50-331

ALLIANT ENERGY
DUANE ARNOLD ENERGY CENTER
TRANSMITTAL/ACKNOWLEDGMENT
MEMORANDUM

Manual #92
NRC-NRR/Document Control Desk
Washington, DC

DATE: __November 20, 2000__

Emergency Action Level

Please make the following changes to your Emergency Action Level Basis Document.

MANUAL CONTENTS	
Remove	**Insert**
EAL Basis Document INDEX, Rev. 6	**EAL Basis Document INDEX, Rev. 7**
EAL Basis Document INTRODUCTION, Rev. 0	**EAL Basis Document INTRODUCTION, Rev. 1**
EBD-A, Page 1 of 26 (no "Effective Date:")	**EBD-A, Page 1 of 26 ("Effective Date: 10/26/00")**
EBD-F, Rev. 2 (All Pages)	**EBD-F, Rev. 3 (All Pages)**

If you do not have an EAL Basis Document, return to Kathryn Dunlap.

Please acknowledge that the above action has been taken by signing below and returning this memorandum to K. Dunlap, 3313 DAEC Rd, Emergency Planning, PSC, Palo, IA 52324

PLEASE RETURN WITHIN 20 DAYS.

I have made the above revisions as indicated.

_____ _____

Print Name Signature/Date

A045

Effective Date: 11/20/2000

PROCEDURE	TITLE	REV #	REV. DATE
	Introduction	1	2/01/2000
	Definitions	1	2/01/2000
	Organization of Basis Information	1	2/01/2000
EBD-A	Abnormal Rad Levels/Radiological Effluent Category	2	10/26/2000
EBD-F	Fission Product Barrier Degradation Category	3	11/20/2000
EBD-H	Hazards and Other Conditions Affecting Plant Safety Category	2	11/02/2000
EBD-S	System Malfunction Category	2	9/15/2000

Effective Date: 2/1/00

Alliliant Energy 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 Plant*s, 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.

EAL BASES DOCUMENT	EBD-A Rev. 2
ABNORMAL RAD LEVELS/RADIOLOGICAL EFFLUENT CATEGORY	PAGE 1 of 26

Effective Date: 10/26/00

TECHNICAL REVIEW

Prepared by: _____ Date: 8/2/00

Reviewed by: _____ Date: 8/28/00
Independent Reviewer

PROCEDURE APPROVAL

I am responsible for the technical content of this procedure and for obtaining the necessary approval from the State and County Emergency Management officials prior to implementation.

Documentation of State and County Emergency Management approval is via NEP-2000-0081.

Approved by: _____ Date: 9-22-00
Manager, Emergency Planning

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Effective Date: 11-20-00

TECHNICAL REVIEW

Prepared by: _____ Date: 11-7-00

Reviewed by: _____ Date: 11-8-00
Independent Reviewer

PROCEDURE APPROVAL

I am responsible for the technical content of this procedure and for obtaining the necessary approval from the State and County Emergency Management officials prior to implementation.

Documentation of State and County Emergency Management approval is via NEP-2000-0114.

Approved by: Kevin Wellins for Date: 11/20/2000
Manager, Emergency Planning

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FU1 Any Loss or Any Potential Loss of Primary Containment Barrier

EVENT TYPE: See Fission Barrier Table

OPERATING MODE APPLICABILITY: Run, Startup, Hot Shutdown

EAL Threshold Values:

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

EAL Threshold Values:

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

EAL Threshold Values:

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.

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

EAL Threshold Values:

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

EAL THRESHOLD VALUE:
Clad Damage Determination

LOSS – Fuel Damage assessment procedures indicate at least 5% fuel clad damage.

POTENTIAL LOSS – Not 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.

REFERENCES:

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

FISSION BARRIER: Fuel Clad

DAEC INDICATOR: Radiation/Core Damage

EAL THRESHOLD VALUE:
Drywell/Torus Radiation Monitoring

LOSS - Valid Drywell Rad Monitor Reading GREATER THAN 7E+2 R/hr
OR
LOSS - Valid Torus Rad Monitor Reading GREATER THAN 3E+1 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

EAL THRESHOLD VALUE:
Primary Coolant Activity Level

LOSS - Coolant activity GREATER THAN 300 $\mu Ci/gm\ I_{131}$ dose equivalent.

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 $\mu Ci/gm\ I_{131}$ 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: RPV Level

EAL THRESHOLD VALUE:
Reactor Vessel Water Level

LOSS – RPV Level Below –25 Inches that cannot be restored.

POTENTIAL LOSS – RPV Level Below15 Inches that cannot be restored.

DAEC INFORMATION:

The loss indicator is based on a 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 -25 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 -25 inches that cannot be restored is used.

The potential 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 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/OSM Judgment

EAL THRESHOLD VALUE:
Emergency Director Judgment

Any condition which in the judgment of the Emergency Director that indicates LOSS or POTENTIAL LOSS of the FUEL CLAD barrier such as, but not limited to:

- Degraded *barrier monitoring* capability from loss of/lack of reliable indicators.
- Consideration for instrumentation operability.
- Portable instrumentation readings.
- Offsite monitoring results.
- Complete loss of 125 VDC.
- Loss of decay heat removal.
- ATWS with failure of Standby Liquid Control.
- Prolonged station blackout.
- Loss of offsite power with early HPCI/RCIC failure

DAEC INFORMATION:

There is no significant deviation from the generic indicator. Per EPIP 2.5, Control Room Emergency Response Operation, the Emergency Coordinator/Operations Shift Manager (EC/OSM) performs the emergency director function at DAEC. EC/OSM 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/OSM 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/OSM when determining if a fission barrier loss or potential loss has occurred.

This assessment should also include consideration for instrumentation operability and portable instrumentation readings.

Offsite monitoring results may be an indication of Fission Product Barrier degradation causing an unmonitored release.

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/OSM should also consult System Malfunction EALs, as appropriate, to assure timely emergency classification declaration.

REFERENCES:

1. Emergency Plan Implementing Procedure (EPIP) 2.5, Control Room Emergency Response Operation
2. Duane Arnold Energy Center Individual Plant Examination (IPE) November 1992

Fuel Clad Barrier
EC/OSM Judgement

FISSION BARRIER: RCS

DAEC INDICATOR: Radiation/Core Damage

EAL THRESHOLD VALUE:
Drywell Radiation Monitoring

LOSS - Valid Drywell Rad Monitor Reading GREATER THAN 5 R/hr after Reactor Shutdown

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 5 R/hr 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

(2.1×10^4) R/hr x $[(1 \times 10^{-2})$ percent / 100 percent$] = (2.1) \times 10^{4-4}$ R/hr = 2.1×10^0 R/hr = 2 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 3.4.5. Drywell Leak Detection Instrumentation

FISSION BARRIER: RCS

DAEC INDICATOR: RPV Level

EAL THRESHOLD VALUE:
Reactor Vessel Water Level

LOSS – RPV Level LESS THAN 15 Inches

POTENTIAL LOSS - Not applicable

DAEC INFORMATION:

There is no significant deviation from the generic indicator. This loss indicator corresponds to the water level at the top of the active fuel (TAF). In order to provide normal means to cool the fuel, water level must be maintained above the top of active fuel otherwise extraordinary means must be taken to assure that adequate core cooling exists. In certain failure event sequences reactor vessel water level may be procedurally lowered to the top of active fuel and the reactor coolant system depressurized to allow for steam cooling of the core. Even though fuel clad damage is not predicted under these conditions several safety system failures need to have occurred to reach the condition where steam cooling would be procedurally required. Therefore this is indicative of a loss of the reactor coolant system boundary. Water levels below this value indicate a challenge to core cooling which is a precursor to more serious events.

REFERENCES:

1. Emergency Operating Procedures (EOP) Basis, Breakpoints

FISSION BARRIER: RCS

DAEC INDICATOR: Leakage

EAL THRESHOLD VALUE:
RCS Leak Rate

LOSS – Not Applicable

POTENTIAL LOSS - RCS leakage GREATER THAN 50 GPM inside the drywell
OR
POTENTIAL LOSS - Unisolable primary system leakage outside drywell as indicated by valid area temperatures or area rad monitor readings above Max Normal Limits per Table 6, EOP 3.

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 is considered a Potential loss of RCS based on RCS leakage outside the drywell. Site specific RCS leakage is determined from temperature or area radiation alarms (ARMs) exceeding the Max Normal limits listed in Table 6, EOP 3. Unisolable primary system leakage in the areas of the steam tunnel, main turbine generator, RCIC, HPCI, etc., indicate a direct path from the RCS to areas outside primary containment. It should be confirmed that the indicators are caused by RCS leakage. Area temperatures or area radiation alarms above Max Normal limits are the criteria for declaration of an Alert classification. An unisolable leak which is indicated by exceeding Max Safe limits escalates to a Site Area Emergency when combined with Primary Containment Barrier loss (after a containment isolation) and a General Emergency when the Fuel Clad Barrier criteria is also exceeded.

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. *NEI Methodology for Development of Emergency Action Levels Revision 4*, May 1999

FISSION BARRIER: RCS

DAEC INDICATOR: Primary Containment Atmosphere

EAL THRESHOLD VALUE:
Drywell Pressure

LOSS - Valid Drywell High Pressure Reading of 2 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 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: Emergency Director Judgment

EAL THRESHOLD VALUE:

Any condition which in the judgment of the Emergency Director that indicates LOSS or POTENTIAL LOSS of the RCS barrier such as, but not limited to:

- Degraded *barrier monitoring* capability from loss of/lack of reliable indicators.
- Consideration for instrumentation operability.
- Portable instrumentation readings.
- Offsite monitoring results.
- Complete loss of 125 VDC.
- Loss of decay heat removal.
- ATWS with failure of Standby Liquid Control.
- Prolonged station blackout.
- Loss of offsite power with early HPCI/RCIC failure.

DAEC INFORMATION:

There is no significant deviation from the generic EAL. Per EPIP 2.5, Control Room Emergency Response Operation, the Emergency Coordinator/Operations Shift Manager (EC/OSM) performs the emergency director function at DAEC. EC/OSM 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/OSM 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/OSM when determining if a fission barrier loss or potential loss has occurred.

This assessment should also include consideration for instrumentation operability and portable instrumentation readings.

Offsite monitoring results may be an indication of Fission Product Barrier degradation causing an unmonitored release.

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/OSM should also consult System Malfunction EALs, as appropriate, to assure timely emergency classification

For the RCS barrier, the EC/OSM 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/OSM 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) 2.5, Control Room Emergency Response Operation
2. Duane Arnold Energy Center Individual Plant Examination (IPE) November 1992
3. *NEI Methodology for Development of Emergency Action Levels NUMARC/NESP-007 Revision 4*, May 1999

FISSION BARRIER: Primary Containment

DAEC INDICATOR: Radiation/Core Damage

EAL THRESHOLD VALUE:
Significant Radioactive Inventory in Containment

LOSS - Not applicable

POTENTIAL LOSS - Drywell Rad Monitor reading GREATER THAN 3E+3 R/hr
OR
POTENTIAL LOSS - Torus Rad Monitor reading GREATER THAN 1E+2 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

EAL THRESHOLD VALUE:
Clad Damage Determination

LOSS – Not Applicable

POTENTIAL LOSS - Fuel Damage assessment procedures indicate at least 20% fuel clad damage.

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

EAL THRESHOLD VALUE:
Reactor Vessel Water Level

LOSS - Not applicable

POTENTIAL LOSS - RPV level less than –39 inches 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 -39 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 -39 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. *NEI Methodology for Development of Emergency Action Levels NUMARC/NESP-007 Revision 4*, May 1999

FISSION BARRIER: Primary Containment

DAEC INDICATOR: Leakage

EAL THRESHOLD VALUE:
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.

OR

LOSS -Unisolable primary system leakage outside drywell as indicated by valid area temperatures or area rad monitor readings above Max Safe Limits per EOP 3, Table 6.

OR

LOSS -Intentional venting per EOPs

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.

This EAL is intended to cover the inability to isolate the containment when containment isolation is required. In addition, the presence of area radiation or temperature alarms above the Max Safe limits listed in Table 6, EOP 3 after a containment isolation, indicate an unisolable primary system leakage outside the drywell. The indicators should be confirmed to be caused by RCS leakage. Also, an intentional venting of primary containment for pressure control per EOPs to the secondary containment and/or the environment is considered a loss of containment. Containment venting for temperature or pressure when not in an accident situation should not be considered.

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

EAL THRESHOLD VALUE:
Drywell Pressure/Atmosphere

LOSS – Rapid unexplained decrease following initial increase.
OR
LOSS – Drywell pressure response not consistent with LOCA conditions.
OR
POTENTIAL LOSS – Torus Pressure of 53 PSIG.
OR
POTENTIAL LOSS – 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% providing the possibility of an explosive mixture.

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" 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/OSM Judgment

EAL THRESHOLD VALUE:

Any condition which in the judgment of the Emergency Director that indicates LOSS or POTENTIAL LOSS of the Primary Containment Barrier such as, but not limited to:

- Degraded *barrier monitoring* capability from loss of/lack of reliable indicators.
- Consideration for instrumentation operability.
- Portable instrumentation readings.
- Offsite monitoring results.
- Complete loss of 125 VDC.
- Loss of decay heat removal.
- ATWS with failure of Standby Liquid Control.
- Prolonged station blackout.
- Loss of offsite power with early HPCI/RCIC failure

DAEC INFORMATION:

There is no significant deviation from the generic indicator. Per EPIP 2.5, Control Room Emergency Response Operation, the Emergency Coordinator/Operations Shift Manager (EC/OSM) performs the emergency director function at DAEC. EC/OSM 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/OSM 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/OSM when determining if a fission barrier loss or potential loss has occurred.

This assessment should also include consideration for instrumentation operability and portable instrumentation readings.

Primary Containment Barrier
EC/OSM Judgement

Offsite monitoring results may be an indication of Fission Product Barrier degradation causing an unmonitored release.

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/OSM should also consult System Malfunction EALs, as appropriate, to assure timely emergency classification

REFERENCES:

1. Emergency Plan Implementing Procedure (EPIP) 2.5, Control Room Emergency Response Operation
2. Duane Arnold Energy Center Individual Plant Examination (IPE) November 1992