DUANE ARNOLD
ENERGY CENTER

EMERGENCY ACTION LEVEL (EAL)
TECHNICAL BASIS DOCUMENT

Revision 2 *(For NRC Review)*
December 1996

Duane Arnold Energy Center EMERGENCY ACTION LEVEL BASES DOCUMENT	Rev. 2 *(for NRC review)*
APPROVAL SHEET	PAGE 1 of 1 EFFECTIVE DATE: TBD

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REQUEST FOR ADDITIONAL INFORMATION

REGARDING DUANE ARNOLD ENERGY CENTER

EAL REVISION TO NUMARC/NESP-007 METHODOLOGY

The NRC has completed its initial review of the proposed Emergency Action Levels (EALs) contained in the September 15, 1995, Duane Arnold Energy Center submittal. The submittal consisted of the proposed EAL procedure, the Duane Arnold EAL Technical Basis Document, letters of agreement from State and local authorities, and copies of applicable Emergency Operating and Abnormal Operating procedures. The EAL procedure contained the EAL statements, the corresponding emergency classifications, a unique designator number for each EAL, the plant Operating Condition Applicability, and any tables or other data necessary for interpretation of the EAL. The Technical Basis Document gave further details on the EAL, provided justification for any deviations from the NUMARC example EALs and cited specific Duane Arnold procedure numbers and other related references.

The proposed EALs were reviewed against the guidance in NUMARC/NESP-007, "Methodology for Development of Emergency Action Levels," Revision 2. This document has been endorsed by the NRC in Regulatory Guide 1.101, "Emergency Planning and Preparedness for Nuclear Power Reactors," Revision 3, as an alternative means by which licensees can meet the requirements in 10 CFR 50.47 (b) (4) and Appendix E to 10 CFR Part 50. Since the staff has previously endorsed the guidance in NUMARC/NESP-007, the review focused on those EALs that deviated from the guidance and those EALs that required the development of site-specific thresholds. As a result of the initial review, a number of EALs were identified which required additional information in order to determine whether the EALs conform to NUMARC/NESP-007. Please provide this additional information as discussed below.

GENERAL

Issued No. 1

The Duane Arnold EAL scheme deviated from the NUMARC methodology by not grouping EALs under initiating conditions (ICs). The Duane Arnold EAL basis document did group the EALs under ICs; however, this arrangement was not maintained in the emergency implementing procedure used for classifying the emergency. The grouping of EALs under the ICs to which the EALs correspond allows the person classifying (and the people being notified of the classification) to more clearly understand the plant condition of concern.

IES Utilities Response

> The NUMARC Initiating Conditions's have been grouped with their applicable DAEC EAL's.

EAL Recognition Category A
Abnormal Rad Levels/Radiological Effluent

Issue No. 2

NUMARC Initiating Condition (IC) AU1 states:

> *AU1 Any unplanned Release of Gaseous or Liquid Radioactivity to the Environment that Exceeds Two Times the Radiological Technical Specifications for 60 Minutes or Longer.*

NUMARC EALs associated with this IC include:

> 1. *A valid reading on one or more of the following monitors that exceeds the "value shown" (site specific monitors) indicates that the release may have exceeded the above criterion and indicates the need to assess the release with (site specific procedure):*
>
> *(site-specific list)*
>
> 2. *Confirmed sample analyses for gaseous or liquid releases indicates concentrations or release rates with a release duration of 60 minutes or longer in excess of two times (site-specific technical specifications).*

A. An EAL corresponding the NUMARC Example EAL 2 was not provided. No justification for this deviation was provided. (This comment also applies to the corresponding Duane Arnold Alert level EAL AA1.)

B. In the Duane Arnold basis document for this EAL it is stated that:

> *The Low Level Radwaste Processing and Storage Facility (LLRPSF) is not considered as an accident release point since the radiation monitor automatically trips the building exhaust at the Technical Specification instantaneous release limit thus terminating the release...*

The NUMARC basis states that this IC "represents an uncontrolled situation and hence, a potential degradation in the level of safety." In formulating the EAL's for this IC, it should not be presumed that safety systems will operate as designed. In fact it is the misoperation of this equipment which will cause the IC to be met. Therefore, the Duane Arnold EAL scheme should include EAL's for the monitored release paths. (This comment also applies to the corresponding Duane Arnold Alert level EAL AA1.)

IES Utilities Response

EAL's have been added to meet the NUMARC condition of concern. The DAEC EAL's read "Confirmed sample analyses for gaseous or liquid releases indicates concentrations in excess of 2 times ODAM limits for greater than 60 minutes" for AU1 and "Confirmed sample analyses for gaseous or liquid releases indicates concentrations in excess of 200 times ODAM limits for greater than 15 minutes" for AA1.

In order to meet the NUMARC condition of concern for issue 2 B, EAL's have been added which read "Valid LLRSPF (Kaman) rad monitor reading above 9 E-4 µCi/cc for more than 60 minutes" for AU1 and "Valid LLRSPF (Kaman) rad monitor reading above 9 E-2 µCi/cc for more than 15 minutes" for AA1.

Issue No. 3

NUMARC Initiating Condition (IC) AA1 states.

AA1 Any unplanned Release of Gaseous or Liquid Radioactivity to the Environment that Exceeds 200 Times the Radiological Technical Specifications for 15 Minutes or Longer.

A NUMARC EAL associated with this IC is:

3. A valid reading on perimeter radiation monitoring system greater than 10.0 mR/hr sustained for 15 minutes or longer.

The equivalent Duane Arnold EAL is:

Valid field survey reading outside the site boundary above 10 mR/hr. (Dose assessment is NOT available.)

A. The addition of the condition "dose assessment NOT available" is not appropriate because exceeding the survey results, in and of itself, is indicative of a loss of control of radioactive material which meets the IC. (This comment also applies to Duane Arnold EAL's AS1 and AG1.)

B. The Duane Arnold EAL did not include the condition "sustained for 15 minutes or longer." No justification was provided for this deviation. (This same comment also applies to Duane Arnold EAL's AS1 and AG1.)

IES Utilities Response

The phrase "dose assessment NOT available" has been removed and the phrase "sustained for 15 minutes or longer" has been added to the EAL's under IC's AA1, AS1, and AG1.

Issue No. 4

NUMARC IC AA2 states:

AA2 Major Damage to Irradiated Fuel or Loss of Water Level that Has or Will Result in the Uncovering of Irradiated Fuel Outside the Reactor Vessel.

NUMARC EAL's associated with this IC include:

2. *Report of visual observation of irradiated fuel uncovered.*

3. *Water level less than (site-specific) feet for the Reactor Refueling Cavity that will result in Irradiated Fuel Uncovering.*

The Duane Arnold EAL scheme includes the following EAL's:

1. *Uncontrolled loss of reactor cavity or fuel pool water level results in a spent fuel assembly that is NOT fully covered by water.*

   OR

2. *Valid Fuel Pool water level indication (LI-3414) below 13 feet 9 inches*

A. The Duane Arnold EAL #1 does not provide for the method of detection of the plant condition as is provided for in NUMARC EAL #2, i.e. "Report of visual observation of irradiated fuel uncovered." This concern may be the result of the Duane Arnold EAL scheme not including EALs under ICs.
B. The Duane Arnold EAL scheme does not include an EAL corresponding the NUMARC EAL #3. No justification was provided for this deviation.

IES Utilities Response

An EAL has been added as identified in the NUMARC document. The EAL reads "Report of visual observation of irradiated fuel uncovered." The phrase "Uncontrolled loss of reactor cavity..." has been removed.

An EAL that reads "Water level reading below 450", as indicated on LI4541 (floodup) for the Reactor Refueling Cavity that will result in Uncovering Irradiated Fuel." has been added.

Issue No. 5

NUMARC IC AA3 states:

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

NUMARC EAL's associated with this IC include:

> 1. *Valid (site-specific) radiation monitor readings GREATER THAN 15 mR/hr in areas requiring continuous occupancy to maintain plant safety functions.*
>
> 2. *Valid (site-specific) radiation monitor readings GREATER THAN <site specific> values in areas requiring infrequent access to maintain plant safety functions.*
>
> *(Site-specific) list*

The corresponding Duane Arnold EAL is:

> *Dose rates prevent occupancy or access to areas required to achieve or maintain safe shutdown.*

A. The Duane Arnold EAL scheme did not include EAL's corresponding to these NUMARC EALs for this IC. The condition provided in the Duane Arnold scheme is closely related to the NUMARC IC but does not contain site-specific thresholds for classifying the event.

IES Utilities Response

Two EAL's have been added. They are "Valid area radiation monitor (RE9162) reading greater than 15 mR/hr in the Control Room" and "Valid area radiation monitor (RE9168) reading greater than 500 mR/hr at the Remote Shutdown Panel 1C388."

Issue No. 6

NUMARC IC AS1 states:

> *AS1 Boundary Dose Resulting from an Actual or Imminent Release of Gaseous Radioactivity Exceeds 100 mR Whole Body or 500 mR Child Thyroid for the Actual or Projected Duration of the Release.*

NUMARC Example EAL's associated with this IC include:

> 1. *A valid reading on one or more of the following monitors that exceeds or is expected the value shown indicates that the release may have exceeded the above criterion and indicates the need to assess the release with (site-specific procedure):*
>
> 4. *Field survey results indicate site boundary dose rates exceeding 100 mR/hr expected to continue for more than one hour; or analyses of field survey samples indicate child thyroid dose commitment of 500 mR for one hour of inhalation.*

The Duane Arnold EAL corresponding to NUMARC EAL #4 is:

> 4. *Valid field survey reading outside the site boundary above 100 mR/hr.*

A. The Duane Arnold EAL scheme did not include the NUMARC condition for the child thyroid dose commitment.

B. The Duane Arnold EAL scheme includes EAL's corresponding to NUMARC EAL #1. The Duane Arnold EAL basis document states that; 'In order to calculate suitable radiation monitor values as described in the generic methodology, use of an assumed source term mixture, use of annual average meteorology, and rounding off is required." Insufficient detail was provided to determine whether the "assumed source term" met the guidance for the source term in the NUMARC basis for this EAL.

IES Utilities Response

Dose assessment using MIDAS is based on the EPA-400 methodology, e.g., use of Total Effective Dose Equivalent (TEDE) and Committed Dose Equivalent (CDE) Thyroid. TEDE is somewhat different from whole body dose received from gaseous effluents, as determined by ODAM methodology, which forms the basis for the radiation monitor readings calculated in AU1. Whole body dose from gaseous effluents is in accordance with the generic methodology.

The gaseous effluent radiation monitors can only detect noble gases. The contribution of iodine's to TEDE and CDE Thyroid could therefore only be determined either by: (1) utilizing MIDAS, or (2) gaseous effluent sampling. DAEC EAL #4 is written in terms of TEDE and CDE Thyroid. Including the child thyroid dose commitment terminology is not applicable in this case.

The source terms used have been pre-loaded into MIDAS and are the default mixes associated with a loss of coolant accident (LOCA) and a control rod drop (CRD) as described in the UFSAR, incorporating EPA-400 guidelines. DAEC has recalculated and rewritten the EAL's associated with AS1 and AG1 to reflect the revised source term. The revised EAL's associated with AS1 and AG1 are "Valid field survey reading outside the site boundary >100 mR/hr or above 500 mR/hr CDE Thyroid." and "Valid field survey reading outside the site boundary >1,000 mR/hr or >5,000 mR/hr CDE Thyroid", respectively.

NUMARC Recognition Category F
Fission Product Barrier Degradation

Issue No. 7

The NUMARC EAL methodology includes a fission product barrier matrix for determining whether or not a barrier (fuel clad, reactor coolant system, or

containment) is lost or potentially lost and for classifying events based on the combination of lost or potentially lost barriers. The fission product barrier matrix provides multiple indications to operators to assess the status of each of the barriers.

Classification of an event is made by determining the combination of barriers which have either been lost or potentially lost. The NUMARC guidance specifies that the following combination of barriers is indicative of a Site Area Emergency.

Loss of BOTH Fuel Clad and RCS
OR
Potential Loss of BOTH Fuel Clad and RCS
OR
Potential Loss of EITHER Fuel Clad OR RCS, and Loss of ANY Additional Barrier

A. The Duane Arnold EAL scheme also contains a fission product barrier matrix. However, the Duane Arnold EAL scheme defines the combination of barriers which is indicative of a Site Area Emergency differently than the NUMARC guidance. The combination of barriers specified in the Duane Arnold EAL scheme for the Site Area Emergency is:

Loss or Potential Loss of Any Two Barriers

The Duane Arnold EAL basis document explains that using this combination of barriers makes the classification easier to understand and that no sequences are significantly affected by the simplified logic. Insufficient detail was provided in the Duane Arnold EAL basis document to verify that the Duane Arnold EAL scheme meets the intent of the NUMARC guidance. The comparison table provided did not identify which EALs were being compared and did not justify the adequacy of those combinations which would result in a Site Area Emergency classification in the Duane Arnold EAL which would not have resulted in a Site Area Emergency classification in the NUMARC guidance.

IES Utilities Response

The potential loss of the primary containment based on radiation/core damage and RPV level indicators can only occur if there is a loss of both the fuel clad and RCS barriers. This is true because the primary containment barrier potential loss value is higher than the corresponding values for the same indicators that indicate loss of both the RCS and fuel clad barriers. For the primary containment atmosphere indicators, potential loss indicators of either torus pressure of [illegible] psig or elevated hydrogen levels can only result from significant core damage that would result from a loss of both the RCS and the fuel clad. Thus, for these conditions, existence of the thresholds for containment potential loss could only lead to a General Emergency declaration. That only leaves EC/OSS judgment indicators, Primary Containment Atmosphere loss indicators and the Leakage indicators to be considered.

By their very nature, the EC/OSS Judgment indicators are judgment calls and use of the NUMARC generic logic or the DAEC logic would make no difference. That leaves the Leakage indicators and the remaining Primary Containment Atmosphere indicators. Since the primary containment barrier indicators are all "loss" indicators, the existence of at least a "potential loss" of either the fuel clad or the RCS barriers will always result in a Site Area Emergency whether or not the NUMARC logic or the logic used at the DAEC and other plants is applied.

Issue No. 8

The NUMARC EAL for the loss and potential loss of the fuel clad barrier based on reactor vessel water level indications are:

Loss:

Level LESS THAN (site-specific) value

Potential Loss:

Level LESS THAN (site-specific) value

The corresponding Duane Arnold EAL's are:

Loss:

RPV Level below -30 inches and cannot be restored

Potential Loss:

RPV Level below -15 inches and cannot be restored

A. The Duane Arnold EAL basis document did not justify the addition of "cannot be restored" to these EALs. It is not clear why the loss or potential loss cannot be based on the level alone. The addition of the condition "cannot be restored" may cause confusion and/or delay classification. (this same comment also applies to the Loss of Reactor Coolant System Barrier EAL based upon reactor vessel level.)

IES Utilities Response

The phrase "cannot be restored" will be removed. This will be applied to all applicable level statements contained on the Fission Barrier Table.

Issue No. 9

The Duane Arnold EAL scheme includes the following EAL:

Core damage assessment determines at least 5% fuel clad damage

The Duane Arnold EAL basis document states:

> *It is intended that determination of barrier loss be made whenever the indicator threshold (for the containment monitor) is reached until such time that core damage assessment is performed, at which time direct use of containment rad monitor readings is no longer required.*

A. The Duane Arnold EAL scheme did not include a statement corresponding to the statement in the Duane Arnold basis document regarding the use of the containment rad monitor EAL. This may cause confusion when classifying an event.

IES Utilities Response

The Fuel Clad Barrier statements have been rewritten as follows:

[L] Fuel damage assessment (PASAP 7.2) determines at least 5% fuel clad damage

OR

Fuel damage is indicated by any of the following:

[L] Valid drywell rad monitor reading above 7E+2 R/hr

OR

[L] Valid torus rad monitor reading above 3E+1 R/hr

OR

[L] Coolant activity above 300uCi/gm DOSE EQUIVALENT I-131

Issue No. 10

The NUMARC EAL for the potential loss of the reactor coolant system barrier based on RCS leak rate indications includes the following conditions:

> *Unisolable primary system leakage outside the drywell as indicated by area temp or area rad alarm*

The corresponding Duane Arnold EAL is:

> *Unisolable primary system leakage outside the drywell as indicated by ARMs or in-plant radiological surveys*

A. The Duane Arnold EAL basis document did not justify not including the condition "as indicated by area temp" in the Duane Arnold EAL. (This comment also applies to the same Duane Arnold EAL listed under the Loss of Containment Barrier column of the fission product barrier table.)

IES Utilities Response

The phrase "as indicated by area temperatures" was added to all applicable locations on the Fission Barrier table.

Issue No. 11

The NUMARC EAL for the potential loss of the RCS barrier based on drywell pressure indications is:

> *Pressure Greater than (site-specific) psig*

The NUMARC basis for this EAL states:

> *The (site-specific) drywell pressure is based on the drywell high pressure alarm setpoint and indicates a LOCA. A higher value may be used if supporting documentation is provided which indicates the chosen value is less than the pressure which would be reached for a 50 gpm Reactor Coolant System leak.*

The corresponding Duane Arnold EAL is:

> *Drywell cooling operating AND drywell pressure above 2 psig*

The Duane Arnold EAL basis document states:

> *There is no significant deviation from the generic indicator. The (site-specific) value for this loss indicator corresponds to the drywell high pressure ECCS initiation signal setpoint of 2.0 psig.*

A. The Duane Arnold EAL basis does not address why the Duane Arnold EAL uses the ECCS initiation drywell pressure setpoint instead of the alarm setpoint as specified in the NUMARC guidance.

B. It is not clear whether drywell cooling operation may be automatically isolated when drywell pressure exceeds 2 psig and whether this may cause confusion when classifying the event.

IES Utilities Response

The DAEC design is that of a GE Mark I Containment. During reactor operation, drywell pressure is maintained between 0.5 and 1.0 psig. The high pressure alarm setpoint of 1.5 psig was not selected, as this is too close to the normal operating pressure band, and could be exceeded for reasons other than a RCS leak. Analysis at the DAEC shows that a 50 gpm RCS leak would result in a 2 to 3 psig pressure rise over a six minute time period. Since a 2 psig rise would be above the ECCS initiation setpoint, it is more practical to use the ECCS initiation setpoint of 2 psig for this EAL. Drywell cooling does not automatically isolate at 2 psig in the drywell, at the DAEC. Therefore, no confusion would result when classifying the event.

Issue No. 12

The NUMARC EAL's for the loss of the Containment barrier based on drywell pressure indications are:

> *Rapid unexplained decrease following initial increase or*
>
> *Drywell pressure response not consistent with LOCA conditions*

A. The Duane Arnold EAL scheme did not include these EAL's and the Duane Arnold EAL basis document did not justify this deviation.

IES Utilities Response

The NUMARC EAL's for Primary Containment Barrier Loss for Drywell Pressure have been added as follows:

> Rapid unexplained decrease following initial increase
>
> **OR**
>
> Drywell pressure response not consistent with LOCA conditions.

The basis information to support the above conditions has been inserted into the DAEC basis document to support these EAL's.

Issue No. 13

The NUMARC EAL for the potential loss of the containment barrier based on reactor pressure vessel water level is:

Reactor vessel water level LESS THAN (site-specific) value and the maximum core uncovery time limit is in the UNSAFE region

The corresponding Duane Arnold EAL is:

RPV Level below -40 inches AND no injections source available

A. The Duane Arnold EAL does not appear to meet the intent of the NUMARC EAL. Two concerns have been identified with the Duane Arnold EAL. One is that the term "not available' has not been defined and may cause confusion when classifying the event. The second concern is that even if the injection source is available, if the water level was to remain below 40 inches for a given amount of time, the barrier should be considered potentially lost. As stated in the NUMARC EAL basis: "if emergency operating procedures have been ineffective in restoring reactor vessel level within the maximum core uncovery time limit, there is not a success path.".. Whether or not the procedures will be effective should be apparent within the time provided. The Emergency Director should make the declaration as soon as it is determined that the procedures have been, or will be, ineffective."

IES Utilities Response

The phrase "...AND no injection source available" has been removed, as this is just a restatement of given information in EOP's. Also the Maximum Core Uncovery Time Limit (MCUTL) is addressd in Q&A Fission Product Barriers - BWR #10. Which states that this is the improper use of this chart considering the input assumptions for the curve.

NUMARC Recognition Category H
Hazards and Other Conditions Affecting Plant Safety

Issue No. 14

NUMARC IC HU1 includes the following EAL:

> 3. *Assessment by the control room that an event has occurred.*

A. The Duane Arnold EAL scheme did not include an EAL corresponding to this EAL and no justification was provided for this deviation.

IES Utilities Response

An EAL stating "Assessment by the control room that an event has occurred" along with supporting basis information has been included.

Issue No. 15

NUMARC IC HU1 includes the following EAL:

> 4. *Vehicle crash into plant structures or systems within protected area boundary.*

The NUMARC Basis for this EAL explains that:

> *Automobiles, trucks, and forklifts are also vehicles within the context of this EAL. The key is whether or not the vehicle can potentially cause significant damage to plant structures.*

The corresponding Duane Arnold EAL is:

> 7. *Vessel or vehicle collision with structures or equipment required for safe shutdown*

The Duane Arnold basis document states that:

> *DAEC EAL 7 addresses vessel (aircraft) or vehicle (truck or train) crashes with structures or equipment required for safe shutdown*

A. The Duane Arnold EAL did not define the term "structures or equipment required for safe shutdown." It is not clear that users of the EAL procedure will be able to ascertain what are the structures or equipment required for safe shutdown. (This comment applies to the other EALs under IC HU1.)

B. The Duane Arnold basis document deviates from the NUMARC guidance by specifically not including automobiles and forklifts as vehicles for this EAL.

IES Utilities Response

The table located adjacent to HU2 titled "Systems & Equipment of Concern" is intended to illustrate the equipment or structures required for safe shutdown.

"Automobiles" and "forklifts" have been added to the basis document.

Issue No. 16

NUMARC IC HU1 includes the following EAL:

> 5. *Report by plant personnel of an unanticipated explosion within protected area boundary resulting in visible damage to permanent structure or equipment.*

The corresponding Duane Arnold EAL is:

> 3. *Visible damage of structures or equipment required for safe shutdown*
>
> 5. *Explosion within plant protected area*

A. Duane Arnold EAL #3 is not specific as to the cause of the damage which would result in the Unusual Event classification. It is not clear whether "damage" to equipment from maintenance errors or operational errors would be classified under this EAL.

B. Duane Arnold EAL #5 does not include the NUMARC condition of "resulting in visible damage...." No justification was provided for this deviation.

IES Utilities Response

EAL's #3 and #5 were combined into a single EAL consistent with the NUMARC EAL which states "Report of an unanticipated explosion within the protected area boundary resulting in visible damage to permanent structures or equipment".

Issue No. 17

NUMARC IC HU1 includes the following EAL:

> 6. *Report of turbine failure resulting in casing penetration or damage to turbine or generator seals*

The corresponding Duane Arnold EAL is:

> 6. *Turbine failure causing observable casing damage*

A. Contrary to the NUMARC EAL, the Duane Arnold EAL did not include the condition "damage to turbine or generator seals."

IES Utilities Response

"Report of turbine failure resulting in casing penetration or damage to turbine or generator seals" has been added to the DAEC EAL's.

Issue No. 18

NUMARC IC HU2 includes the following EAL:

1. *Fire in buildings or area 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*

The corresponding Duane Arnold EAL is:

1. *Fire within safe shutdown area NOT extinguished within 15 minutes of control room notification or verification of control room alarm.*

A. The NUMARC EAL specifies "buildings or areas contiguous to...." The corresponding Duane Arnold EAL limits the areas considered to only "safe shutdown area(s)." This same list of areas is included in the related Alert level EAL. The areas applicable under the Unusual Event EAL is broader than the areas applicable under the Alert level EAL.

IES Utilities Response

The revised DAEC EAL reads "Fire in buildings or areas contiguous to any of the following areas not extinguished within 15 minutes of control room notification or verification of a control room alarm."

- Reactor, Turbine, Control, Administrative/Security buildings
- Pump house
- Intake structure

The basis information has been updated to identify the above facilities.

Issue No. 19

NUMARC IC HU3 includes the following EAL:

> 2. *Report by local, county or State official for potential evacuation of site personnel based on offsite event*

The corresponding Duane Arnold EAL is:

> 2. *Notification of near site release that may require evacuation.*

A. The term "near site" is not defined in the Duane Arnold EAL. In addition, it is not clear that including this term is necessary for the Duane Arnold EAL to meet the intent of this NUMARC EAL.

IES Utilities Response

The DAEC EAL has been reworded as follows: "Report by local, county or State official for potential evacuation of site personnel based on an offsite event".

Issue No. 20

NUMARC IC HU4 includes the following EAL:

> 2. *Other security events as determined from (site-specific) Safeguards Contingency Plan.*

The corresponding Duane Arnold EAL is:

> 1. *Suspected sabotage device discovered in plant switchyard.*

The Duane Arnold EAL basis document states:

> *Other (site-specific) security events of concern at DAEC include discovery of a suspected sabotage device in the plant switchyard, which is located outside the protected area.*

A. It is not clear from the information provided whether Duane Arnold EAL #2 includes all the applicable security events in the Duane Arnold Safeguards Contingency Plan.

IES Utilities Response

The information provided is complete and consistent with the DAEC Security Contingency Plan, Revision 33. Due to the nature of the safeguards information contained in that plan it is exempt from public disclosure pursuant to 10CFR73.21.

Issue No. 21

NUMARC IC HA1 includes the following EAL:

3. *Report of any visible structural damage on any of the following plant structures:*

- *Reactor Building*
- *Intake Building*
- *Ultimate Heat Sink*
- *Refueling Water Storage Tank*
- *Diesel Generator Building*
- *Turbine Building*
- *Condensate Storage Tank*
- *Control Rooms*
- *Other (Site-Specific) Structures*

A. The Duane Arnold EAL scheme did not include an EAL corresponding to this EAL and did not justify this deviation.

IES Utilities Response

The DAEC EAL corresponding to NUMARC IC HA1 EAL #3 is; "Report to control room of damage affecting safe shutdown areas". The DAEC basis document addresses this in Section H page 9. The DAEC basis document also identifies all locations applicable to this EAL in the table labeled "Safe Shutdown Areas".

Issue No. 22

NUMARC IC HA1 includes the following EAL:

3. *Vehicle crash affecting plant vital areas*

The corresponding Duane Arnold EAL is:

6. *Vessel or vehicle collision affecting ability to achieve or maintain safe shutdown*

A. The Duane Arnold EAL deviates from the NUMARC EAL by including the condition that the collision affects the ability to achieve or maintain safe shutdown. No justification was provided for this deviation. It may be difficult to make a definitive determination whether the vehicle collision did affect the ability to achieve or maintain safe shutdown. It is not appropriate to delay classification in order to make this determination. (This comment also applies to Duane Arnold EAL HA1, #5)

IES Utilities Response

HA1 is intended to be a "confirmed" collision affecting a plant safe shutdown area (vital area). The EAL has been rewritten as follows: "Vehicle crash affecting plant vital areas".

Issue No. 23

NUMARC IC HA4 includes the following EAL:

> 2. *Other security events as determined from (site-specific) Safeguards Contingency Plan.*

A. The Duane Arnold EAL scheme did not include an EAL corresponding to this EAL. The Duane Arnold EAL basis document states that: "Based on information provided by DAEC Security, generic EAL 2 is unnecessary at DAEC." It is not clear what, if any, security events were considered in making this determination. This comment also applies to the corresponding Site Area Emergency IC HS1. For HS1 it appears that a sabotage device discovered in the plant vital are as should be included as an EAL.

IES Utilities Response

"Sabotage device discovered in the plant protected area" as an EAL under HA4, has been added. The EAL "Sabotage device discovered in the plant vital area" has also been added under HS1.

Issue No. 24

NUMARC IC HA5 includes the following EAL:

> 1. *Entry into (site-specific) procedure for control room evacuation.*

The corresponding Duane Arnold EAL is:

Control Room evacuation procedures have been initiated

A. Contrary to the NUMARC guidance, the specific Duane Arnold procedure for control room evacuation is not identified in the EAL.

IES Utilities Response

A reference to Abnormal Operating Procedure 915, "Shutdown Outside Control Room", was incorporated into this EAL.

Issue No. 25

NUMARC IC HS2 includes the following EAL:

1. *The following conditions exist:*
    - a. *Control room evacuation has been initiated*

      AND
    - b. *Control of the plant cannot be established per (site-specific) procedure within (site-specific) minutes.*

The corresponding Duane Arnold EAL is:

> *Control room has been evacuated AND control of plant from Remote Shutdown Panel 1C388 NOT established within 20 minutes.*

The basis for the NUMARC EAL states:

> *(Site-specific) time for transfer based on analysis or assessments as to how quickly control must be reestablished without core uncovering and/or core damage.* ***This time should not exceed 15 minutes****.* (emphasis added)

The Duane Arnold basis document states:

> *Operator control within 20 minutes would not impact the integrity of the fuel clad, the reactor pressure vessel, and the primary containment.*

A. The Duane Arnold EAL basis did not justify why the time limit to classify this event should exceed 15 minutes. For instance, the Duane Arnold basis did not describe why more than 15 minutes is needed for determining whether control is established at the remote shutdown panel

IES Utilities Response

Support has been added to the DAEC basis document explaining why the basis for 20 minutes vs. 15 minutes, as recommended by the generic document. DAEC physically has satellite panels associated with the remote shutdown panel at various locations within the plant. General Electric Report MDE-44-0386, "Safe Shutdown Appendix R Analysis for DAEC", March 83, identifies that for the DAEC, it is not possible to reposition all the required switches at all the remote locations in less than 15 minutes. Therefore, DAEC has allowed 20 minutes to line up the remote shutdown panel.

Issue No. 26

NUMARC IC HG1 contains the following EAL's:

1. *Loss of physical control of the control room due to security event.*

   OR

2. *Loss of physical control of the remote shutdown capability due to security event.*

The corresponding Duane Arnold EAL's are:

1. *Loss of physical control of the control room*

   OR

2. *Loss of physical control of remote shutdown capability*

A. Contrary to the NUMARC guidance, the Duane Arnold EAL's do not include the condition "due to security event." Therefore an event where the control room must be evacuated for reasons other than due to a security event may erroneously be classified under this EAL.

IES Utilities Response

Due to the layout of the DAEC EAL table, the words, "...due to security event" is unnecessary because the associated EAL's are identified within the "Security" event type. This is consistant with current plant practice and consistent with the NUREG - 0654 based EAL tables, currently in use at the DAEC.

NUMARC Recognition Category S
System Malfunction

Issue No. 27

NUMARC IC EAL SU1 contains the following EAL's:

1. *The following conditions exist:*
   - a. *Loss of power to (site-specific) transformers for greater than 15 minutes*

     AND
   - b. *At lease (site-specific) emergency generators are supplying power to emergency busses.*

The corresponding Duane Arnold EAL is:

Loss of offsite power lasting more than 15 minutes

A. Contrary to the NUMARC guidance, the Duane Arnold EAL does not identify site-specific transformers, loss of power to which constitutes "loss of all offsite power".

IES Utilities

"Loss of Offsite Power" is consistent with the DAEC Operation's Department terminology for the conditions of SU1. The NUMARC conditions as outlined in the NUMARC example EAL's would be less clear to the operators at the DAEC.

Issue No. 29

NUMARC IC SU3 includes the following EAL:

1. *The following conditions exist:*

    a. *Loss of most or all (site-specific) annunciators associated with safety systems for greater than 15 minutes.*

    AND

    b. *Compensatory non-alarming indications are available*

    AND

    c. *In the opinion of the Shift Supervisor, the loss of the annunciators or indicators requires increased surveillance to safely operate the unit(s)*

    AND

    d. *Annunciator or indicator loss does not result from planned action.*

The corresponding Duane Arnold EAL is:

Unplanned Loss of annunciators on panels 1C03, 1C04, and 1C05 lasting more than 15 minutes AND compensatory non-alarming indications are available.

A. The Duane Arnold EAL is inconsistent with the NUMARC guidance in that it specifies loss of <u>all</u> annunciators. The Duane Arnold EAL basis document states that the annunciators share a common power supply and therefore it is not necessary to include the condition of "most annunicators." It is not clear that there is no event which could result in a loss of most annunciators and no reason was given for why a loss of most annunciators would not meet the intent of the NUMARC guidance.

IES Utilities Response

DAEC rewrote the EAL as follows:

> Unplanned loss of most annunciators on Panels 1C03, 1C04, and 1C05 lasting more than 15 minutes.
>
> **AND**
>
> Compensatory non-alarming indications are available.

Issue No. 29

NUMARC IC SU4 includes the following EAL's:

1. *(Site-specific) radiation monitor readings indicating fuel clad degradation greater than technical specification limits.*
2. *(Site-specific) coolant sample activity value indicating fuel clad degradation greater than technical specification limits.*

The corresponding Duane Arnold EAL's are:

1. *Valid Pretreat RM-4104 rad monitor reading above 4E+3 mR/hr*
2. *Coolant activity above 1.2 µCi/ml DOSE EQUIVALENT I-131*

A. The Duane Arnold basis document describes the technical specification basis used for developing these EAL's. It is not clear why technical specification 3.6.b.2 was used for the basis for Duane Arnold EAL #1 whereas technical specification 3.6.b.1 was used for the basis for Duane Arnold EAL #2.

IES Utilities Response

The DAEC basis document has veen revised to reflect that both DAEC EAL's are based on Technical Specification 3.6.B.1.a.

Issue No. 30

NUMARC IC SU5 contains the following EAL:

> *1. The following conditions exist:*
>
> *a. Unidentified or pressure boundary leakage greater than 10 gpm*
>
> *OR*
>
> *b. Identified leakage greater than 25 gpm*

The corresponding Duane Arnold EAL is:

> *Unidentified leakage above 10 GPM*
> *OR*
> *Total RCS leakage above 35 GPM*
> *OR*
> *Main steam line break as determined from annunciators or plant personnel report*

A. The Duane Arnold EAL is not consistent with the NUMARC guidance in that it does not specify a value for pressure boundary leakage. No justification was provided for this deviation.

B. The Duane Arnold EAL is not consistent with the NUMARC guidance in that it specifies 35 gpm for the total RCS leakage instead of 25 gpm as is specified in the NUMARC guidance. The NUMARC guidance states that this IC is included as an Unusual Event because it may be a precursor of more serious conditions. The Duane Arnold basis document does not address why a 25 gpm total RCS is not indicative of a potential degradation of the level of safety at Duane Arnold and therefore is not an Unusual Event.

IES Utilities Response

This EAL has been revised to be consistent with the NUMARC guidance. The EAL now reads:

> Unidentified or pressure boundary leakage greater than 10 gpm.
> **OR**
> Identified leakage greater than 25 gpm

Issue No. 31

NUMARC IC SU6 contains the following EAL's:

1. *Loss of all (site-specific list) onsite communication capability affecting the ability to perform routine operations.*

2. *Loss of all (site-specific) offsite communications capability*

The corresponding Duane Arnold EAL's are:

1. *Loss of ALL onsite electronic communication methods*

2. *Loss of ALL electronic communication methods with government agencies*

A. Contrary to the NUMARC guidance, a site-specific list of communication capabilities was not included in these EAL's. The concern with this deviation is that the user of the classification procedure may not be readily able to ascertain whether the EAL's are met or not because of the lack of site-specific information.

IES Utilities Response

Consistent with the NUMARC basis information, the EAL's have been revised to state:

1) Loss of all onsite telephone and radio communication methods (PABX, direct-ring, UHF, and radiological survey radio systems.)

2) Loss of ALL electronic communication methods with government agencies (PABX, direct-ring, ENS, microwave and police radio).

Issue No. 32

NUMARC IC SU7 contains the following EAL:

1. *Either of the following conditions exist:*

   a. *Unplanned Loss of Vital DC power to required DC busses based on (site-specific) bus voltage indications.*

   AND

   b. *Failure to restore power to at least one required DC bus within 15 minutes from the time of loss.*

The corresponding Duane Arnold EAL is:

Complete Loss of 125 VDC lasting more than 15 minutes

A. Contrary to the NUMARC guidance, the Duane Arnold EAL did not specify the applicable vital buses in this EAL and did not specify what voltage level constitutes a loss of DC. The concern with this deviation is that classification may be delayed or an event improperly classified due to the lack of specific information. (This comment also applies to IC SS3)

IES Utilities Response

The DAEC EAL for SU7 has been revised to read "The following conditions exist:

> Unplanned Loss of Div 1 and Div 2 125 VDC busses based on bus voltage less than 105 VDC indicated.
>
> **AND**
>
> Failure to restore power to at least one required 125 VDC bus within 15 minutes from time of loss.

The basis has been revised as follows: There is no significant deviation from the generic EAL. *Unplanned* loss of Div 1 and Div 2 125 VDC busses excludes scheduled maintenance and testing activities. Under the conditions of concern, AOP 302.1, "Loss of 125 VDC Power", would be entered. The DAEC EAL's address the loss of both divisions of the 125 VDC systems consistent with AOP 302.1.

The 125 VDC system is divided into two independent divisions - Division I (1D1) and Division II (1D2) - each supplied by separate AC and DC (battery) power supplies. Loss of both 125 VDC Divisions could compromise the ability to monitor and control the removal of decay heat during cold shutdown or refueling operations. These EAL's are intended to be anticipatory in as much as the operating crew may not have necessary indication and control of equipment needed to respond to the loss. If this loss results in the inability to maintain cold shutdown, the escalation to an Alert will be per SA3 "RCS temperature rise that is not allowed by procedures that will result in RCS temperature above 212 F".

Bus voltage is based on the minimum bus voltage necessary for the operation of safety related equipment and may be indicated by the illumination of annunciators "125 VDC System 1 Trouble" on 1C08A A-9 and/or "125 VDC System 2 Trouble" on 1C08B A-4.

The EAL for SS3 has been rewritten as follows "Unplanned Loss of Div I and Div II 125 VDC Busses Lasting More Than 15 Minutes".

Issue No. 33

NUMARC IC SA3 contains the following EAL:

1. *The following conditions exist:*

    a. *Loss of (site-specific) technical specification required functions to maintain cold shutdown.*

    AND

    b. *Temperature increase that either:*

        - *Exceeds technical specification cold shutdown temperature limit*

        OR

        - *Results in uncontrolled temperature rise approaching cold shutdown technical specification limit.*

The corresponding Duane Arnold EAL is:

> *RCS temperature rise that is not allowed by procedures or Tech Specs that will result in RCS temperature above 212 F.*

A. The Duane Arnold EAL deviates from the NUMARC EAL by including the condition that the temperature rise is "not allowed by procedures or Tech Specs" rather than "the loss of tech spec functions." The concern is that the conditions specified in the Duane Arnold EAL will make classifying events more difficult and that some events classified under the NUMARC EAL scheme may not be classified under the Duane Arnold EAL scheme.

B. The Duane Arnold EAL deviates from the NUMARC guidance by not including the condition of "uncontrolled temperature rise approaching cold shutdown technical specification limit." This may result in delaying classifications. This deviation was not justified in the Duane Arnold EAL basis document.

IES Utilities Response

The revised wording of this EAL is:

Loss of decay heat removal systems required to maintain cold shutdown
AND
Temperature rise that exceeds 212^0 F
OR
Uncontrolled temperature rise approaching 212^0 F

The loss of monitoring and removal of decay heat during shutdown conditions is currently governed by DAEC's procedure AOP 149, "Loss of Decay Heat Removal." The DAEC EAL is written to imply a RCS temperature rise above 212^0 F that is not allowed under plant procedures. This corresponds to the inability to maintain required temperature conditions for Cold Shutdown. "Uncontrolled" means that system temperature increase is not the result of planned actions by the plant staff. Minor cooling interruptions occurring at the transition between Hot Shutdown and Cold Shutdown or temperature changes that are permitted to occur during the establishment of alternate core cooling would not require an unnecessary declaration of an Alert.

Issue No. 34

NUMARC IC SS2 includes the following EAL:

> *(Site-specific) indications exist that automatic and manual scram were not successful*

The corresponding Duane Arnold EAL is:

> *All control rods NOT inserted to at least position 02 AND boron injection with SBLC is required.*

A. The Duane Arnold EAL deviates by including the condition that "boron injection with SBLC is required." This condition may result in delaying classification. If the reactor is producing more heat than the maximum decay heat load for which the safety systems are designed then conditions exist that lead to imminent loss or potential loss of both fuel clad and the RCS and therefore a Site Area Emergency classification is warranted. It is not appropriate to wait until boron injection is procedurally mandated to classify the event.

IES Utilities Response

This EAL has been reworded as follows:

Failure of automatic and manual scram
AND
Power remains above 5%
OR
Boron injection is required.

This change addresses the issue where an automatic and manual scram are not considered successful if actions away from the reactor control console are required to scram the reactor. Under these conditions, the reactor is producing more heat than the maximum decay heat load for which the safety systems are designed. These conditions could lead to imminent loss or potential loss of both fuel clad and RCS.

Issue No. 35

NUMARC IC SS4 states:

Complete loss of Function Needed to Achieve or Maintain Hot Shutdown

NUMARC EAL's associated with this IC include:

1. *Complete loss of any (site-specific) function required for hot shutdown*

The corresponding Duane Arnold EAL is:

Adequate core cooling conditions CANNOT be achieved or maintained
OR
Reactor CANNOT be brought subcritical

A. The Duane Arnold EAL does not include plant specific indication for determining whether adequate core cooling conditions exist. This could make this EAL difficult to use.

IES Utilities Response

This EAL has been reworded as follows:

EOP Graph 4, Heat Capacity Limit is exceeded
OR
Reactor CANNOT be brought subcritical

This EAL addresses complete loss of functions, including ultimate heat sink and reactivity control, required for hot shutdown with the reactor at pressure and temperature. Under these conditions, there is an actual major failure of a system intended for the protection of the public. If the main condenser is unavailable and the Torus is threatened, this would be a plant condition that would correspond to an actual major failure of a system intended for the protection of the public. This condition would also impact "adequate core cooling" conditions. The reactivity condition criteria is addressed by maintenance of required shutdown margin.

Issue No. 36

NUMARC IC SG1 contains the following EAL:

1. *Prolonged loss of all offsite and onsite AC power as indicated by:*

   a. *Loss of power to (site-specific) transformers.*

   AND

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

   AND

   c. *At least one of the following conditions exists:*

      * *Restoration of at least one emergency bus within (site-specific) hours is NOT likely*

      OR

      * *(Site-specific) indication of continuing degradation of core cooling based on Fission Product Barrier monitoring.*

The corresponding Duane Arnold EAL is:

Loss of Voltage on Buses 1A3 and 1A4 and ANY of the following
- *Restoration of power to either Bus 1A3 or 1A4 is NOT likely within 4 hours*
- *RPV level remains indeterminate*
- *RPV level remains below -30 inches*

A. The terms "remains indeterminate" and "remains below -30 inches" are not defined in the Duane Arnold EAL. Using undefined terms such as these may result in confusion when classifying an event. In addition, if a station blackout condition occurred and water level reached the top of active fuel, plant conditions warrant classifying the event as a General Emergency without waiting to determine if the level is going to "remain" less than top of active fuel.

IES Utilities Response

"Loss of Voltage on Buses 1A3 and 1A4" addresses the NUMARC EAL statements of "Loss of Power to (site-specific) transformers AND failure of (site-specific) emergency diesel generators to supply power to emergency busses." "Restoration of power to either Bus 1A3 or 1A4 is NOT likely within 4 hours" is synonymous with "Restoration of at

least one emergency bus within (site-specific) hours is NOT likely". The DAEC removed the word "remains" from the EAL table under SG1. The EAL wording has been revised to read, "RPV level indeterminate, RPV level below 15 inches. This wording is synonymous with "(site-specific)" indication of continuing degradation of core cooling based on Fission Product Barrier monitoring. DAEC has stated the EAL's in terms that are familiar to our operators by extracting those words directly from our EOP's.

The EOP basis document for DAEC identifies that adequate core cooling is assured for the DAEC at a level of 15 inches.

Issue No. 37

NUMARC IC SG2 contains the following EAL's:

1. *(Site-specific) indications exist that automatic and manual scram were not successful*

   AND

2. *Either of the following:*

   a. *(Site-specific) indications exist that the core cooling is extremely challenged*

      OR

   b. *(site-specific) indications exist that heat removal is extremely challenged.*

The corresponding Duane Arnold EAL is:

Entry into ATWS EOP-RPV Control is required and BOTH of the following:

- *Reactor power is expected to remain above 5% or CANNOT be determined*

  AND

- *Main condenser is NOT available*

A. It is not clear that the condition of the main condenser not being available is a sufficient indication of an extreme challenge to heat removal. The NUMARC EAL guidance state that "For BWRs (site-specific) considerations include inability to remove heat via the main condenser, or via the suppression pool or torus (e.g. due to high pool water temperature). The Duane Arnold EAL did not include indications regarding heat removal via the suppression pool.

B. No condition equivalent to the NUMARC condition "(Site-specific) indications exist that the core cooling is extremely challenged" was provided in the Duane Arnold EAL. The NUMARC guidance states, "For BWRs, the extreme challenge of the ability of cool the core is intended to mean that the reactor vessel water level is below 2/3 coverage of active fuel." The Duane Arnold EAL did not include a comparable EAL for this condition."

C. Contrary to the NUMARC guidance the Duane Arnold EAL includes the condition "Reactor power is expected to remain above 5% or CANNOT be determined." Further justification is needed to determine whether the addition of this condition meets the intent of the NUMARC EAL. In addition, the term "is expected to remain above 5%" is not defined in the Duane Arnold EAL.

<u>IES Utilities Response</u>

This EAL has been reworded as follows:

Entry into ATWS EOP-RPV Control is required
AND
RPV level cannot be maintained above -30 inches
OR
EOP Graph 4 Heat Capacity Limit is exceeded.

The NUMARC condition, (Site-specific) indications exist that automatic and manual scram were not successful is addressed by "Entry into ATWS EOP-RPV Control is required."

The NUMARC condition, (Site-specific) indications exist that the core cooling is extremely challenged is addressed by "RPV level cannot be maintained above -30 inches."

The NUMARC condition, (Site-specific) indications exist that heat removal is extremely challenged is addressed by "EOP Graph 4 Heat Capacity Limit is exceeded"

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INTRODUCTION

IES Utilities has revised the Duane Arnold Energy Center (DAEC) Emergency Plan to incorporate guidance from NUMARC/NESP-007, Revision 2 (January 1992), *Methodology for Development of Emergency Action Levels*. The NUMARC (now Nuclear Energy Institute or NEI) guidance was developed to replace Emergency Action Levels (EAL) guidance contained in NUREG-0654/FEMA-REP-1 (Revision 1), *Criteria for Preparation and Evaluation of Radiological Emergency Response Plans and Preparedness in Support of Nuclear Power Plants*, November 1980. The NEI-sponsored methodology was used to develop a set of generic EAL guidelines, together with the basis, so that they could be used and adapted by each utility in a consistent manner. The NRC has endorsed use of the NEI generic guidance as an acceptable alternative method to NUREG-0654 for developing plant-specific EALs in Regulatory Guide 1.101, "Emergency Planning and Preparedness for Nuclear Power Reactors," Revision 3, August 1992. This Regulatory Guide further states that: "Licensees may use either NUREG-0654/FEMA-REP-1 or NUMARC/NESP-007 in developing their EAL scheme but may not use portions of both methodologies."

This EAL basis document was developed to: (1) provide clear documentation of how NEI generic guidance was applied in the development of DAEC upgraded EALs, (2) provide justification of any exceptions or additions to NEI generic guidance as it is applied to DAEC, and (3) facilitate the regulatory approval of the upgraded EALs that is required under 10 CFR 50 Appendix E.

Although there are many similarities, there are some basic differences from the previous EALs based on NUREG-0654 guidance. These include:

1. Events that are explicitly covered under 10 CFR 50.72 such as one-hour or four-hour reports are no longer classified under the Unusual Event emergency classification. Items such as contaminated injured person transported off-site, partial communications losses, meteorological measurement losses, shutdown within the requirements of technical specifications, and inadvertent actuation of ECCS are no longer treated as emergencies because they are explicitly defined in 10 CFR 50.72 as "non-emergency" conditions to report.

2. Precursor conditions are explicitly included in the Unusual Event emergency classification. This includes EALs addressing RCS leakage and loss of off-site power.

3. Conditions such as fire, explosion, gas releases, flooding, low river water level, tornado, or earthquake can be directly escalated only up to the Alert emergency classification. Escalation to Site Area Emergency or General Emergency is based on degraded system response as would be determined by fission product barrier, loss of AC power, or projected effluent release EALs.

4. Core damage sequences are addressed by determining their level of challenge to each of the three primary fission product barriers - fuel clad, reactor coolant system, and the primary containment. The level of challenge is determined in accordance with the Emergency Operating Procedures (EOPs), Integrated Plant Operating Instructions (IPOIs), Abnormal Operating Procedures (AOPs) and core damage assessment methodology. This allows the operations crew to readily recognize the corresponding emergency classification and allows for ready escalation to Site Area Emergency or General Emergency as conditions may worsen.

5. Offsite radiological releases that can be expected to exceed Environmental Protection Agency (EPA) Protective Action Guide (PAG) levels for inhalation doses - 1,000 mrem TEDE or 5,000 mrem CDE Thyroid - will result in declaration of a General Emergency.

DEFINITIONS

AC - Alternating Current

Affecting (in regard to events such as fire, flood, or missiles) - Causing degraded equipment performance as determined by physical observation or by indications in the Control Room or at local control stations.

Alert - Events are in process or have occurred which involve an actual or potential substantial degradation of the level of safety of the plant. Any releases are expected to be limited to small fractions of the EPA Protective Action Guide (PAG) exposure levels.

All - Initiating Condition applies to all Technical Specification operating modes as well as defueled operation.

AOP - Abnormal Operating Procedure

APRM - Average Power Range Monitor

ARM - Area Radiation Monitor

ATWS - Anticipated Transient Without Scram

Barrier - Same as "Fission Product Barrier", below.

Barrier Monitoring Ability - This is a judgment factor in determining whether a fission product barrier is lost or potentially lost. Decreased ability to monitor a barrier results from a loss of/lack of reliable indicators, including instrumentation operability concerns, readings from portable instrumentation, and consideration for offsite monitoring results.

Becquerel - A measurement of radioactive decay rate equal to one disintegration per second.

BOP - Balance of Plant

BWR - Boiling Water Reactor

CAM - Continuous Air Monitor

CDE - Committed Dose Equivalent as defined in 10 CFR 20.1003

CEDE - Committed Effective Dose Equivalent as defined in 10 CFR 20.1003

CFM - Cubic Feet per Minute

CFS - Cubic Feet per Second

Cold condition - As defined in Technical Specification 1.0, this refers to the condition where the reactor coolant temperature is less than or equal to 212°F.

Cold shutdown - As defined in Technical Specification 1.0, the reactor is in the shutdown mode, the reactor coolant temperature is less than or equal to 212°F, and the reactor is vented to atmosphere.

Compensatory non-alarming indications - Information displayed in the main control room including analog and digital parameter displays, trend recorders, the Safety Parameter Display System (SPDS), and the plant process computer.

Control - As applied to remote shutdown capability, this is the ability to manipulate plant parameters without reliance on control room devices or instrumentation using components and methods specified by Abnormal Operating Procedure 915, Shutdown Outside Control Room.

CPS - Counts Per Second

CRD - Control Rod Drive

CSCS - Core Standby Cooling System

CST - Condensate Storage System

Curie (Ci) - A measurement of radioactive decay rate equal to 3.70E+10 disintegration's per second (becquerels).

CW - Circulating Water

DAEC - Duane Arnold Energy Center

DC - Direct Current

DEQ - Dose Equivalent

Dominant accident sequences - These will lead to degradation of all fission product barriers. Dominant accident sequences leading to core damage at DAEC include complete loss of 125 VDC, loss of decay heat removal, ATWS with failure of Standby Liquid Control, prolonged station blackout, and loss of offsite power with early HPCI/RCIC failure.

DW - Drywell

EC - Emergency Coordinator

ECCS - Emergency Core Cooling System

EDE - Effective Dose Equivalent as defined in 10 CFR 20.1003

Emergency Action Level (EAL) - A pre-determined, site-specific, observable threshold for a plant Initiating Condition that places the plant in a given Emergency Class. An EAL can be: an instrument reading, an equipment status indicator, a measurable parameter (on-site or offsite), a discrete observable event, results of analyses, entry into specific emergency operating procedures, or another phenomenon which, if it occurs, indicates entry into a particular Emergency Class.

Emergency Class - Same as "Emergency Classification Level" below.

Emergency Classification Level - These are taken from 10 CFR 50, Appendix E. They are, in escalating order: (Notification of) Unusual Event (UE), Alert, Site Area Emergency (SAE), and General Emergency (GE).

EOP - Emergency Operating Procedure

EPA - Environmental Protection Agency

EPIP - Emergency Plan Implementing Procedure

ESF - Engineered Safety Features

ESS - Engineered Safety Systems

Establish - Make arrangements for a stated condition, e.g., establish communications with control room.

ESW - Emergency Service Water

Fission Product Barrier - One of the three principal barriers to uncontrolled release of radionuclides: Fuel Clad, Reactor Coolant System (RCS), and the Primary Containment.

FP - Fuel Pool

Fuel Clad (Barrier) - The zirconium alloy tubes that contain the fuel pellets.

General Emergency (GE) - Events are in process or have occurred which involve actual or *imminent* substantial core degradation or melting with potential for loss of containment integrity. Releases can reasonably be expected to exceed EPA Protective Action Guide (PAG) exposure levels offsite for more than the immediate site area.

GPM - Gallons Per Minute

GSW - General Service Water

Hot shutdown - As defined in Technical Specification 1.0, the reactor is in shutdown mode and the reactor coolant temperature is greater than 212°F.

Hot standby condition - As defined in Technical Specification 1.0, this refers to operation with the reactor coolant temperature greater than 212°F, reactor pressure vessel less than 1055 psig, and the mode switch in Startup position.

HPCI - High Pressure Coolant Injection (system).

Identified Leakage - Identified Leakage shall be:

a. Leakage into collection systems, such as pump seal or valve packing leaks, that is captured and conducted to a sump or collecting tank, or

b. Leakage into the containment atmosphere from sources that are both specifically located and known not to interfere with the operation of the leakage detection systems.

IDLH - Immediately Dangerous to Life and Health

Inadvertent - Accidental or unintentional, *e.g.*, the event occurred because procedures were not strictly adhered to.

Imminent - No turnaround in safety system performance is expected and escalation to a higher emergency classification level is expected to occur within two hours.

Implement - Commence a required program or series of procedures.

In service - A component or system in the appropriate configuration for normal operation and is considered *operable* as defined in the Technical Specifications.

Indicator - The name for the row on the fission barrier table that is used for convenient grouping of similar symptoms.

Initiate - Take action to begin a process

Initiating Condition (IC) - One of a predetermined subset of nuclear power plant conditions where either the potential exists for a radiological emergency or such an emergency has occurred.

IPE - Individual Plant Examination

IPOI - Integrated Plant Operating Instruction

IRM - Intermediate Range Monitor

Isolate - Remove from service by closing off the flow path

kV - Kilovolt(s)

LCO - Limiting Condition for Operation

LLRPSF - Low Level Radwaste Processing and Storage Facility

LOCA - Loss of Coolant Accident

LOOP - Loss of Offsite Power

Loss (of a fission product barrier) - A severe challenge to a fission product barrier exists such that the barrier is considered incapable of performing its safety function.

LPCI - Low Pressure Coolant Injection

MCC - Motor Control Center

MCUTL - Maximum Core Uncovery Time Limit

Microcurie (μCi) - One millionth of a curie, *i.e.*, 3.7E+4 disintegration's per second (becquerels).

MIDAS - Meteorological Information and Dose Assessment System, primary method for detecting and quantifying gaseous releases at the DAEC.

Millicurie (mCi) - One thousandth of a curie, *i.e.*, 3.7E+7 disintegration's per second (becquerels).

Millirem (mrem) - One thousandth of a rem

MPH - Miles Per Hour

mR - milliroentgen, *i.e.*, one thousandth of a roentgen (R)

MSIV - Main Steam Isolation Valve

MSL - Main Steam Line

NEI - Nuclear Energy Institute (formerly NUMARC)

Notification of Unusual Event (NOUE) - Same as "Unusual Event", below.

NPSH - Net Positive Suction Head

NUMARC - Nuclear Utility Management and Resources Council (now NEI)

OBE - Operating Basis Earthquake

ODAM - Offsite Dose Assessment Manual

Operating Mode - As defined by Technical Specification Table 1.0-1, Operating Mode describes the operating status of the unit. Mode designations (and the associated Reactor Mode Switch Positions) used at DAEC are: RUN/POWER OPERATION (Run), STARTUP (Startup or Refuel), HOT SHUTDOWN (Shutdown), COLD SHUTDOWN (Shutdown), and REFUELING (Shutdown or Refuel).

Operable - A system is considered capable of performing its function in accordance with the applicable Technical Specification requirements. Implicit in this definition is the assumption that all auxiliary equipment required for the system is also operable.

OSS - Operations Shift Supervisor

PAG - Protective Action Guide

Planned - Loss of a component or system due to expected events such as scheduled maintenance and testing activities.

Potential Loss (of a fission product barrier) - A challenge to a fission product barrier exists such that the barrier is considered degraded in its ability to perform its safety function.

Primary Containment (Barrier) - The drywell, the torus, their respective interconnecting paths, and other connections up to and including the outermost containment isolation valves.

PSIG - Pounds per Square Inch Gauge

RB - Reactor Building

RBCCW - Reactor Building Closed Cooling Water (system)

RCIC - Reactor Core Isolation Cooling (system)

RCS - Reactor Coolant System

RCS Barrier - The reactor coolant system pressure boundary including the reactor pressure vessel and all reactor coolant system piping up to and including the outermost isolation valves.

Recognition Category - A logical grouping of Initiating Conditions, *e.g.*, System Malfunctions.

Rem - Unit of radiation dose as defined in 10 CFR 20.1004

Required - Action taken (such as entry into emergency operating procedure) is neither optional nor merely suggested; rather, it is imperative based on existing conditions.

RHR - Residual Heat Removal (system)

RHRSW - Residual Heat Removal Service Water (system)

Roentgen (R) - Unit of ionizing radiation energy absorbed in a cubic centimeter of air

RPV - Reactor Pressure Vessel

RWCU - Reactor Water Clean-Up (system)

SBDG - Standby Diesel Generator

SBGT - Standby Gas Treatment (system)

SBLC - Standby Liquid Control (system)

SBO - Station Blackout

S/D - Shutdown

SDC - Shutdown Cooling

SDV - Scram Discharge Volume

Shutdown - As defined in Technical Specification 1.0, the reactor is in a shutdown condition when the reactor mode switch is in the Shutdown position.

Significant transient - (See also, "Transient", below.) Includes response to automatic or manually initiated functions such as scrams, runbacks involving greater than 25% thermal power change, ECCS injections, or thermal power oscillations of 10% or greater.

Site Area Emergency (SAE) - Events are in process or have occurred which involve actual or likely major failures of plant functions needed for protection of the public. Any releases are not expected to result in

exposure levels which exceed EPA Protective Action Guide (PAG) exposure levels except near the site boundary.

SPDS - Safety Parameter Display System

SRM - Startup Range Monitor

SRO - Senior Reactor Operator

SRV - Safety-Relief Valve

Sustained wind speed - Baseline wind speed measured by meteorological tower that does not include gusts

TAF - Top of Active Fuel (344.5 inches above bottom of RPV)

TEDE - Total Effective Dose Equivalent as defined in 10 CFR 20.1003

Total Leakage - Total leakage is the sum of Identified Leakage and Unidentified Leakage.

Transient - A condition that: (1) is beyond the expected steady-state fluctuations in temperature, pressure, power level, or water level, and (2) is beyond the normal manipulations of the Control Room operating crew, and (3) is expected to require actuation of fast-acting automatic control or protection systems to bring the reactor to a new safe, steady-state condition.

TSC - Technical Support Center

Uncontrolled - Condition is not the result of planned actions by the plant staff in accordance with procedures.

Unisolable - Actions taken from the Main Control Board or locally are not successful in eliminating the leakage path.

Unidentified Leakage - Unidentified Leakage shall be all leakage which is not identified leakage.

Unplanned - Used to preclude the declaration of an emergency where a component or system has been removed intentionally from service (e.g., for maintenance and/or testing activities). As used in the context of radioactive releases, "unplanned" includes any release for which a radioactive 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.

Unusual Event (UE) - Events are in process or have occurred which indicate a potential degradation of the level of safety of the plant. No releases of radioactive material requiring offsite response or monitoring are expected unless further degradation of safety systems occurs.

VAC - Volt(s) Alternating Current

VDC - Volt(s) Direct Current

Valid - Indication is from instrumentation determined to be operable in accordance with the Technical Specifications or has been verified by other independent methods such as indications displayed on the control panels, reports from plant personnel, or radiological survey results.

WEC - Water Effluent Concentration

ORGANIZATION OF BASIS INFORMATION

The format of the EAL Basis information was developed to address training needs, to facilitate NRC approval, and to facilitate future revisions and 10 CFR 50.54(q) evaluations. Each EAL Basis is organized in the following manner:

1. Emergency Action Level (EAL) Basis Information Organized by Initiating Condition (IC)

Initiating Condition Identifier

For consistency, DAEC has chosen to make its Initiating Condition (IC) identifiers identical to those used in NEI document NUMARC/NESP-007. The EAL Technical Basis information is organized by generic IC identifier number and name. NUMARC/NESP-007 organized the generic information into four Recognition Categories. These are:

A - Abnormal Rad Levels/Radiological Effluent
F - Fission Product Barrier Degradation
H - Hazards and Other Conditions Affecting Plant Safety
S - System Malfunctions

For the A, H, and S recognition categories, all EAL basis information is organized by IC identifier in escalating emergency class order from Unusual Event through General Emergency. For the F recognition category, the initiating conditions are the combinations of fission product barrier losses and potential losses that correspond to each emergency classification level. The individual indicators used on the fission barrier table are separately discussed below. The generic IC identifiers use two letters followed by one number. The first letter corresponds to the event category as shown above. The second letter corresponds to the emergency classification level for the IC:

U - (Notification of) Unusual Event
A - Alert
S - Site Area Emergency
G - General Emergency

The number designates whether the IC is the first, second, third, etc., IC for that recognition category under that emergency classification. For example, SU2 is the designator for the second System Malfunction recognition category IC in the Unusual Event classification, etc. Generic information is quoted directly from NUMARC/NESP-007, Revision 2, dated January, 1992. **Changes from the NUMARC/NESP-007**

text are denoted by caret marks (< >). Such changes are based on correction of typographical errors such as those mentioned in the NUMARC Questions and Answers dated June 1993, reflect changes made in 10 CFR Part 20, or to put the information in proper context.

Event Type

This is the label of the applicable row for the EAL Table shown in EPIP-1.1, *Determination of Emergency Action Levels*. The event type lists the general area of concern and includes Offsite Rad Conditions, Onsite Rad Conditions, Natural Disasters, Fire, Other Hazards and Failures, Security, Control Room Evacuation, EC/OSS Judgment, Loss of Power, RPS Failure, Inability to Maintain Shutdown Conditions, Instrumentation/Communication, Coolant Activity, and Coolant Leak. This structure was chosen to be consistent with the previous EAL presentation which is already familiar to the Emergency Coordinators and Operations Shift Supervisors. It is also permissible to organize the generic information in this manner based on the response to Question 5 contained in the *NUMARC Methodology for Development of Emergency Action Levels NUMARC/NESP-007 Revision 2 Questions and Answers June 1993.*

Applicable Operating Modes

The applicable operating modes for each Initiating Condition/Emergency Action Level is then listed based on NUMARC/NESP-007 mode descriptions. The DAEC EALs use the operating modes defined in Technical Specifications Table 1.0-1. These are:

1 - Run/Power Operation
2 - Startup
3 - Hot Shutdown
4 - Cold Shutdown
5 - Refueling

To conserve space, the EAL displays use "Run" to mean "Run/Power Operation" and "S/D" as an abbreviation for "Shutdown."

Operating mode applicability of EALs is based on the operating mode that the plant was in immediately before the event sequence leading to entry into the emergency classification. For example, events/conditions addressed by EALs applicable to Run mode are expected to lead to reactor trip which should bring the plant to Hot Shutdown. However, the appropriate emergency classification would still be based on the applicable EALs for Run/Power Operation for these events/conditions. *If "ALL" operating modes are specified for the EAL, then the EAL applies to all modes identified above plus defueled conditions.*

Generic Example EAL(s)

The generic example EALs are then listed. When more than one is provided, logic phrasing is used to describe whether all EALs are suggested or whether at least one EAL should be chosen.

DAEC EAL Information

This contains the plant-specific information used to implement the generic EALs. This section will also include the basis, as appropriate, for deviation from generic EALs. For example, DAEC does not have a Independent Spent Fuel Storage Installation for on-site dry storage of spent fuel. Thus, DAEC does not have EALs corresponding to the generic guidance for this item. As appropriate, description of any supporting calculations, their underlying bases and assumptions, and their results are included in this section.

References

The references used to develop the DAEC EAL Information are listed here, as appropriate.

2. **Fission Product Barrier Table Indicators**

The basis information for the fission barrier table indicators is organized similarly to the other basis information described above. For each barrier - fuel clad, RCS, and primary containment - basis information is organized by "Indicator." The indicator is the name for the row on the fission barrier table and is used for convenient grouping of similar symptoms, similar to the "Event Type" used for the A, H, and S EALs described above. Indicators include Radiation/Core Damage, RPV Level, Leakage, Primary Containment Atmosphere, and EC/OSS Judgment.

After the DAEC Indicator, the applicable generic BWR fission product barrier indicators are then displayed, showing both the generic loss and potential loss conditions, as applicable. Next displayed is the appropriate DAEC information and references. These are displayed in the same manner as the A, H, and S recognition category basis information described above.

EMERGENCY PLAN IMPLEMENTING PROCEDURE	No. EPIP - 1.1 Rev. 2 PAGE 1 of 1 *(For NRC Review)*
ABNORMAL RAD LEVELS/RADIOACTIVE EFFLUENT	EFFECTIVE DATE: TBD

EVENT TYPE	UNUSUAL EVENT	ALERT	SITE AREA EMERGENCY	GENERAL EMERGENCY
OFFSITE RAD CONDITIONS	***AU1*** **Any Unplanned Release of Gaseous or Liquid Radioactivity to the Environment That Exceeds Two Times the Radiological Technical Specifications For 60 Minutes or Longer** Valid Reactor Building or Turbine Building ventilation (Kaman) rad monitor reading above 1 E-3 µCi/cc for more than 60 minutes. **OR** Valid Offgas Stack (Kaman) rad monitor reading above 6 E-1 µCi/cc for more than 60 minutes. **OR** Valid LLRPSF (Kaman) rad monitor reading above 9 E-4 µCi/cc for more than 60 minutes. **OR** Valid GSW rad monitor reading above 3E+3 CPS for more than 60 minutes. **OR** Valid RHRSW & ESW rad monitor reading above 8E+2 CPS for more than 60 minutes. **OR** Valid RHRSW & ESW Discharge Canal rad monitor reading above 1E+3 CPS for more than 60 minutes. **OR** Confirmed sample analyses for gaseous or liquid releases indicates concentrations in excess of 2 times ODAM limits for greater than 60 minutes. **OR** Dose assessment determines hourly dose outside the site boundary above 0.1 mrem TEDE. **Op. Modes: ALL**	***AA1*** **Any Unplanned Release of Gaseous or Liquid Radioactivity to the Environment That Exceeds 200 Times the Radiological Technical Specifications For 15 Minutes or Longer** Valid Reactor Building or Turbine Building ventilation (Kaman) rad monitor reading above 3 E-2 µCi/cc for more than 15 minutes. **OR** Valid Offgas Sta[illegible]aman) rad monitor reading above 2 E[illegible] µCi/cc for more than 15 minutes. **OR** Valid LLRPSF (Kaman) rad monitor reading above 9 E-2 µCi/cc for more than 15 minutes. **OR** Valid GSW rad monitor reading above 3E+5 CPS for more than 15 minutes. **OR** Valid RHRSW & ESW rad monitor reading above 8E+4 CPS for more than 15 minutes. **OR** Valid RHRSW & ESW Discharge Canal rad monitor reading above 1E+5 CPS for more than 15 minutes. **OR** Confirmed sample analyses for gaseous or liquid releases indicates concentrations in excess of 200 times ODAM limits for greater than 15 minutes. **OR** Valid field survey reading outside the site boundary >10 mR/hr or >50 mR/hr CDE Thyroid. **OR** Dose assessment determines hourly dose outside the site boundary above 10 mrem TEDE. **Op. Modes: ALL**	***AS1*** **Site Boundary Dose Resulting from an Actual or Imminent Release of Gaseous Radioactivity Exceeds 100 mrem TEDE or 500 mrem CDE Thyroid for the Actual or Projected Duration of the Release** Valid Reactor Building or Turbine Building ventilation (Kaman) rad monitor reading above 6 E-2 µCi/cc for more than 15 minutes. (Dose assessment not available) **OR** Valid Offgas Stack (Kaman) rad monitor reading above 4 E+1 µCi/cc for more than 15 minutes. (Dose assessment not available) **OR** Valid field survey reading outside the site boundary >100 mR/hr or >500 mR/hr CDE Thyroid. **OR** Dose assessment determines integrated accident dose projection outside the site boundary above 100 mrem TEDE or above 500 mrem CDE Thyroid. **Op. Modes: ALL**	***AG1*** **Site Boundary Dose Resulting from an Actual or Imminent Release of Gaseous Radioactivity Exceeds 1000 mrem TEDE or 5000 mrem CDE Thyroid for the Actual or Projected Duration of the Release** Valid Reactor Building or Turbine Building ventilation (Kaman) rad monitor reading above 6 E-1 µCi/cc for more than 15 minutes. (Dose assessment not available) **OR** Valid Offgas Stack (Kaman) rad monitor reading above 4 E+2 µCi/cc for more than 15 minutes. (Dose assessment not available) **OR** Valid field survey reading outside the site boundary >1,000 mR/hr or >5,000mR/hr CDE Thyroid. **OR** Dose assessment determines integrated accident dose projection outside the site boundary above 1,000 mrem TEDE or above 5,000 mrem CDE Thyroid. **Op. Modes: ALL**
	AU2 **Unexpected Increase in Plant Radiation** Uncontrolled loss of reactor cavity or fuel pool water level with all spent fuel assemblies remaining water covered as indicated by ANY of the following:	***AA2*** **Major Damage to Irradiated Fuel or Loss of Water Level that Has or Will Result in the Uncovering of Irradiated Fuel Outside the Reactor Vessel** Report of ANY of the following: • Valid ARM HI RAD alarm for the Refueling Floor North End, Refueling Floor South End, New Fuel Storage		

ONSITE RAD CONDITIONS	- Report to control room - Valid fuel pool level indication (LI-3413) below 36 feet and lowering - Valid WR GEMAC Floodup indication (LI-4541) coming on scale. **OR** Unexpected ARM reading offscale high or above 1000 times normal reading. **Op. Modes: ALL**	Area, or Spent Fuel Storage Area - Valid Refueling Floor North End, Refueling Floor South End, or New Fuel Storage Area ARM Reading above 10 mR/hr - Valid Spent Fuel Storage Area ARM Reading above 100 mR/hr **OR** Report of visual observation of Irradiated Fuel uncovered **OR** Water level reading below 450" as indicated on LI4541 (floodup) for the Reactor Refueling Cavity that will result in Irradiated Fuel uncovering **OR** Valid Fuel Pool water level indication (LI-3413) below 16 feet. **Op. Modes: ALL** *AA3* **Release of Radioactive Material or Increases in Radiation Levels Within the Facility That Impedes Operation of Systems Required to Maintain Safe Operations or to Establish or to Maintain Cold Shutdown** Vaild area radiation monitor (RE9162) reading greater than 15 mR/hr in the Control Room. **OR** Vaild area radiation monitor (RE9168) reading greater than 500 mR/hr at the Remote Shutdown Panel, 1C388. **Op. Modes: ALL**		

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ABNORMAL RAD LEVELS/RADIOLOGICAL EFFLUENT CATEGORY

AU1 Any Unplanned Release of Gaseous or Liquid Radioactivity to the Environment That Exceeds Two Times the Radiological Technical Specifications For 60 Minutes or Longer

EVENT TYPE: Offsite Rad Conditions

OPERATING MODE APPLICABILITY: All

EXAMPLE EMERGENCY ACTION LEVELS: (1 or 2 or 3 or 4)

1. A valid reading on <site-specific> monitors that < > indicates that the release may have exceeded <2 x site-specific technical specifications for 60 minutes or longer.>
2. Confirmed sample analyses for gaseous or liquid releases indicates concentrations or release rates with a release duration of 60 minutes or longer in excess of two times (site-specific technical specifications).
3. Valid reading on perimeter radiation monitoring system greater than 0.10 mR/hr above normal background for 60 minutes [for sites having telemetered perimeter monitors].
4. Valid indication on automatic real-time dose assessment capability greater than (site-specific value) for 60 minutes or longer [for sites having such capability].

DAEC EAL INFORMATION:

Valid means that the reading is from instrumentation determined to be operable in accordance with the Technical Specifications or has been verified by other independent methods such as indications displayed on the control panels, reports from plant personnel, or radiological survey results.

The primary method for declaration is by means of dose assessment using the MIDAS computer model. This is listed as DAEC EAL 4. However, if the monitor readings are sustained for longer than 60 minutes and the required dose assessments cannot be completed within this period, then the declaration must be made based on the valid reading.

The approach taken for calculation of gaseous radioactive effluent EAL setpoints includes use of the ODAM Table 3-2 source term computed by BWR-GALE for the DAEC Base Case. The release is assumed to be from a single release point. Multiple release points would be difficult to present as explicit EAL threshold values and in any case, are addressed by off-site dose assessment by MIDAS, which is the preferred method for determining this condition. The calculation methods for setpoint determination are from ODAM Section 3.4 and are based on Regulatory Guide 1.109 methodology. The table below lists the

results of the gaseous effluent EAL calculations. The Kaman extended range capability is used because the General Electric Offgas Stack monitor has a limited range.

	Gaseous Effluent EALs			
	Offgas Stack Kaman 9/10		Turbine Bldg (Kaman 1/2) and Reactor Bldg (Kaman 3/4, 5/6, 7/8)	
Maximum flow (CFM)	10,000		72,000	
Release Limits	Concentration (μCi/cc)	Release Rate (μCi/sec)	Concentration (μCi/cc)	Release Rate (μCi/sec)
Tech Spec	3.2E-1	1.5E+6	6.2E-4	2.1E+4
Unusual Event (2 x TS)	6.4E-1	3.0E+6	1.2E-3	4.2E+4
Alert (60 x TS)	1.9E+1	8.9E+7	3.7E-2	1.3E+6
	LLRPSF Kaman 12			
Maximum flow (CFM)	99,000			
Release Limits	Concentration (μCi/cc)		Release Rate (μCi/sec)	
Tech Spec	4.5E-4		2.1E+5	
Unusual Event (2 x TS)	9.0E-4		4.2E+5	
Alert (200 x TS)	9.0E-2		4.2E+7	

The off-gas stack is treated as an elevated release and the turbine building and reactor building vents are treated as mixed-mode releases. The ground level setpoints are taken from the default setpoint calculations from the quarterly surveillance tests performed by DAEC Chemistry technicians. Reactor Building, Turbine Building, LLRPSF (Low Level Radwaste Processing and Storage Facility) and Offgas Stack Noble Gas Monitor alarm setpoints are calculated based on achieving the Tech Spec instantaneous release limit, assuming annual average meteorology as defined in the ODAM. The Tech Spec Limit currently corresponds to a reactor building or turbine building ventilation alarm setpoint of 6.2 E-04 μCi/cc. The monitor alarm setpoint can be periodically adjusted but typically does not vary by much. The DAEC EAL therefore addresses valid radiation levels exceeding 2 times the alarm setpoint for greater than 60 minutes. Rounded off, this corresponds to 1 E-3 μCi/cc. The corresponding offgas stack monitor value is 0.64 μCi/cc, rounded off to 6 E-1 μCi/cc. The Tech Spec Limit currently for the LLRPSF building ventilation alarm setpoint is 4.5 E-04 μCi/cc. The DAEC EAL therefore addresses valid radiation levels exceeding 2 times the alarm setpoint for greater than 60 minutes. This corresponds to 9 E-4 μCi/cc.

Technical specification setpoints for radioactive liquid radiation monitors are 10 times the 10 CFR 20 Appendix B, Table 2, Water Effluent Concentration (WEC) limits. It is the policy of DAEC to process all

liquid radwaste so that no release of radioactive liquid to the environment is allowed. The radwaste effluent line which could be used as a batch release mechanism has a trip function that prevents exceeding the DAEC release limit, however, an EAL has been provided. The other pathways to the environment (RHRSW - to cooling tower, RHRSW - to discharge canal) have radiation monitors with readouts going to the Control Room. These systems could become contaminated if heat exchanger leaks develop; however, historically this has not occurred in the service water systems at DAEC. These monitors are displayed on panels 1C02 and 1C10.

Reactor water is the likely source of contamination through the service water systems as opposed to floor drain, detergent drain, and chemical waste discharge. The floor drain and detergent drains go to Radwaste Processing and would be batch released to the Radwaste effluent discharge line (if such a release were to occur). The chemical discharge sump is normally a radioactivity clean system and is tested by Chemistry to ensure no contamination prior to discharging to the canal.

The setpoints for the three service water radiation effluent monitors vary because of differences in detector efficiencies and background. Setpoints based on the same reactor water sample are listed below to show the differences. The rounded off readings will be used for the EALs for ease of reading the monitor scales.

Monitor	TS Limit	Reading	UE Level	Alert Level
GSW	1,555 CPS	1.5E+3 CPS	3E+3 CPS	3E+5 CPS
RHRSW & ESW to cooling tower	413 CPS	4E+2 CPS	8E+2 CPS	8E+4 CPS
RHRSW & ESW to Discharge Canal	507 CPS	5E+2 CPS	1E+3 CPS	1E+5 CPS

There are no significant deviations from the generic EALs. However, DAEC does not have a telemetered radiation monitoring system. As an alternative, use of field instruments was considered. It is not practical to establish an EAL based on field survey readings of 0.1 mR/hr for greater than 60 minutes because field instruments in use for emergency response do not have a threshold of detection to meet such criteria. Thus, DAEC does not have an EAL corresponding to generic EAL 3.

Hourly Whole Body Dose Corresponding to 2 x ODAM Limit for Gaseous Release

ODAM limit = 500 mrem/year Whole Body Dose

2 x **ODAM** limit = [2 x 500 mrem/year]/8760 hours/year = 0.114 mrem Whole Body in one hour

Rounded off to 0.1 mrem

Dose assessment using MIDAS is based on the EPA-400 methodology, *e.g.*, use of Total Effective Dose Equivalent (TEDE). This is somewhat different from whole body dose from gaseous effluents determined by ODAM methodology which forms the basis for the radiation monitor readings calculated in accordance with the generic methodology. The gaseous effluent radiation monitors can only detect noble gases. The contribution of iodine's to TEDE could therefore only be determined either by: (1) utilizing MIDAS, or (2) gaseous effluent sampling. DAEC EAL 4 is written in terms of TEDE and the gaseous effluent radiation monitor readings are determined based on ODAM.

REFERENCES:

1. Offsite Dose Assessment Manual Section 6.1.2 and 7.1.2 Bases
2. Emergency Plan Implementing Procedure (EPIP) 3.3, Dose Assessment and Protective Action
3. Radiation Protection Calculation No. 95-001-C, Emergency Actions Levels Based on Effluent Radiation Monitors, January 24, 1995
4. UFSAR Section 11.5, Process and Effluent Radiation Monitoring and Sampling Systems
5. EPA 400-R-92-001, *Manual of Protective Action Guides and Protective Actions for Nuclear Incidents*
6. *NUMARC Methodology for Development of Emergency Action Levels NUMARC/NESP-007 Revision 2 Questions and Answers*, June 1993

AU2 Unexpected Increase in Plant Radiation< >

EVENT TYPE: Onsite Rad Conditions

OPERATING MODE APPLICABILITY. All

EXAMPLE EMERGENCY ACTION LEVELS: (1 or 2 or 3 or 4)

1. (Site-specific) indication of uncontrolled water level decrease in the reactor refueling cavity with all irradiated fuel assemblies remaining covered by water.
2. Uncontrolled water level decrease in the spent fuel pool < > with all irradiated fuel assemblies remaining covered by water.
3. (Site-specific) radiation reading for irradiated spent fuel in dry storage.
4. Valid Direct Area Radiation Monitor readings increases by a factor of 1000 over normal* levels.

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

DAEC EAL INFORMATION:

There are no significant deviations from the generic EALs. DAEC does not have a spent fuel transfer canal or on-site dry storage of spent fuel.

Uncontrolled means that the condition is not the result of planned actions by the plant staff in accordance with procedures. *Valid* means that the reading is from instrumentation determined to be operable in accordance with the Technical Specifications or has been verified by other independent methods such as indications displayed on the control panels, reports from plant personnel, or radiological survey results.

There are three methods to determine water level decreases of concern. The first method is by report to the control room. The other methods include use of the Floodup level indicator and the spent fuel pool level indicator. These are further described below.

During preparation for reactor cavity flood up prior to entry into refuel mode, reactor vessel level instrument LI-4541 (WR GEMAC, FLOODUP) on control room panel 1C04 is placed in service by I&C personnel connecting a compensating air signal after the reference leg is disconnected from the reactor head. Normal refuel water level is above the top of the span of this flood up level indicator. A valid indication (*e.g.*, not due to loss of compensating air signal or other instrument channel failure) of reactor

cavity level coming on span for this instrument is used at DAEC as an indicator of uncontrolled reactor cavity level decrease.

DAEC Technical Specifications require a minimum of 36 feet of water in the spent fuel pool. During refueling, the gates between the reactor cavity and the refueling cavity are removed and the spent fuel pool level indicator LI 3413 is used to monitor refueling water level. Procedures require that a normal refueling water level be maintained at 37 feet 5 inches. A low level alarm actuates when spent fuel pool level drops below 37 feet 1 inch. Symptoms of inventory loss at DAEC include visual observation of decreasing water levels in reactor cavity or spent fuel storage pool, Reactor Building (RB) fuel storage pool radiation monitor or refueling area radiation monitor alarms, observation of a decreasing trend on the spent fuel pool water level recorder, and actuation of the spent fuel pool low water level alarm. To eliminate minor level perturbations from concern, DAEC uses LI 3413 indicated water level below 36 feet and lowering.

Increased radiation levels can be detected by the local refueling floor area radiation monitors, the refueling floor Continuous Air Monitor (CAM) alarm, refueling areas radiation monitors, fuel pool ventilation exhaust monitors, and by Standby Gas Treatment (SGBT) System automatic start. Applicable area radiation monitors include those that are displayed on Panel 1C02 and alarmed on Panel 1C04B. The DAEC EAL has also been written to reflect the case where an ARM may go offscale high prior to reaching 1,000 times the normal reading.

NOTE: On Annunciator Panel 1C04B, the indicators listed below are expected alarms during pre-planned transfers of highly radioactive material through the affected area. If an HP Technician is present, sending an Operator is not required. Radiation levels other than those expected should be promptly investigated. The indicators are high radiation alarms from the Hot Laboratory or Administrative Building, the new fuel storage area, and the radwaste building.

REFERENCES:

1. Alarm Response Procedure (ARP) 1C04B, Reactor Water Cleanup and Isolation
2. Technical Specification 3.9C, Spent Fuel Pool Water Level
3. Emergency Plan Implementing Procedure (EPIP) 3.1, Inplant Radiological Monitoring, Attachment 1, ARM Locations
4. Emergency Operating Procedures (EOP) Basis Document, Breakpoints for RC/L & L
5. Surveillance Test Procedure (STP) 42A-0001, Daily and Shift Instrument Checks
6. Integrated Plant Operating Instruction (IPOI) 8 , Outage and Refueling Operations
7. Fuel & Reactor Component Handling Procedure (F&RCHP) 5, Procedure for Moving Core Components Between Reactor Core and Spent Fuel Pool, Within the Reactor Core, or Within the Spent Fuel Pool
8. *NUMARC Metnodology for Development of Emergency Action Levels NUMARC/NESP-007 Revision 2 Questions and Answers*, June 1993

AA1 Any Unplanned Release of Gaseous or Liquid Radioactivity to the Environment that Exceeds 200 Times Radiological Technical Specifications for 15 Minutes or Longer

EVENT TYPE: Offsite Rad Conditions

OPERATING MODE APPLICABILITY: All

EXAMPLE EMERGENCY ACTION LEVELS: (1 or 2 or 3 or 4)

1. A valid reading on <site-specific> monitors that < > indicates that the release may have exceeded <200 x site-specific technical specifications for 15 minutes or longer.>
2. Confirmed sample analyses for gaseous or liquid releases indicates concentrations or release rates in excess of (200 x site-specific technical specifications) for 15 minutes or longer.
3. A valid reading on perimeter radiation monitoring system greater than 10.0 mR/hr sustained for 15 minutes or longer. [for sites having telemetered perimeter monitors]
4. Valid indication on automatic real-time dose assessment capability greater than (200 x site-specific Technical Specifications value) for 15 minutes or longer. [for sites having such capability]

DAEC EAL INFORMATION:

Valid means that the reading is from instrumentation determined to be operable in accordance with the Technical Specifications or has been verified by other independent methods such as indications displayed on the control panels, reports from plant personnel, or radiological survey results.

The primary method for declaration is by means of dose assessment using the MIDAS computer model. This is listed as DAEC EAL 4. However, if the monitor readings are sustained for longer than 15 minutes and the required dose assessments cannot be completed within this period, then the declaration must be made based on the valid reading.

Gaseous Effluent EALs

	Offgas Stack Kaman 9/10		Turbine Bldg (Kaman 1/2) and Reactor Bldg (Kaman 3/4, 5/6, 7/8)	
Maximum flow (CFM)	10,000		72,000	
Release Limits	Concentration (µCi/cc)	Release Rate (µCi/sec)	Concentration (µCi/cc)	Release Rate (µCi/sec)
Tech Spec	3.2E-1	1.5E+6	6.2E-4	2.1E+4
Unusual Event (2 x TS)	6.4E-1	3.0E+6	1.2E-3	4.2E+4
Alert (60 x TS)	1.9E+1	8.9E+7	3.7E-2	1.3E+6
	LLRPSF Kaman 12			
Maximum flow (CFM)	99,000			
Release Limits	Concentration (µCi/cc)		Release Rate (µCi/sec)	
Tech Spec	4.5E-4		2.1E+5	
Unusual Event (2 x TS)	9.0E-4		4.2E+5	
Alert (200 x TS)	9.0E-2		4.2E+7	

The off-gas stack is treated as an elevated release and the turbine building and reactor building vents are treated as mixed-mode releases. The ground level setpoints are taken from the default setpoint calculations from the quarterly surveillance tests performed by DAEC Chemistry technicians. Reactor Building, Turbine Building, LLRPSF (Low Level Radwaste Processing and Storage Facility) and Offgas Stack Noble Gas Monitor alarm setpoints are calculated based on achieving the Tech Spec instantaneous release limit assuming annual average meteorology as defined in the ODAM. The Tech Spec Limit currently corresponds to a reactor building or turbine building ventilation alarm setpoint of 6.2 E-4 µCi/cc. The monitor alarm setpoint can be periodically adjusted but typically does not vary by much. For the Offgas Stack, Reactor Building and Turbine building KAMAN monitor readings, DAEC chose to multiply the technical specification concentration by a factor of 60 (instead of 200) in order to allow for a logical step progression in monitor setpoints from the AU1 through AA1 to AS1. The DAEC EAL therefore addresses valid radiation levels exceeding 60 times the alarm setpoint for greater than 15 minutes. Rounded off, this corresponds to 3 E-2 µCi/cc. The corresponding offgas stack monitor value is 19.2 µCi/cc, rounded off to 2 E+1 µCi/cc. The Tech Spec Limit currently for the LLRPSF building ventilation alarm setpoint is 4.5 E-04 µCi/cc. The DAEC EAL therefore addresses valid radiation levels exceeding 200 times the alarm setpoint for greater than 15 minutes. This corresponds to 9 E-2 µCi/cc.

Technical specification setpoints for radioactive liquid radiation monitors are 10 times the 10 CFR 20 Appendix B, Table 2, Water Effluent Concentration (WEC) limits. It is the policy of DAEC to process all

liquid radwaste so that no release of radioactive liquid to the environment is allowed. The radwaste effluent line which could be used as a batch release mechanism has a trip function that prevents exceeding the DAEC release limit, and therefore no EAL limits are provided. The other pathways to the environment (RHRSW - to cooling tower, RHRSW - to discharge canal) have radiation monitors with readouts going to the Control Room. These systems could become contaminated if heat exchanger leaks develop; however, historically this has not occurred in the service water systems at DAEC. These monitors are displayed on panels 1C02 and 1C10.

Reactor water is the likely source of contamination through the service water systems as opposed to floor drain, detergent drain, and chemical waste discharge. The floor drain and detergent drains go to Radwaste Processing and would be batch released to the Radwaste effluent discharge line (if such a release were to occur). The chemical discharge sump is normally a radioactivity clean system and is tested by Chemistry to ensure no contamination prior to discharging to the canal.

The setpoints for the three service water radiation effluent monitors vary because of differences in detector efficiencies and background. Setpoints based on the same reactor water sample are listed below to show the differences. The rounded off readings will be used for the EALs for ease of reading the monitor scales.

Monitor	TS Limit	Reading	UE Level	Alert Level
GSW	1,555 CPS	1.5E+3 CPS	3E+3 CPS	3E+5 CPS
RHRSW & ESW to cooling tower	413 CPS	4E+2 CPS	8E+2 CPS	8E+4 CPS
RHRSW & ESW to Discharge Canal	507 CPS	5E+2 CPS	1E+3 CPS	1E+5 CPS

DAEC does not have a telemetered radiation monitoring system. As an alternative, DAEC uses valid field survey readings outside the site boundary greater than 10 mR/hr or greater than 50 mR/hr CDE Thyroid.

Hourly Whole Body Dose Corresponding to 200 x ODAM Limit for Gaseous Release

ODAM limit = 500 mrem/year Whole Body

200 x **ODAM** limit = [200 x 500 mrem/year]/8760 hours/year = 11.4 mrem Whole Body in one hour

Rounded off to 10 mrem

Dose assessment using MIDAS is based on the EPA-400 methodology, *e.g.*, use of Total Effective Dose Equivalent (TEDE). This is somewhat different from whole body dose from gaseous effluents determined by ODAM methodology which forms the basis for the radiation monitor readings calculated in AU1 in accordance with the generic methodology. The gaseous effluent radiation monitors can only detect noble gases. The contribution of iodine's to TEDE could therefore only be determined either by: (1) utilizing MIDAS, or (2) gaseous effluent sampling. DAEC EAL 4 is written in terms of TEDE and the gaseous effluent radiation monitor readings are determined based on ODAM.

REFERENCES:

1. Offsite Dose Assessment Manual Section 6.1.2 and 7.1.2 Bases
2. Emergency Plan Implementing Procedure (EPIP) 3.3, Dose Assessment and Protective Action
3. Radiation Protection Calculation No. 95-001-C, Emergency Actions Levels Based on Effluent Radiation Monitors, January 24, 1995
4. UFSAR Section 11.5, Process and Effluent Radiation Monitoring and Sampling Systems
5. EPA 400-R-92-001, *Manual of Protective Action Guides and Protective Actions for Nuclear Incidents*
6. *NUMARC Methodology for Development of Emergency Action Levels NUMARC/NESP-007 Revision 2 Questions and Answers*, June 1993

AA2 Major Damage to Irradiated Fuel or Loss of Water Level that Has or Will Result in the Uncovering of Irradiated Fuel Outside the Reactor Vessel

EVENT TYPE: Onsite Rad Conditions

OPERATING MODE APPLICABILITY: All

EXAMPLE EMERGENCY ACTION LEVELS: (1 or 2 or 3 or 4)

1. <Valid <Site-specific> radiation monitor readings for the refuel floor area, fuel handling area, and the fuel bridge area.>
2. Report of Visual observation of irradiated fuel uncovered.
3. Water Level less than (site-specific) feet for the Reactor Refueling Cavity that will result in Irradiated Fuel Uncovering.
4. Water Level less than (site-specific) feet for the Spent Fuel Pool < > that will result in Irradiated Fuel uncovering.

DAEC EAL INFORMATION:

Valid means that the reading is from instrumentation determined to be operable in accordance with the Technical Specifications or has been verified by other independent methods such as indications displayed on the control panels, reports from plant personnel, or radiological survey results. Valid alarms are solely due to damage to irradiated fuel or loss of water level that has or will result in the uncovering of irradiated fuel.

There are no significant deviations from the generic EALs. Increased radiation levels can be detected by the local radiation monitors, in-plant radiological surveys, new fuel and spent fuel storage area radiation monitor alarms displayed on panel 1C04B, fuel pool ventilation exhaust monitors, and by Standby Gas Treatment (SBGT) System automatic start. Applicable area radiation monitors include RT 9163, RT 9164, RT 9153, and RT 9178. These monitors are located in the north end of the refuel floor, the south end of the refuel floor, the new fuel vault area, and near the spent fuel pool, respectively.

Per ARP 1C04B, the applicable area radiation monitor alarms actuate when radiation levels increase above 100 mR/hr in the spent fuel pool area or above 10 mR/hr in the other three areas of concern. If a valid actuation of these alarms were to occur, the refueling floor would be immediately evacuated. Thus, a report of a fuel handling accident with either valid actuation of the fuel area alarms on panel 1C04B or with

measured radiation levels in the spent fuel pool or north fuel area are used to address the generic concern consistent with DAEC design and procedures.

During preparation for reactor cavity flood up prior to entry into refuel mode, reactor vessel level instrument LI-4541 (WR GEMAC, FLOODUP) on control room panel 1C04 is placed in service by I&C personnel connecting a compensating air signal after the reference leg is disconnected from the reactor head. Normal refuel water level is above the top of the span of this flood up level indicator. A valid on-scale indication (*e.g.*, not due to loss of compensating air signal or other instrument channel failure) from this instrument can be used to determine uncontrolled loss of water level in the reactor cavity.

During refueling, the gates between the reactor cavity and the refueling cavity are removed and the spent fuel pool level indicator LI 3413 is used to monitor refueling water level. This measures the common water level in the reactor cavity and the fuel pool. The bottom of the fuel transfer slot between the spent fuel pool and the reactor cavity is 16 feet above the bottom of the spent fuel pool. The top of the active fuel in the spent fuel storage racks is slightly less than 13 feet 9 inches above the bottom of the spent fuel pool. Therefore, postulated failures which drain the reactor cavity through the reactor vessel cannot uncover fuel in the spent fuel storage racks. However, valid indication of spent fuel pool level less than 16 feet would indicate that spent fuel in the storage racks may potentially become uncovered.

F&RCHP 5 requires that upon a loss of water level situation, that the refueling crew on the refueling floor shall discharge any fuel assembly on the fuel grapple as follows:

- If a fuel assembly is currently being withdrawn from a slot in the core or spent fuel pool, immediately reinsert it into that slot.
- If a fuel assembly is being transferred and is still over or near the core, insert it into the closest available slot in the core.
- If a fuel assembly is being transferred and is over or near the spent fuel pool, insert it into the closest available slot in the spent fuel racks.

Following these actions, the refueling floor is to be evacuated of all personnel. The DAEC EAL is written to address the generic concern that a spent fuel assembly was not fully covered by water. This can either be by visual observation of an uncovered spent fuel assembly or by trending fuel pool level in the control room if a spent fuel assembly could not be placed in a safe storage location specified by F&RCHP 5 as described above.

REFERENCES:

1. Alarm Response Procedure (ARP) 1C04B, Reactor Water Cleanup and Isolation
2. Technical Specification 3.9C, Spent Fuel Pool Water Level
3. Emergency Operating Procedures (EOP) Basis Document, Breakpoints for RC/L & L
4. Emergency Plan Implementing Procedure (EPIP) 3.1, Inplant Radiological Monitoring, Attachment 1, ARM Locations
5. Surveillance Test Procedure (STP) 42A-0001, Daily and Shift Instrument Checks
6. Integrated Plant Operating Instruction (IPOI) 8 , Outage and Refueling Operations
7. Fuel & Reactor Component Handling Procedure (F&RCHP) 5, Procedure for Moving Core Components Between Reactor Core and Spent Fuel Pool, Within the Reactor Core, or Within the Spent Fuel Pool
8. Bechtel Drawing C-492, Reactor Building - Reactor Well, Spent Fuel & Dryer-Separator Pool General Arrangement, Rev. 6
9. Bechtel Drawing C-493, Reactor Building - Spent Fuel Liner Plan Elevations and Details, Sheet 1, Rev. 6
10. Holtec International Drawing No. 1045, Rack Construction - Spent Fuel Storage Racks, Rev. 3
11. *NUMARC Methodology for Development of Emergency Action Levels NUMARC/NESP-007 Revision 2 Questions and Answers*, June 1993

AA3 Release of Radioactive Material or Increases in Radiation Levels Within the Facility That Impedes Operation of Systems Required to Maintain Safe Operations or to Establish or to Maintain Cold Shutdown

EVENT TYPE: Onsite Rad Conditions

OPERATING MODE APPLICABILITY: All

EXAMPLE EMERGENCY ACTION LEVELS: (1 or 2)

1. Valid (site-specific) radiation monitor readings GREATER THAN (site-specific) values in areas requiring continuous occupancy to maintain plant safety functions < >
2. Valid (site-specific) radiation monitor readings GREATER THAN (site-specific) values in areas requiring infrequent access to maintain plant safety functions < >

DAEC EAL INFORMATION:

Valid means that the reading is from instrumentation determined to be operable in accordance with the Technical Specifications or has been verified by other independent methods such as indications displayed on the control panels, reports from plant personnel, or radiological survey results.

There are no significant deviations from the generic EALs. Per the UFSAR, the control room is the only area that is required to be continuously occupied to achieve and maintain safe shutdown following design basis accidents. DAEC EAL 1 is directly applicable to NUMARC EAL 1. However, the capability exists for plant shutdown from outside the main control room in the event that the control room becomes uninhabitable using remote shutdown panel 1C388. DAEC EAL 2 is directly applicable to NUMARC EAL 2.

The EC/OSS should determine the cause of the increase in radiation levels and review other EALs for applicability. Expected increases in monitor readings due to controlled evolutions (such as lifting the steam dryer during refueling) do not result in emergency declaration. Nor should momentary increases due to events such as resin transfers or controlled movement of radioactive sources result in emergency declaration. In-plant radiation level increases that would result in emergency declaration, are also *unplanned*, e.g., outside the limits established by an existing radioactive discharge permit.

REFERENCES:

1. Alarm Response Procedure (ARP) 1C04B, Reactor Water Cleanup and Isolation
2. Abnormal Operating Procedure (AOP) 913, Fire
3. Abnormal Operating Procedure (AOP) 914, Security
4. Abnormal Operating Procedure (AOP) 915, Shutdown Outside Control Room
5. Surveillance Test Procedure (STP) 42A-0001, Daily and Shift Instrument Checks
6. Integrated Plant Operating Instruction (IPOI) 8 , Outage and Refueling Operations
7. Emergency Plan Implementing Procedure (EPIP) 3.1, Inplant Radiological Monitoring
8. UFSAR Section 6.4, Habitability Systems
9. Bechtel Calculation DA-4, Project Number 265-002, Control Room Habitability, 9/3/80
10. *NUMARC Methodology for Development of Emergency Action Levels NUMARC/NESP-007 Revision 2 Questions and Answers*, June 1993

AS1 Site Boundary Dose Resulting from an Actual or Imminent Release of Gaseous Radioactivity Exceeds 100 <mrem TEDE> or 500 <mrem CDE> Thyroid for the Actual or Projected Duration of the Release

EVENT TYPE: Offsite Rad Conditions

OPERATING MODE APPLICABILITY: All

EXAMPLE EMERGENCY ACTION LEVELS: (1 or 2 or 3 or 4)

1. A valid reading on <site-specific> monitors <for greater than 15 minutes which corresponds to an offsite dose of 100 mrem or 500 mrem Thyroid in an hour>.
2. A valid reading sustained for 15 minutes or longer on perimeter radiation monitoring system greater than 100 mR/hr. [for sites having telemetered perimeter monitors].
3. Valid dose assessment capability indicates dose consequences greater than 100 <mrem TEDE> or 500 <mrem CDE> thyroid.
4. Field survey results indicate site boundary dose rates exceeding 100 <mrem>/hr expected to continue for more than one hour; or analyses of field survey samples indicate <CDE> thyroid of 500 <mrem> for one hour of inhalation.

DAEC EAL INFORMATION:

Valid means that the reading is from instrumentation determined to be operable in accordance with the Technical Specifications or has been verified by other independent methods such as indications displayed on the control panels, reports from plant personnel, or radiological survey results.

There are no significant deviations from the generic EALs.

DAEC's Meteorological Information and Dose Assessment System (MIDAS) was utilized to determine the KAMAN monitor limits. Eight separate combinations of release point, source term, meteorological conditions and equipment status were analyzed. Pathways considered were the offgas stack, the turbine building exhaust vent and a single reactor building exhaust vent. Multiple release points were not considered. In this same vein, it was assumed that only one of the three reactor building vents is on during the release.

The source terms used have been pre-loaded into MIDAS and are the default mixes associated with a loss of coolant accident (LOCA) and a control rod drop (CRD). The LOCA mix was used in conjunction with a release via the offgas stack while the CRD mix was used for releases via the turbine or reactor building vents. The source term for a release via the offgas stack is further impacted by the status of the standby gas treatment system. The status of that system was also taken into consideration.

Based of 1995 data (NG-96-0987), the atmospheric stability was classified as Pascal E 33% of the time. Consequently, both classifications were evaluated. Based on the same report, the most common wind speeds were:

Pascal Class	Altitude	Speed (mph)
D	156'	8 - 12
D	33'	8 - 12
E	156'	8 - 12
E	33'	4 - 7

Though the temperature setting has no impact on the MIDAS calculations, a value must be entered in order for the program to run. Consequently, the temperature was arbitrarily set at 50 F.

The rain estimate was set at zero, to eliminate any on site washout of radioactive material.

For the first MIDAS runs a 1Ci/cc concentration was assumed. The results of these runs were then normalized to the limits, thus generating a theoretical KAMAN limit. Additional MIDAS runs were made with these theoretical limits as input to verify the normalization process.

In addition to the total integrated dose, MIDAS calculates a peak whole body DDE rate resulting from the plume and a peak thyroid CDE rate resulting from inhalation. Because the AS1 and AG1 KAMAN limits are to be based on a one hour exposure, establishing concentration limits so these peak values match the NUMARC limits is acceptable.

ABNORMAL RAD LEVELS/RADIOLOGICAL EFFLUENT CATEGORY

Initiating Condition	Site Area Emergency AS1	General Emergency AG1
Valid Turbine or Reactor Building ventilation rad monitor (KAMAN) reading for more than 15 minutes above:	0.06 μCi/cc	0.6 μCi/cc
Valid Offgas Stack ventilation rad monitor (KAMAN) reading for more than 15 minutes above:	40 μCi/cc	400 μCi/cc

The primary method for declaration is by means of dose assessment using the MIDAS computer model. However, if the monitor readings are sustained for longer than 15 minutes and the required dose assessments cannot be completed within this period, then the declaration must be made based on the valid reading.

DAEC does not have a telemetered radiation monitoring system. As an alternative, DAEC uses valid field survey readings outside the site boundary greater than 100 mR/hr or greater than 500 mR/hr CDE Thyroid.

Dose assessment using MIDAS is based on the EPA-400 methodology, *e.g.*, use of Total Effective Dose Equivalent (TEDE) and Committed Dose Equivalent (CDE) Thyroid. TEDE is somewhat different from whole body dose from gaseous effluents determined by ODAM methodology which forms the basis for the radiation monitor readings calculated in AU1. These factors can introduce differences that are at least as large as those introduced by using TEDE versus whole body dose. The gaseous effluent radiation monitors can only detect noble gases. The contribution of iodine's to TEDE and CDE Thyroid could therefore only be determined either by: (1) utilizing the source term mixture in MIDAS, or (2) gaseous effluent sampling. Therefore, DAEC EAL 4 is written in terms of TEDE and CDE Thyroid.

REFERENCES:

1. Offsite Dose Assessment Manual, Section 6.1.2 and 7.1.2, Bases
2. Emergency Plan Implementing Procedure (EPIP) 3.3, Dose Assessment and Protective Action
3. Radiation Protection Calculation No. 95-001-C, Emergency Actions Levels Based on Effluent Radiation Monitors, January 24, 1995
4. Radiation Engineering Calculation No. 96-007-A, Determination of DAEC Radioactive Release Initiating Conditions for AS1 & AG1 Emergency Classifications, July 3, 1996
5. UFSAR Section 11.5, Process and Effluent Radiation Monitoring and Sampling Systems
6. EPA 400-R-92-001, *Manual of Protective Action Guides and Protective Actions for Nuclear Incidents*

7. *NUMARC Methodology for Development of Emergency Action Levels NUMARC/NESP-007 Revision 2 Questions and Answers*, June 1993

AG1 Site Boundary Dose Resulting from an Actual or Imminent Release of Gaseous Radioactivity that Exceeds <1,000 mrem TEDE> or <5,000 mrem CDE> Thyroid for the Actual or Projected Duration of the Release < >

EVENT TYPE: Offsite Rad Conditions

OPERATING MODE APPLICABILITY: All

EXAMPLE EMERGENCY ACTION LEVELS: (1 or 2 or 3 or 4)

1. A valid reading on <site-specific> monitors <for greater than 15 minutes which corresponds to an offsite dose of 1,000 mrem or 5,000 mrem Thyroid in an hour>.
2. A valid reading sustained for 15 minutes or longer on perimeter radiation monitoring system greater than 1,000 mR/hr. [for sites having telemetered perimeter monitors].
3. Valid dose assessment capability indicates dose consequences greater than 1,000 <mrem TEDE> or 5,000 <mrem CDE> thyroid.
4. Field survey results indicate site boundary dose rates exceeding 1,000 <mrem>/hr expected to continue for more than one hour; or analyses of field survey samples indicate <CDE thyroid> of 5,000 <mrem> for one hour of inhalation.

DAEC EAL INFORMATION:

Valid means that the reading is from instrumentation determined to be operable in accordance with the Technical Specifications or has been verified by other independent methods such as indications displayed on the control panels, reports from plant personnel, or radiological survey results.

There are no significant deviations from the generic EALs.

DAEC's Meteorological Information and Dose Assessment System (MIDAS) was utilized to determine the KAMAN monitor limits. Eight separate combinations of release point, source term, meteorological conditions and equipment status were analyzed. Pathways considered were the offgas stack, the turbine building exhaust vent and a single reactor building exhaust vent. Multiple release points were not considered. In this same vein, it was assumed that only one of the three reactor building vents is on during the release.

The source terms used have been pre-loaded into MIDAS and are the default mixes associated with a loss of coolant accident (LOCA) and a control rod drop (CRD). The LOCA mix was used in conjunction with a release via the offgas stack while the CRD mix was used for releases via the turbine or reactor building vents. The source term for a release via the offgas stack is further impacted by the status of the standby gas treatment system. The status of that system was also taken into consideration.

Based of 1995 data (NG-96-0987), the atmospheric stability was classified as Pascal E 33% of the time. Consequently, both classifications were evaluated. Based on the same report, the most common wind speeds were:

Pascal Class	Altitude	Speed (mph)
D	156'	8 - 12
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E	156'	8 - 12
E	33'	4 - 7

Though the temperature setting has no impact on the MIDAS calculations, a value must be entered in order for the program to run. Consequently, the temperature was arbitrarily set at 50 F.

The rain estimate was set at zero, to eliminate any on site washout of radioactive material.

For the first MIDAS runs a 1Ci/cc concentration was assumed. The results of these runs were then normalized to the limits, thus generating a theoretical KAMAN limit. Additional MIDAS runs were made with these theoretical limits as input to verify the normalization process.

In addition to the total integrated dose, MIDAS calculates a peak whole body DDE rate resulting from the plume and a peak thyroid CDE rate resulting from inhalation. Because the AS1 and AG1 KAMAN limits are to be based on a one hour exposure, establishing concentration limits so these peak values match the NUMARC limits is acceptable.

Initiating Condition	Site Area Emergency AS1	General Emergency AG1
Valid Turbine or Reactor Building ventilation rad monitor (KAMAN) reading for more than 15 minutes above:	0.06 μCi/cc	0.6 μCi/cc
Valid Offgas Stack ventilation rad monitor (KAMAN) reading for more than 15 minutes above:	40 μCi/cc	400 μCi/cc

The preferred method for declaration is by means of dose assessment using the MIDAS computer model and is therefore is listed as DAEC EAL 4. However, if the monitor readings are sustained for longer than 15 minutes and the required dose assessments cannot be completed within this period, then the declaration must be made based on the valid reading.

DAEC does not have a telemetered radiation monitoring system. As an alternative, DAEC uses valid field survey readings outside the site boundary greater than 1,000 mR/hr or greater than 5,000 mR/hr CDE Thyroid.

Dose assessment using MIDAS is based on the EPA-400 methodology, *e.g.*, use of Total Effective Dose Equivalent (TEDE) and Committed Dose Equivalent (CDE) Thyroid. TEDE is somewhat different from whole body dose from gaseous effluents determined by ODAM methodology which forms the basis for the radiation monitor readings calculated in AU1. These factors can introduce differences that are at least as large as those introduced by using TEDE versus whole body dose. The gaseous effluent radiation monitors can only detect noble gases. The contribution of iodine's to TEDE and CDE Thyroid could therefore only be determined either by: (1) utlizing the source term mixture in MIDAS, or (2) gaseous effluent sampling. Therefore, DAEC EAL 4 is written in terms of TEDE and CDE Thyroid.

REFERENCES:

1. Offsite Dose Assessment Manual, Section 6.1.2 and 7.1.2, Bases
2. Emergency Plan Implementing Procedure (EPIP) 3.3, Dose Assessment and Protective Action
3. Radiation Protection Calculation No. 95-001-C, Emergency Actions Levels Based on Effluent Radiation Monitors, January 24, 1995
4. Radiation Engineering Calculation No. 96-007-A, Determination of DAEC Radioactive Release Initiating Conditions for AS1 & AG1 Emergency Classifications, July 3, 1996
5. UFSAR Section 11.5, Process and Effluent Radiation Monitoring and Sampling Systems
6. EPA 400-R-92-001, *Manual of Protective Action Guides and Protective Actions for Nuclear Incidents*

7. *NUMARC Methodology for Development of Emergency Action Levels NUMARC/NESP-007 Revision 2 Questions and Answers*, June 1993

FISSION PRODUCT BARRIER DEGRADATION CATEGORY

EMERGENCY PLAN IMPLEMENTING PROCEDURE	No. EPIP - 1.1 Rev. 2 PAGE 1 of 1 *(For NRC Review)*
FISSION BARRIER TABLE	EFFECTIVE DATE: TBD

INDICATORS	**FUEL CLAD BARRIER**	**RCS BARRIER**
RADIATION / CORE DAMAGE	*Loss* L Fuel damage assessment (PASAP 7.2) determines at least 5% fuel clad damage **OR** Fuel damage is indicated by any of the following: L Valid drywell rad monitor reading above 7E+2 R/hr **OR** L Valid torus rad monitor reading above 3E+1 R/hr **OR** L Coolant activity above 300µCi/gm DOSE EQUIVALENT I-131 *Potential Loss* - None	*Loss* L Valid drywell rad monitor reading above 5 R/hr after reactor shutdown *Potential Loss* - None
RPV LEVEL	*Loss* L RPV Level below -30 Inches *Potential Loss* P RPV Level below 15 inches	*Loss* L RPV Level below 15 inches *Potential Loss* - None
LEAKAGE	**None**	*Loss* - None *Potential Loss* P RCS Leakage is above 50 GPM **OR** P Unisolable primary system leakage outside the drywe as indicated by area temps or ARMs
PRIMARY CONTAINMENT ATMOSPHERE	**None**	*Loss* L Drywell pressure above 2 psig and not caused by a loss of DW Cooling *Potential Loss* - None
EC/OSS JUDGMENT	Any condition which in the EC/OSS's judgment indicates loss or potential loss of the fuel clad barrier due to: • Imminent barrier degradation • Degraded fission barrier monitoring capability	Any condition which in the EC/OSS's judgment indicates loss or potential loss of the RCS barrier due to: • Imminent barrier degradation • Degraded fission barrier monitoring capability

IMMINENT - No turnaround in safety system performance is expected and escalation to General Emergency conditions is expected within 2 hours

***NOTE**: Step 1; for all indicators, move from left to right across table, marking all applicable "L's" and "P's" for each barrier, based on plant indications. Then "L's" and "P's" marked on Barrier Table to the Logic Diagram (at right). "L's" and "P's" should be marked for each affected barrier (working top to bottom) on Step 3, an "L" or "P" marked for each associated barrier will constitute a Logic I input. When coincidence is met, then the EAL can be declared.*

L = *Loss* (of a fission product barrier) - A severe challenge to a fission product barrier exists such that the barrier is considered incapable of performing its

P = *Potential Loss* (of a fission product barrier) - A challenge to a fission product barrier exists such that the barrier is considered degraded in its ability to p

PRIMARY CONTAINMENT BARRIER

Loss - None

Potential Loss

P Valid drywell rad monitor reading above 3E+3 R/hr

OR

P Valid torus rad monitor reading above 1E+2 R/hr

OR

P Core damage assessment determines at least 20% fuel clad damage

---

Loss - None

Potential Loss

P RPV Level below -40 inches

Loss

L Failure of both isolation valves and a downstream pathway to the environment exists

OR

L Unisolable primary system leakage outside the drywell as indicated by area temps or ARMs

OR

L Primary containment venting performed per EOPs

Potential Loss - None

Loss - None

L Rapid unexplained decrease following initial increase

OR

☐ Drywell pressure response not consistent with LOCA conditions

Potential Loss

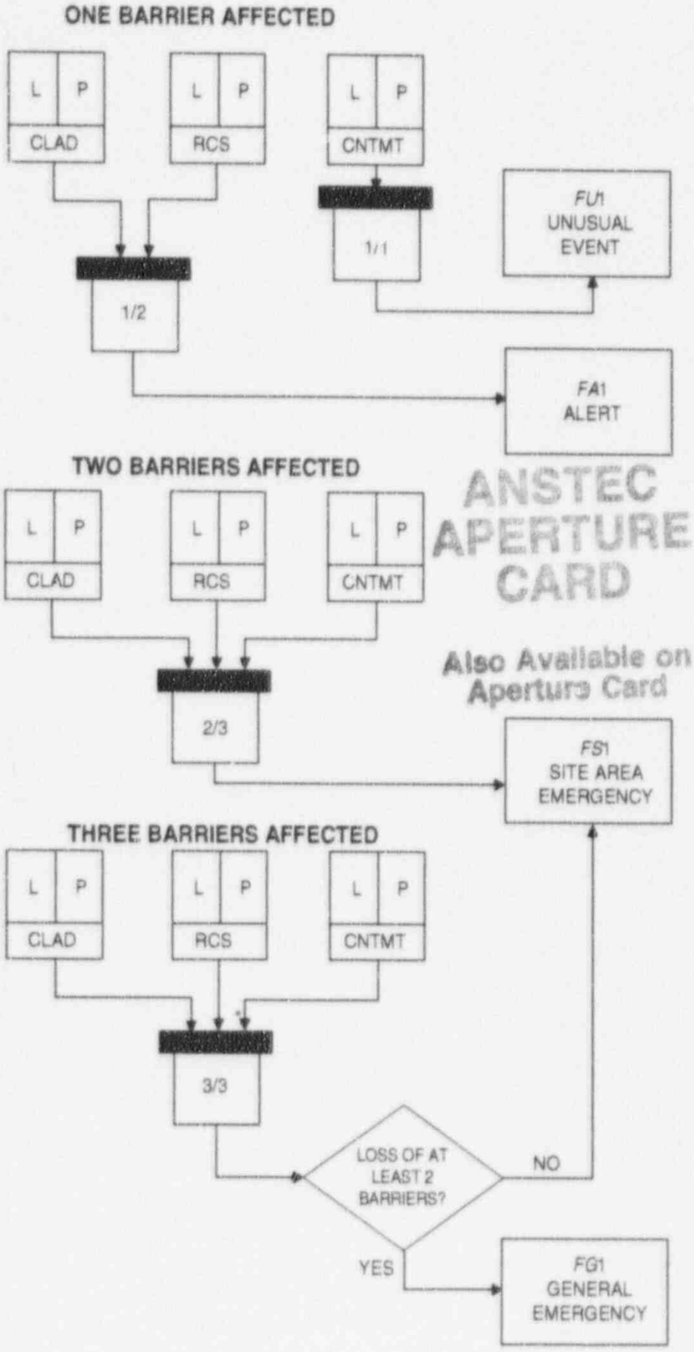
P Torus pressure reaches 53 psig

OR

P Drywell or torus H_2 CANNOT be determined to be below 6% **AND** Drywell or torus O_2 CANNOT be determined to be below 5%

Any condition which in the EC/OSS's judgment indicates loss or potential loss of the primary containment barrier due to:

- Imminent barrier degradation
- Degraded fission barrier monitoring capability



Op. Modes: Run, Startup, Hot S/D

step 2, transcribe all
he flowchart.

afety function.

rform its safety function.

9701230408-02

FISSION PRODUCT BARRIER DEGRADATION CATEGORY

FU1 Any Loss or Any Potential Loss of <Primary> Containment <Barrier>

EVENT TYPE: See Fission Barrier Table

OPERATING MODE APPLICABILITY: Run, Startup, Hot Shutdown

EXAMPLE EMERGENCY ACTION LEVELS:

See the Fission Barrier Table indicators discussed later in this section.

DAEC INFORMATION:

See the Fission Barrier Table indicators discussed later in this section. The entry conditions for this Initiating Condition are shown by the logic chart located to the right of the Fission Barrier Table.

REFERENCES:

See the Fission Barrier Table indicators discussed later in this section.

FA1 Any Loss or Any Potential Loss of Either Fuel Clad Or RCS <Barrier>

EVENT TYPE: See Fission Barrier Table

OPERATING MODE APPLICABILITY: Run, Startup, Hot Shutdown

EXAMPLE EMERGENCY ACTION LEVELS:

See the Fission Barrier Table indicators discussed later in this section.

DAEC INFORMATION:

See the Fission Barrier Table indicators discussed later in this section. The entry conditions for this Initiating Condition are shown by the logic chart located to the right of the Fission Barrier Table.

REFERENCES:

See the Fission Barrier Table indicators discussed later in this section.

FS1 <Loss Or Potential Loss of Any Two Barriers>

EVENT TYPE: See Fission Barrier Table

OPERATING MODE APPLICABILITY: Run, Startup, Hot Shutdown

EXAMPLE EMERGENCY ACTION LEVELS:

See the Fission Barrier Table indicators discussed later in this section.

DAEC INFORMATION:

The entry conditions for this Initiating Condition are shown by the logic chart located to the right of the Fission Barrier Table. DAEC uses "Loss Or Potential Loss of Any Two Barriers." This logic is simplified from the generic logic based on the following considerations:

1. Human Factors - It is easier to understand and to remember the escalation from Alert to Site Area Emergency to General Emergency using the simpler logic.

2. Comprehensiveness - A comparison was made of the combinations of barrier losses and potential losses corresponding to Site Area Emergency between the DAEC logic and the NUMARC/NESP-007 logic. All six generic barrier loss/potential loss combinations are addressed in the DAEC logic that addresses 12 combinations of barrier loss/potential loss. No sequences addressed by the NUMARC/NESP-007 logic are significantly affected by the simplified logic when applied to a BWR. See the table below.

REFERENCES:

See the Fission Barrier Table indicators discussed later in this section.

COMPARISON OF DAEC FS1 BARRIER COMBINATIONS WITH NUMARC/NESP-007 FS1 BARRIER COMBINATIONS

FUEL CLAD BARRIER		**RCS BARRIER**		**PRIMARY CONTAINMENT BARRIER**	
LOSS	**POTENTIAL LOSS**	**LOSS**	**POTENTIAL LOSS**	**LOSS**	**POTENTIAL LOSS**
1. D, N		D, N			
2. D, N			D, N		
3. D				D	
4. D					D
5.	D, N	D, N			
6.	D, N		D, N		
7.	D, N			D, N	
8.	D				D
9.		D		D	
10		D			D
11			D, N	D, N	
12		D			D

D - Barrier status addressed by DAEC simplified logic (Loss Or Potential Loss of Any Two Barriers)

N - Barrier status addressed by NUMARC/NESP-007 generic logic (Loss of BOTH Fuel Clad AND RCS **OR** Potential Loss of BOTH Fuel Clad AND RCS **OR** Potential Loss of EITHER Fuel Clad OR RCS AND Loss of ANY Additional Barrier)

FG1 Loss of Any Two Barriers AND Potential Loss of <the> Third Barrier

EVENT TYPE: See Fission Barrier Table

OPERATING MODE APPLICABILITY: Run, Startup, Hot Shutdown

EXAMPLE EMERGENCY ACTION LEVELS:

See the Fission Barrier Table indicators discussed later in this section.

DAEC INFORMATION:

See the Fission Barrier Table indicators discussed later in this section. The entry conditions for this Initiating Condition are shown by the logic chart located to the right of the Fission Barrier Table.

REFERENCES:

See the Fission Barrier Table indicators discussed later in this section.

FISSION BARRIER: Fuel Clad

DAEC INDICATOR: Radiation/Core Damage

GENERIC INDICATOR:
Drywell Radiation Monitoring
LOSS - <Valid > Drywell Rad Monitor Reading GREATER THAN (site-specific) R/hr
POTENTIAL LOSS - Not Applicable

DAEC INFORMATION:

Valid means that the reading is from instrumentation determined to be operable in accordance with the Technical Specifications or has been verified by other independent methods such as indications displayed on the control panels, reports from plant personnel, coolant sampling or radiological survey results.

There is no significant deviation from the generic "loss" indicator. Per NUMARC/NESP-007, the (site-specific) reading is a value which indicates release into the drywell of reactor coolant with elevated activity corresponding to about 2% to 5% fuel clad damage. This activity level is well above that expected from iodine spiking. *It is intended that determination of barrier loss be made whenever the indicator threshold is reached until such time that core damage assessment is performed, at which time direct use of containment rad monitor readings is no longer required.*

As documented by NG-88-0966, General Electric performed a study to predict dose rate readings from fuel damage calculations for emergency planning. The calculations were performed to obtain gamma ray dose rates at the locations of the containment atmospheric monitoring system radiation detectors in the drywell and torus locations for assumed releases of gap activity form the core. These calculations were based on "nominal" estimates of fuel rod gap fission product inventory fractions, which are considered to be more appropriate for determining a minimum threshold reading than inventory assumptions found in the NRC Regulatory Guides. The Regulatory Guide inventory assumptions applicable to dose assessments are larger and therefore non-conservative for determination of this EAL threshold. Two separate cases were evaluated. In the first case, the released activity was assumed to be contained in the drywell atmosphere. This case is considered representative of conditions following a line break in which activity is released directly into the drywell. In the second case, the released activity was assumed to be contained in the torus. This could be applied for an event which results in vessel isolation and blowdown to the suppression chamber. The results for each case were provided for each case in the form of gamma ray dose rate versus time profiles for assumed releases of 100% and 20% of the gap activity from the core. The dose rate calculations were carried out independent of any specific information on details of construction or response

characteristics of the detector systems. The figures show a drywell reading of about 2.9×10^3 R/hr or a torus reading of about 1.1×10^2 R/hr associated with 20% gap release at two hours after shutdown. Scaling this down to 5% gap release:

Calculation of Drywell and Torus Monitor Readings Assuming 5% Gap Release

NG-88-0966 value 20% Gap Release at 2 hours for drywell = 2.9×10^3 R/hr

Drywell reading = 2.9×10^3 R/hr x [5 % / 20 %] = 7.25×10^2 R/hr, round off as 7 E+2 R/hr

NG-88-0966 value 20% Gap Release at 2 hours for torus = 1.1×10^2 R/hr

Torus reading = 1.1×10^2 R/hr x [5 % / 20 %] = 2.75×10^1 R/hr, round off as 3 E+1 R/hr

The results are rounded off for ease of reading the respective radiation monitors' scales. The two hour point was picked because it allows ample time for the Technical Support Center to be operational and core damage assessment to begin. These indicators correspond to about 2.5% gap release if they occur immediately after shutdown. Thus, the indicators address the 2%-5% fuel clad damage range of concern described by the generic guidance.

REFERENCES:

1. Office Memo NG-88-0966, G.E. Fuel Damage Documentation/Dose Rate Calculations, 03/18/88

FISSION BARRIER: Fuel Clad

DAEC INDICATOR: Radiation/Core Damage

GENERIC INDICATOR:
Primary Coolant Activity Level
LOSS - Coolant activity GREATER THAN (site-specific) value
POTENTIAL LOSS - Not Applicable

DAEC INFORMATION:

There is no significant deviation from the generic indicator. Consistent with the generic methodology, DAEC uses a coolant activity value of 300 μCi/gm I_{131} equivalent. This value is well above that expected for iodine spikes and would indicate fuel clad damage has occurred.

REFERENCES:

1. Post Accident Sampling and Analysis Procedure (PASAP) 7.2, Fuel Damage Assessment

FISSION BARRIER: Fuel Clad

DAEC INDICATOR: Radiation/Core Damage

GENERIC INDICATOR:
Other (Site-Specific) Indications
LOSS - (Site-specific) as applicable
POTENTIAL LOSS - (Site-specific) as applicable

DAEC INFORMATION:

As a site-specific loss indicator, DAEC uses determination of at least 5% fuel clad damage, which is consistent with the containment rad monitor reading indicators described previously. This can be determined from the appropriate fuel damage assessment procedures.

No other reliable indications of Fuel Clad "loss" or "potential loss" could be determined.

REFERENCES:

1. Post Accident Sampling and Analysis Procedure (PASAP) 7.2, Fuel Damage Assessment

FISSION BARRIER: Fuel Clad

DAEC INDICATOR: RPV Level

GENERIC INDICATOR:
Reactor Vessel Water Level
LOSS - Level LESS THAN (site-specific) value
POTENTIAL LOSS - Level LESS THAN (site-specific) value

DAEC INFORMATION:

There are no significant deviations from the generic indicators. The generic loss indicator is based on a (site-specific) value that corresponds to the minimum value to assure core cooling without further degradation of the fuel clad. DAEC uses the Minimum Steam Cooling RPV Water Level of -30 inches. This is defined to be the lowest RPV water level at which the covered portion of the reactor core will generate sufficient steam to preclude any clad temperature in the uncovered portion of the core from exceeding 1500°F. Consistent with the EOPs, an indicated RPV level below -30 inches that cannot be restored is used.

The potential loss indicator corresponds to the (site-specific) water level at the top of the active fuel (TAF). Consistent with the EOPs, an indicated RPV level below 15 inches that cannot be restored is used.

REFERENCES:

1. Emergency Operating Procedure (EOP)-1, RPV Control, Sheet 1 of 1
2. ATWS Emergency Operating Procedure (EOP)-RPV Control, Sheet 1 of 1
3. Emergency Operating Procedure (EOP) Basis, Curves and Limits, C5, Minimum Steam Cooling RPV Water Level

FISSION BARRIER: Fuel Clad

DAEC INDICATOR: EC/OSS Judgment

GENERIC INDICATOR:
Emergency Director Judgment
Any condition which in the judgment of the Emergency Director that indicates LOSS or POTENTIAL LOSS of the FUEL CLAD barrier

DAEC INFORMATION:

There is no significant deviation from the generic indicator. Per EPIP 7.1, Emergency Coordinator Duties, the Emergency Coordinator/Operations Shift Supervisor (EC/OSS) performs the emergency director function at DAEC. EC/OSS considerations for determining whether any barrier "Loss" or "Potential Loss" include *imminent* barrier degradation, degraded *barrier monitoring* capability, and consideration of *dominant accident sequences*.

Imminent means that no turnaround in safety system performance is expected and that General Emergency conditions can be expected to occur within two hours. *Imminent* fission barrier degradation must be considered by the EC/OSS to assure timely declaration of a General Emergency and to better assure that offsite protective actions can be effectively accomplished. Degraded *barrier monitoring* capability from loss of/lack of reliable indicators must also be considered by the EC/OSS when determining if a fission barrier loss or potential loss has occurred. This assessment should also include consideration for instrumentation operability, portable instrumentation readings, and offsite monitoring results. *Dominant accident sequences* can lead to loss of all Fission Barriers. Based on the IPE, the dominant accident sequences leading to core damage at DAEC include complete loss of 125 VDC, loss of decay heat removal, ATWS with failure of Standby Liquid Control, prolonged station blackout, and loss of offsite power with early HPCI/RCIC failure. The EC/OSS should also consult System Malfunction EALs, as appropriate, to assure timely emergency classification declaration.

REFERENCES:

1. Emergency Plan Implementing Procedure (EPIP) 7.1, Emergency Coordinator Duties
2. Duane Arnold Energy Center Individual Plant Examination (IPE) November 1992

FISSION BARRIER: RCS

DAEC INDICATOR: Radiation/Core Damage

GENERIC INDICATOR:
Drywell Radiation Monitoring
LOSS - <Valid> Drywell Rad Monitor Reading GREATER THAN (site-specific) R/hr
POTENTIAL LOSS - Not applicable

DAEC INFORMATION:

Valid means that the reading is from instrumentation determined to be operable in accordance with the Technical Specifications or has been verified by other independent methods such as indications displayed on the control panels, reports from plant personnel, coolant sampling, or radiological survey results.

There is no significant deviation from the generic indicator. This loss indicator is based on conditions after reactor shutdown to assure that it is not misapplied, *i.e.*, to exclude readings due to N-16 effects which are typically 5 to 8 R/hr at full power conditions.

The (site-specific) value for this loss indicator corresponds to instantaneous release and dispersal of the reactor coolant noble gas and iodine inventory associated with normal operating concentrations (*i.e.*, within Technical Specifications) into the drywell atmosphere. The reading will be less than that specified for the loss indicator for Radiation/Core Damage that applies to the Fuel Clad barrier. Thus, this indicator would be indicative of a RCS leak only. If the radiation monitor reading increased to that value specified by the Radiation/Core indicator applying to the Fuel Clad barrier, fuel damage would also be indicated.

As documented by NG-88-0966, General Electric performed a study to predict dose rate readings from fuel damage calculations for emergency planning. The calculations were performed to obtain gamma ray dose rates at the locations of the containment atmosphere monitoring system radiation detectors in the drywell and torus locations for assumed releases of gap activity form the core. These calculations were based on "nominal" estimates of fuel rod gap fission product inventory fractions, which are considered to be more appropriate for determining a minimum threshold reading than inventory assumptions found in the NRC Regulatory Guides. The Regulatory Guide inventory assumptions applicable to dose assessments are larger and therefore non-conservative for determination of this EAL threshold. Two separate cases were evaluated. In the first case, the released activity was assumed to be contained in the drywell atmosphere. This case is considered representative of conditions following a line break in which activity is released directly into the drywell. In the second case, the released activity was assumed to be contained in the torus.

This could be applied for an event which results in vessel isolation and blowdown to the suppression chamber. The results for each case were provided for each case in the form of gamma ray dose rate versus time profiles for assumed releases of 100% and 20% of the gap activity from the core. The dose rate calculations were carried out independent of any specific information on details of construction or response characteristics of the detector systems. The figures show a drywell reading of about 2.1×10^4 R/hr associated with a 100% gap release immediately after shutdown. Assuming 99.99% fuel clad integrity (0.01% gap release) and uniform dispersal of radionuclides into the drywell immediately after shutdown, a drywell monitor reading is calculated:

Calculation of Drywell Monitor Reading Assuming 0.01% Gap Release

NG-88-0966 value for 100% Gap Release at 0.01 minutes = 2.1×10^4 R/hr

$$(2.1 \times 10^4)\ \text{R/hr} \times [(1 \times 10^{-2})\ \text{percent} / 100\ \text{percent}] = (2.1) \times 10^{4-4}\ \text{R/hr} = 2.1 \times 10^0\ \text{R/hr} = 2\ \text{R/hr}$$

To assure an indicator that is readily discernible on the drywell radiation monitor scale, DAEC uses a valid reading above 5 R/hr after reactor shutdown.

REFERENCES:

1. Office Memo NG-88-0966, G.E. Fuel Damage Documentation/Dose Rate Calculations, 03/18/88
2. Technical Specification 3.2E, Drywell Leak Detection Instrumentation

FISSION BARRIER: RCS

DAEC INDICATOR: RPV Level

GENERIC INDICATOR:
Reactor Vessel Water Level
LOSS - Level LESS THAN (site-specific) value
POTENTIAL LOSS - Not applicable

DAEC INFORMATION:

There is no significant deviation from the generic indicator. This (site-specific) loss indicator corresponds to the water level at the top of the active fuel (TAF). Consistent with the EOPs, an indicated RPV level below 15 inches that cannot be restored is used.

REFERENCES:

1. Emergency Operating Procedures (EOP) Basis, Breakpoints

FISSION BARRIER: RCS

DAEC INDICATOR: Leakage

GENERIC INDICATOR:
RCS Leak Rate

LOSS - <Valid> (site-specific) indication of Main Steamline Break

POTENTIAL LOSS - RCS leakage GREATER THAN 50 GPM inside the drywell OR unisolable primary system leakage outside drywell as indicated by <valid> area temp or area rad monitor alarm

DAEC INFORMATION:

Valid means that the reading is from instrumentation determined to be operable in accordance with the Technical Specifications or has been verified by other independent methods such as indications displayed on the control panels, reports from plant personnel, or radiological survey results.

There are no significant deviations from the generic potential loss indicators applying to RCS leakage and indications of unisolable primary system leakage. *Please note that RCS leakage inside the drywell excludes Safety-Relief Valve (SRV) discharge through the SRV discharge piping into the torus below the water line*. SRV leakage is addressed by SU5, RCS Leakage.

Unisolable primary system leakage outside the drywell includes leakage through portions of the main steam lines, portions of the Reactor Water Cleanup System (RWCU), and through the Scram Discharge Volumes (SDVs) detected per EOP 3. It is possible to have relatively small amounts of leakage result in radiation monitor alarms, therefore it is treated as a potential loss of the RCS barrier and loss of the Primary Containment barrier (see the discussion under Primary Containment Leakage indicator).

DAEC does not use the generic "loss" indicator for main steam line break. *NUMARC Methodology for Development of Emergency Action Levels NUMARC/NESP-007 Revision 2 Questions and Answers*, June 1993, discloses that the main steam line break with isolation does not have to be included as a fission barrier table indicator. This event can be appropriately classified in the System Malfunction Recognition Category. This event was classified as a RCS barrier loss indicator in the generic guidance because this event typically results in a puff release with dose consequences greater than 10 millirem whole body, *i.e.*, offsite dose consequences consistent with declaration of an Alert in accordance with AA1, Any Unplanned Release of Gaseous or Liquid Radioactivity to the Environment that Exceeds 200 Times Radiological Technical Specifications for 15 Minutes or Longer. However, UFSAR Section 15.6.6, Table 15.6-1, Steam-Line Break - Radiological Effects for Puff Release at 47 Meters, Total Dose, shows a maximum

dose of 0.58 mrem (5.8E-04 rem) passing cloud whole body dose using conservative assumptions. Therefore, because this event at DAEC has dose consequences similar to those of AU1, Any Unplanned Release of Gaseous or Liquid Radioactivity to the Environment that Exceeds 2 Times Radiological Technical Specifications for 60 Minutes or Longer, it has been added as an Unusual Event EAL in SU5, RCS Leakage.

REFERENCES:

1. Alarm Response Procedure (ARP) 1C04B, Reactor Water Cleanup and Recirculation
2. Alarm Response Procedure (ARP) 1C04C, Reactor Water Cleanup and Recirculation
3. Emergency Operating Procedure (EOP) 3, Secondary Containment Control
4. UFSAR Section 15.6.6, Loss-of-Coolant-Accident
5. *NUMARC Methodology for Development of Emergency Action Levels NUMARC/NESP-007 Revision 2 Questions and Answers*, June 1993

FISSION BARRIER: RCS

DAEC INDICATOR: Primary Containment Atmosphere

GENERIC INDICATOR:
Drywell Pressure
LOSS - <Valid> Pressure <Reading> GREATER THAN (site-specific) psig
POTENTIAL LOSS - Not applicable

DAEC INFORMATION:

Valid means that the reading is from instrumentation determined to be operable in accordance with the Technical Specifications or has been verified by other independent methods such as indications displayed on the control panels, reports from plant personnel, or radiological survey results.

There is no significant deviation from the generic indicator. The (site-specific) value for this loss indicator corresponds to the drywell high pressure ECCS initiation signal setpoint of 2.0 psig. DAEC also specifies that drywell cooling is operating to assure that the indicator is not misapplied to conditions that do not indicate RCS leakage into the drywell, *i.e.*, the drywell pressure increase is not due to loss of drywell cooling.

DAEC uses a GE Mark I Containment. During reactor operation, with drywell cooling in operation and the drywell inerted, the normal operating pressure in the drywell is between 0.5 and 1.0 psig. Analysis at the DAEC shows that a 50 gpm RCS leak would result in a 2 to 3 psig pressure rise over a six minute time period. Since a 2 psig rise would place DAEC above the ECCS initiation setpoint, (2 psig) it is necessary to select the DAEC ECCS initiation setpoint of 2 psig to indicate an actual loss of the RCS. Drywell cooling is not isolated at the 2 psig ECCS initiation setpoint, therefor further pressure rise would be indicative of a RCS leak.

REFERENCES:

1. Emergency Operating Procedures (EOP) Bases, Breakpoints
2. Emergency Operating Procedures (EOP) -1, RPV Control
3. Emergency Operating Procedures (EOP) -2, Primary Containment Control

FISSION BARRIER: RCS

DAEC INDICATOR: Primary Containment Atmosphere

GENERIC INDICATOR:
Emergency Director Judgment

Any condition which in the judgment of the Emergency Director that indicates LOSS or POTENTIAL LOSS of the RCS barrier

DAEC INFORMATION:

There is no significant deviation from the generic EAL. Per EPIP 7.1, Emergency Coordinator Duties, the Emergency Coordinator/Operations Shift Supervisor (EC/OSS) performs the emergency director function at DAEC. EC/OSS considerations for determining whether any barrier "Loss" or "Potential Loss" include *imminent* barrier degradation, degraded *barrier monitoring* capability, and consideration of *dominant accident sequences*.

Imminent means that no turnaround in safety system performance is expected and that General Emergency conditions will occur within two hours. *Imminent* fission barrier degradation must be considered by the EC/OSS to assure timely declaration of a General Emergency and to better assure that offsite protective actions can be effectively accomplished. Degraded *barrier monitoring* capability from loss of/lack of reliable indicators must also be considered by the EC/OSS when determining if a fission barrier loss or potential loss has occurred. This assessment should also include consideration for instrumentation operability, portable instrumentation readings, and offsite monitoring results. *Dominant accident sequences* can lead to loss of all Fission Barriers. Based on the IPE, the dominant accident sequences leading to core damage at DAEC include complete loss of 125 VDC, loss of decay heat removal, ATWS with failure of Standby Liquid Control, prolonged station blackout, and loss of offsite power with early HPCI/RCIC failure. The EC/OSS should also consult System Malfunction EALs, as appropriate, to assure timely emergency classification declaration.

For the RCS barrier, the EC/OSS should also consider safety-relief valves (SRVs) open or cycling. If an SRV is stuck open or is cycling and no other emergency conditions exist, an emergency declaration may not be appropriate. However, *if the fuel is damaged and the SRV is allowing fission products to escape into primary containment, a loss of RCS should be determined as having occurred.* The EC/OSS should also consult SU5, RCS Leakage, to determine if RCS leakage exceeds the threshold required for declaration of an Unusual Event.

REFERENCES:

1. Emergency Plan Implementing Procedure (EPIP) 7.1, Emergency Coordinator Duties
2. Duane Arnold Energy Center Individual Plant Examination (IPE) November 1992
3. *NUMARC Methodology for Development of Emergency Action Levels NUMARC/NESP-007 Revision 2 Questions and Answers*, June 1993

FISSION BARRIER: RCS

DAEC INDICATOR: None

GENERIC INDICATOR:
Other (Site-Specific) Indications
  LOSS - (Site-specific) as applicable
  POTENTIAL LOSS - (Site-specific) as applicable

DAEC INFORMATION:

Other indicators were also considered. No other reliable indicators of RCS barrier "loss" or "potential loss" could be determined.

REFERENCES:

None

FISSION BARRIER: Primary Containment

DAEC INDICATOR: Radiation/Core Damage

GENERIC INDICATOR:
Significant Radioactive Inventory in Containment
LOSS - Not applicable
POTENTIAL LOSS - Containment Rad Monitor reading GREATER THAN (site-specific) R/hr

DAEC INFORMATION:

There is no significant deviation from the generic indicators. The "potential loss" (site-specific) indicator value corresponds to at least 20% fuel clad damage with release into the primary containment. This indicator corresponds to loss of both the Fuel Clad and RCS barriers with Potential Loss of the Primary Containment barrier, and would result in declaration of a General Emergency. The basis for the 20% fuel clad damage threshold is described under the 20% core damage assessment indicator. *It is intended that determination of barrier potential loss be made whenever the indicator threshold is reached until such time that core damage assessment is performed, at which time direct use of containment rad monitor readings is no longer required.*

As documented by NG-88-0966, General Electric performed a study to predict dose rate readings from fuel damage calculations for emergency planning. The calculations were performed to obtain gamma ray dose rates at the locations of the containment atmospheric monitoring system radiation detectors in the drywell and torus locations for assumed releases of gap activity form the core. These calculations were based on "nominal" estimates of fuel rod gap fission product inventory fractions, which are considered to be more appropriate for determining a minimum threshold reading than inventory assumptions found in the NRC Regulatory Guides. The Regulatory Guide inventory assumptions applicable to dose assessments are larger and therefore non-conservative for determination of this EAL threshold. Two separate cases were evaluated. In the first case, the released activity was assumed to be contained in the drywell atmosphere. This case is considered representative of conditions following a line break in which activity is released directly into the drywell. In the second case, the released activity was assumed to be contained in the torus. This could be applied for an event which results in vessel isolation and blowdown to the suppression chamber. The results for each case were provided for each case in the form of gamma ray dose rate versus time profiles for assumed releases of 100% and 20% of the gap activity from the core. The dose rate calculations were carried out independent of any specific information on details of construction or response characteristics of the detector systems. The figures show a drywell reading of about 2.9×10^3 R/hr and a torus reading of about 1.1×10^2 R/hr associated with 20% gap release at two hours after shutdown. These

values are rounded to 3 E+3 R/hr and 1 E+2 R/hr , respectively. The two hour point was picked because it allows ample time for the Technical Support Center to be operational and core damage assessment to begin.

REFERENCES:

1. Office Memo NG-88-0966, G.E. Fuel Damage Documentation/Dose Rate Calculations, 03/18/88

FISSION BARRIER: Primary Containment

DAEC INDICATOR: Radiation/Core Damage

GENERIC INDICATOR:
Other (Site-Specific) Indications
LOSS - (Site-specific) as applicable
POTENTIAL LOSS - (Site-specific) as applicable

DAEC INFORMATION:

As a site-specific "potential loss" indicator, DAEC uses determination of at least 20% fuel clad damage, which is consistent with the level of fuel damage indicated by the drywell and torus radiation monitor readings above. This can be determined using appropriate fuel damage assessment procedures. *Regardless of whether primary containment integrity is challenged, it is possible for significant radioactivity within the primary containment to result in EPA PAG plume exposure levels being exceeded even assuming that the primary containment is within technical specification allowable leakage rates.* With or without primary containment challenge, however, a major release of radioactivity requiring off-site protective actions from core damage is not possible unless a major failure of the fuel clad barrier allows radioactive material to be released from core into the reactor coolant. NUREG-1228 indicates that such conditions do not exist when the amount of fuel clad damage is less than 20%.

Other indicators were also considered. No other reliable indicators for Primary Containment "loss" or "potential loss" could be determined.

REFERENCES:

1. Post Accident Sampling and Analysis Procedure (PASAP) 7.2, Fuel Damage Assessment
2. NUREG-1228, *Source Term Estimations During Incident Response to Severe Nuclear Power Plant Accidents,* October 1988

FISSION BARRIER: Primary Containment

DAEC INDICATOR: RPV Level

GENERIC INDICATOR:
Reactor Vessel Water Level
LOSS - Not applicable
POTENTIAL LOSS - <RPV> level less than (site-specific) value and <no injection source is available>

DAEC INFORMATION:

The underlying concern for this indicator is a threshold that represents significant uncovering of the core and *imminent* core damage. *Imminent* means that no turnaround in safety system performance would be expected and that General Emergency conditions would be expected within two hours.

Consistent with the underlying concern, the DAEC indicator addresses conditions where the water level is below the Minimum Zero-Injection RPV Water Level of -40 inches with no injection source available. The Minimum Zero-Injection RPV Water Level is defined to be the lowest RPV water level at which the covered portion of the reactor core will generate sufficient steam to preclude any fuel clad temperature in the uncovered portion of the core from exceeding 1800 °F. The Minimum Zero-Injection RPV Water Level is utilized to preclude significant fuel clad damage and hydrogen generation for as long as possible when no sources of RPV makeup water are available.

Thus, for RPV water level below -40 inches, if no source of injection water was available, water levels would continue to decrease and the fuel clad temperature would be expected to continue to rise. Due to large uncertainties in severe accident progression, it should be assumed that severe core melt is *imminent* if this condition were to occur. It would not be acceptable to delay the declaration of the General Emergency and issuance of Protective Action Recommendations beyond this point.

REFERENCES:

1. Emergency Operating Procedure (EOP) Bases Document, Curves and Limits
2. Emergency Operating Procedure (EOP) RPV/F - RPV Flooding
3. *NUMARC Methodology for Development of Emergency Action Levels NUMARC/NESP-007 Revision 2 Questions and Answers*, June 1993

FISSION BARRIER: Primary Containment

DAEC INDICATOR: Leakage

GENERIC INDICATOR:
Containment Isolation Valve Status After Containment Isolation Signal

LOSS - Failure of both valves in any one line to close AND downstream pathway to the environment exists <u>OR</u> Intentional venting per EOPs <u>OR</u> unisolable primary system leakage outside drywell as indicated by <valid> area temp or area rad alarm

POTENTIAL LOSS - Not applicable

DAEC INFORMATION:

Valid means that the reading is from instrumentation determined to be operable in accordance with the Technical Specifications or has been verified by other independent methods such as indications displayed on the control panels, reports from plant personnel, or radiological survey results.

The "loss" indicators used at DAEC directly correspond to the generic indicators. Venting of the primary containment can be performed in accordance with EOP 2 irrespective of the offsite radioactivity release rate that will occur and by defeating isolation interlocks as necessary. The consequences of not doing so may be the loss of primary containment integrity, core damage, and an uncontrolled radioactive release much greater than might otherwise occur. Primary containment venting is performed only as necessary to reduce and then maintain torus pressure below the Primary Containment Pressure Limit (PCPL) of 53 psig.

Unisolable primary system leakage outside the drywell includes leakage through portions of the main steam lines, portions of the Reactor Water Cleanup System (RWCU), and through the Scram Discharge Volumes (SDV's) detected per EOP 3. It is possible to have relatively small amounts of leakage result in radiation monitor alarms, therefore it is treated as a "potential loss" of the RCS (see the discussion under RCS Barrier Leakage indicator) and "loss" of the Primary Containment.

REFERENCES:

1. Emergency Operating Procedure (EOP) 2, Primary Containment Control
2. Emergency Operating Procedure (EOP) 3, Secondary Containment Control
3. Emergency Operating Procedures (EOP) Bases, Breakpoints

FISSION BARRIER: Primary Containment

DAEC INDICATOR: Primary Containment Atmosphere

GENERIC INDICATOR:

Drywell Pressure
  LOSS - Rapid unexplained decrease following initial increase OR Drywell pressure response not consistent with LOCA conditions
  POTENTIAL LOSS - (site-specific) PSIG OR explosive mixture exists

DAEC INFORMATION:

There are no significant deviations from the generic indicators. The "loss" indicators used at DAEC directly correspond to the generic indicators.

The first "potential loss" (site-specific) indicator is torus pressure of 53 psig, which is the Primary Containment Pressure Limit (PCPL) used in the EOPs. The second "potential loss" indicator is based on determination of explosive mixture in accordance with the EOPs. DAEC EOPs require control of drywell and torus atmosphere gas concentrations to less than 6% H_2 and less than 5% O_2 to assure that an explosive mixture does not exist. This "potential loss" indicator is written to be consistent with the EOPs.

REFERENCES:

1. Emergency Operating Procedure (EOP) 2, Primary Containment Control
2. Emergency Operating Procedure (EOP) PCH - Primary Containment Hydrogen

FISSION BARRIER: Primary Containment

DAEC INDICATOR: EC/OSS Judgment

GENERIC INDICATOR:
Emergency Director Judgment

> Any condition which in the judgment of the Emergency Director that indicates LOSS or POTENTIAL LOSS of the RCS barrier

DAEC INFORMATION:

There is no significant deviation from the generic indicator. Per EPIP 7.1, Emergency Coordinator Duties, the Emergency Coordinator/Operations Shift Supervisor (EC/OSS) performs the emergency director function at DAEC. EC/OSS considerations for determining whether any barrier "Loss" or "Potential Loss" include *imminent* barrier degradation, degraded *barrier monitoring* capability, and consideration of *dominant accident sequences*.

Imminent means that no turnaround in safety system performance is expected and General Emergency conditions will occur within two hours. *Imminent* fission barrier degradation must be considered by the EC/OSS to assure timely declaration of a General Emergency and to better assure that offsite protective actions can be effectively accomplished. Degraded *barrier monitoring* capability from loss of/lack of reliable indicators must also be considered by the EC/OSS when determining if a fission barrier loss or potential loss has occurred. This assessment should also include consideration for instrumentation operability, portable instrumentation readings, and offsite monitoring results. *Dominant accident sequences* can lead to loss of all Fission Barriers. Based on the IPE, the dominant accident sequences leading to core damage at DAEC include complete loss of 125 VDC, loss of decay heat removal, ATWS with failure of Standby Liquid Control, prolonged station blackout, and loss of offsite power with early HPCI/RCIC failure. The EC/OSS should also consult System Malfunction EALs, as appropriate, to assure timely emergency classification declaration.

REFERENCES:

1. Emergency Plan Implementing Procedure (EPIP) 7.1, Emergency Coordinator Duties
2. Duane Arnold Energy Center Individual Plant Examination (IPE) November 1992

EMERGENCY PLAN IMPLEMENTING PROCEDURE	No. EPIP - 1.1 Rev. 2 PAGE 1 of 1 *(For NRC Review)*
HAZARDS AND OTHER CONDITIONS AFFECTING PLANT SAFETY	EFFECTIVE DATE: TBD

EVENT TYPE	UNUSUAL EVENT	ALERT	SITE AREA EMERGENCY	GENERAL EMERGENCY
NATURAL DISASTERS	***HU1*** **Natural and Destructive Phenomena Affecting the Protected Area** Earthquake detected per AOP 901, Earthquake. **OR** Report of tornado touching down within plant protected area or within switchyard. **OR** Assessment by the control room that an event has occurred. **OR** Vehicle crash into plant structures or systems within protected area boundary. **OR** Report of an unanticipated explosion within the protected area boundary resulting in visible damage to structures or equipment. **OR** Turbine failure resulting in casing penetration or damage to turbine or generator seals. **OR** River level above 757 feet. **OR** Any area water level above Max Normal Operating Limit. **OR** River level below 725 feet 6 inches. **Op. Modes: ALL**	***HA1*** **Natural and Destructive Phenomena Affecting the Plant Vital Area** Earthquake peak horizontal acceleration above ± 0.06 Gravity. **OR** Report of tornado striking plant vital area. **OR** Report to control room of damage affecting safe shutdown areas. **OR** Vehicle crash affecting plant vital areas. **OR** Sustained wind speed above 95 MPH. **OR** Missiles affecting safe shutdown areas. **OR** River level above 767 feet. **OR** Water level above Max Safe Operating Limit in 2 or more areas AND Reactor shutdown is required. **OR** River level below 724 feet 6 inches. **Op. Modes: ALL**		
FIRE	***HU2*** **Fire Within Safe Shutdown Areas Not Extinguished Within 15 Minutes of Detection** Fire in buildings or areas contiguous to any of the following areas not extinguished within 15 minutes of control room notification or verification of a control room alarm: • Reactor, turbine, control, admin/security • Intake structure • Pump house **Op. Modes: ALL**	***HA2*** **Fire Affecting the Operability of Plant Safety Systems Required to Establish or Maintain Safe Shutdown** Fire or explosion in any of the following areas: • Reactor, turbine, control, admin/security • Intake structure • Pump house **AND** Affected system parameter indications show degraded performance or plant personnel report visible damage to permanent structures or equipment within the specified area. **Op. Modes: ALL**		
	HU3 **Release of Toxic or Flammable Gases Deemed Detrimental to Safe Operation of the Plant**	***HA3*** **Release of Toxic or Flammable Gases Within a Facility Structure Which Jeopardizes Operation of Systems Required to Maintain Safe Operations or to**		

Safe Shutdown Areas	
Category	Area
Electrical Power	Switchyard, 1G31 DG and Day Tank Rooms, 1G21 DG and Day Tank Rooms, Battery Rooms, Essential Switchgear Rooms, Cable Spreading Room
Heat Sink/ Coolant Supply	Torus Room, Intake Structure, Pumphouse
Containment	Drywell, Torus
Emergency Systems	NE, NW, SE Corner Rooms, HPCI Room, RCIC Room, RHR Valve Room, North CRD Area, South CRD Area
Other	Control Building, Remote Shutdown Panel 1C388 Area, Panel 1C56 Area, SBGT Room

Water Level Operating Limits			
Room Area	Indicator	Max Normal Operating Limit (inches)	Max Safe Operating Limit (inches)
HPCI Room Area	LI 3768	6	24
RCIC Room Area	LI 3769	6	18
A RHR Corner Room SE Area	LI 3770	6	23
B RHR Corner Room NW Area	LI 3771	6	23
Torus Area	LI 3772	12	24

Systems & Equipment of Concern
• Reactivity Control • Containment (Drywell/Torus) • RHR/Core Spray/SRV's • HPCI/RCIC • RHRSW/River Water/ESW

OTHER HAZARDS AND FAILURES	Toxic or flammable gas release affecting normal operation. **OR** Report by local, county or State official for potential evacuation of site personnel based on offsite event. **Op. Modes: ALL**	Toxic or flammable gas making safe shutdown areas uninhabitable or inaccessible. **Op. Modes: ALL**	- Offsite AC Power - Instrument AC - DC Power - Remote Shutdown Capability	
SECURITY	*HU4* **Confirmed Security Event Which Indicates a Potential Degradation in the Level of Safety of the Plant** Suspected sabotage device discovered within plant protected area and outside plant vital area. **OR** Suspected sabotage device discovered in plant switchyard. **Op. Modes: ALL**	*HA4* **Security Event in a Plant Protected Area** Intrusion into plant protected area by a hostile force. **OR** Sabotage device discovered in the plant protected area. **Op. Modes: ALL**	*HS1* **Security Event in a Plant Vital Area** Intrusion into plant vital area by a hostile force. **OR** Sabotage device discovered in the plant vital area. **Op. Modes: ALL**	*HG1* **Security Event Resulting in Loss of Ablity to Reach and Maintain Cold Shutdown** Loss of physical control of the Control Room. **OR** Loss of physical control of remote shutdown capability. **Op. Modes: ALL**
CONTROL ROOM EVACUATION	None	*HA5* **Control Room Evacuation Has Been Initiated** Control room evacuation initiated per AOP 915, Shutdown Outside Control Room. **Op. Modes: ALL**	*HS2* **Control Room Evacuation Has Been Initiated and Plant Control Cannot Be Established** Control room has been evacuated AND control of plant from Remote Shutdown Panel 1C388 NOT established within 20 minutes. **Op. Modes: ALL**	None
EC/OSS JUDGMENT	*HU5* **Other Conditions Existing Which in the Judgment of the EC/OSS Warrant Declaration of an Unusual Event** Other conditions exist which in the judgment of the EC/OSS indicate potential degradation of the level of safety of the plant. **Op. Modes: ALL**	*HA6* **Other Conditions Existing Which in the Judgment of the EC/OSS Warrant Declaration of an Alert** Other conditions exist which in the judgment of the EC/OSS indicate that plant systems may be degraded and that increased monitoring of plant functions is warranted. **Op. Modes: ALL**	*HS3* **Other Conditions Existing Which in the Judgment of the EC/OSS Warrant Declaration of a Site Area Emergency** Other conditions exist which in the judgment of the EC/OSS indicate actual or likely major failures of plant functions needed for protection of the public. **Op. Modes: ALL**	*HG2* **Other Conditions Existing Which in the Judgment of the EC/OSS Warrant Declaration of a General Emergency** Other conditions exist which in the judgment of the EC/OSS indicate EITHER: - Actual or imminent substantial core degradation with potential for loss of containment. - Potential for uncontrolled radionuclide releases which can reasonably be expected to exceed EPA PAG plume exposure levels outside the site boundary. **Op. Modes: ALL**

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HAZARDS AND OTHER CONDITIONS AFFECTING PLANT SAFETY CATEGORY

HAZARDS AND OTHER CONDITIONS AFFECTING PLANT SAFETY CATEGORY

HU1 Natural and Destructive Phenomena Affecting the Protected Area

EVENT TYPE: Natural Disasters, Other Hazards and Failures

OPERATING MODE APPLICABILITY: All

EXAMPLE EMERGENCY ACTION LEVELS: (1 or 2 or 3 or 4 or 5 or 6 or 7)

1. (Site-Specific) method indicates felt earthquake.
2. Report by plant personnel of tornado striking within protected area boundary.
3. Assessment by the control room that an event has occurred.
4. Vehicle crash into plant structures or systems within protected area boundary.
5. Report by plant personnel of an unanticipated explosion within the protected area boundary resulting in visible damage to permanent structures or equipment.
6. Report of turbine failure resulting in casing penetration or damage to turbine or generator seals.
7. (Site-Specific) occurrences.

DAEC EAL INFORMATION:

There are no significant deviations from the generic EALs. EAL 1 addresses earthquakes that are detected in accordance with AOP 901. For DAEC, a minimum detectable earthquake that is indicated on panel 1C35 is an acceleration greater than ± 0.01 Gravity. DAEC EAL 2 addresses report of a tornado striking within the protected area or within the plant switchyard. DAEC EAL 3 allows for the control room to determine that and event has occurred and take appropriate action based on personal assessment as opposed to verification. No attempt is made to assess the actual magnitude of the damage. Such damage can be due to collision, tornadoes, missiles, or any other cause. Damage can be indicated by report to the control room, physical observation, or by Control Room/local control station instrumentation. Such items as scorching, cracks, dents, or discoloration of equipment or structures required for safe shutdown are addressed by this EAL. DAEC EAL 4 addresses a vehicle (automobile, aircraft, forklift, truck or train) crash that may potentially damage plant structures containing functions and systems required for safe shutdown of the plant. This does not include vehicle crashes with each other or damage to office or warehouse structures. Escalation to Alert under HA1 would occur if damage was sufficient to affect the ability to achieve or maintain safe shutdown, *e.g.*, damage made required equipment inoperable or structural damage was observed such as bent supports or pressure boundary leakage.

DAEC EAL 5 addresses explosions within the protected area. As used here, an explosion is a rapid, violent, unconfined combustion, or a catastrophic failure of pressurized equipment, that potentially imparts

significant energy to near-by structures or equipment. Damage can be indicated by report to the control room, physical observation, or by Control Room/local control station instrumentation. Such items as scorching, cracks, dents, or discoloration of equipment or structures required for safe shutdown are addressed by this EAL. The EC/OSS needs to consider the security aspects of the explosion, if applicable. DAEC EAL 6 addresses turbine failure causing observable damage to the turbine casing or to the seals of the generator.

EALs 7 through 9 address site-specific occurrences of concern. These concerns include external flood water levels, internal flooding, and low river water level affecting the ultimate heat sink. DAEC EAL 7 addresses the observed effects of flooding in accordance with AOP 902. Plant site finished grade is at elevation 757.0 ft. Personnel doors and railroad and truck openings at or near grade would require protection in the event of a flood above elevation 757.0 ft. Therefore, EAL 7 uses a threshold of flood water levels above 757.0 ft.

DAEC EAL 8 addresses internal flooding can be due to system malfunctions, component failures, or repair activity mishaps (such as failed freeze seal) that can threaten safe operation of the plant. Therefore, this EAL is based on a valid indication that the water level is higher than the maximum normal operating limits. The Maximum Normal Operating Limits are defined as the highest values of the identified parameter expected to occur during normal plant operating conditions with all directly associated support and control systems functioning properly. Exceeding these limits is an entry condition into EOP 3, Secondary Containment Control and may be an indication that water from a primary system is discharging into secondary containment. Exceeding the maximum normal operating limit is interpreted as a potential degradation in the level of the safety of the plant and is appropriately treated as an Unusual Event emergency classification. The maximum normal operating water level limits are taken from AOP 902 and EOP 3 and are shown in the table below:

Maximum Operating Limits - Water Levels			
Affected Location	Indicator	Maximum Normal OL	Maximum Safe OL
HPCI Room Area	LI 3768	6 inches	24 inches
RCIC Room Area	LI 3769	6 inches	18 inches
A RHR Corner Room SE Area	LI 3770	6 inches	23 inches
B RHR Corner Room NW Area	LI 3771	6 inches	23 inches
Torus Area	LI 3772	12 inches	24 inches

EAL 9 addresses the effects of low river water level. The intake structure for the safety-related water supply systems (river water, RHR service water, and emergency service water) is located on the west bank

of the Cedar River. An overflow-type barrier across the river was designed and constructed in accordance with Seismic Category I criteria to intercept the streambed flow and divert it to the intake structure. This makes the entire flow of the river available to the safety-related water supply systems. A minimum flow of 13 cubic feet per second (cfs) from a minimum 1000-year river flow of 60 cfs must be diverted. The top of the barrier wall is at elevation 725 ft. 6 in. River water level below this level represents a potential degradation in the level of safety of the plant and is addressed by EAL 9.

REFERENCES:

1. Abnormal Operating Procedure (AOP) 901, Earthquake
2. Abnormal Operating Procedure (AOP) 902, Flood
3. Abnormal Operating Procedure (AOP) 903, Tornado
4. Emergency Operating Procedure (EOP)-3, Secondary Containment Control
5. EOP Basis Document, EOP-3, Secondary Containment Control
6. UFSAR Chapter 3, Design of Structures, Components, Equipment, and Systems
7. Bechtel Drawing BECH-M017, Equipment Location - Intake Structure Plans at Elevations, Rev. 6

HU2 Fire Within Protected Area BoundaryNot Extinguished Within 15 Minutes of Detection

EVENT TYPE: Fire

OPERATING MODE APPLICABILITY: All

EXAMPLE EMERGENCY ACTION LEVEL:

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

DAEC EAL INFORMATION:

There is no significant deviation from the generic EAL. The purpose of this EAL is to address the magnitude and extent of fires that may be potentially significant precursors to damage to safety systems. This includes such items as fires within the administration building, and security building (buildings contiguous to the reactor building, turbine building and control building), yet, excludes fires in the warehouse or construction support center, waste-basket fires, and other small fires of no safety consequence.

Per AOP 913, the location of a fire can be determined by observing XL3 alarm messages, Zone Indicating Unit (ZIU) alarms, or fire annunciators on panels 1C40 and 1C40A. The location of a fire can also be determined by verbal report of the person discovering the fire. *Verification* of the alarm in this context means those actions taken to determine that the control room alarm is not spurious.

REFERENCES:

1. Abnormal Operating Procedure (AOP) 913, Fire
2. Abnormal Operating Procedure (AOP) 914, Security

HU3 Release of Toxic or Flammable Gases Deemed Detrimental to Safe Operation of the Plant

EVENT TYPE: Other Hazards and Failures

OPERATING MODE APPLICABILITY: All

EXAMPLE EMERGENCY ACTION LEVELS: (1 or 2)

1. Report or detection of toxic or flammable gases that could enter within the site area boundary in amounts that can affect normal operation of the plant.

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

DAEC EAL INFORMATION:

There is no significant deviation from the generic EALs. This IC is based on releases in concentrations within the site boundary that will affect the health of plant personnel or affecting the safe operation of the plant with the plant being within the evacuation area of an offsite event (i.e., tanker truck accident releasing toxic gases, etc.) The evacuation area is as determined from the DOT Evacuation Tables for Selected Hazardous Materials, in the DOT Emergency Response Guide for Hazardous Materials.

For the purposes of this IC, CO_2 (such as is discharged by the fire suppression system) is not toxic. CO_2 can be lethal if it reduces oxygen to low concentrations that are immediately dangerous to life and health (IDLH). *CO_2 discharge into an area is not basis for emergency classification under this IC unless: (1) Access to the affected area is required, and (2) CO_2 concentration results in conditions that make the area uninhabitable or inaccessible (i.e., IDLH).*

REFERENCES:

1. UFSAR Section 2.2, Nearby Industrial, Transportation, and Military Facilities
2. UFSAR Section 6.4, Habitability Systems

HU4 Confirmed Security Event Which Indicates a Potential Degradation in the Level of Safety of the Plant

EVENT TYPE: Security

OPERATING MODE APPLICABILITY: All

EXAMPLE EMERGENCY ACTION LEVELS: (1 or 2)

1. Bomb device discovered within plant Protected Area and outside the plant Vital Area.

2. Other security events as determined from (site-specific) Safeguards Contingency Plan.

DAEC EAL INFORMATION:

There is no significant deviation from the generic EALs. Security events which do not represent at least a potential degradation in the level of safety of the plant are reported under 10 CFR 73.71 or in some cases under 10 CFR 50.72. The term "suspected sabotage device" is used in place of "bomb device" for consistency with the DAEC Safeguards Contingency Plan.

Other (site-specific) security events of concern at DAEC include discovery of a suspected sabotage device in the plant switchyard, which is located outside the protected area.

Suspected sabotage devices discovered within the plant Vital Area would result in escalation via other Security Event ICs.

REFERENCES:

1. Abnormal Operating Procedure (AOP) 914, Security Events

HU5 Other Conditions Existing Which in the Judgment of the <EC/OSS> Warrant Declaration of an Unusual Event

EVENT TYPE: EC/OSS Judgment

OPERATING MODE APPLICABILITY: All

EXAMPLE EMERGENCY ACTION LEVEL:

1. Other conditions exist which in the judgment of the Emergency Director indicate a potential degradation of the level of safety of the plant.

DAEC EAL INFORMATION:

There is no significant deviation from the generic EAL.

Per EPIP 7.1, the Emergency Coordinator/Operations Shift Supervisor (EC/OSS) is the title for the emergency director function at DAEC. The EAL addresses conditions that fall under the Notification of Unusual Event emergency classification description contained in NUREG-0654, Appendix 1 that is retained under the generic methodology.

REFERENCES:

1. Emergency Plan Implementing Procedure (EPIP) 7.1, Emergency Coordinator Duties
2. NUREG-0654/FEMA-REP-1, *Criteria for Preparation and Evaluation of Radiological Emergency Response Plans and Preparedness in Support of Nuclear Power Plants*, Revision 1, October 1980, Appendix 1

HA1 Natural and Destructive Phenomena Affecting the Plant Vital Area

EVENT TYPE: Natural Disasters, Other Hazards and Failures

OPERATING MODE APPLICABILITY: All

EXAMPLE EMERGENCY ACTION LEVELS: (1 or 2 or 3 or 4 or 5 or 6 or 7)

1. (Site-Specific) method indicates Seismic Event greater than Operating Basis Earthquake (OBE).
2. Tornado or high winds striking plant vital areas: Tornado or high winds greater than (site-specific) mph strike within protected area boundary.
3. Report of any visible structural damage on <site-specific structures>
4. (Site-Specific) indications in the control room.
5. Vehicle crash affecting plant vital areas.
6. Turbine failure generated missiles result in any visible structural damage to or penetration of any of the following <site-specific areas>
7. (Site-Specific) occurrences.

DAEC EAL INFORMATION:

There are no significant deviations from the generic EALs. *For the events of concern here, the key issue is not the wind speed, earthquake intensity, etc., but whether there is resultant damage to equipment or structures required to achieve or maintain safe shutdown, regardless of the cause.* Determination of damage affecting the ability to achieve or maintain safe shutdown can be indicated by reports to the control room, physical observation or by Control Room/local control station instrumentation.

EAL 1 addresses OBE events that are detected in accordance with AOP 901. For DAEC, the OBE is associated with a peak horizontal acceleration of ± 0.06 Gravity. DAEC EAL 2 addresses report of a tornado striking a plant vital area. DAEC EAL 3 addresses a report to the control room of damage affecting safe shutdown areas. The reported damage can be from tornadoes, high winds, flooding, missiles, collisions, or any other cause.

DAEC EAL 4 addresses vehicle (automobile, aircraft, forklift, truck or train) confirmed crashes affecting plant vital areas. This does not include vehicle crashes with each other or damage to office or warehouse structures. DAEC EAL 5 addresses sustained high wind speeds as measured by the 33-Foot or 156-Foot elevations on the Meteorological Tower. *Sustained wind speed* means the baseline wind speed measured by meteorological tower that does not include gusts. The design basis wind speed is 105 miles per hour.

However, the meteorological instrumentation is only capable of measuring wind speeds up to 100 miles per hour. Thus the alert level for sustained high wind speed, 95 miles per hour, is selected to be on-scale for the meteorological instrumentation and to conservatively account for potential measurement errors. DAEC EAL 6 addresses missiles affecting safe shutdown areas. Such missiles can be from any cause, *e.g.*, tornado-generated; turbine, pump or other rotating machinery catastrophic failure; or generated from an explosion.

Per AOPs 913 and 914, the following areas are identified as safe shutdown areas and are shown on the EAL tables. *This table is displayed as an aid to the Emergency Coordinator in determining appropriate areas of concern.*

Safe Shutdown Areas	
Category	Area
Electrical Power	Switchyard, 1G31 DG and Day Tank Rooms, 1G21 DG and Day Tank Rooms, Battery Rooms, Essential Switchgear Rooms, Cable Spreading Room
Heat Sink/ Coolant Supply	Torus Room, Intake Structure, Pumphouse
Containment	Drywell, Torus
Emergency Systems	NE, NW, SE Corner Rooms, HPCI Room, RCIC Room, RHR Valve Room, North CRD Area, South CRD Area
Other	Control Building, Remote Shutdown Panel 1C388 Area, Panel 1C56 Area, SBGT Room

DAEC EALs 7, 8, and 9 address site-specific occurrences of concern. These concerns include external flood water levels, internal flooding, and low river water level affecting the ultimate heat sink. DAEC EAL 7 addresses river water levels exceeding design flood water levels. All Seismic Category I structures and non-seismic structures housing Seismic Category I equipment are designed to withstand the hydraulic head resulting from the "maximum probable flood" to which the site could be subjected. The design flood water is at elevation 767.0 ft. Major equipment penetrations in the exterior walls are located above elevation 767.0 ft. Openings below the flood level are either watertight or are provided with means to control the inflow of water in order to ensure that a safe shutdown can be achieved and maintained. Consideration has also been given to providing temporary protection for openings in the exterior walls up to flood levels of 769.0 ft. All buildings were also checked for uplift (buoyancy) for a flood level at elevation 767.0 ft, and the minimum factor of safety used was 1.2. Therefore, DAEC EAL 7 uses as its threshold flood water levels above 767 feet.DAEC EAL 8 addresses internal flooding consistent with the

requirements of EOP 3, Secondary Containment Control. If RPV pressure reduction will have no effect on leakage into secondary containment, then EOP 3 requires that reactor shutdown be performed in accordance with Integrated Plant Operating Instruction (IPOI) 3, 4, or 5 as necessary if the water level exceeds its maximum safe operating limits in two or more areas. If RPV pressure reduction will decrease leakage into secondary containment then this is due to leakage from the primary system, which is addressed by the Fission Barrier Table indicators and System Malfunction EALs, and is not addressed here. Therefore, EAL 8 addresses conditions in which water level in two or more areas is above Maximum Safe Operating Limits and reactor shutdown is *required.*

Required means that the reactor shutdown was procedurally mandated by EOP 3 and is not merely performed as a precaution or inadvertently. *Maximum Safe Operating Limits* are defined as the highest parameter value at which neither (1) equipment necessary for safe shutdown of the plant will fail nor (2) personnel access necessary for the safe shutdown of the plant will be precluded. The internal flooding can be due to system malfunctions, component failures, or repair activity mishaps (such as failed freeze seal) that can threaten safe operation of the plant. This includes water intrusion on equipment that is not designed to be submerged (*e.g.*, motor control centers).

The maximum safe operating water level limits are taken from EOP 3 and are shown on the table below:

Maximum Operating Limits - Water Levels			
Affected Location	Indicator	Maximum Normal OL	Maximum Safe OL
HPCI Room Area	LI 3768	6 inches	24 inches
RCIC Room Area	LI 3769	6 inches	18 inches
A RHR Corner Room SE Area	LI 3770	6 inches	23 inches
B RHR Corner Room NW Area	LI 3771	6 inches	23 inches
Torus Area	LI 3772	12 inches	24 inches

DAEC EAL 9 addresses the effects of low river water level. The intake structure for the safety-related water supply systems (river water, RHR service water, and emergency service water) is located on the west bank of the Cedar River. The overflow weir is at elevation 724 feet 6 inches. River level at or below this elevation will result in all river flow being diverted to the safety related water supply systems. The top of the intake structure around the pump wells is at elevation 724 feet. If the river water level dropped to this level, the pump suction would have no continuous supply. Therefore, this EAL uses a threshold of water level below 724 feet 6 inches as a potential substantial degradation of the ultimate heat sink capability.

REFERENCES:

1. Abnormal Operating Procedure (AOP) 901, Earthquake
2. Abnormal Operating Procedure (AOP) 902, Flood
3. Abnormal Operating Procedure (AOP) 903, Tornado
4. Abnormal Operating Procedure (AOP) 913, Fire
5. Abnormal Operating Procedure (AOP) 914, Security Events
6. UFSAR Chapter 3, Design of Structures, Components, Equipment, and Systems
7. Bechtel Drawing BECH-M017, Equipment Location - Intake Structure Plans at Elevations, Rev. 6
8. EOP Basis Document, EOP 3 - Secondary Containment Control
9. Emergency Operating Procedure (EOP) 3, Secondary Containment Control

HA2 Fire Affecting the Operability of Plant Safety Systems Required to Establish or Maintain Safe Shutdown

EVENT TYPE: Fire

OPERATING MODE APPLICABILITY: All

EXAMPLE EMERGENCY ACTION LEVEL:

1. The following conditions exist:
   a. Fire or explosion in <site-specific areas>

   AND

   b. Affected system parameter indications show degraded performance or plant personnel report visible damage to permanent structures or equipment within the specified area.

DAEC EAL INFORMATION:

There is no significant deviation from the generic EAL. Of particular concern for this EAL are fires that may be detected in the reactor building, control building, turbine building, pumphouse, and intake structure as shown in Tabs 1 and 3 of AOP 913. Damage from fire or explosion can be indicated by physical observation, or by Control Room/local control station instrumentation.. *No attempt is made in this EAL to assess the actual magnitude of the damage.*

Per AOP 913, the location of a fire can be determined by observing XL3 alarm messages, Zone Indicating Unit (ZIU) alarms, or fire annunciators on panels 1C40 and 1C40A.

This table is displayed as an aid to the Emergency Coordinator in determining appropriate areas of concern.

Systems & Equipment of Concern
• Reactivity Control
• Containment (Drywell/Torus)
• RHR/Core Spray/SRV's
• HPCI/RCIC
• RHRSW/River Water/ESW
• Onsite AC Power/EDG's
• Offsite AC Power
• Instrument AC
• DC Power
• Remote Shutdown Capability

NOTE:

Scope of Systems and Equipment of concern established by review of Appendix R Safe Shutdown credited systems. Only those systems directly affecting safe shutdown or heat removal are listed for consideration, due to fire damage. Support Systems and equipment such as HVAC and specific instrumentation, while included in Appendix R analysis is not considered an immediate threat to the ability to shutdown the plant and remove decay heat.

With regard to explosions, *only those explosions of sufficient force to damage permanent structures or identified equipment required for safe operation, should be considered.* As used here, an explosion is a rapid, violent, unconfined combustion, or a catastrophic failure of pressurized equipment, that potentially imparts significant energy to near-by structures and materials. The occurrence of the explosion with reports of evidence of damage (e.g., deformation, scorching) is sufficient for the declaration. *The EC/OSS also needs to consider any security aspects of the explosions, if applicable.*

Per the UFSAR, the control room is the only area that is required to be continuously occupied to achieve and maintain safe shutdown following design basis accidents. However, the capability exists for plant shutdown from outside the main control room in the event that the control room becomes uninhabitable. If the control room becomes uninhabitable, remote shutdown panel 1C388 is utilized in accordance with AOP 915.

REFERENCES:

1. Abnormal Operating Procedure (AOP) 913, Fire
2. Abnormal Operating Procedure (AOP) 914, Security Events
3. Abnormal Operating Procedure (AOP) 915, Shutdown Outside Control Room
4. UFSAR Section 6.4, Habitability Systems

HA3 Release of Toxic or Flammable Gases Within a Facility Structure Which Jeopardizes Operation of Systems Required to Maintain Safe Operations or to Establish or Maintain Cold Shutdown

EVENT TYPE: Other Hazards and Failures

OPERATING MODE APPLICABILITY: All

EXAMPLE EMERGENCY ACTION LEVELS: (1 or 2)

1. Report or detection of toxic gases within a Facility Structure in concentrations that will be life threatening to plant personnel.
2. Report or detection of flammable gases within a Facility Structure in concentrations that will affect the safe operation of the plant.

DAEC EAL INFORMATION:

There is no significant deviation from the generic EALs. This IC, in addition to IC HA5 below, also addresses entry of toxic gases that may result in control room evacuation in accordance with AOP 915.

For the purposes of this IC, CO_2 (such as is discharged by the fire suppression system) is not toxic. CO_2 can be lethal if it reduces oxygen to low concentrations that are immediately dangerous to life and health (IDLH). *CO_2 discharge into an area is not basis for emergency classification under this IC unless: (1) Access to the affected area is required, and (2) CO_2 concentration results in conditions that make the area uninhabitable or inaccessible (i.e., IDLH).*

Per the UFSAR, the control room is the only area that is required to be continuously occupied to achieve and maintain safe shutdown following design basis accidents. However, the capability exists for plant shutdown from outside the main control room in the event that the control room becomes uninhabitable. If the control room becomes uninhabitable, remote shutdown panel 1C388 is utilized to achieve and maintain cold shutdown.

Per AOPs 913 and 914, the following areas are identified as safe shutdown areas. *This table is displayed as an aid to the Emergency Coordinator in determining appropriate areas of concern.*

Safe Shutdown Areas	
Category	Area
Electrical Power	Switchyard, 1G31 DG and Day Tank Rooms, 1G21 DG and Day Tank Rooms, Battery Rooms, Essential Switchgear Rooms, Cable Spreading Room
Heat Sink/Coolant Supply	Torus Room, Intake Structure, Pumphouse
Containment	Drywell, Torus
Emergency Systems	NE, NW, SE Corner Rooms, HPCI Room, RCIC Room, RHR Valve Room, North CRD Area, South CRD Area
Other	Control Building, Remote Shutdown Panel 1C388 Area, Panel 1C56 Area, SBGT Room

REFERENCES:

1. Abnormal Operating Procedure (AOP) 913, Fire
2. Abnormal Operating Procedure (AOP) 914, Security Events
3. Abnormal Operating Procedure (AOP) 915, Shutdown Outside Control Room
4. UFSAR Section 6.4, Habitability Systems

HA4 Security Event in a Plant Protected Area

EVENT TYPE: Security

OPERATING MODE APPLICABILITY: All

EXAMPLE EMERGENCY ACTION LEVELS: (1 or 2)

1. Intrusion into plant protected area by a hostile force.

2. Other security events as determined from (site-specific) Safeguards Contingency Plan.

DAEC EAL INFORMATION:

There is no significant deviation from generic EALs.

This class of security events represents an escalated threat to plant safety above that contained in the Unusual Event. *For the purposes of this EAL a civil disturbance which penetrates that protected area boundary can be considered a hostile force.* Under this EAL, adversaries within the protected area are not yet affecting nuclear safety systems, engineered safety features, or reactor shutdown capability that are located within the vital area. Intrusion into a vital area by a hostile force will escalate the event to a Site Area Emergency.

REFERENCES:

1. Abnormal Operating Procedure (AOP) 914, Security Events

HA5 Control Room Evacuation Has Been Initiated

EVENT TYPE: Control Room Evacuation

OPERATING MODE APPLICABILITY: All

EXAMPLE EMERGENCY ACTION LEVEL:

1. Entry into (site-specific) procedure for control room evacuation.

DAEC EAL INFORMATION:

There is no significant deviation from the generic EAL. The applicable procedure for control room evacuation at DAEC is AOP 915.

REFERENCES:

1. Abnormal Operating Procedure (AOP) 915, Shutdown Outside Control Room
2. UFSAR Section 6.4, Habitability Systems

HA6 Other Conditions Existing Which in the Judgment of the <EC/OSS> Warrant Declaration of an Alert

EVENT TYPE: EC/OSS Judgment

OPERATING MODE APPLICABILITY: All

EXAMPLE EMERGENCY ACTION LEVEL:

1. Other conditions exist which in the Judgment of the Emergency Director indicate that plant safety systems may be degraded and that increased monitoring of plant functions is warranted.

DAEC EAL INFORMATION:

There is no significant deviation from the generic EAL.

Per EPIP 7.1, the Emergency Coordinator/Operations Shift Supervisor (EC/OSS) is the title for the emergency director function at DAEC. The EAL addresses conditions that fall under the Alert emergency classification description contained in NUREG-0654, Appendix 1.

REFERENCES:

1. Emergency Plan Implementing Procedure (EPIP) 7.1, Emergency Coordinator Duties
2. NUREG-0654/FEMA-REP-1, *Criteria for Preparation and Evaluation of Radiological Emergency Response Plans and Preparedness in Support of Nuclear Power Plants*, Revision 1, October 1980, Appendix 1

HS1 Security Event in a Plant Vital Area

EVENT TYPE: Security

OPERATING MODE APPLICABILITY: All

EXAMPLE EMERGENCY ACTION LEVELS: (1 or 2)

1. Intrusion into plant vital area by a hostile force.
2. Other security events as determined from (site-specific) Safeguards Contingency Plan.

DAEC EAL INFORMATION:

There is no significant deviation from generic EAL 1.

This class of security events represents an escalated threat to plant safety above that contained in HA4, Security Event in a Plant Protected Area, in that a hostile force has progressed from the Protected Area to the Vital Area. *Under the condition of concern here, the adversaries are considered to be in a position to directly and negatively affect nuclear safety systems, engineered safety features, or reactor shutdown capability.*

REFERENCES:

1. Abnormal Operating Procedure (AOP) 914, Security Events

HS2 Control Room Evacuation Has Been Initiated and Plant Control Cannot Be Established

EVENT TYPE: Control Room Evacuation

OPERATING MODE APPLICABILITY: All

EXAMPLE EMERGENCY ACTION LEVEL:

1. The following conditions exist:
   a. Control room evacuation has been initiated.
   AND
   b. Control of the plant cannot be established per (site-specific) procedure within (site-specific) minutes.

DAEC EAL INFORMATION:

There is no significant deviation from the generic EAL. The applicable procedure for control room evacuation at DAEC is AOP 915. Based on the results of the analysis described below, DAEC uses 20 minutes as the site-specific time limit for establishing control of the plant. DAEC has satellite panels associated with the remote shutdown panel at various locations through out the plant. It physically takes an operator longer than 15 minutes to lineup all the controls at the various panels. Control of the plant from outside the control room is assumed when the controls are transferred to remote shutdown panel 1C388 in accordance with AOP 915.

The EC/OSS is expected to make a reasonable, informed judgment within the 20 minute time limit that control of the plant from the remote shutdown panel has been established. The intent of the EAL is that control of important plant equipment and knowledge of important plant parameters has been achieved in a timely manner. Primary emphasis should be placed on those components and instruments that provide protection of and information about safety functions. At a minimum, consistent with the Appendix R safe shutdown analysis described above, these safety functions include reactivity control, maintaining reactor water level, and decay heat removal.

General Electric performed analyses to demonstrate compliance with the requirements of 10 CFR 50 Appendix R for DAEC. The evaluation of Reactor Coolant Inventory was performed using the GE evaluation model (SAFE). The SAFE code determines if the reactor coolant inventory is above the TAF during the safe shutdown operation. If core uncovery occurs, the fuel clad integrity evaluation is performed

by determining the duration of the core uncovery and the resulting peak cladding temperature (PCT). The PCT calculations were performed by incorporating the SAFE output into the Core Heatup Analysis code (CHASTE). The details of these calculations are provided in Section 4 of the final report for DAEC Appendix R analyses ("Safe Shutdown Appendix R Analyses for Duane Arnold Energy Center", MDE-44-036).

The required analyses include evaluation of the safe shutdown capability of the remote shutdown system for various control room fire events assuming: (1) no spurious operation of equipment, (2) spurious operation of a safety-relief valve (SRV) for 20 minutes, (3) spurious operation of a SRV for 10 minutes, and (4) spurious leakage from a one-inch line. The analyses show that the worst case spurious operation of SRV or isolation valves on a one-inch liquid line (high-low pressure interface) will not affect the safe shutdown ability of the remote shutdown system for DAEC in case of a fire requiring control room evacuation before the identified time limit for the necessary operator actions at the auxiliary shutdown panels. For the limiting cases of worst case spurious leakage from a one-inch line and spurious operation of a SRV, operator control within 20 minutes would not impact the integrity of the fuel clad, the reactor pressure vessel, and the primary containment.

REFERENCES:

1. Abnormal Operating Procedure (AOP) 915, Shutdown Outside Control Room
2. General Electric Report MDE-44-0386, *Safe Shutdown Appendix R Analysis for DAEC*, March 1986
3. UFSAR Section 6.4, Habitability Systems
4. *NUMARC Methodology for Development of Emergency Action Levels NUMARC/NESP-007 Revision 2 Questions and Answers*, June 1993

HS3 Other Conditions Existing Which in the Judgment of the <EC/OSS> Warrant Declaration of Site Area Emergency

EVENT TYPE: EC/OSS Judgment

OPERATING MODE APPLICABILITY: All

EXAMPLE EMERGENCY ACTION LEVEL:

1. Other conditions exist which in the Judgment of the Emergency Director indicate actual or likely major failures of plant functions needed for protection of the public.

DAEC EAL INFORMATION:

There is no significant deviation from the generic EAL.

Per EPIP 7.1, the Emergency Coordinator/Operations Shift Supervisor (EC/OSS) is the title for the emergency director function at DAEC. The EAL addresses conditions that fall under the Site Area Emergency classification description contained in NUREG-0654, Appendix 1.

REFERENCES:

1. Emergency Plan Implementing Procedure (EPIP) 7.1, Emergency Coordinator Duties
2. NUREG-0654/FEMA-REP-1, *Criteria for Preparation and Evaluation of Radiological Emergency Response Plans and Preparedness in Support of Nuclear Power Plants*, Revision 1, October 1980, Appendix 1

HG1 Security Event Resulting in Loss Of Ability to Reach and Maintain Cold Shutdown

EVENT TYPE: Security

OPERATING MODE APPLICABILITY: All

EXAMPLE EMERGENCY ACTION LEVELS: (1 or 2)

1. Loss of physical control of the control room due to security event.

2. Loss of physical control of the remote shutdown capability due to security event.

DAEC EAL INFORMATION:

There are no significant deviations from the generic EALs. The EALs encompass conditions under which a hostile force has taken physical control of vital area required to reach and maintain safe shutdown. This also includes areas where any switches that transfer control of safe shutdown equipment to outside the control room are located.

REFERENCES:

1. Abnormal Operating Procedure (AOP) 914, Security Events
2. UFSAR Section 6.4, Habitability Systems

HG2 Other Conditions Existing Which in the Judgment of the <EC/OSS> Warrant Declaration of General Emergency

EVENT TYPE: EC/OSS Judgment

OPERATING MODE APPLICABILITY: All

EXAMPLE EMERGENCY ACTION LEVEL:

1. Other conditions exist which in the Judgment of the Emergency Director indicate: (1) actual or imminent substantial core degradation with potential for loss of containment, or (2) potential for uncontrolled radionuclide releases. These releases can reasonably be expected to exceed EPA PAG plume exposure levels outside the site boundary.

DAEC EAL INFORMATION:

There is no significant deviation from the generic EAL.

Per EPIP 7.1, the Emergency Coordinator/Operations Shift Supervisor (EC/OSS) is the title for the emergency director function at DAEC. The EAL addresses conditions that fall under the General Emergency classification description contained in NUREG-0654, Appendix 1 and is consistent with FG1, Loss of Any Two Barriers AND Potential Loss of Third Barrier, and AG1, Site Boundary Dose Resulting from an Actual or Imminent Release of Gaseous Radioactivity that Exceeds 1000 mrem TEDE or 5000 mrem CDE Thyroid for the Actual or Projected Duration of the Release.

REFERENCES:

1. Emergency Plan Implementing Procedure (EPIP) 7.1, Emergency Coordinator Duties
2. NUREG-0654/FEMA-REP-1, *Criteria for Preparation and Evaluation of Radiological Emergency Response Plans and Preparedness in Support of Nuclear Power Plants*, Revision 1, October 1980, Appendix 1

EVENT TYPE	UNUSUAL EVENT	ALERT	SITE AREA EMERGENCY	GENERAL EMERGENCY
LOSS OF POWER	***SU1*** **Loss of All Offsite Power to Essential Busses for Greater Than 15 Minutes** Loss of Offsite Power Lasting More Than 15 Minutes. **Op. Modes: ALL** ***SU7*** **Unplanned Loss of Required DC Power During Cold Shutdown or Refuel Mode For Greater Than 15 Minutes** Unplanned Loss of Div 1 and Div 2 125 VDC busses based on bus voltage less than 105 VDC indicated. **AND** Failure to restore power to at least one required 125 VDC bus within 15 minutes from time of loss. **Op. Modes: Cold S/D, Refuel**	***SA1*** **Loss of All Offsite Power and Loss of All Onsite AC Power to Essentail Busses During Cold Conditions** Loss of Voltage on Buses 1A3 and 1A4 lasting more than 15 minutes. **Op. Modes: Cold S/D, Refuel, Defueled** ***SA5*** **AC Power Capability to Essential Busses Reduced to a Single Power Source for Greater Than 15 Minutes Such That Any Additional Single Failure Would Result in Station Blackout** Only one AC power source remains available to supply Bus 1A3 or Bus 1A4 AND if it is lost, a Station Blackout will occur. **Op. Modes: Run, Startup, Hot S/D**	***SS1*** **Loss of All Offsite Power and Loss of All Onsite AC Power to Essential Busses** Loss of Voltage on Buses 1A3 and 1A4 lasting more than 15 minutes. **Op. Modes: Run, Startup, Hot S/D** ***SS3*** **Loss of All Vital DC Power** Unplanned Loss of Div 1 and Div 2 125 VDC busses Lasting More Than 15 Minutes. **Op. Modes: Run, Startup, Hot S/D**	***SG1*** **Prolonged Loss of All Offsite Power and Prolonged Loss of All Onsite AC Power** Loss of Voltage on Buses 1A3 and 1A4 and ANY of the following: • Restoration of power to either Bus 1A3 or 1A4 is NOT likely within 4 hours. • RPV level indeterminate • RPV Level below +15 inches. **Op. Modes: Run, Startup, Hot S/D**
RPS FAILURE	None	***SA2*** **Failure of 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 Successful** Failure of automatic scram. **Op. Modes: Run, Startup**	***SS2*** **Failure of Reactor Protection System Instrumentation to Complete or Initiate on Automatic Reactor Scram Once a Reactor Protection System Setpoint Has Been Exceeded and Manual Scram Was NOT Successful** Failure of automatic and manual scram **AND** Power remains above 5% **OR** Boron injection required. **Op. Modes: Run, Startup**	***SG2*** **Failure of the Reactor Protection System to Complete an Anutomatic Scram and Manual Scram was NOT successful and There is Indication of an Extreme Challenge to the Ability to Cool the Core** Entry into ATWS EOP- RPV Control is required **AND** RPV level cannot be maintained above -30 inches. **OR** EOP Graph 4 Heat Capacity Limit is exceeded **Op. Modes: Run, Startup**
INABILITY TO MAINTAIN SHUTDOWN CONDITIONS	***SU2*** **Inability to Reach Required Shutdown Within Technical Specification Limits** Plant NOT brought to required mode within applicable LCO Action Statement Time Limits.	***SA3*** **Inability to Maintain Plant in Cold Shutdown** Loss of decay heat removal systems required to maintain cold shutdown. **AND** Temperature rise that exceeds 212°F. **OR** Uncontrolled temperature rise approaching 212°F.	***SS4*** **Complete Loss of Function Needed to Achieve or Maintain Hot Shutdown** EOP Graph 4 Heat Capacity Limit is exceeded **OR** Reactor CANNOT be brought subcritical. **Op. Modes: Run, Startup, Hot S/D** ***SS5*** **Loss of Water Level in the Reactor Vessel That Has or Will Uncover Fuel in the Reactor Vessel** NO cooling method lined up or available AND RPV Level below 15 inches.	See Fission Barrier Table

	Op. Modes: Run, Startup, Hot S/D	**Op. Modes: Cold S/D, Refuel**	**Op. Modes: Cold S/D, Refuel**	
INSTRUMENTATION / COMMUNICATION	*SU3* **Unplanned Loss of All Safety System Annunication or Indication in the Control Room for Greater Than 15 Minutes** Unplanned loss of most annunciators on panels 1C03, 1C04 and 1C05 lasting more than 15 minutes AND compensatory non-alarming indications are available. **Op. Modes: Run, Startup, Hot S/D** *SU6* **Unplanned Loss of All Onsite or Offsite Communications Capabilities** Loss of ALL onsite telephone and radio communication methods (PABX, direct-ring, UHF, and radiological survey radio systems). **OR** Loss of ALL electronic communication methods with government agencies (PABX, direct-ring, ENS, microwave and police radio). **Op. Modes: ALL**	*SA4* **Unplanned Loss of Most or All Safety System Annunication or Indication in Control Room With Either (1) a Significant Transient in Progress, or (2) Compensatory Non-Alarming Indicators are Unavailable** Unplanned loss of most annunciators on panels 1C03, 1C04 and 1C05 lasting more than 15 minutes and EITHER: • Significant transient in progress. • Loss of compensatory non-alarming indications. **Op. Modes: Run, Startup, Hot S/D**	*SS6* **Inability to Monitor a Significant Transient in Progress** Significant transient in progress and BOTH of the following: • Loss of annunciators on panels 1C03, 1C04 and 1C05 AND • Loss of compensatory non-alarming indications. **Op. Modes: Run, Startup, Hot S/D**	See Fission Barrier Table
COOLANT ACTIVITY	*SU4* **Fuel Clad Degradation** Valid Pretreat RM-4104 rad monitor reading above 4E+3 mR/hr **OR** Coolant activity above 1.2 μCi/ml DOSE EQUIVALENT I-131 **Op. Modes: Run, Startup, Hot S/D**	See Fission Barrier Table	See Fission Barrier Table	See Fission Barrier Table
COOLANT LEAKAGE	*SU5* **RCS Leakage** Unidentified or pressure boundary leakage greater than 10 GPM. **OR** Identified leakage greater than 25 GPM. **OR** Main steam line break as determined from annunciators or plant personnel report. **Op. Modes: Run, Startup, Hot S/D**	See Fission Barrier Table	See Fission Barrier Table	See Fission Barrier Table

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SYSTEM MALFUNCTION CATEGORY

SU1 Loss of All Offsite Power to Essential Busses for Greater Than 15 Minutes

EVENT TYPE: Loss of Power

OPERATING MODE APPLICABILITY: All

EXAMPLE EMERGENCY ACTION LEVEL:

1. The following conditions exist:
   a. Loss of power to (site-specific) transformers for greater than 15 minutes.
   AND
   b. At least (site-specific) emergency generators are supplying power to emergency busses.

DAEC EAL INFORMATION:

There is no significant deviation from the generic EAL. *This event is a precursor of a more serious Station Blackout condition and is thus considered as a potential degradation of the level of safety of the plant. It is possible to be operating within Technical Specification LCO Action Statement time limits and make a declaration of an Unusual Event in accordance with this EAL.*

Under the conditions of concern, entry into AOP 301, Loss of Essential Electrical Power, would be made under Tab 3, Loss of Offsite Power. Indications/alarms related to loss of offsite AC power are displayed on control room panel 1C08 and are listed in the procedure under "Probable Indications." Under these conditions, Essential 4160V Buses 1A3 and 1A4 would indicate zero volts until A diesel generator 1G-31 4kV breaker 1A311 and B diesel generator 1G-21 4kV breaker 1A411, respectively, close for each bus.

REFERENCES:

1. Abnormal Operating Procedure (AOP) 301, Loss of Essential Electrical Power
2. UFSAR Section 8.2, Offsite Power System
3. *NUMARC Methodology for Development of Emergency Action Levels NUMARC/NESP-007 Revision 2 Questions and Answers*, June 1993

SU2 Inability to Reach Required Shutdown Within Technical Specification Limits

EVENT TYPE: Inability to Maintain Shutdown Conditions

OPERATING MODE APPLICABILITY: Run, Startup, Hot Shutdown

EXAMPLE EMERGENCY ACTION LEVEL:

1. Plant is not brought to required operating mode within (site-specific) Technical Specifications LCO Action Statement Time.

DAEC EAL INFORMATION:

There is no significant deviation from the generic EAL. LCO Action Statement time limits for placing the unit in the required OPCON are provided in the DAEC Technical Specifications.

An immediate Notification of an Unusual Event is required when the plant is not brought to the required OPCON within the Technical Specifications LCO Action Statement time limits. *Declaration of an Unusual Event is based on the time at which the LCO-specified action statement time period elapses under the site Technical Specifications and is not related to how long a condition may have existed.*

REFERENCES:

1. DAEC Technical Specifications

SU3 Unplanned Loss of All Safety System Annunciation or Indication in the Control Room for Greater Than 15 Minutes

EVENT TYPE: Instrumentation/Communication

OPERATING MODE APPLICABILITY: Run, Startup, Hot Shutdown

EXAMPLE EMERGENCY ACTION LEVEL:

1. The following conditions exist:
   a. Loss of most annunciators < > associated with safety systems for greater than 15 minutes.
   AND
   b. Compensatory non-alarming indications are available.
   AND
   c. In the opinion of the <Operations> Shift Supervisor, the loss of annunciators or indicators requires increased surveillance to safely operate the unit < >.
   AND
   d. Annunciator or indicator loss does not result from planned action.

DAEC EAL INFORMATION:

Control room panels 1C03, 1C04, and 1C05 contain the annunciators associated with safety systems at DAEC. Therefore, the DAEC EAL addresses unplanned loss of most annunciators on these panels. *Compensatory non-alarming indications* includes the plant process computer, SPDS, plant recorders, or plant instrument displays in the control room. *Unplanned* loss of annunciators or indicators excludes scheduled maintenance and testing activities.

Under the conditions of concern, entry into AOP 302.2, Loss of Alarm Panel Power, would be made. The procedure requires alerting operators on shift to the nature of the lost annunciation. It further requires that operators be attendant and responsive to abnormal indications that relate to those systems and components that have lost annunciation. Therefore, the generic criterion related to specific opinion of the Operations Shift Supervisor that additional operating personnel will be required to safely operate the unit is not included in the DAEC EAL because the concern is addressed by the AOP.

Annunciators on 1C03, 1C04, and 1C05 share a common power supply from 125 VDC Division I that is fed through circuit breaker 1D13.

Indications of loss of annunciators associated with safety systems include:

- 125 VDC charger, battery, or system annunciators on control room panel 1C08
- Loss of "sealed in" annunciators at affected panels
- Failure of affected annunciator panels shiftily testing by plant operators
- Expected alarms are not received
- Computer point ID B350 indicates "NSS ANN DC LOSS TRBL." (Loss of DC power to panels 1C03, 1C04, and 1C05)

REFERENCES:

1. Operating Instruction (OI) No. 317.2 Annunciator System
2. Abnormal Operating Procedure (AOP) 302.1, Loss of 125 VDC Power
3. Abnormal Operating Procedure (AOP) 302.2, Loss of Alarm Panel Power

SU4 Fuel Clad Degradation

EVENT TYPE: Coolant Activity

OPERATING MODE APPLICABILITY: Run, Startup, Hot S/D

EXAMPLE EMERGENCY ACTION LEVELS: (1 or 2)

1. (Site-Specific) <valid> radiation monitor readings indicating fuel clad degradation greater than Technical Specification allowable limits.
2. (Site-Specific) coolant sample activity value indicating fuel clad degradation greater than Technical Specification allowable limits.

DAEC EAL INFORMATION:

There are no significant deviations from the generic EALs. *These EALs are precursors of more serious fuel clad degradation and are thus considered as indicating a potential degradation of the level of safety of the plant. Thus, it is possible to be operating within Technical Specification LCO Action Statement time limits for iodine spikes and make a declaration of an Unusual Event.* DAEC mode applicability for these EALs are consistent with the Tech Specs.

EAL 1 addresses valid pretreat rad monitor exceeding (RM-4104) above 4E+3 mR/hr. The calculation supporting this value is described below. *Valid* means that the pretreat rad monitor reading is determined to be operable in accordance with the Technical Specifications or has been verified by other independent methods such as indications displayed on the control panels, reports from plant personnel, or coolant sampling results. This reading would be displayed on Control Room panels 1C-02 and 1C-10 on pretreat rad recorder RR-4104.

As specified in the generic methodology, DAEC EAL 2 addresses coolant samples exceeding technical specification 3.6.B.1.a, coolant activity less than or equal to 1.2 μCi/ml dose equivalent I-131.

Radiological Engineering Calculation 94-014A and UFSAR Table 15.4-1 were reviewed to determine a suitable EAL threshold for the pretreat rad monitor reading corresponding to the Tech Spec 3.6.B.1.a coolant activity limit of 1.2 μCi/ml of dose equivalent I-131. Using the condenser noble gas source term for the control rod drop accident of 2.38 E +06 Curies shown on UFSAR Table 15.4-1 and the condenser free volume of 55,000 cubic feet, an initial noble gas concentration in the condenser offgas line is determined. Because the offgas flow rate is very small (about 50 standard cubic feet per minute) compared to the total condenser free volume, dilution of the condenser noble gas concentration due to offgas flow is

not considered in the calculation shown below. Decrease in the noble gas source term due to decay of short-lived noble gas radioisotopes and offgas flow dilution effects are addressed by rounding down the value calculated as shown below.

Calculation 94-014A used an exposure rate method based on using a source term consisting of a defined mixture of noble gases and iodine from the control rod drop accident as described in the DAEC UFSAR, Section 15.4. The calculation assumed that the activity is released instantly and immediately reached in equilibrium with the reactor coolant inventory. Using this calculation, using dose correction factors (DCFs) for child thyroid dose from Reg. Guide 1.109, and adjusting for the specific gravity (0.736) of saturated water at 1050 psia (fluid conditions assumed in the calculation) to adjust for standard conditions, the I-131 dose equivalent (in units of μCi/ml assuming 1 cc equals 1 ml) is determined for this event. This result is then linearly scaled for rad monitor readings corresponding to the Tech Spec 3.6.B.1.a allowable primary coolant activity of 1.2 μCi/ml I-131 dose equivalent, *i.e.*, the relative mixture of noble gases and iodine is assumed to remain constant. I-129 is ignored because it has no effect on the calculation result.

Isotope	DCF (mrem/pci)	Concentration (μCi/cc)	Correction Factor [$DCF_{ISOTOPE}$ / DCF_{I-131}] / 0.736	I-131 DEQ (μCi/cc)
I-131	4.39 E-03	1.6 E+01	1.4 E+00	2.2 E+01
I-132	5.23 E-05	2.2 E+01	1.6 E-02	3.6 E-01
I-133	1.04 E-03	3.1 E+01	3.2 E-01	1.0 E+01
I-134	1.37 E-05	3.4 E+01	4.2 E-03	1.4 E-01
1-135	2.14 E-04	2.9 E+01	6.6 E-02	1.9 E+00
TOTAL	- -	- -	- -	3.4 E+01

Therefore, for this event, a coolant activity of 34 μCi/cc I-131 dose equivalent is calculated. Scaling the results for 1.2 μCi/cc I-131 dose equivalent, a suitable condenser source term and corresponding initial concentration in the offgas flow is then determined. This is then converted to a pretreat rad monitor reading by use of the monitor efficiency factor:

Pretreat Rad Monitor (RM-4104) Reading

$$\text{NG concentration}_{\text{clad damage}} = \text{NG concentration}_{\text{ROD DROP}} \times [1.2\ \mu\text{Ci/cc} / 34\ \mu\text{Ci/cc}]$$
$$= [2.38\text{ E}+6\text{ Ci} \times 1\text{ E}+6\ \mu\text{Ci/Ci}] / [5.5\text{ E}+4\text{ ft}^3 \times 2.83\text{ E}+4\text{ cc/ft}^3] \times [1.2\ \mu\text{Ci/cc} / 34\ \mu\text{Ci/cc}]$$
$$= 1529\ \mu\text{Ci} \times 0.0353 = 54.0\ \mu\text{Ci/cc}$$

Pretreat rad monitor reading = NG concentration X Rad monitor efficiency

Rad monitor efficiency = 89.2 mR/hr / μCi/cc, therefore:
Pretreat rad monitor reading = 89.2 X 54.0 = 4800 mR/hr

To account for isotopic decay and dilution effects of offgas flow, round down to 4E+03 mR/hr.

The calculation results were also reviewed to determine if suitable values for the main steam line (MSL) radiation monitors could be developed. As shown above, the rod drop accident corresponds to coolant activity of 34 μCi/cc I-131 dose equivalent. As determined by the reference calculation, this corresponds to a MSL radiation monitor reading of about 5.7 R/hr. Scaling the results for 1.2 μCi/ml I-131 dose equivalent:

MSL Reading Corresponding to 1.2 μCi/ml I-131 dose equivalent

[[1.2 μCi/cc] / [34 μCi/cc]] X 5.7 R/hr = 0.2 R/hr = 200 mR/hr

200 mR/hr is at the lower end of the normal MSL monitor readings during full power Because this value is not distinguishable and hydrogen water chemistry system malfunctions that result in increased production of N-16 can also result in increased main steam line radiation levels, it is not appropriate at DAEC to use the main steam line monitor readings.

REFERENCES:

1. Abnormal Operating Procedure (AOP) 672.2, Offgas Radiation/Reactor Coolant High Activity
2. Technical Specification 3/4.6.B, Coolant Chemistry
3. Radiological Engineering Calculation No. 94-014A, Main Steam Line Radiation Monitor Setpoint Calculation, August 29, 1994
4. Surveillance Test Procedure (STP) No. 46B001, Reactor Coolant Gamma and Iodine Activity
5. Annunciator Response Procedure (ARP) 1C03A, Reactor and Containment Cooling and Isolation
6. Annunciator Response Procedure (ARP) 1C05B, Reactor Control
7. *NUMARC Methodology for Development of Emergency Action Levels NUMARC/NESP-007 Revision 2 Questions and Answers*, June 1993

SU5 RCS Leakage

EVENT TYPE: Coolant Leak

OPERATING MODE APPLICABILITY: Run, Startup, Hot Shutdown, Cold Shutdown

EXAMPLE EMERGENCY ACTION LEVELS< >: < (1 or 2 or 3) >

<1.> Unidentified or pressure boundary leakage greater than 10 gpm.

OR

<2.> Identified leakage greater than 25 gpm.

<OR>

<3.> <Valid (site-specific) indication of Main Steamline Break>

DAEC EAL INFORMATION:

EALs 1 and 2 are precursors of more serious RCS barrier challenges and are thus considered as a potential degradation of the level of safety of the plant. Thus, it is possible to be operating within Technical Specification LCO Action Statement time limits and make a declaration of an Unusual Event in accordance with these EALs. Credit for the action statement time limit should only be given when leakage exceeds technical specification limits but has not yet exceeded the Unusual Event EAL thresholds described above. In addition, indication of main steam line break ;has been added here as previously discussed in the basis for Fission Barrier Table RCS Barrier EAL 1, RCS Leak Rate, and is further discussed below. This is in accordance with *NUMARC Methodology for Development of Emergency Action Levels NUMARC/NESP-007 Revision 2 Questions and Answers,* June 1993, Fission Product Barrier-BWR section, response to question 4 which states that the main steam line break with isolation can be classified under System Malfunctions.

Valid means that the reading is from instrumentation determined to be operable in accordance with the Technical Specifications or has been verified by other independent methods such as indications displayed on the control panels, reports from plant personnel, or radiological survey results.

The DAEC Tech Spec Section 3.6.C.1 coolant system leakage LCO limits are: (1) 5 gpm unidentified leakage, (2) 2 gpm increase in unidentified leakage within a 24 hour period, and (3) 25 gpm total leakage. Total leakage is defined as the sum of identified and unidentified leakage.

DAEC EAL 1 uses the generic value of 10 GPM for unidentified leakage or pressure boundary leakage. The 10 gpm value for the unidentified or pressure boundary leakage was selected as it is observable with

normal control room indications. DAEC EAL 2 uses identified leakage set at a higher value due to the lesser significance of identified leakage in comparison to unidentified or pressure boundary leakage.

REFERENCES:

1. Technical Specification 3.6.C, Coolant Leakage
2. Surveillance Test Procedure No. (STP) 42A001, Reactor Coolant System Leak Rate Calculation
3. Operating Instruction No. (OI) 920, Drywell Sump System
4. Alarm Response Procedure (ARP) 1C04B, Reactor Water Cleanup and Recirculation
5. Alarm Response Procedure (ARP) 1C04C, Reactor Water Cleanup and Recirculation
6. UFSAR Section 5.2.5, Detection of Leakage through Reactor Coolant Pressure Boundary
7. UFSAR Section 15.6.6, Loss-of-Coolant-Accident
8. *NUMARC Methodology for Development of Emergency Action Levels NUMARC/NESP-007 Revision 2 Questions and Answers*, June 1993

SU6 Unplanned Loss of All Onsite or Offsite Communications Capabilities

EVENT TYPE: Instrumentation/Communication

OPERATING MODE APPLICABILITY: All

EXAMPLE EMERGENCY ACTION LEVEL:

1. Either of the following conditions exist:
   a. Loss of all (site-specific list) onsite communications capability affecting the ability to perform routine operations.

   OR

   b. Loss of all (site-specific list) offsite communications capability.

DAEC EAL INFORMATION:

There is no significant deviation from the generic EAL. The communications methods used at DAEC are described in the Emergency Plan. In-plant and external agency telephone communication methods include PABX lines, direct-ring lines, and NRC telephones which are extensions for the Emergency Notification System. There is also a microwave system to provide backup emergency telephone communications.

The availability of one method of ordinary offsite communication is sufficient to inform state and local authorities of plant problems. *This EAL is intended to be used only when extraordinary means (relaying of information from radio transmissions, individuals being sent to offsite locations, etc.) are being utilized to make communications possible.*

The DAEC plant operations radio system is a UHF system with consoles located in the Control Room, Technical Support Center, Operational Support Center, and the Central Alarm Station. Hand-held transceivers are used in this system to provide simplex communications within the plant and onsite. The DAEC Radiological Survey Radio System is an 800 MHz trunked/conventional repeater system that provides base-to-portable communications throughout the DAEC EPZ. A secondary high-band system provides back-up capability for the 800 MHz radio. Consoles are located in the Technical Support Center and the Emergency Operations Facility at the IES Tower. The DAEC Security (backup radiological survey) Radio System provides base-to-portable security communication within the plant and with the Linn County Sheriff's Office using a mobile relay (repeater) type base station and two VHF frequencies. Control consoles are located in the Secondary Alarm Station, Central Alarm Station, Security Control Point, Technical Support Center, and Emergency Operations Facility. The DAEC also has a base station licensed for operation in the Police Radio Service on the law enforcement state-wide, point-to-point VHF

frequency. The transmitter and one control console are located at the Secondary Alarm Station and in the Central Alarm Station. This station is for communications with Iowa Department of Public Safety radio station, Linn County Sheriff's office, and the Benton County Sheriff's office. This point-to-point channel is also used by the Linn County Emergency Management and other public-safety organizations throughout the state of Iowa.

REFERENCES:

1. Emergency Plan, Section F, Emergency Communications
2. *NUMARC Methodology for Development of Emergency Action Levels NUMARC/NESP-007 Revision 2 Questions and Answers*, June 1993

SU7 Unplanned Loss of Required DC Power During Cold Shutdown or Refuel < > Mode For Greater Than 15 Minutes

EVENT TYPE: Loss of Power

OPERATING MODE APPLICABILITY: Cold Shutdown, Refuel

EXAMPLE EMERGENCY ACTION LEVEL:

1. <T>he following conditions exist:
    a. Unplanned Loss of Vital DC power to required DC busses based on (site-specific) bus voltage indications.

    AND

    b. Failure to restore power to at least one required DC bus within 15 minutes from time of loss.

DAEC EAL INFORMATION:

There is no significant deviation from the generic EAL. *Unplanned* loss of Div. 1 and Div. 2. 125 VDC busses excludes scheduled maintenance and testing activities. Under the conditions of concern, AOP 302.1, Loss of 125 VDC Power, would be entered. The DAEC EAL's address the loss of both divisions of the 125 VDC systems consistent with AOP 302.1.

The 125 VDC system is divided into two independent divisions - Division I (1D1) and Division II (1D2) - each with separate AC and DC (battery) power supplies. Loss of both 125 VDC Divisions could compromise the ability to monitor and control the removal of decay heat during cold shutdown or refueling operations. These EAL's are intended to be anticipatory in as much as the operating crew may not have necessary indication and control of equipment needed to respond to the loss. If this loss results in the inability to maintain cold shutdown, the escalation to an Alert will be per SA3 "RCS temperature rise that is not allowed by procedures or Technical Specifications that will result in RCS temperature above 212 F".

Bus voltage is based on the minimum bus voltage necessary for the operation of safety related equipment and may be indicated by the illumination of annunciators "125 VDC System 1 Trouble" on 1C08A A-9 and/or "125 VDC System 2 Trouble" on 1C08B A-4.

REFERENCES:

1. Abnormal Operating Procedure (AOP) 302.1, Loss of 125 VDC Power
2. Abnormal Operating Procedure (AOP) 388, Loss of 250 VDC Power
3. Technical Specification 3.8.B, DC Power Systems
4. UFSAR Section 8.3, Onsite Power Systems
5. UFSAR Table 8.3-6, Plant Battery System - DC Power, Instrumentation, and Control, Principle DC Loads (125V)
6. ARP 1C08A A-9
7. ARP 1C08B A-4

SA1 Loss of All Offsite Power and Loss of All Onsite AC Power to Essential Busses During Cold <Conditions>

EVENT TYPE: Loss of Power

OPERATING MODE APPLICABILITY: Cold Shutdown, Refuel, Defueled

EXAMPLE EMERGENCY ACTION LEVEL:

1. The following conditions exist:
   a. Loss of power to (site-specific) transformers.

   AND

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

   AND

   c. Failure to restore power to at least one emergency bus within 15 minutes from the time of loss of both offsite and onsite AC power.

DAEC EAL INFORMATION:

There is no significant deviation from the generic EAL. Under the conditions of concern, entry into AOP 301.1, Station Blackout, would be made under Tab 1. Indications/alarms related to station blackout are displayed on control room panel 1C08 and are listed in the procedure under "Probable Indications."

At DAEC, the Essential Buses of concern are the 4160V Buses 1A3 and 1A4. Each of these buses feed their associated 480V and 120V AC buses through step down transformers.

REFERENCES:

1. Abnormal Operating Procedure (AOP) 301.1, Station Blackout
2. Abnormal Operating Procedure (AOP) 301, Loss of Essential Electrical Power
3. Technical Specifications Section 3.8, Auxiliary Electrical Systems

SA2 Failure of 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 Successful

EVENT TYPE: RPS Failure

OPERATING MODE APPLICABILITY: Run, Startup

EXAMPLE EMERGENCY ACTION LEVEL:

1. (Site-specific) indication(s) exist that indicate that reactor protection system setpoint was exceeded and automatic scram did not occur, and a successful manual scram occurred.

DAEC EAL INFORMATION:

The DAEC EAL is written in terms of failure of automatic scram. IPOI 5 specifies manual scram insertion immediately following any automatic scram signal, and therefore separately specifying successful manual scram is not required. The reactor is considered successfully shutdown if either: (1) all control rods are inserted to least position 02, or (2) it has been determined that the reactor will remain shutdown under ALL conditions without boron. If these conditions are not achieved, entry into the ATWS - RPV Control EOP will be made where additional manual actions to be performed at panel 1C05 are specified to quickly shutdown the reactor. These actions include reducing recirculation pumps to minimum speed and inserting Alternate Rod Insertion (ARI).

If the mode switch is in Startup and the rods are fully inserted (i.e., the reactor is shutdown) prior to the automatic signal failure, then declaration of an Alert would not be required. In this case, the event would be reported under 10 CFR 50.72 (b) (2) (I) as a four hour report.

The condition of concern is failure of the automatic protection system to 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 plant safety has been compromised and design limits of the fuel may have been exceeded. In the generic EAL, reactor protection system setpoint being exceeded (rather than limiting safety system setpoint being exceeded) is specified to emphasize that failure of the automatic protection system to complete the scram following generation of a scram signal is the issue of concern.

REFERENCES:

1. Integrated Plant Operating Instruction (IPOI) No. 5, Reactor Scram
2. ATWS Emergency Operating Procedure (EOP) - RPV Control
3. Emergency Operating Procedure (EOP) 1 - RPV Control
4. *NUMARC Methodology for Development of Emergency Action Levels NUMARC/NESP-007 Revision 2 Questions and Answers*, June 1993

SA3 Inability to Maintain Plant in Cold Shutdown

EVENT TYPE: Inability to Maintain Shutdown Conditions

OPERATING MODE APPLICABILITY: Cold Shutdown, Refuel

EXAMPLE EMERGENCY ACTION LEVEL:

1. Loss of <decay heat removal systems required> to maintain cold shutdown

AND

Temperature increase that either:

- Exceeds Technical Specification cold shutdown temperature limit

OR

- Results' in uncontrolled temperature rise approaching cold shutdown technical specification limit.

DAEC EAL INFORMATION:

Under the conditions of concern for EAL 1, AOP 149, Loss of Decay Heat Removal, would be entered under Tab 1, Loss of Shutdown Cooling. Indications/alarms related to loss of shutdown cooling are displayed on control room panels 1C03 and 1C05 and are listed in the procedure under "Probable Indications." The procedure requires that shutdown cooling be re-established. If this cannot be done, then the following actions are to be performed:

- Re-establish primary and/or secondary containment.
- Increase reactor water level to between 240 inches and 250 inches to improve natural circulation.
- Start one reactor recirculation pump, if available.
- Monitor reactor temperatures per STP 46A003 noting that some points will lag behind the bulk coolant temperature.
- Notify Health Physics to begin increased monitoring of the Reactor Building.
- Evaluate plant conditions to determine need to achieve or maintain the plant in cold shutdown.
- Initiate alternate means of shutdown cooling which includes feed and bleed to radwaste or condenser, feed and bleed to the torus through the SRVs, using the reactor water cleanup heat exchanger, or reactor cavity flood up and use of fuel pool cooling.

The procedure provides curves of maximum water heat up rates which provide an upper bound of the heatup until an estimated time to boil calculation can be completed by Engineering.

The DAEC EAL is written to imply a RCS temperature rise above 212 °F that is not allowed by plant procedures. This corresponds to the inability to maintain required temperature conditions for Cold Shutdown. "Uncontrolled" means that system temperature increase is not the result of planned actions by the plant staff. The wording is also intended to eliminate minor cooling interruptions occurring at the transition between Hot Shutdown and Cold Shutdown or temperature changes that are permitted to occur during establishment of alternate core cooling so that an unnecessary declaration of an Alert does not occur. The uncontrolled temperature rise is necessary to preserve the anticipatory philosophy of NUREG-0654 for events starting from temperatures much lower than the cold shutdown temperature limit.

REFERENCES:

1. Abnormal Operating Procedure (AOP) 149, Loss of Decay Heat Removal
2. DAEC Technical Specifications
3. Surveillance Test Procedure (STP) 46A003, Heatup and Cooldown Rate Log
4. NUREG 1449, *Shutdown and Low-Power Operation at Commercial Nuclear Power Plants in the United States*, September 1993
5. *NUMARC Methodology for Development of Emergency Action Levels NUMARC/NESP-007 Revision 2 Questions and Answers*, June 1993

SA4 Unplanned Loss of Most or All Safety System Annunciation or Indication in Control Room With Either (1) a Significant Transient in Progress, or (2) Compensatory Non-Alarming Indicators are Unavailable

EVENT TYPE: Instrumentation/Communication

OPERATING MODE APPLICABILITY: Run, Startup, Hot Shutdown

EXAMPLE EMERGENCY ACTION LEVEL:

1. The following conditions exist:
   a. Loss of most< > annunciators associated with safety systems for greater than 15 minutes.

   AND

   b. In the opinion of the <Operations> Shift Supervisor, the loss of all annunciators or indicators requires increased surveillance to safely operate the unit< >.

   AND

   c. Annunciator or Indicator loss does not result from planned action.

   AND

   d. Either of the following:
      - A significant plant transient in progress.

      OR

      - Compensatory non-alarming indications are unavailable.

DAEC EAL INFORMATION:

Control room panels 1C03, 1C04, and 1C05 contain the annunciators associated with safety systems at DAEC. Therefore, the DAEC EAL addresses unplanned loss of annunciators on these panels. *Compensatory non-alarming indications* includes the plant process computer, SPDS, plant recorders, or plant instrument displays in the control room. *Unplanned* loss of annunciators or indicators excludes scheduled maintenance and testing activities. *Significant transient* includes response to automatic or manually initiated functions such as scrams, runbacks involving greater than 25% thermal power change, ECCS injections, or thermal power oscillations of 10% or greater.

Under the conditions of concern , entry into AOP 302.2, Loss of Alarm Panel Power, would be made. The procedure requires alerting operators on shift to the nature of the lost annunciation. It further requires that operators be attendant and responsive to abnormal indications that relate to those systems and components that have lost annunciation. Therefore, the generic criterion related to specific opinion of the Operations

Shift Supervisor that additional operating personnel will be required to safely operate the unit is not included in the DAEC EAL because the concern is addressed by the AOP.

Annunciators on 1C03, 1C04, and 1C05 share a common power supply from 125 VDC Division I that is fed through circuit breaker 1D13. Therefore, DAEC does not specify a loss of "most" annunciators as specified in the generic methodology.

Indications of loss of annunciators associated with safety systems include:

- 125 VDC charger, battery, or system annunciators on control room panel 1C08
- Loss of "sealed in" annunciators at affected panels
- Failure of affected annunciator panels shiftily testing by plant operators
- Expected alarms are not received
- Computer point ID B350 indicates "NSS ANN DC LOSS TRBL." (Loss of DC power to panels 1C03, 1C04, and 1C05)

REFERENCES:

1. Operating Instruction (OI) No. 317.2 Annunciator System
2. Abnormal Operating Procedure (AOP) 302.1, Loss of 125 VDC Power
3. Abnormal Operating Procedure (AOP) 302.2, Loss of Alarm Panel Power

SA5 AC Power Capability to Essential Busses Reduced to a Single Power Source for Greater Than 15 Minutes Such That Any Additional Single Failure Would Result in Station Blackout

EVENT TYPE: Loss of Power

OPERATING MODE APPLICABILITY: Run, Startup, Hot Shutdown

EXAMPLE EMERGENCY ACTION LEVEL:

1. The following conditions exist:
   a. Loss of power to (site-specific) transformers for greater than 15 minutes.
   AND
   b. Onsite power capability has been degraded to one (train of) emergency bus(ses) powered from a single onsite power source due to loss of:
   (Site-specific list)

DAEC EAL INFORMATION:

The DAEC EAL is written to address the underlying concern, *i.e.*, only one AC power source remains and if it is lost, a Station Blackout will occur. Under the conditions of concern, entry into AOP 301, Loss of Essential Electrical Power, would be made under Tab 1, Loss of One Essential 4160V Bus, and/or under Tab 3, Loss of Offsite Power. Indications/alarms related to degraded AC power are displayed on control room panel 1C08 and are listed in the procedure under "Probable Indications."

At DAEC, the Essential Buses of concern are 4160V Buses 1A3 and 1A4. Each of these buses feed their associated 480V and 120V AC busses through step down transformers. Onsite power sources at DAEC include the A and B Diesel Generators, 1G-31 and 1G-21, respectively.

REFERENCES:

1. Abnormal Operating Procedure (AOP) 301, Loss of Essential Electrical Power
2. UFSAR Chapter 8 Electrical Power
3. Technical Specifications Section 3.8. Auxiliary Electrical Systems

SS1 Loss of All Offsite Power and Loss of All Onsite AC Power to Essential Busses

EVENT TYPE: Loss of Power

OPERATING MODE APPLICABILITY: Run, Startup, Hot Shutdown

EXAMPLE EMERGENCY ACTION LEVEL:

1. Loss of all offsite and onsite AC power as indicated by:
   a. Loss of power to (site-specific) transformers.

   AND

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

   AND

   c. Failure to restore power to at least one emergency bus within <15 minutes> minutes from the time of loss of both offsite and onsite AC power.

DAEC EAL INFORMATION:

There is no significant deviation from the generic EAL. In accordance with the generic guidance, DAEC is using a threshold of 15 minutes for Station Blackout to exclude transient or momentary power losses.

Under the conditions of concern, entry into AOP 301.1, Station Blackout, would be made under Tab 1. Indications/alarms related to station blackout are displayed on control room panel 1C08 and are listed in the procedure under "Probable Indications."

At DAEC, the Essential Buses of concern are the 4160V Buses 1A3 and 1A4. Each of these buses feed their associated 480V and 120V AC buses through step down transformers.

REFERENCES:

1. Abnormal Operating Procedure (AOP) 301.1, Station Blackout
2. Technical Specifications Section 3.8, Auxiliary Electrical Systems
3. UFSAR Chapter 8, Electric Power

SS2 Failure of 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

EVENT TYPE: RPS Failure

OPERATING MODE APPLICABILITY: Run, Startup

EXAMPLE EMERGENCY ACTION LEVEL:

1. (Site-specific) indications exist that automatic and manual scram were not successful.

DAEC EAL INFORMATION:

The DAEC EAL addresses conditions where failure of an automatic scram has occurred and manual actions performed at panel 1C05 to quickly shutdown the reactor do not meet the success criteria of IPOI 5 and the ATWS - RPV Control EOP, where power remains above 5% or boron injection is required.

Under the conditions of concern for this EAL, the reactor may be producing more heat than the maximum decay heat load for which the safety systems are designed. A Site Area Emergency is indicated because conditions exist that lead to imminent loss or potential loss of the primary containment and the fuel clad. In addition, if the SRV's are open, the RCS is no longer capable of retaining fission products and therefore is not acting as a fission product barrier. Although this EAL may be viewed as redundant to the Fission Barrier Table, its inclusion is necessary to better assure timely recognition and emergency response.

REFERENCES:

1. Integrated Plant Operating Instruction (IPOI) No. 5, Reactor Scram
2. ATWS Emergency Operating Procedure (EOP) - RPV Control
3. *NUMARC Methodology for Development of Emergency Action Levels NUMARC/NESP-007 Revision 2 Questions and Answers*, June 1993

SS3 Loss of All Vital DC Power

EVENT TYPE: Loss of Power

OPERATING MODE APPLICABILITY: Run, Startup, Hot Shutdown

EXAMPLE EMERGENCY ACTION LEVEL:

1. Loss of All Vital DC Power based on (site-specific) bus voltage indications for greater than 15 minutes.

DAEC EAL INFORMATION:

There is no significant deviation from the generic EAL. Under the conditions of concern, AOP 302.1, Loss of 125 VDC Power, would be entered under Tab 3, Complete Loss of 125 VDC. Consequently, the DAEC EAL addresses loss of both divisions of the 125V DC system consistent with AOP.

At DAEC, the 250V/125V DC Systems ensure power is available for the reactor to be shutdown safely and maintained in a safe condition. The 125V System is divided into two independent divisions - Division I and Division II - with separate DC power supplies. These power supplies consist of two separate 125V batteries and chargers serving systems such as RCIC, RHR, EDGs, and HPCI.

Complete loss of both 125V DC Divisions could compromise the ability to monitor and control the removal of decay heat during cold shutdown or refueling operations.

REFERENCES:

1. Abnormal Operating Procedure (AOP) 302.1, Loss of 125 VDC Power
2. Abnormal Operating Procedure (AOP) 388, Loss of 250 VDC Power
3. Technical Specification 3.8.B, DC Power Systems
4. UFSAR Section 8.3, Onsite Power Systems
5. UFSAR Table 8.3-6, Plant Battery System - DC Power, Instrumentation, and Control, Principle DC Loads (125V)

SS4 Complete Loss of Function Needed to Achieve or Maintain Hot Shutdown

EVENT TYPE: Inability to Maintain Shutdown Conditions

OPERATING MODE APPLICABILITY: Run, Startup, Hot Shutdown.

EXAMPLE EMERGENCY ACTION LEVEL:

1. <EOP Graph 4 Heat Capacity Limit is exceeded>

<OR>

<2. Reactor CANNOT be brought subcritical.>

DAEC EAL INFORMATION:

This EAL addresses complete loss of functions, including ultimate heat sink and reactivity control, required for hot shutdown with the reactor at pressure and temperature. Under these conditions, there is an actual major failure of a system intended for protection of the public. The reactivity condition criteria is addressed by maintenance of required shutdown margin. If inadvertent criticality could not be eliminated by performing the actions of AOP 255.1, AOP 255.2, or the ATWS EOP, it corresponds to a failure of a system intended for the protection of the public and thus classification as a Site Area Emergency is warranted.

This EAL represents an escalation from the conditions of concern in SA3, Inability to Maintain Cold Shutdown, because the reactor is at operating pressure and temperature and decay heat levels are higher.

Per DAEC Technical Specifications, the following systems are necessary to achieve or maintain Hot Shutdown conditions:

- Reactor Protection System Instrumentation (T.S. 3.1)
- Core and Containment Cooling Systems Instrumentation (T.S. 3.2)
- Reactivity Control (T.S. 3.3)
- Standby Liquid Control System (T.S. 3.4)
- Core and Containment Cooling Systems (T.S. 3.5)
- Primary System Boundary (T.S. 3.6)
- Auxiliary Electrical Systems (T.S. 3.8)

Loss of instrumentation is addressed by SS6, Inability to Monitor a Significant Transient in Progress, below. The Auxiliary Electrical System is addressed by SS1, Station Blackout, and SS3, Loss of 125V DC, above and are therefore not covered here. Failure of the primary system boundary is covered by the Fission Barrier Table and SU5, RCS Leakage, above.

REFERENCES:

1. Abnormal Operating Procedure (AOP) 149, Loss of Decay Heat Removal
2. Abnormal Operating Procedure (AOP) 255.1, Control Rod Movement/Indication Abnormal
3. Abnormal Operating Procedure (AOP) 255.2, Power/Reactivity Abnormal Change
4. Emergency Operating Procedure (EOP) 1 - RPV Control
5. ATWS Emergency Operating Procedure (EOP) - RPV Control
6. Emergency Operating Procedure ALC - Alternate Level Control
7. Emergency Operating Procedure (EOP) Basis, EOP Breakpoints
8. *NUMARC Methodology for Development of Emergency Action Levels NUMARC/NESP-007 Revision 2 Questions and Answers*, June 1993

SS5 Loss of Water Level in the Reactor Vessel That Has or Will Uncover Fuel in the Reactor Vessel

EVENT TYPE: Inability to Maintain Shutdown Conditions

OPERATING MODE APPLICABILITY: Cold Shutdown, Refuel

EXAMPLE EMERGENCY ACTION LEVEL:

1. Loss of Reactor Vessel Water Level as indicated by:
   a. Loss of all decay heat removal cooling as determined by (site-specific) procedure.

   AND

   b. (Site-specific) indicators that the core is or will be uncovered.

DAEC EAL INFORMATION:

There is no significant deviation from the generic EAL. The DAEC EAL is written in terms of the general concern that no cooling water source is lined up or available for injection into the RPV and water level is decreasing below the top of the active fuel (TAF). Under the conditions of concern for EAL 1, AOP 149, Loss of Decay Heat Removal, would be entered under Tab 1, Loss of Shutdown Cooling. Indications/alarms related to loss of shutdown cooling are displayed on control room panels 1C03 and 1C05 and are listed in the procedure. Consistent with the value used in the EOPs, the EAL uses an indicated RPV level of 15 inches for the water level corresponding to TAF.

REFERENCES:

1. Abnormal Operating Procedure (AOP) 149, Loss of Decay Heat Removal
2. Emergency Operating Procedure (EOP)-1, RPV Control, Sheet 1 of 1
3. Emergency Operating Procedure (EOP) Basis, EOP Breakpoints

SS6 Inability to Monitor a Significant Transient in Progress

EVENT TYPE: Instrumentation/Communication

OPERATING MODE APPLICABILITY: Run, Startup, Hot Shutdown

EXAMPLE EMERGENCY ACTION LEVEL:

1. The following conditions exist:
   a. Loss of < > annunciators associated with safety systems.
   AND
   b. Compensatory non-alarming indications are unavailable.
   AND
   c. Indications needed to monitor (site-specific) safety functions are unavailable.
   AND
   d. <Significant> transient in progress.

DAEC EAL INFORMATION:

The DAEC EAL is written in terms of a *significant transient* in progress with loss of both safety system annunciators and loss of compensatory non-alarming instrumentation. The DAEC EAL structure, which addresses all the key points in the generic EAL, better assures that the condition of concern for this EAL will be readily recognized.

Significant transient includes response to automatic or manually initiated functions such as scrams, runbacks involving greater than 25% thermal power change, ECCS injections, or thermal power oscillations of 10% or greater.

Control room panels 1C03, 1C04, and 1C05 contain the annunciators associated with safety systems at DAEC. Annunciators on 1C03, 1C04, and 1C05 share a common power supply from 125 VDC Division I that is fed through circuit breaker 1D13. Therefore, DAEC does not specify a loss of "most" annunciators as specified in the generic methodology.

Compensatory non-alarming indications include the plant process computer, SPDS, plant recorders, or plant instrument displays in the control room. These indications are needed to monitor (site-specific) safety functions that are of concern in the generic EAL.

Indications of loss of annunciators associated with safety systems include:

- 125 VDC charger, battery, or system annunciators on control room panel 1C08
- Loss of "sealed in" annunciators at affected panels
- Failure of affected annunciator panels shiftily testing by plant operators
- Expected alarms are not received
- Computer point ID B350 indicates "NSS ANN DC LOSS TRBL." (Loss of DC power to panels 1C03, 1C04, and 1C05)

REFERENCES:

1. Operating Instruction (OI) No. 317.2, Annunciator System
2. Abnormal Operating Procedure (AOP) 302.1, Loss of 125 VDC Power
3. Abnormal Operating Procedure (AOP) 302.2, Loss of Alarm Panel Power

SG1 Prolonged Loss of All Offsite Power and Prolonged Loss of All Onsite AC Power

EVENT TYPE: Loss of Power

OPERATING MODE APPLICABILITY: Run, Startup, Hot Shutdown

EXAMPLE EMERGENCY ACTION LEVEL:

1. Prolonged loss of all offsite and onsite AC power as indicated by:
   a. Loss of power to (site-specific) transformers.

   AND

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

   AND

   c. At least one of the following conditions exist:
      - Restoration of at least one emergency bus within (site-specific) hours is *NOT* likely

      OR

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

DAEC EAL INFORMATION:

There is no significant deviation from the generic EAL. Under prolonged Station Blackout (SBO) 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 EC/OSS a reasonable idea of how quickly a General Emergency should be declared based on the following considerations:

- Are there any present indications that core cooling is already degraded to the point where a General Emergency is IMMINENT (*i.e.*, loss of two barriers and a potential loss of the third barrier)?

- If there are presently no indications of degraded core cooling, how likely is it that power can be restored prior to occurrence of a General Emergency?

The first part of this EAL corresponds to the threshold conditions for Initiating Condition SS1, Station Blackout - namely, entry into AOP 301.1, Station Blackout. The second part of the EAL addresses the conditions that will escalate the SBO to General Emergency. Occurrence of any of the following is sufficient for escalation: (1) SBO coping capability exceeded, or (2) loss of drywell cooling that continues

to make RPV water level measurements unreliable, or (3) indications of inadequate core cooling. Each of these conditions is discussed below:

1. SBO Coping Capability Exceeded

DAEC has a SBO coping duration of four hours. *The likelihood of restoring at least one emergency 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.*

2. RPV Water Level Measurements Remaining Unreliable

Flashing of the reference leg water will result in erroneously high RPV water level readings giving a false indication of actual water inventory and potentially indicating adequate core cooling when it may not exist. EOP Graph 1, RPV Saturation Temperature, defines the conditions under which RPV level instrument leg boiling may occur.

3. Indications of Inadequate Core Cooling

DAEC uses the RPV level that is used for the Fuel Clad EAL 2 "potential loss" condition. This is RPV level below +15 inches.

REFERENCES:

1. Abnormal Operating Procedure (AOP) 301.1, Station Blackout
2. Letter NG-92-0283, John F. Franz, Jr. to Dr. Thomas E. Murley, Response to Safety Evaluation by NRC-NRR "Station Blackout Evaluation Iowa Electric Light and Power Company Duane Arnold Energy Center," February 10, 1992
3. Emergency Operating Procedure (EOP)1 - RPV Control
4. Emergency Operating Procedure (EOP) ALC - Alternate Level Control

SG2 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

EVENT TYPE: RPS Failure

OPERATING MODE APPLICABILITY: Run, Startup

EXAMPLE EMERGENCY ACTION LEVEL:

1. The following conditions exist:
   a. (Site-specific) indications exist that automatic and manual scram were NOT successful.

   AND

   b. Either of the following:
      - (Site-specific) indication exists that the core cooling is extremely challenged.

      OR

      - (Site-specific) indication exists that heat removal is extremely challenged.

DAEC EAL INFORMATION:

Automatic and manual scram are not considered successful if action away from the reactor control console is required to scram the reactor. Consistent with the EOPs, the ATWS conditions of concern in this EAL are reactor power that is expected to remain above 5% or that is indeterminate.

Escalation to the General Emergency classification requires extreme challenge to core or containment cooling, *i.e.*, *imminent* barrier loss. If the Main Condenser is available for steam release from the reactor, sufficient heat removal capability exists. However, without the main condenser being available as a heat sink, heat removal capability under these conditions is insufficient and a threat to the Fuel Clad barrier exists. In addition, the SRV's will lift and thus the RCS barrier will not retain fission products. Eventually, the torus water will be heated to the point where the containment function will become ineffective. Thus, the resultant combination of barrier conditions warrants a declaration of a General Emergency if ATWS reactor power-level control methods are ineffective in reducing reactor power level.

REFERENCES:

1. Emergency Operating Procedure ATWS EOP - RPV Control