building SE Corner Room
Reactor Building SW Corner Room (RBEDT Rm)
Reactor Building NW Corner Room (CRD Pp Rm)

OR

6. Abnormal Intake Structure level, as indicated by **EITHER**:

   - **> 4.5 ft. MSL** (> 4.26 psig on PI-533-1172 and PI-533-1173 or > 4.5 ft MSL on CR-423-11 pt 24 and pt 23).

   OR

   - **≤ -3.0 ft. MSL** (≤ 0.94 psig on PI-533-1172 and PI-533-1173 or ≤ -3.0 ft MSL on CR-423-11 pt 24 and pt 23).

   MSL = Mean Sea Level

Basis:

PROTECTED AREA: is an area, which normally encompasses all controlled areas within the security protected area fence.

RECOGNITION CATEGORY
HAZARDS AND OTHER CONITIONS AFFECTING PLANT SAFETY

HU5 (cont)

Basis: (cont)

Threshold #1 Basis:

Confirmation from the National Earthquake center shall not delay declaration in the presence of other reliable confirming indications.

A felt earthquake is an earthquake of sufficient intensity such that the vibratory ground motion is felt at the nuclear plant site. An earthquake of this magnitude may be sufficient to cause minor damage to plant structures or equipment within the Protected Area. Damage is considered to be minor, as it would not affect physical or structural integrity. This event is not expected to affect the capabilities of plant safety functions.

The method of earthquake detection relies on the agreement of the shift operators on duty in the Control Room that the suspected ground motion is a “felt earthquake” followed by one or more reports confirming seismic activity near the station. Consensus of the Control Room operators with respect to ground motion helps avoid unnecessary classification if the motion were not due to seismic motion. As defined in the EPRI-sponsored “Guidelines for Nuclear Plant Response to an Earthquake”, dated October 1989, a “felt earthquake” is:

“An earthquake of sufficient intensity such that: (a) the inventory ground motion is felt at the nuclear plant site and recognized as an earthquake based on a consensus of Control Room operators on duty at the time, and (b) for plants with operable seismic instrumentation, the seismic switches of the plant are activated. For most plants with seismic instrumentation, the seismic switches are set at an acceleration of about 0.01 g.”

Threshold #2 Basis:

A tornado touching down within the Protected Area or sustained wind speeds > 99 mph within the Owner Controlled Area are of sufficient velocity to have the potential to cause damage to Vital Areas. These criteria are indicative of unstable weather conditions and represent a potential degradation in the level of safety of the plant. Verification of a tornado will be by direct observation and reporting by station personnel. Verification of wind speeds > 99 mph will be via meteorological data in the control room. This event will be escalated to an Alert if the tornado or high wind speeds result in damage to Vital Areas.

Threshold #3 Basis:

In this context, a "vehicle crash" is intended to address crashes of vehicle types large enough to cause significant damage to plant structures containing functions and systems required for safe shutdown of the plant.

Table D-2: OCGS EAL Technical Basis

RECOGNITION CATEGORY
HAZARDS AND OTHER CONITIONS AFFECTING PLANT SAFETY

HU5 (cont)

Basis: (cont)

Threshold #4 Basis:

This Threshold addresses main turbine rotating component failures of sufficient magnitude to cause observable damage to the turbine casing or to the seals of the turbine generator. Of major concern is the potential for significant leakage of combustible fluids (lubricating oils) and gases (hydrogen cooling) to the plant environs. It is not the intent of this EAL to classify minor operational leakage.

Threshold #5 Basis:

This threshold addresses the effect of flooding caused by internal events such as component failures, equipment misalignment, fire suppression system actuation or outage activity mishaps. The Internal Flooding Areas of concern for the Unusual Event declaration are those Table H3 areas that have the potential to affect safety related equipment needed for the current operating mode.

Threshold #6 Basis:

Both pressure gauges listed (PI-533-1172 and 1173) are located inside each of the intake structure bays; they should both be considered to determine the overall effect on plant operations due to water level changes, since one structure could be isolated or have its associated screens clogged resulting in a low intake level condition on one side of the intake structure and therefore not provide positive indication of actual water level trend.

High Intake Structure level, > 4.5 feet MSL (> 4.26 psig on PI-533-1172 [CR-423-11 pt 24] and PI-533-1173 [CR-423-11 pt 23]) is sufficiently high to require plant shutdown per ABN-32, Abnormal Intake Level. This event will be escalated to an Alert classification based on water level reaching the elevation of the Intake Structure lower deck. PI-533-1172 and PI-533-1173 are local pressure indicators that provide a reading in psig that corresponds to Mean Sea Level (MSL) in feet. CR-423-11 points 23 and 24 are Main Control Room indications of intake bay levels displayed directly in feet MSL. CR-423-11 points 23 and 24 receive the same instrument signals and provide the same related indications as their associated local indicators discussed above.

Low Intake Structure level ≤ -3.0 feet MSL (< 0.94 psig on PI-533-1172 [CR-423-11 pt 24] and PI-533-1173 [CR-423-11 pt 23]) indicates the possible loss of Radwaste Service Water pumps and is approaching levels, which may result in a loss of vital cooling equipment. This event will be escalated to an Alert based upon water level dropping to ≤ -4.0 feet MSL. PI-533-1172 and PI-533-1173 are local pressure indicators that provide a reading in psig that corresponds to Mean Sea Level (MSL) in feet. CR-423-11 points 23 and 24 are Main Control Room indications of intake bay levels displayed directly in feet MSL. CR-423-11 points 23 and 24 receive the same instrument signal and provide the same related indications as their associated local indicators discussed above.

HU5 (cont)

Basis Reference(s):

1. ABN-38, Station Seismic Event
2. FSAR Update 3.3.7 (Seismic)
3. FSAR Update 3.3.1 (High winds)
4. ABN-31, High Winds
5. ABN-32, Abnormal Intake Level
6. ABN-29, Plant Fire
7. NEI 99-01, Rev. 4 HU1

Table D-2: OCGS EAL Technical Basis

RECOGNITION CATEGORY
HAZARDS AND OTHER CONITIONS AFFECTING PLANT SAFETY

HA6

Initiating Condition

FIRE or EXPLOSION Affecting the Operability of Plant Safety Systems Required to Establish or Maintain Safe Shutdown.

Operating Mode Applicability:

1, 2, 3, 4, D

EAL Threshold Values:

1. FIRE or EXPLOSION in any Table H2 area.

Table H2 – Vital Areas
Reactor Bldg
Control Room Complex
Main Transformer/Condensate Transfer Pad
Intake Structure
#1 EDG Vault
#2 EDG Vault
EDG Fuel Oil Storage Tank

AND

2. a. Affected safety system parameter indications show degraded performance.

   OR

   b. Plant personnel report VISIBLE DAMAGE to permanent structures or safety system equipment within the specified area.

Basis:

FIRE: is 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.

EXPLOSION: is a rapid, violent, unconfined combustion, or catastrophic failure of pressurized equipment that imparts energy of sufficient force to potentially damage permanent structures, systems, or components.

Table D-2: OCGS EAL Technical Basis

RECOGNITION CATEGORY
HAZARDS AND OTHER CONITIONS AFFECTING PLANT SAFETY

HA6 (cont.)

Bases: (cont)

VISIBLE DAMAGE: is damage to equipment or structure that is readily observable without measurements, testing, or analysis. Damage is sufficient to cause concerns regarding the continued operability or reliability of affected safety structures, systems, or components. Example damage includes: deformation due to heat or impact, denting, penetration, rupture, cracking, paint blistering. Surface blemishes (e.g., paint chipping, scratches) should not be included.

The areas listed in Table H2 are VITAL AREAs that house equipment the operation of which may be needed to ensure the reactor safely reaches and is maintained in its lowest energy state. Personnel access to these areas may be an important factor in monitoring and controlling equipment operability. This EAL addresses fires and explosions that challenge the operability of systems necessary for safe shutdown of the plant.

The only fires and explosions that should be considered are those of sufficient force to visibly damage permanent structures or equipment required for safe shutdown. Visual observation of damage infers the ability to approach or enter the affected Vital Areas. Lacking the ability to adequately inspect the area for damage, the Alert classification is appropriate if relevant plant parameters indicate that the performance of safety systems in the affected Vital Areas has been degraded. No attempt should be made to fully inventory the actual magnitude of the damage or quantify the degradation of safety system performance prior to declaration of an Alert under this EAL. The declaration of an Alert and the activation of the TSC provide the Emergency Director with the resources needed to perform detailed damage assessments.

A steam line break or steam explosion that damages permanent structures or equipment in a Vital Area would be classified under this EAL. The method of damage is not as important as the degradation of plant structures or equipment.

Basis Reference(s):

1. ABN-29, Plant Fires
2. NEI 99-01, Rev. 4 HA2

Table D-2: OCGS EAL Technical Basis

RECOGNITION CATEGORY
HAZARDS AND OTHER CONITIONS AFFECTING PLANT SAFETY

HU6

Initiating Condition

FIRE Not Extinguished Within 15 Minutes of Detection, or EXPLOSION, within PROTECTED AREA Boundary.

Operating Mode Applicability:

1, 2, 3, 4, D

EAL Threshold Values:

1. FIRE in any Table H2 area **not** extinguished within **15 minutes** of Control Room notification or verification of a Control Room alarm.

OR

2. FIRE outside any Table H2 area with the potential to damage safety systems in any Table H2 area **not** extinguished within **15 minutes** of Control Room notification or verification of a Control Room alarm.

Table H2 – Vital Areas
Reactor Bldg
Control Room Complex
Main Transformer/Condensate Transfer Pad
Intake Structure
#1 EDG Vault
#2 EDG Vault
EDG Fuel Oil Storage Tank

OR

3. Report by plant personnel of an unanticipated EXPLOSION within PROTECTED AREA boundary resulting in VISIBLE DAMAGE to permanent structure or equipment.

Basis:

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

PROTECTED AREA: is an area, which normally encompasses all controlled areas within the security protected area fence.

EXPLOSION: is a rapid, violent, unconfined combustion, or catastrophic failure of pressurized equipment that imparts energy of sufficient force to potentially damage permanent structures, systems, or components.

Basis: (cont)

VISIBLE DAMAGE: is damage to equipment or structure that is readily observable without measurements, testing, or analysis. Damage is sufficient to cause concern regarding the continued operability or reliability of affected safety structure, system, or component. Example damage includes: deformation due to heat or impact, denting, penetration, rupture, cracking, paint blistering. Surface blemishes (e.g., paint chipping, scratches) should not be included.

Threshold #1 and #2 Basis

For the purposes of declaring an emergency event, the term "extinguished" means no visible flames.

The purpose of this threshold is to address the magnitude and extent of fires that may be potentially significant precursors to damage to safety systems. As used here, notification is visual observation and report by plant personnel or sensor alarm indication. The 15-minute period begins with a credible notification that a fire is occurring or indication of a valid fire detection system alarm. A verified alarm is assumed to be an indication of a fire unless personnel dispatched to the scene disprove the alarm within the 15-minute period. The report, however, shall not be required to verify the alarm.

The intent of the 15-minute period is to size the fire and discriminate against small fires that are readily extinguished (e.g., smoldering waste paper basket, etc.). Such fires are excluded from consideration in this EAL since they have no safety consequence.

Areas directly associated with a fire that may be considered to have a hazardous environment (due to smoke or administrative controls awaiting ventilation and/or testing) do not warrant declaration of an Alert under EAL HA7, Release of Toxic or Flammable Gases. However, an IDLH atmosphere resulting from the discharge of a fire-extinguishing agent (Cardox or Halon) should be evaluated under EAL HA7.

Threshold #3 Basis:

The only EXPLOSIONS that should be considered are those of sufficient force to visibly damage permanent structures or equipment in the PROTECTED AREA.

A steam line break or steam explosion that damages permanent structures or equipment in a PROTECTED AREA would be classified under this EAL. The method of damage is not as important as the degradation of plant structures or equipment.

Basis Reference(s):

1. ABN-29, Plant Fires
2. Station Security Plan – Appendix C
3. NEI 99-01, Rev. 4 HU2

Table D-2: OCGS EAL Technical Basis

RECOGNITION CATEGORY
HAZARDS AND OTHER CONITIONS AFFECTING PLANT SAFETY

HA7

Initiating Condition

Release of Toxic or Flammable Gases Within or Contiguous to a VITAL AREA Which Jeopardizes Operation of Systems Required to Maintain Safe Operations or Establish or Maintain Safe Shutdown.

Operating Mode Applicability:

1, 2, 3, 4, D

EAL Threshold Values:

1. Report or detection of toxic or asphyxiant gases within a Table H2 area (or area that restricts access to a Table H2 area) in concentrations that result in an atmosphere IMMEDIATELY DANGEROUS TO LIFE AND HEALTH (IDLH).

OR

2. Report or detection of flammable gases within a Table H2 area (or area that restricts access to a Table H2 area) in concentration greater than the LOWER FLAMMABILITY LIMIT (LFL).

Table H2 – Vital Areas
Reactor Bldg
Control Room Complex
Main Transformer/Condensate Transfer Pad
Intake Structure
#1 EDG Vault
#2 EDG Vault
EDG Fuel Oil Storage Tank

Basis:

VITAL AREA: is any area, normally within the PROTECTED AREA, which contains equipment, systems, components, or material, the failure, destruction, or release of which could directly or indirectly endanger the public health and safety by exposure to radiation.

IMMEDIATELY DANGEROUS TO LIFE AND HEALTH (IDLH): A condition that either poses an immediate threat to life and health or an immediate threat of severe exposure to contaminants which are likely to have adverse delayed effects on health.

LOWER FLAMMABILITY LIMIT (LFL): The minimum concentration of a combustible substance that is capable of propagating a flame through a homogenous mixture of the combustible and a gaseous oxidizer.

Table D-2: OCGS EAL Technical Basis

RECOGNITION CATEGORY
HAZARDS AND OTHER CONITIONS AFFECTING PLANT SAFETY

HA7 (cont.)

Bases: (cont)

This EAL is based on toxic, asphyxiant or flammable gases that have entered a plant structure in concentrations that are unsafe for plant personnel and, therefore, preclude access to equipment necessary for the safe operation of the plant. Toxic or flammable gases detected outside of these areas need not be considered for this EAL unless there is a spread of the gases into one of these areas.

Concentrations above life-threatening or flammable concentrations that result from planned maintenance or repair activities on-site, where planned contingency measures are identified to monitor and control gas(es), do not require classification.

Threshold #1:

Declaration should not be delayed for conformation from atmospheric testing if it is reasonable to conclude that the IDLH concentrations have been met (e.g. documented analysis, indication of personal ill effects from exposure, or operating experience with the hazards).

Areas directly associated with a fire that may be considered to have a hazardous environment (due to smoke or administrative controls awaiting ventilation and/or testing) do not warrant declaration of an Alert under this EAL. However, an IDLH atmosphere resulting from the discharge of a fire-extinguishing agent (Cardox or Halon) should be evaluated under this EAL.

The first condition is met if measurement of toxic gas concentration results in an atmosphere that is immediately dangerous to life and health (IDLH) within a Table H2 area. Non-Toxic Gases which displace oxygen (Halon or Nitrogen) to a life threatening level due to asphyxiation (oxygen deprivation) should also be considered for this EAL.

An asphyxiant is a material 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.

Threshold #2:

The second condition is met when the flammable gas concentration in a Table H2 area exceeds the lower flammability limit. Flammable gases such as hydrogen and acetylene are routinely used to maintain plant systems (hydrogen – main generator cooling, reactor coolant chemistry control) or repair equipment/components (acetylene - welding). This condition addresses concentrations at which gases can ignite or support combustion. An uncontrolled release of flammable gases within a Table H2 area has the potential to affect safe operation of the plant by limiting either operator or equipment operations due to the potential for ignition and resulting equipment damage or personnel injury. Once it has been determined that an uncontrolled release of flammable gas is occurring, sampling must be done to determine if the gas concentration exceeds the lower flammability limit.

Table D-2: OCGS EAL Technical Basis

RECOGNITION CATEGORY
HAZARDS AND OTHER CONITIONS AFFECTING PLANT SAFETY

HA7 (cont.)

Basis Reference(s):

1. ABN-33, Toxic or Flammable Gas Release
2. NEI 99-01, Rev. 4 HA3

Table D-2: OCGS EAL Technical Basis

RECOGNITION CATEGORY
HAZARDS AND OTHER CONITIONS AFFECTING PLANT SAFETY

HU7

Initiating Condition

Release of Toxic or Flammable Gases Deemed Detrimental to Normal Operation of the Plant.

Operating Mode Applicability:

1, 2, 3, 4, D

EAL Threshold Values:

1. Report or detection of toxic, asphyxiant or flammable gases that has or could enter the site area boundary in amounts that can affect NORMAL PLANT OPERATIONS.

OR

2. Report by Local, County or State Officials for evacuation or sheltering of site personnel based on an offsite event.

Basis:

NORMAL PLANT OPERATIONS: activities at the plant site associated with routine testing, maintenance, or equipment operations, in accordance with normal operating or administrative procedures. Entry into abnormal or emergency operating procedures, or deviation from normal security or radiological controls posture, is a departure from NORMAL PLANT OPERATIONs.

This EAL is based on the existence of uncontrolled releases of toxic, asphyxiant or flammable gas affecting safe plant operations or the health of plant personnel. The release may have originated within the Protected Area boundary, or it may have originated offsite and subsequently drifted inside the Protected Area boundary. Offsite events (e.g., tanker truck accident releasing toxic gases, etc.) resulting in the plant being within the evacuation area should also be considered in this EAL because of the adverse affect on normal plant operations.

It is intended that releases of toxic, asphyxiant or flammable gases are of sufficient quantity and the release point of such gases is such that safe plant operations would be affected. This would preclude small or incidental releases, or releases that do not impact structures needed for safe plant operation. The EAL is not intended to require significant assessment or quantification. The EAL assumes an uncontrolled process that has the potential to affect safe plant operations or plant personnel safety.

An asphyxiant is a material 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.

Basis Reference(s):

1. ABN-33, Toxic or Flammable Gas Release
2. NEI 99-01, Rev. 4 HU3