PECO NUCLEAR
A Unit of PECO Energy

10CFR50, Appendix E

Nuclear Group Headquarters
200 Exelon Way
Kennett Square, PA 19348

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

Subject: Peach Bottom Atomic Power Station, Units 2 & 3
Emergency Response Procedure Revisions

Dear Sir/Madam:

Enclosed are the following procedure revisions to the Emergency Response Procedures (ERPs) for Peach Bottom Atomic Power Station (PBAPS), Units 2 and 3. The procedures are required to be submitted within thirty (30) days of their revision in accordance with 10CFR50, Appendix E, and 10CFR50.4.

- ERP-101 Bases, Revision 0, "PBAPS EAL Technical Basis Manual"
- ERP-680, Revision 7, "Control of Thyroid Blocking Potassium Iodide (KI) Tablets"
- ERP-700, Revision 10, "Technical Support Team"

Also, enclosed is a copy of a computer generated report index identifying the latest revisions of the PBAPS ERPs.

If you have any questions or require additional information, please do not hesitate to contact us.

Very truly yours,

G. C. Heller /for

James A. Hutton
Director - Licensing

Attachments

cc: H. J. Miller, Administrator, Region I, USNRC (2 copies)
A. C. McMurtray, USNRC Senior Resident Inspector, PBAPS

A045

PECO NUCLEAR
PEACH BOTTOM UNITS 2 AND 3
EMERGENCY RESPONSE PROCEDURE

ERP-680 CONTROL OF THYROID BLOCKING POTASSIUM IODIDE (KI) TABLETS

1.0 RESPONSIBILITIES

1.1 The Emergency Director (ED) is responsible for authorizing the use of Potassium Iodide (KI). This is a non-delegable responsibility of the ED.

1.2 The Health Physics Team Leader (HPTL)is responsible for recommending when potassium iodide (KI) administration is warranted, and advises the ED.

1.3 The HPTL is responsible for distribution and administration of KI tablets.

1.4 The Dose Assessment Coordinator is responsible for performing calculations for issuance of KI to field survey personnel.

CAUTION

1. THE TAKING OF KI TABLETS IS STRICTLY VOLUNTARY FOR EACH INDIVIDUAL. HOWEVER, ONCE ADMINISTERED, DOSAGE SHOULD CONTINUE FOR A MINIMUM OF THREE DAYS BUT, PREFERABLY FOR 10 CONSECUTIVE DAYS, UNLESS WHOLE BODY COUNTING VERIFIES THE ABSENCE OF RADIOACTIVE IODINE IN THE BODY.
2. PERSONNEL HAVING KNOWN ALLERGY REACTIONS TO IODINE SHALL NOT BE ADMINISTERED KI UNLESS ABSOLUTELY NECESSARY AND ONLY WITH SPECIFIC MEDICAL DIRECTION.
3. KI IS MOST EFFECTIVE IF ADMINISTERED WITHIN 1 HOUR OF EXPECTED EXPOSURE OR SHORTLY AFTER EXPOSURE BEGINS. USE SEVERAL HOURS BEFORE EXPECTED EXPOSURE WILL SIGNIFICANTLY REDUCE EFFECTIVENESS OF PROTECTIVE EFFECT.

2.0 INITIAL ACTIONS

2.1 The HPTL shall:

2.1.1 Determine the need for administering KI by completing or reviewing ERP-680, Appendix 1, "Potassium Iodide Worksheet".

2.1.2 Recommend to the ED that KI administration will be beneficial in reducing projected thyroid CDE and request authorization to administer KI.

2.1.3 Upon ED authorization to issue KI, assign an individual(s) to be responsible for KI distribution and administration.

2.1.3.1 Direct that KI be administered using ERP-680, Appendix 4, "KI Authorization".

2.1.4 Ensure the names of each individual approved for KI administration is provided to the appropriate group leaders and direct them to send these individuals to the personnel assigned distribution and administration responsibilities.

> NOTE:
>
> NO CREDIT IS GIVEN OR ALLOWED FOR KI USE IN EVALUATION FOR COMPLIANCE WITH THE NRC EXPOSURE LIMIT.

2.1.5 Consider whether the dose contribution from exposure to airborne radionuclides will require the need for Emergency Dose Authorization. Advise the ED, per ERP-670, "Emergency Radiation Exposure Guidelines and Controls".

2.2 The DAC shall:

> NOTE:
>
> THE FOLLOWING STEPS MAY BE PERFORMED BY THE HPTL OR OTHER DESIGNEE IF THE DAC HAS OTHER PRIORITIES.

2.2.1 <u>IF</u> requested by the Field Survey Group Leader (FSGL) or Dose Assessment Team Leader (DATL), to evaluate KI administration for field teams, <u>THEN</u> complete ERP-680, Appendix 1, "Potassium Iodide Worksheet".

2.2.2 Submit completed calculation to the HPTL and obtain HPTL and ED approval for issuance.

2.2.3 Advise the FSGL or DATL when KI issuance has been approved.

2.3 The ED shall:

2.3.1 Evaluate the HPTL recommendation and review the data from Appendix 1, "Potassium Iodide Worksheet".

2.3.2 If appropriate, authorize the distribution of KI by signing Appendix 4.

3.0 CONTINUING ACTIONS

3.1 Individual responsible for KI distribution and administration shall:

3.1.1 Assemble the personnel to be treated.

3.1.2 Obtain an adequate supply of tablets from:

a. OSC Equipment Locker
b. Unit #1 Emergency Equipment Room
c. Field Survey Kits
d. Evacuation Assembly Area Kit (guardhouse exit)

3.1.3 Brief personnel taking KI concerning the following and obtain their signature on ERP-680, Appendix 2, "Potassium Iodide Consent Form".

3.1.3.1 That taking KI is strictly voluntary for each individual.

3.1.3.2 That side effects noticed shall be reported immediately.

3.1.4 Discuss cases of individuals with known allergy to iodine with the HPTL. If possible, these individuals should not be assigned to duties where radioiodine exposure is likely.

3.1.5 Administer tablets to personnel who already have been exposed to radioiodine first or, preceding exposure, preferably no more than 1 hour before expected exposure.

3.1.6 Provide each individual receiving KI with a copy of ERP-680, Appendix 3, "Instruction and Record Sheet".

3.1.7 Inform the HPTL when completed.

3.1.8 Inform the HPTL of any reported side effects.

3.2 The HPTL shall notify the Medical Director of all reported side effects.

4.0 FINAL CONDITIONS

4.1 The HPTL shall ensure that:

4.1.1 Thyroid uptake of iodine is evaluated and resultant radiation doses estimated and entered into personnel monitoring records.

4.1.2 Reports and evaluations are completed and any exposure in excess of the applicable limits in 10CFR20.2203 are reported to the NRC pursuant to 10CFR20.2204.

4.1.3 Exposure data is reported to the individual pursuant to 10CFR19.13.

4.2 The ED shall verify reports required by the Reportability Reference Manual are initiated.

5.0 ATTACHMENTS AND APPENDICES

5.1 Appendix 1 - "Potassium Iodide Worksheet"

5.2 Appendix 2 - "Potassium Iodide Consent Form"

5.3 Appendix 3 - "Instruction and Record Sheet"

5.4 Appendix 4 - "KI Authorization"

6.0 SUPPORTING INFORMATION

6.1 PURPOSE

This procedure provides guidelines for administration of potassium iodide (KI) as a radio-protective drug to emergency workers for protection against airborne radioiodine.

6.2 CRITERIA FOR USE

This procedure may be utilized at any emergency event classification or at the discretion of the ED whenever anticipated thyroid doses to emergency workers from radioiodines may exceed 10 rem.

6.3 SPECIAL EQUIPMENT

None

6.4 REFERENCES

6.4.1 Code of Federal Regulations, Title 10, Energy, Parts 19 and 20

6.4.2 ERP-301, "Dose Assessment Coordinator (DAC)"

6.4.3 NUREG-0654, section II.J.6.c

6.4.4 Nuclear Emergency Plan

6.4.5 ERP-600, "Health Physics Team Leader (HPTL)"

6.4.6 ERP-670, "Emergency Radiation Exposure Guidelines and Controls"

6.4.7 ERP-200, "Emergency Director (ED)"

6.5 COMMITMENT ANNOTATION

None

PECO NUCLEAR
PEACH BOTTOM UNITS 2 AND 3
EMERGENCY RESPONSE PROCEDURE

ERP-700 TECHNICAL SUPPORT TEAM

1.0 RESPONSIBILITIES

1.1 The Technical Support Team Leader (TSTL) is responsible for directing the activities of the Technical Support Team and advising the Emergency Director (ED) on technical matters.

1.2 The Technical Support Group Leader (TSGL) is responsible for supervising technical support engineering activities.

1.3 The Technical Support Team Members (TSTM) are responsible for analyzing plant conditions and providing recommendations for mitigating emergencies.

1.4 The Severe Accident Management Evaluators provide strategies and technical solutions for resolution of plant problems utilizing SAM plans and Technical Support Guidelines (TSG).

2.0 INITIAL ACTIONS

2.1 The Technical Support Team:

2.1.1 Reports to the Technical Support Center (TSC), obtain badge, dosimetry and sign in.

2.2 The TSTL and/or TSGL:

2.2.1 Sign in on status board in the TSC.

2.2.2 Start and maintain an Emergency Log, listing all significant actions, decisions, and communications, and their times.

2.2.3 Assign a TSTM to the position of NRC Communicator to perform the following per the EP Aid for the NRC Communicator:

a. Complete the Event Notification Worksheet.

b. Verify Emergency Response Data System (ERDS) activated and if not, activate ERDS link per ERP-110.

c. Relieve the Control Room NRC Communicator, with permission of the NRC.

2.2.4 Assign a TSTM to the position of Control Room Communicator to perform the following:

a. Report to the Control Room and obtain a cordless headset or alternate communications preferably from the PRO desk and maintain open communications with the TSC Status Board Keeper.

2.2.5 Assign one or more TSTM to the position of Status Board Keepers to perform the following:

a. Establish and maintain open communications with the Control Room Communicator.

b. Maintain and update assigned status boards with information obtained from the Control Room Communicator, the TSTL, the TSGL, and/or the Plant Monitoring System.

c. Continuously seek the most up-to-date, accurate information available on both plant and emergency conditions.

2.2.6 Verify personnel available with expertise in core/thermal hydraulics, electrical engineering, and mechanical engineering.

2.2.7 IF plant conditions warrant,
THEN assign a qualified TSTM to the position of Severe Accident Management Evaluator. (Qualified Severe Accident Management Evaluators are listed in the ERO Directory on the ED Communicator's table.)

2.2.8 Direct additional personnel call-outs as necessary utilizing the ERO Directory on the ED Communicator's table.

2.2.9 Brief the Technical Support Team on plant conditions and preliminary problem solving strategies.

2.3 TSTL:

2.3.1 Report status of the Technical Support Team activation to the Emergency Director (ED).

3.0 CONTINUING ACTIONS

3.1 TSTL:

3.1.1 Communicate engineering activities by priority from the ED or assistant ED to the Status Board Keeper and the TSGL.

3.1.2 Submit Emergency Special Procedures to the ED for review and authorization.

3.1.3 Periodically, when requested, provide briefings to the TSC on progress and strategies being utilized to mitigate the emergency.

3.2 TSTL and/or TSGL:

3.2.1 Coordinate both the preparation and review of any Emergency Special Procedures as needed utilizing the Emergency Special Procedure book.

3.2.2 If the EOF is activated, maintain communications with the EOF Engineering Support Team utilizing the Status Board Keeper conference circuit #36. (Additional EOF engineering support telephone numbers are available in the "Peach Bottom ERO Facility Directory".)

3.2.3 Ensure all recommendations from the EOF Engineering Support Team are reviewed and authorized by the Emergency Director.

3.2.4 Periodically provide briefings to the Technical Support Team on the status of the emergency.

3.3 Severe Accident Management Evaluators:

3.3.1 Evaluate plant symptoms and provide recommendations for potential strategies utilizing the Severe Accident Management Plans (SAMP) and Technical Support Guidelines (TSG).

4.0 FINAL CONDITIONS

4.1 TSTL and TSGL:

4.1.1 <u>WHEN</u> informed by the ED of termination or recovery,
<u>THEN</u> deactivate the Technical Support Team.

4.2 NRC Communicator:

4.2.1 <u>WHEN</u> permitted by the NRC,
<u>THEN</u> de-activate the ERDS link per ERP-110 and hang up the ENS phone.

4.3 Control Room Communicator:

4.3.1 Debrief with the TSTL or TSGL.

4.4 Status Board Keepers:

4.4.1 Retain necessary information and clean status boards unless otherwise directed.

4.5 Technical Support Team Members:

4.5.1 Debrief with the TSTL or TSGL and return procedures and drawings to their stored location.

4.5.2 Forward all records and documents to an Emergency Preparedness Coordinator for review and filing.

5.0 ATTACHMENTS AND APPENDICES

None

6.0 SUPPORTING INFORMATION

6.1 PURPOSE

To provide guidelines for the activities of the Technical Support Team.

6.2 CRITERIA FOR USE

This procedure shall be implemented at the Alert or higher emergency classification, or at the discretion of the ED.

6.3 SPECIAL EQUIPMENT

None

6.4 REFERENCES

6.4.1 ERP-110, "Emergency Notifications"

6.4.2 Nuclear Emergency Plan

6.4.3 NEI 91-04, "Severe Accident Issue Closure Guidelines"

6.4.4 Severe Accident Management Plans (SAMP)

6.4.5 Technical Support Guidelines (TSG)

6.5 COMMITMENT ANNOTATION

None

Effective 9/22/00

PBAPS EAL Technical Basis Manual
Table of Contents

Section I - Introduction

This manual contains the technical basis for the Emergency Action Levels as utilized in ERP-101, Classification of Emergencies. The format and use of this manual is as follows.

1. Heading and Sub-Heading

There are nine major headings each containing one or more sub-headings. These are as follows:

1.0 Reactor Fuel
- 1.1 Coolant Activity
- 1.2 Irradiated Fuel or New Fuel

2.0 Reactor Pressure Vessel
- 2.1 Reactor Water Level
- 2.2 Reactor Power

3.0 Fission Product Barrier
- 3.1 Initiating Condition Matrix
- 3.2 Fuel Clad Barrier Thresholds
- 3.3 Reactor Coolant System Barrier Thresholds
- 3.4 Primary Containment Barrier Thresholds
- 3.5 Fission Product Barrier Table

4.0 Secondary Containment Bypass
- 4.1 Main Steam Line

5.0 Radioactivity Release
- 5.1 Effluent Release and Dose
- 5.2 In-Plant Radiation

6.0 Loss of Power
- 6.1 Loss of AC or DC Power

7.0 Internal Events
- 7.1 Technical Specifications & Control Room Evacuation
- 7.2 Loss of Decay Heat Removal Capability
- 7.3 Loss of Assessment/Communications Capability

8.0 External Events
- 8.1 Security Events
- 8.2 Fire/Explosion and Toxic/Flammable Gases
- 8.3 Man-Made Events
- 8.4 Natural Events

9.0 Other
- 9.1 General

2. Emergency Classification Level and Number Identification

The classifications range from Unusual Event through Alert, Site Area Emergency to General Emergency. For each sub-heading, there may not be an EAL in every classification level. Each EAL is individually and uniquely numbered. No two numbers are the same.

3. INITIATING CONDITION

The Initiating Condition or IC (as described in NUMARC NESP-007) is contained in this section. ICs are a predetermined subset of conditions where either the potential exists for a radiological emergency or such an emergency has occurred. Additionally, ICs are the means by which EALs for different nuclear power plants are standardized.

4. EAL

Each Emergency Action Level exactly as it is contained in ERP-101.

5. MODE

The mode that the EAL is applicable in is contained here. There are six MODEs (1, 2, 3, 4 and 5 and defueled) that are used. PBAPS also uses mode switch position. These positions are stated below and are Run, Startup, Shutdown and Refueling. It should be noted that these MODEs are entry level conditions. The EAL is applicable if the plant was in the MODE at the start of the event. Subsequent positions of the mode selector switch should be ignored for purposes of classification.

MODE	MODE SWITCH POSITION
1	Run
2	Startup
3	Shutdown (hot)
4	Shutdown (cold)
5	Refueling
D	N/A (defueled)

6. BASIS

The technical basis of each EAL is contained in this section. This includes any necessary calculations and also includes escalation references.

7. DEVIATION

Any deviations from the NUMARC NESP-007 methodology are contained in this section. If there are no deviations, NONE is used.

8. REFERENCES

All applicable references used in developing the technical basis for each EAL are contained in this section.

9. GENERAL EAL IMPLEMENTATION PHILOSOPHY

The following guidance is provided to describe the philosophy used in the implementation of ERP-101 by the Emergency Director (ED) in making emergency classifications. **CM-1** (ERP-101)

In most cases, the emergency classification process is a straight-forward comparison of important plant parameters to the emergency action levels (EAL's). The instruments and annunciators referred to in the Emergency Classification Tables are presented as primary indicators and should be validated by plant conditions or event conditions.

A broad spectrum of discretion in classifying events is provided to the ED under the "General Conditions" category. In using the "General Conditions" category and in classifying emergencies under circumstances which are not straight-forward use of the EAL's, the ED should be mindful than an approach is needed which is conservative with respect to public, plant, and personnel safety and with respect to ensuring the adequacy of personnel and technical support. Conservative decisions must be made if the ED has any doubt regarding the health and safety of the public.

The ED should be mindful that declaring Unusual Events provide the Company and off-site agencies the opportunity for early information regarding the event and for early activation of resources and may be considered a "no consequence decision." Conversely, not declaring an Unusual Event when there is credible (but, not clear) bases for doing so, would appear to be less than open or candid and could have serious adverse consequences. Although the consequences of declaring an Unusual Event are limited, inappropriate classifications do not accurately indicate the significance of the event to the public and emergency responders and should be avoided.

At the Alert, Site Area and General Emergency levels, clearly the threat to the plant and to the public is at a heightened level. Rapid application of resources and preparation for providing for the public health and safety are appropriate. Because of the magnitude of resource mobilization and the potential disruption of normal public activities, an overly conservative or an inappropriately early declaration of these levels is not advisable.

Events that meet the Emergency Action Level criteria for event declaration, but which are terminated before they are identified and declared, should still be classified and reported, but not declared to implement the Emergency Plan.

All EAL's may not consider trends, rates of change, or status changes in equipment availability. In the event of rapidly changing parameters trending toward an increased emergency classification, the ED can appropriately decide that the higher level EAL will be exceeded and escalate the classification early. In the event of trends toward a decreased emergency classification, parameter values must be below the EAL's to de-escalate.

In the event of a "spike" which rapidly exceeds and then decreases below an EAL, entry into the Emergency Plan or escalation to the higher classification "in retrospect" is not appropriate unless the "spike" is indicative of continuing degrading conditions which will lead to an escalated emergency classification level. This statement does not apply if the EAL includes a "spike". Spurious alarms or parameters which are known to be invalid indicators of actual plant conditions or of the emergency classification, should not be used to declare emergency classifications.

Section II - Acronyms

AC	-	Alternating Current
ADS	-	Automatic Depressurization System
APRM	-	Average Power Range Monitor
ARI	-	Alternate Rod Insertion
ARM	-	Area Radiation Monitor
ATWS	-	Anticipated Transient Without Scram
BRP	-	Bureau of Radiation Protection
CAC	-	Containment Atmosphere Control
CAD	-	Containment Atmosphere Dilution
CDE	-	Committed Dose Equivalent
CFM	-	Cubic Feet Per Minute
CFR	-	Code of Federal Regulations
CRD	-	Control Rod Drive
CS	-	Core Spray
DBA	-	Design Basis Accident
DC	-	Direct Current
DEI	-	Dose Equivalent Iodine
EAL	-	Emergency Action Level
ECCS	-	Emergency Core Cooling Systems
ECW	-	Emergency Cooling Water
EDG	-	Emergency Diesel Generator
EPA	-	Environmental Protection Agency
ERP-C	-	Emergency Response Procedure - Common
ESW	-	Emergency Service Water
FC	-	Fuel Clad (Barrier)
FTS	-	Federal Telephone System
GPM	-	Gallons Per Minute
HCTL	-	Heat Capacity Temperature Limit
HPCI	-	High Pressure Coolant Injection
HPSW	-	High Pressure Service Water
IC	-	Initiating Condition
IRM	-	Intermediate Range Monitor
KV	-	KiloVolt
LCO	-	Limiting Condition for Operation
LOCA	-	Loss of Coolant Accident
LPCI	-	Low Pressure Coolant Injection
MPH	-	Miles Per Hour
mR/hr	-	Milli Roentgen Per Hour
MSIV	-	Main Steam Isolation Valve
NFPB	-	Normal Full Power Background
NPSH	-	Net Positive Suction Head
NRC	-	Nuclear Regulatory Commission
NUMARC	-	Nuclear Management and Resources Council
ODCM	-	Offsite Dose Calculation Manual
OPCON	-	Operating Condition
PBAPS	-	Peach Bottom Atomic Power Station
PEMA	-	Pennsylvania Emergency Management Agency
PC	-	Primary Containment (Barrier)
PCIS	-	Primary Containment Isolation System
PSIG	-	Pounds Square Inch Gauge
RC	-	Reactor Coolant (Barrier)
RCIC	-	Reactor Core Isolation Cooling

RCS	-	Reactor Coolant System
RHR	-	Residual Heat Removal
RPS	-	Reactor Protection System
RPV	-	Reactor Pressure Vessel
SBGTS	-	Standby Gas Treatment System
SBO	-	Station Blackout
SJAE	-	Steam Jet Air Ejector
SRM	-	Source Range Monitor
SRV	-	Safety Relief Valve
TAF	-	Top of Active Fuel
TPARD	-	Total Protective Action Recommendation Dose
TRIPs	-	Transient Response Implementation Plan Procedures
μCi/cc	-	Micro Curie Per Cubic Centimeter
μCi/gm	-	Micro Curie Per Gram
UFSAR	-	Updated Final Safety Analysis Report
VDC	-	Volts Direct Current

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Section III - EAL Technical Basis

1.0 Reactor Fuel

1.1 Coolant Activity

UNUSUAL EVENT - 1.1.1.a

IC Fuel Clad Degradation

EAL

Reactor Coolant activity **> *4 μCi/gm*** Dose Equivalent Iodine 131

MODE All

BASIS

Coolant activity in excess of Technical Specifications (> 4 μCi/gm) is considered to be a precursor of more serious problems. The Technical Specification limit reflects a degrading or degraded core condition. This level is chosen to be above any possible short duration spikes under normal conditions. An Unusual Event is only warranted when actual fuel clad damage is the cause of the elevated coolant sample (as determined by laboratory confirmation). However, fuel clad damage should be assumed to be the cause of elevated Reactor Coolant activity unless another cause is known, e.g., Reactor Coolant System chemical decontamination evolution (during shutdown) is ongoing with resulting high activity levels.

This event will be escalated to an Alert when Reactor Coolant activity exceeds 300 μCi/gm Dose Equivalent Iodine 131 per Fission Product Barrier Table.

DEVIATION

None

REFERENCES

Technical Specification Section 3.6.B
NUMARC NESP-007, SU4.2

1.0 Reactor Fuel

1.1 Coolant Activity

UNUSUAL EVENT - 1.1.1.b

IC Fuel Clad Degradation

EAL

SJAE Discharge Radiation > ***$2.5x10^3$ mR/hr***

MODE 1, 2, 3

BASIS

The steam jet air ejector discharge (Offgas) radiation monitor RR-2(3)-17-152 in the Control Room would be one of the first indicators of a degrading core. The high-high alarm is set at the Technical Specification limit of $2.5x10^3$ mR/hr. This instrument takes a sample before the recombiner. This indicator of elevated activity is considered to be a precursor of more serious problems. The Technical Specification limit reflects a degrading or degraded core condition.

Escalation of this IC to the Alert level is via the Fission Product Barrier Degradation Monitoring ICs.

DEVIATION

The MODE applicability [1,2,3] is a deviation from NUMARC [all] in that the SJAE Radiation Monitor and Main Steam Line Radiation Monitors will only be a valid indication of Fuel Clad Degradation in those MODE's. At Peach Bottom, there are no other monitors which can be an indicator of Fuel Clad Degradation. Degradation in cold shutdown or refueling will be first indicated by ventilation release monitor's which are covered by EAL on Effluent Release and Dose.

REFERENCES

Technical Specifications Section 3.8.C.7.a
NUMARC NESP-007, SU4.1

1.0 Reactor Fuel

1.2 Irradiated Fuel or New Fuel

UNUSUAL EVENT - 1.2.1.a

IC Unexpected Rise in Plant Radiation or Airborne Concentration.

EAL

Uncontrolled water level drop in the spent fuel pool with all irradiated fuel assemblies remaining covered by water

MODE All

BASIS

This event tends to have a long lead time relative to potential for radiological release outside the site boundary, thus impact to public health and safety is very low.

In light of Reactor Cavity Seal failure incidents at two different PWRs and loss of water in the Spent Fuel Pit/Fuel Transfer Canal at a BWR all occurring since 1984, explicit coverage of these types of events via this EAL is appropriate given their potential for increased doses to plant staff. Classification as an Unusual Event is warranted as a precursor to a more serious event.

This event will be escalated to an Alert as a result of uncovery of a fuel assembly and/or indication of high radiation levels on the refueling floor.

DEVIATION

None

REFERENCES

NUMARC NESP-007, AU2.2
Technical Specifications

1.0 Reactor Fuel

1.2 Irradiated Fuel or New Fuel

UNUSUAL EVENT - 1.2.1.b

IC Unexpected Rise in Plant Radiation or Airborne Concentration.

EAL

Unexpected Skimmer Surge Tank low level alarm **AND** Visual observation of an uncontrolled water level drop below the fuel pool skimmer surge tank inlet

MODE All

BASIS

A drop in the Spent Fuel Pool level or the RPV [when in refueling and flooded up with the gates removed] will result in a control room annunciator Fuel Pool Cooling and Cleanup System Trouble Alarm. This Control Room alarm directs an operator to be dispatched to a local alarm panel which will identify the Skimmer Surge Tank low level alarm. This alarm is validated with visual observation of a decreasing Spent Fuel Pool level. If the spent fuel pool level decreases below the inlet to the skimmer surge tank, without a planned event such as removing a large piece of equipment, there must be a leak in the spent fuel pool or the RPV. This event has a long lead time relative to potential for radiological release outside the site boundary, thus the impact to public health and safety is very low. Classification as an Unusual Event is warranted as a precursor to a more serious event.

In light of Reactor Cavity Seal failure incidents at two different PWRs and loss of water in the Spent Fuel Pit/Fuel Transfer Canal at a BWR all occurring since 1984, explicit coverage of these types of events via this EAL is appropriate given their potential for increased doses to plant staff. Classification as an Unusual Event is warranted as a precursor to a more serious event.

This event will be escalated to an Alert as a result of uncovery of a fuel assembly and/or indication of high radiation levels on the refueling floor.

DEVIATION

None

REFERENCES

NUMARC NESP-007, AU2.1

1.0 Reactor Fuel

1.2 Irradiated Fuel or New Fuel

UNUSUAL EVENT - 1.2.1.c

IC Unexpected Rise in Plant Radiation

EAL

Radiological readings exceed **600 mR/hr** one foot away <u>OR</u> **1200 mR/hr** at the external surface of any dry storage system

MODE All

BASIS

This EAL applies to potential emergency conditions which might develop during use of the Independent Spent Fuel Storage Installation and dry cask storage system. This EAL provides for an Unusual Event classification, which may be entered in the event that conditions occur which have the potential for damaging or degrading the fuel, but no releases of radioactive material requiring offsite response or monitoring are expected. Consistent with the NUMARC guidance, escalations above the Unusual Event are not warranted.

Accidents associated with the dry cask storage system include natural and man-made events that are postulated to affect the storage system. The limiting impacts to the system include loss of shielding capability and loss of confinement. The loss of shielding results in higher direct radiation to the environment from the cask while the loss of confinement results in a release of materials from within the cask to the environment at a postulated leak rate.

Loss of confinement for the dry storage system is evaluated in TN-68, Safety Analysis Report, Section 7. Two scenarios are considered, one for off-normal conditions and one for hypothetical accident conditions. Dose calculations are included in section 7.3.2.1. In the extremely unlikely event that one of these scenarios did occur, the event would be addressed by the Radioactivity Release EALs contained in Table 5.

Loss of shielding for the dry storage system is evaluated in TN-68, Safety Analysis Report, Section 5. Dose calculations are included in Table 5.1-2 for both normal and accident conditions. The value of **600 mR/hr** one foot away <u>OR</u> **1200 mR/hr** at the external surface are determined for several reasons. According to the TN-68, Safety Analysis Report, Table 5.1-2, Summary of Average Dose Rates, the maximum expected surface dose rates will be 529.5 mR/hr (see note 2). Consequently, the value of 1200 mR/hr is sufficiently above normal conditions as to preclude inappropriate classifications.

Also, the value of 1200 mR/hr is sufficiently below the 1467 mR/hr found in Table 5.1-2 for the cask surface radiological reading for accident conditions. Therefore, 1200 mR/hr from a loss of shielding accident would trigger an Unusual Event classification.

DEVIATION

None

REFERENCES

NUMARC NESP-007, AU2.3

1.0 Reactor Fuel

1.2 Irradiated Fuel or New Fuel

ALERT - 1.2.2.a

IC 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

EAL

Unplanned general area radiation **> *500 mR/hr*** on the refuel floor (Table 1-1)

MODE All

BASIS

This IC applies to spent fuel requiring water coverage and is not intended to address spent fuel which is licensed for dry storage, which is discussed in NUMARC/NESP-007 IC AU2, "Unexpected Rise in Plant Radiation or Airborne Concentration."

NUREG-0818, "Emergency Action Levels for Light Water Reactors," forms the basis for this EAL. The areas where Irradiated fuel is located forms the basis for the radiation monitors listed in Table 1-1.

Unexpected radiation levels which are at least 100 times higher than the normal background will generally indicate a fuel handling accident or loss of water covering the irradiated fuel. Readings may be from refuel floor Area Radiation Monitors or taken during a qualified radiological survey. Table 1-1 monitors are as follows:

Table 1-1 Refuel Floor ARMs

3-7 (7-9)	Steam Separator Pool
3-8 (7-10)	Refuel Slot
3-9 (7-11)	Fuel Pool
3-10 (7-12)	Refueling Bridge

There is time available to take corrective actions, and there is little potential for substantial fuel damage. In addition, NUREG/CR-4982, "Severe Accident in Spent Fuel Pools in Support of Generic Safety Issue 82," July 1987, indicates that even if corrective actions are not taken, no prompt fatalities are predicted, and that risk of injury is low. In addition, NRC Information Notice No. 90-08, "Kr-85 Hazards from Decayed Fuel" presents the following in its discussion:

> In the event of a serious accident involving decayed spent fuel, protective actions would be needed for personnel on site, while offsite doses (assuming an exclusion area radius of one mile from the plant site) would be well below the Environmental Protection Agency's Protective Action Guides. Accordingly, it is important to be able to properly survey and monitor for Kr-85 in the event of an accident with decayed spent fuel.
>
> Licensees may wish to reevaluate whether Emergency Action Levels specified in the emergency plan and procedures governing decayed fuel-handling activities appropriately focus on concern for onsite workers and Kr-85 releases in areas where

decayed spent fuel accidents could occur, for example, the spent fuel pool working floor. Furthermore, licensees may wish to determine if emergency plans and corresponding exposures of onsite personnel who are in other areas of the plant. Among other things, moving onsite personnel away from the plume and shutting off building air intakes downwind from the source may be appropriate.

Offsite doses during these accidents would be well below the EPA Protective Action Guidelines and the classification as an Alert is therefore appropriate. This radiation level could also be caused by an inadvertent criticality and is included even though the probability of this event occurring is low. Radiation increases above 500 mR/hr which were expected should not cause an Alert to be declared during a planned evolution. Additionally, surveys which identify "hot spots" greater than 500 mR/hr should not cause an Alert to be declared.

Escalation, if appropriate, would occur via Effluent Release, In-plant radiation, or Emergency Director Judgment.

DEVIATION

None

REFERENCES

NUMARC NESP-007, AA2.1
NUREG-1228, Source Term Estimation During Incident Response to Severe Nuclear Power Plant Accidents

1.0 Reactor Fuel

1.2 Irradiated Fuel or New Fuel

ALERT - 1.2.2.b

IC 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

EAL

Report of visual observation of irradiated fuel uncovered

MODE All

BASIS

This IC applies to spent fuel requiring water coverage and is not intended to address spent fuel which is licensed for dry storage, which is discussed in NUMARC/NESP-007 IC AU2, "Unexpected Rise in Plant Radiation or Airborne Concentration."

NUREG-0818, "Emergency Action Levels for Light Water Reactors," forms the basis for this EAL.

Studies of the loss of fuel pool water level indicate that a significant release may occur if rapid oxidation of the fuel clad occurs due to prolonged fuel uncovery. Offsite doses are not; however, expected to exceed EPA PAGs. In addition, NRC Information Notice No. 90-08, "Kr-85 Hazards from Decayed Fuel" presents the following in its discussion:

> In the event of a serious accident involving decayed spent fuel, protective actions would be needed for personnel on site, while offsite doses (assuming an exclusion area radius of one mile from the plant site) would be well below the Environmental Protection Agency's Protective Action Guides. Accordingly, it is important to be able to properly survey and monitor for Kr-85 in the event of an accident with decayed spent fuel.
>
> Licensees may wish to reevaluate whether Emergency Action Levels specified in the emergency plan and procedures governing decayed fuel-handling activities appropriately focus on concern for onsite workers and Kr-85 releases in areas where decayed spent fuel accidents could occur, for example, the spent fuel pool working floor. Furthermore, licensees may wish to determine if emergency plans and corresponding exposures of onsite personnel who are in other areas of the plant. Among other things, moving onsite personnel away from the plume and shutting off building air intakes downwind from the source may be appropriate.

Thus, an Alert Classification for this event is appropriate. Escalation, if appropriate, would occur via Effluent Release, In-plant radiation, or Emergency Director Judgment.

DEVIATION

None

REFERENCES

NUMARC NESP-007, AA2.2

1.0 Reactor Fuel

1.2 Irradiated Fuel or New Fuel

ALERT - 1.2.2.c

IC 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

EAL

Water Level ***< 458 " above RPV instrument zero*** for the Reactor Refueling Cavity that will result in Irradiated Fuel uncovering

MODE 5 (With Reactor Refueling Cavity Flooded)

BASIS

This IC applies to spent fuel requiring water coverage and is not intended to address spent fuel which is licensed for dry storage, which is discussed in NUMARC/NESP-007 IC AU2, "Unexpected Rise in Plant Radiation or Airborne Concentration."

NUREG-0818, "Emergency Action Levels for Light Water Reactors," forms the basis for this EAL.

There is time available to take corrective actions, and there is little potential for substantial fuel damage. In addition, NUREG/CR-4982, "Severe Accident in Spent Fuel Pools in Support of Generic Safety Issue 82," July 1987, indicates that even if corrective actions are not taken, no prompt fatalities are predicted, and that risk of injury is low. In addition, NRC Information Notice No. 90-08, "Kr-85 Hazards from Decayed Fuel" presents the following in its discussion:

> In the event of a serious accident involving decayed spent fuel, protective actions would be needed for personnel on site, while offsite doses (assuming an exclusion area radius of one mile from the plant site) would be well below the Environmental Protection Agency's Protective Action Guides. Accordingly, it is important to be able to properly survey and monitor for Kr-85 in the event of an accident with decayed spent fuel.
>
> Licensees may wish to reevaluate whether Emergency Action Levels specified in the emergency plan and procedures governing decayed fuel-handling activities appropriately focus on concern for onsite workers and Kr-85 releases in areas where decayed spent fuel accidents could occur, for example, the spent fuel pool working floor. Furthermore, licensees may wish to determine if emergency plans and corresponding exposures of onsite personnel who are in other areas of the plant. Among other things, moving onsite personnel away from the plume and shutting off building air intakes downwind from the source may be appropriate.

The value 458" above RPV instrument zero is the Tech. Spec. Limit and an uncontrolled level decrease that would uncover irradiated fuel is an indicator of a decrease in the level of safety of the plant.

Thus, an Alert Classification for this event is appropriate. Escalation, if appropriate, would occur via Effluent Release, In-plant radiation, or Emergency Director Judgment.

DEVIATION

The MODE applicability [5 With Reactor Refueling Cavity Flooded] is a deviation from NUMARC [all] in that the EAL is only applicable in that plant condition. This adds clarity to the EAL to ensure that it will not be applied under plant conditions where a classification is not warranted.

REFERENCES

NUMARC NESP-007, AA2.3
Tech Spec 3.9.6

1.0 Reactor Fuel

1.2 Irradiated Fuel or New Fuel

ALERT - 1.2.2.d

IC 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

EAL

Water Level ***< 232 ft 3 inches plant elevation*** for the Spent Fuel Pool that will result in Irradiated Fuel uncovering

MODE All

BASIS

This IC applies to spent fuel requiring water coverage and is not intended to address spent fuel which is licensed for dry storage, which is discussed in NUMARC/NESP-007 IC AU2, "Unexpected Rise in Plant Radiation or Airborne Concentration."

NUREG-0818, "Emergency Action Levels for Light Water Reactors," forms the basis for this EAL.

There is time available to take corrective actions, and there is little potential for substantial fuel damage. In addition, NUREG/CR-4982, "Severe Accident in Spent Fuel Pools in Support of Generic Safety Issue 82," July 1987, indicates that even if corrective actions are not taken, no prompt fatalities are predicted, and that risk of injury is low. In addition, NRC Information Notice No. 90-08, "Kr-85 Hazards from Decayed Fuel" presents the following in its discussion:

> In the event of a serious accident involving decayed spent fuel, protective actions would be needed for personnel on site, while offsite doses (assuming an exclusion area radius of one mile from the plant site) would be well below the Environmental Protection Agency's Protective Action Guides. Accordingly, it is important to be able to properly survey and monitor for Kr-85 in the event of an accident with decayed spent fuel.
>
> Licensees may wish to reevaluate whether Emergency Action Levels specified in the emergency plan and procedures governing decayed fuel-handling activities appropriately focus on concern for onsite workers and Kr-85 releases in areas where decayed spent fuel accidents could occur, for example, the spent fuel pool working floor. Furthermore, licensees may wish to determine if emergency plans and corresponding exposures of onsite personnel who are in other areas of the plant. Among other things, moving onsite personnel away from the plume and shutting off building air intakes downwind from the source may be appropriate.

The value 232 ft 3 inches plant elevation is the Tech. Spec. Limit and an uncontrolled level decrease that would uncover irradiated fuel is an indicator of a decrease in the level of safety of the plant.

Thus, an Alert Classification for this event is appropriate. Escalation, if appropriate, would occur via Effluent Release, In-plant radiation, or Emergency Director Judgment.

DEVIATION

None

REFERENCES

NUMARC NESP-007, AA2.4
Tech Spec 3.7.7

2.0 Reactor Pressure Vessel

2.1 Reactor Pressure Boundary

UNUSUAL EVENT - 2.1.1

IC Reactor Coolant System Leakage

EAL

The following conditions exist: Unidentified Primary System Leakage > ***10 gpm*** into the Drywell <u>OR</u> Identified Primary System Leakage > ***25 gpm*** into the Drywell

MODE 1, 2, 3, 4

BASIS

Utilizing the leak before break methodology, it is anticipated that there will be indication(s) of minor reactor coolant system boundary integrity loss prior to this fault escalating to a major leak or rupture. Detection of low levels of leakage while pressurized is utilized to monitor for the potential of catastrophic failures.

This EAL is included as an Unusual Event because it may be a precursor of more serious conditions and, as a result, it is considered to be a potential degradation of the level of safety of the plant. The value of 10 gpm unidentified leakage is significantly higher than the expected pressurized leak rate from the reactor coolant system. The 10 gpm value for the unidentified pressure boundary leakage was selected as it is twice the Technical Specification value, indicating an increase beyond that assumed in Safety Analysis. It also is observable with normal control room indications. The EAL for identified leakage is set at a higher value (25 gpm) due to the lesser significance of identified leakage in comparison to unidentified or pressure boundary leakage.

Technical Specification LCO required actions would necessitate a plant shutdown and subsequent depressurization, unless the source of the leak can be isolated, identified, and/or stopped. Actions initiated by plant staff would include close monitoring of the calculated break size such that any sudden or gradual increase in leak rate would be identified. A slow power reduction and gradual depressurization would be necessitated due to the possibility that a sudden power and/or pressure surge could potentially worsen the break or cause a catastrophic failure.

The leak rate of 10 gpm may cause a high drywell pressure indication. Other indications of a leak of this magnitude would include an increase in drywell temperature or radiation.

This event will escalate to an Alert based upon high Drywell pressure per Fission Product Barrier Table.

DEVIATION

NUMARC Example EAL SU5.1.a identifies pressure boundary leakage. There is no Peach Bottom EAL listed for pressure boundary leakage specifically since it is a subset of unidentified leakage. Peach Bottom Tech. Specs. requires a shutdown if any pressure boundary leakage is found.

REFERENCES

NUMARC NESP-007, SU5
Technical Specifications 3.6.C.1
T-101, RPV Control
T-102, Primary Containment Control

2.0 Reactor Pressure Vessel

2.1 Reactor Water Level

SITE AREA EMERGENCY - 2.1.3

IC Loss of Water Level in the Reactor Vessel That Has or Will Uncover fuel in the Reactor Vessel

EAL

RPV level **< *-172 "***

MODE 4, 5

BASIS

The indicator for "core is or will be uncovered" is Reactor Pressure Vessel Water level below the Top of Active Fuel (TAF) -172 inches as indicated on RPV Fuel Zone Level Instruments LI-2(3)-02-3-091 or LI-2(3)-02-3-113. Core submergence ensures adequate core cooling. When RPV level decreases below the top of active fuel the ability to remove the decay heat generated from the nuclear fuel becomes suspect and the Fuel Clad Fission Product barrier can no longer be considered intact. Sustained partial or total core uncovery can result in the release of a significant amount of fission products to the reactor coolant.

Under the conditions specified by this IC, severe core damage can occur and reactor coolant system pressure boundary integrity may not be assured. It is intended to address concerns raised by NRC Office for Analysis and Evaluation of Operational Data (AEOD) report AEOD/EG09, "BWR Operating Experience Involving Inadvertent Draining of the Reactor Vessel," dated August 8, 1986. This report states:

> In broadest terms, the dominant causes of inadvertent reactor vessel draining are related to the operational and design problems associated with the residual heat removal system when it is entering into or exiting form the shutdown cooling mode. During this transitional period, water is drawn from the reactor vessel, cooled by the residual heat removal system heat exchangers (from the cooling provided by the service water system), and returned to the reactor vessel. First, there are piping and valves in the residual heat removal system which are common to both the shutdown cooling mode and other modes of operation such as low pressure coolant injection and suppression pool cooling. These valves, when improperly positioned, provide a drain path for reactor coolant to flow from the reactor vessel to the suppression pool or the radwaste system. Second, establishing or making such evolutions vulnerable to personnel and procedural errors. Third, there is no comprehensive valve interlock arrangement for all shutdown cooling. Collectively, these factors have contributed to the inadvertent draining of the reactor vessel.

Thus, declaration of a Site Area Emergency is warranted under the conditions specified by the IC. Escalation to a General Emergency is via effluent release EAL.

DEVIATION

During EAL review and approval process, it was determined that the condition stated in NUMARC NESP-007, SS5, 1.a "Loss of all decay heat removal cooling as determined by (site-specific) procedure" is not necessary to conclude that the plant condition warrants a Site Area Emergency. Therefore, that sample NUMARC EAL was not included in this EAL.

REFERENCES

NUMARC NESP-007, SS5

2.0 Reactor Pressure Vessel

2.2 Reactor Power

ALERT - 2.2.2

IC 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

EAL

Automatic RPS SCRAM should occur due to RPS Setpoint being exceeded
<u>AND</u>
Failure of Automatic RPS SCRAM to make Reactor shutdown

MODE 1, 2

BASIS

Entry into this EAL is based on a reactor parameter actually exceeding a RPS setpoint and the reactor is not brought to a subcritial condition and maintained at that state with automatic RPS functions. The parameter must exceed the RPS setpoint by a significant margin eliminating minor setpoint drifts which are accounted for in the Technical Specification Margin of Safety. Subsequent manual scram actions were successful in bringing the reactor to a shutdown condition. Confirmation indications include control room annunciators, APRM/WRPM power level, and Control rod position indication.

A failure of the Reactor Protection System (RPS) to initiate and complete a reactor scram may indicate that the design limits of the nuclear fuel has been compromised. RPS is designed to automatically detect and generate a reactor scram signal when a limiting safety system setpoint is reached or exceeded. Control rod insertion following a scram signal is designed to be passive (i.e., system de-energizes, control rod motive energy source is previously charged).

Assuming that shutdown (subcritial) conditions cannot be established/maintained, an automatic scram signal failure followed by a successful manual scram would still constitute a scram failure and should be classified under this event.

Although the reactor may be brought initially subcritical based on partial control rod insertion, there is a possibility that positive reactivity may be introduced by a number of factors. Xenon decay and factors associated with cooldown, lower fuel temperature (doppler), lower moderator temperature, and a lower presence of steam bubbles (voids) may all contribute to cause a power increase.

Subcritical conditions can be assured even with the most reactive control rod fully withdrawn from the core if the remaining 184 control rods fully insert. Any other control rod pattern resulting from partial control rod insertion must be carefully analyzed and/or monitored to detect the possibility of re-criticality or local criticality.

Due to the buildup of Xenon in areas of the core that have previously been operating at high power levels, attention should be applied to the possibility that control rods which previously had low worth (e.g., peripheral control rods) may now have significant control rod worth.

When the reactor is not shutdown as identified in the Transient Response Implementing Plan Procedures (TRIPs), then entry into this EAL is warranted. When partial control rod insertion occurs following a scram signal (either manual or automatic) judgment should be applied as to whether classification should occur. Multiple control rods failing to insert beyond notch position 02 may require actions to fully insert the control rods. However, the reactor has been made subcritical, and for all intent the reactor will remain subcritical. TRIP guidance will govern the insertion of these control rods.

This EAL would be escalated with a failure of both manual and automatic scram signals with the Reactor remaining critical.

DEVIATION

None

REFERENCES

NUMARC NESP-007, SA2
T-101, RPV Control, RC-1

2.0 Reactor Pressure Vessel

2.2 Reactor Power

SITE AREA EMERGENCY - 2.2.3

IC 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

EAL

RPS SCRAM should occur due to RPS Setpoint being exceeded **AND** Failure of Automatic RPS, ARI **AND** Manual SCRAM to reduce reactor power < *4%*

MODE 1, 2

BASIS

A valid automatic and/or manual scram signal is present as indicted by control room indications and/or alarms and APRM indication is greater than 4% power. The Reactor Protection System (RPS) is designed to function to shut down the reactor (either manually or automatically). The system is "fail safe," that is, it de-energizes to function. An Anticipated Transient Without Scram (ATWS) event can be caused either by a failure of RPS (electrical failure) or a failure of the Control Rod Drive system to permit the control rods to insert (hydraulic failure).

A failure of the Reactor Protection System to shut down the reactor (as indicated by reactor power remaining above 4%) is a degraded plant condition that together with suppression pool temperature approaching 110°F requires the injection of boron to shut down the reactor.

The RPV Control TRIP Procedure establishes 4% power coincident with loss of scram capability as the initiating condition for various plant responses to ATWS events. With Reactor Power less than 4% the heat being generated in the core can be removed from the RPV and containment while actions are taken to bring the reactor subcritical.

A manual scram is defined as any set of actions by the reactor operator(s) at the reactor control console which causes control rods to be rapidly inserted into the core and brings the reactor subcritical (i.e., mode switch to shutdown, manual scram push buttons, or manual ARI initiation). Taking the mode switch to shutdown as part of the actions required by TRIP procedure is considered a manual scram action, although the mode switch in shutdown will generate a scram signal.

While the plant is being shutdown, significant heat is being generated in the core and the heat up rate of the Torus (due to heat rejection through SRVs) can increase which could approach the Torus temperature limit prior to shutting down. As the Torus heat increases towards the limiting temperature, the probability of causing a major over-pressure event increases substantially.

After an ATWS event, there is a potential that the Main Steam Isolation Valves (MSIV) will remain open. There is additional guidance in the TRIP procedures to ensure that the MSIVs remain open even if RPV level is intentionally lowered to below the normal MSIV isolation level. This situation would allow the plant to remove heat and provide makeup through the normal steam/feed cycle. If this path is not available, or becomes unavailable during the transient, heat rejection will be to the Torus.

With Standby Liquid Control initiated and with partial or no control rod insertion, there is a possibility that the neutron flux profile in the reactor core may become uneven or distorted. Localized clad damage is possible, if localized power levels increase significantly.

With reactor power remaining above 4% containment integrity is threatened, as the ability of systems to remove all of the heat transferred to the containment may be exceeded. As the energy contained in the containment increases there may be a degradation in the ability to remove heat generated by the "at power" reactor core. There is therefore a potential loss of the containment or the fuel cladding (caused by overheating).

This event will be escalated based on Torus Temperature on the "UNSAFE" side of the Heat Capacity Temperature Limit (HCTL) curve (T-102, T/T-1) or RPV level <-200".

DEVIATION

None

REFERENCES

NUMARC NESP-007, SS2
T-100, Scram
T-101, RPV Control, RC/L-2
T-117, Level/Power Control

2.0 Reactor Pressure Vessel

2.2 Reactor Power

GENERAL EMERGENCY - 2.2.4

IC 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

EAL

RPS SCRAM should occur due to RPS Setpoint being exceeded **AND** Failure of Automatic RPS, ARI **AND** Manual SCRAM to reduce reactor power **< 4%** **AND** Torus Temperature is on the "UNSAFE" side of the Heat Capacity Temperature Limit (HCTL) curve (T-102, T/T-1) **OR** RPV level **<-200 "**

MODE 1, 2

BASIS

A valid automatic or manual scram signal is present as indicted by control room indications and/or alarms and APRM indication is greater than 4% power. In addition, control room instrumentation indicates that Torus temperature is on the "UNSAFE" side of the Heat Capacity Temperature Limit (HCTL) curve (T-102, T/T-1) or RPV level is <-200".

Failure of all automatic and manual trip functions coincident with a high Torus temperature will place the plant in a condition where reactivity control capability is jeopardized and heat removal capability is severely limited.

ECCS systems which may be used to cool the core, transfer heat from the reactor, and/or supply cooling water to the reactor all take a suction of the Torus. Operation with sustained high Torus temperatures may render these systems inoperable due to Net Positive Suction Head (NPSH) considerations.

RPV level **<-200 "** indicates an extreme challenge to the ability to cool the core.

The RPV Control TRIP Procedure establishes 4% power coincident with loss of scram capability as the initiating condition for various plant responses to ATWS events. The timely initiation of Standby Liquid Control (prior to Torus temperature reaching 110°F) would bring the reactor to < 4% power before Torus temperature approaches the heat capacity temperature limit curve limitations.

Under ATWS conditions, it is important to assure continuous, stable steam condensation capability. An elevated Torus temperature on the "UNSAFE" side of the HCTL curve would result in unstable steam condensation should rapid reactor depressurization occur (ADS activation). Maintaining the ability to condense steam will preclude the pressurization of the containment and prevent possible containment failure.

Containment over-pressurization, which would be an eventual result of sustained operation with heat being added to the containment and high Torus temperature would result in the loss of containment integrity and the inability to remove the heat generated from the fuel. Fuel clad failure would result from the overheating of the fuel.

DEVIATION

None

REFERENCES

NUMARC NESP-007, SG2.1, SG2.2
T-101, RPV Control
T-102, Primary Containment Control, T/T-1
T-117, Level/Power Control, RC/L-2

3.0 Fission Product Barrier

3.1 Initiating Condition Matrix

Determine which combination of the three barriers (Fuel Clad, Reactor Coolant, Primary Containment) are lost or have a potential loss and use the following key to classify the event. Also, an event for multiple events could occur which result in the conclusion that exceeding the loss or potential loss thresholds is IMMINENT (i.e., within 1 to 2 hours). In this IMMINENT LOSS situation, use judgment and classify as if the thresholds are exceeded.

UNUSUAL EVENT

IC ANY Loss or ANY Potential Loss of Primary Containment

EAL

ANY Loss **OR** ANY Potential Loss of Primary Containment

ALERT

IC ANY Loss or ANY Potential Loss of EITHER Fuel Clad OR RCS

EAL

ANY Loss **OR** ANY Potential Loss of EITHER Fuel Clad **OR** RCS

SITE AREA EMERGENCY

IC 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

EAL

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

GENERAL EMERGENCY

IC Loss of ANY Two Barriers
AND
Potential Loss of Third Barrier

EAL

Loss of ANY Two Barriers **AND** Potential Loss of Third Barrier

MODE 1, 2, 3

NOTES:

1. Although the logic used for these initiating conditions appears overly complex, it is necessary to reflect the following considerations:

   - The Fuel Clad barrier and the RCS barrier are weighted more heavily than the Containment barrier. Unusual Event ICs associated with RCS and Fuel Clad barriers are addressed under the other plant condition EALs.

   - At the Site Area Emergency level, there must be some ability to dynamically assess how far present conditions are from General Emergency. For example, if the Fuel Clad barrier and RCS barrier "Loss" EALs existed, this would indicate to the Emergency Director that, in addition to offsite dose assessments, must focus on continual assessments of radioactive inventory and containment integrity. If, on the other hand, both Fuel Clad barrier and RCS barrier "Potential Loss" EALs existed, the Emergency Director would have more assurance that there was no immediate need to escalate to a General Emergency.

   - The ability to escalate to higher emergency classes as an event gets worse must be maintained. For example, RCS leakage steadily increasing would represent an increasing risk to public health and safety.

2. Fission Product Barrier ICs must be capable of addressing event dynamics. Thus, the EAL Reference Table states that IMMINENT (i.e., within 1 to 2 hours) Loss or Potential Loss should result in a classification as if the affected threshold(s) are already exceeded, particularly for the higher emergency classes.

3. The Fuel Clad barrier is the cladding tubes that contain the fuel pellets.

4. The RCS Barrier is the reactor coolant system pressure boundary and includes the reactor vessel and all reactor coolant system piping up to the isolation valves.

5. The Primary Containment Barrier includes the drywell, the wetwell, their respective interconnecting paths, and other connections up to and including the outermost containment isolation valves.

6. If a "Loss" condition is satisfied, the "Potential Loss" category can be considered satisfied. This is also applicable to conditions where this is a "Loss" indication with no corresponding "Potential Loss" condition.

7. For all conditions listed in Fission Product Barrier Table, the barrier failure column is only satisfied if it fails when called upon to mitigate an accident. For example, failure of both containment isolation valves to isolate with a downstream pathway to the environment is only a concern during an accident. If this condition exists during normal power operations, it will be an active Technical Specification Action Statement. However, during accident conditions, this will represent a breach of containment.

DEVIATION

None

REFERENCES

NUMARC NESP-007, Recognition Category F, Table 3

3.0 Fission Product Barrier

3.2 Fuel Clad Barrier

FC.1 Primary Coolant Activity Level

EAL

LOSS
Reactor Coolant activity > ***300 μCi/gm*** Dose Equivalent Iodine 131
POTENTIAL LOSS
Not Applicable

MODE 1, 2, 3

BASIS

A reactor coolant sample activity of greater than > 300 μCi/gm was determined to indicate significant clad heating and is indicative of the loss of the fuel clad barrier. This concentration is well above that expected for Iodine spikes and corresponds to 2.6% clad damage. 2.6% fuel clad damage is based upon NUREG-1228 core damage analysis.

Calculation of 300 μCi/cc equivalence to percent fuel clad damage is as follows (for purposes of this calculation, cc and gm are considered equivalent):

Iodine Isotope	Dose Factors (Reg Guide 1.109)	Ci/MWe Values (Time After Shutdown = 0) (NUREG-1228)
I-131	4.39E-3	85000
I-132	5.23E-5	120000
I-133	1.04E-3	170000
I-134	1.37E-5	190000
I-135	2.14E-4	150000

Time After Shutdown (T = 0) Ratios
R_{132} = 120000/85000(I-131) = 1.41(I-131)
R_{133} = 170000/85000(I-131) = 2.00(I-131)
R_{134} = 190000/85000(I-131) = 2.24(I-131)
R_{135} = 150000/85000(I-131) = 1.76(I-131)

Equation for Dose Equivalent Iodine (DEI_{131})

$$DEI_{131} = \frac{A_{131} DF_{131} + (R_{132}) A_{131} DF_{132} + (R_{133}) A_{131} DF_{133} + (R_{134}) A_{131} DF_{134} + (R_{135}) A_{131}}{DF_{131}}$$

Solve for A_{131} assuming $DEI_{131} = 300\ \mu Ci/cc$

$$300 = \frac{A_{131}4.39E-3+1.41\,A_{131}5.23E-5+2.00\,A_{131}1.04E-3+2.24\,A_{131}1.37E-5+1.76\,A_{131}2.14E}{4.39E-3}$$

$$300 = \frac{6.95E-3\,A_{131}}{4.39E-3}$$

Therefore: A_{131} = 189 μCi/cc I-131

Clad damage fraction (NUREG-1228, Table 4.1) = .02
Full Power = 1150 MWe

Clad Activity I-131 = (Ci/MWe) (MWe) (Clad Damage Fraction)
= (85000Ci/MWe) (1150MWe) (.02)
= 1.96E6 Ci

Reactor Water Volume = 2.67E8 cc (ERP-C-1410)

Total Coolant Activity I-131 = (A_{131}) (Rx Water Volume) (Ci/μCi)
= (189 μCi/cc) (2.67E8cc) (1.0E-6Ci/ μCi)
= 5.05E4Ci

Percent Clad Damage = Total Coolant Activity/Clad Activity I-131
= (5.05E4) / (1.96E6)
= 2.6%

This event will be escalated to an Site Area Emergency when additional fission product barriers are lost.

DEVIATION

None

REFERENCES

NUMARC NESP-007, FC EAL #1
NUREG 1228 - Source Term Estimation During Incident Response to Severe Nuclear Power Plant Accidents, Table 2.2
Reg. Guide 1.109, Table E-9
ERP-C-1410

3.0 Fission Product Barrier

3.2 Fuel Clad Barrier

FC.2 Reactor Vessel Water Level

EAL

LOSS RPV level < ***-200 "***
POTENTIAL LOSS RPV level < ***-172 "***

MODE 1, 2, 3

BASIS

The "Loss" EAL ***-200 "*** value corresponds to the level which is used in the TRIPS to indicate challenge of core cooling. This is the minimum value to assure core cooling without further degradation of the clad. The "Potential Loss" EAL is the same as the RCS barrier "Loss" EAL 4 and corresponds to the fuel zone water level at the top of the active fuel. Thus, this EAL indicates a "Loss" of RCS barrier and a "Potential Loss" of the Fuel Clad Barrier. This EAL appropriately escalates the emergency class to a Site Area Emergency.

Core submergence is the preferred method of core cooling and as such, the failure to re-establish RPV water level above the top of active fuel for an extended period of time could lead to significant fuel damage. This condition, -200 " as read on instruments LI-2(3)-02-3-091 or LI2(3)-02-3-113, could be indicative of a large break Loss Of Coolant Accident (LOCA) (where ECCS Systems are designed to maintain level at 2/3 core height) or a small LOCA with the inability of emergency core cooling systems to reflood the RPV. The value of -200" was chosen as it represents 2/3 core height.

DEVIATION

None

REFERENCES

NUMARC NESP-007, FC EAL #2 , RC EAL #4
T-101, RPV Control
T-111, Level Restoration/Steam Cooling, LR-11
T-112, Rapid Depressurization
T-117, Level/Power Control
T-116, RPV Flooding

3.0 Fission Product Barrier

3.2 Fuel Clad Barrier

FC.3 Drywell Radiation Monitoring

EAL

LOSS
Drywell Rad Monitor reading > ***$8x10^4$ R/hr***
POTENTIAL LOSS
Not Applicable

MODE 1, 2, 3

BASIS

The $8x10^4$ R/hr reading on a containment high range radiation monitor RI-8(9)103A,B,C,D is a value which indicates the release of reactor coolant, with elevated activity indicative of fuel damage, into the drywell. The reading was calculated assuming an instantaneous release and dispersal of the Reactor Coolant noble gas and iodine inventory into the Primary Containment (direct reading not shine) at a coolant concentration of 300 μCi/gm Dose Equivalent Iodine 131. This calculation is as follows:

Using Curve 3 [1%] of ERP-C-1410

Time after Shutdown = 1 hour (more conservative due to lower value for EAL)

1% fuel clad damage
the dose rate = 30,000 R/hr

Extrapolating to 2.6%
(30,000 R/hr/1%)(2.6) = 78,000 R/hr

This is rounded conservatively to 80,000 R/hr for human factors considerations

2.6% clad damage is based upon NUREG-1228 core damage analysis, and by virtue of its release into containment, the loss of the Reactor Coolant barrier (detailed calculations are contained in the Basis for Fission Product Barrier EAL FC #1).

Reactor coolant concentrations of this magnitude are several times larger than the maximum concentrations (including iodine spiking) allowed within technical specifications and are therefore indicative of fuel damage. This value is higher than that specified for RCS barrier Loss EAL #3. Thus, this EAL indicates a loss of both Fuel Clad barrier and RCS barrier.

There is no "Potential Loss" EAL associated with this item.

DEVIATION

None

REFERENCES

NUMARC NESP-007, FC EAL #3 and RC EAL #3
NUREG 1228 - Source Term Estimation During Incident Response to Nuclear Power Plant Accidents
ERP-C-1410

3.0 Fission Product Barrier

3.2 Fuel Clad Barrier

FC.4 Other Indications

EAL

<u>LOSS</u> Not Applicable <u>POTENTIAL LOSS</u> Not Applicable

MODE 1, 2, 3

BASIS

There are no other indications at PBAPS for loss of the Fuel Clad Barrier.

DEVIATION

None

REFERENCES
NUMARC NESP-007, FC EAL #4 and RC EAL #5

3.0 Fission Product Barrier

3.2 Fuel Clad Barrier

FC.5 Emergency Director Judgment

EAL

Any condition in the judgment of the Emergency Director that indicates Loss or Potential Loss of the FUEL CLAD barrier

MODE 1, 2, 3

BASIS

This EAL addresses any other factors that are to be used by the Emergency Director in determining whether the Fuel Clad barrier is lost or potentially lost. In addition, the inability to monitor the barrier should also be incorporated in this EAL, as a factor in Emergency Director judgment, that the barrier may be considered lost or potentially lost. (See also IC, "Prolonged Loss of All Offsite Power and Prolonged Loss of All Onsite AC Power", for additional information.)

DEVIATION

None

REFERENCES

NUMARC NESP-007, FC EAL #5

3.0 Fission Product Barrier

3.3 Reactor Coolant System Barrier

RC.1 RCS Leak Rate

EAL

<u>LOSS</u> Not Applicable <u>POTENTIAL LOSS</u> RCS leakage ***>50 gpm*** <u>**OR**</u> Unisolable primary system leakage outside drywell as indicated by T-103, **Temperature Action Level** is exceeded in ONE area requiring a SCRAM <u>**OR**</u> Unisolable primary system leakage outside drywell as indicated by T-103, **Radiation Action Level** is exceeded in ONE area requiring a SCRAM

MODE 1, 2, 3

BASIS

Potential loss of RCS based on primary system leakage outside the drywell is determined from T-103 area temperatures or radiation levels. TRIP guidance stipulates that when the Temperature or Radiation Action Level limits have been exceeded for one area, that the reactor be manually SCRAMmed.

There are two ways that the temperatures in the Secondary Containment can reach these levels; i.e., primary leak into secondary and a fire within the secondary containment. As the temperatures rise above normal conditions, the plant staff will isolate the containment and all systems, except those required for shutdown and cooling, at the Temperature Action Levels Isolation levels. If the temperatures continue to rise to the Temperature Action Levels it is indicative that an unisolable leak has occurred. If the radiation levels rise above the Radiation Action Levels, it also indicates that an unisolable leak has occurred.

This event signifies that there is a direct path established for the transfer of main steam to inside the Turbine Building. Assumptions made in dose calculations regarding radioactive material transport (e.g., hold up, plate out, scrubbing, and retention) may be invalid. Additionally the transport time associated with a radiological release may be significantly shortened and there may be a higher percentage of short lived radioisotopes in any release.
As both the reactor coolant pressure boundary and the primary containment are degraded; the extent of radioactive release is dependent on fuel clad integrity. Should a rapid reactor depressurization occur as a result of this event then there is a potential that a large amount of radioiodine may be released.

DEVIATION

None

REFERENCES

NUMARC NESP-007, RC EAL #1 PC EAL #2
T-103 Secondary Containment Control

3.0 Fission Product Barrier

3.3 Reactor Coolant System Barrier

RC.2 Drywell Pressure

EAL

LOSS Drywell Pressure > ***2.0 psig*** **AND** Indication of a leak inside drywell POTENTIAL LOSS Not Applicable

MODE 1, 2, 3

BASIS

The ***2.0 psig*** drywell pressure is based on the drywell high pressure alarm set point and indicates a LOCA.

If drywell pressure exceeds 2 psig, there is a clear indication that a leak of sufficient magnitude exists that prevents drywell pressure stabilization.

DEVIATION

The NUMARC EAL contains only the drywell pressure. A qualifying:

> "**AND**
> Indication of a leak inside drywell"

was added as a human factor reminder to the Emergency Director that use of this EAL is for accident scenarios only. Thus, a Drywell pressure increase due to the loss of Drywell cooling will not require an emergency classification.

REFERENCES

NUMARC NESP-007, RC EAL #2
T-101, RPV Control
T-102, Primary Containment Control

3.0 Fission Product Barrier

3.3 Reactor Coolant System Barrier

RC.3 Drywell Radiation Monitoring

EAL

LOSS Drywell Rad Monitor reading > ***15 R/hr*** POTENTIAL LOSS Not Applicable

MODE 1, 2, 3

BASIS

The ***15 R/hr*** reading is a value which indicates the release of reactor coolant to the drywell. The value assumes an instantaneous release and dispersal of the reactor coolant noble gas and iodine inventory associated with concentrations corresponding to 0.001% Total Isotopic Distribution (TID) into the drywell atmosphere.

Using attachment 5 of ERP-C-1410, Curve 6

Time after Shutdown = 0.1 hour

0.001% TID = 17 R/hr

This is rounded to 15 R/hr for human factors considerations

This reading is less than that specified for Fuel Clad Barrier EAL #3. Thus, this EAL would be indicative of a RCS leak only. If the radiation monitor reading increases to that value specified by Fuel Clad Barrier EAL #3, fuel damage would also be indicated.

There is no "Potential Loss" EAL associated with this item.

DEVIATION

None

REFERENCES

NUMARC NESP-007, FC EAL #3 and RC EAL #3
NUREG 1228 - Source Term Estimation During Incident Response to Nuclear Power Plant Accidents
ERP-C-1410, Attachment 5

3.0 Fission Product Barrier

3.3 Reactor Coolant System Barrier

RC.4 Reactor Vessel Water Level

EAL

LOSS RPV level **< *-172* "** POTENTIAL LOSS Not Applicable

MODE 1, 2, 3

BASIS

This "Loss" EAL is the same as "Potential Loss" Fuel Clad Barrier EAL #2. The ***-172* "** water level corresponds to the level which is used in TRIPS to indicate challenge of core cooling. This EAL appropriately escalates the emergency class to a Site Area Emergency. Thus, this EAL indicates a loss of the RCS barrier and a Potential Loss of the Fuel Clad Barrier.

DEVIATION

None

REFERENCES

NUMARC NESP-007, FC EAL #2 , RC EAL #4
T-101, RPV Control
T-111, Level Restoration/Steam Cooling, LR-11
T-112, Rapid Depressurization
T-117, Level/Power Control
T-116, RPV Flooding

3.0 Fission Product Barrier

3.3 Reactor Coolant System Barrier

RC.5 Other Indications

EAL

LOSS Not Applicable POTENTIAL LOSS RPV level cannot be determined

MODE 1, 2, 3

BASIS

Inability to determine Reactor Pressure Vessel level may be due to reference leg boil-off, instrument power failure, or conflicting information on uncontrolled parameter oscillations. TRIP procedure guidance will require the flooding of the Reactor Pressure Vessel, thus ensuring core submergence. Based on differences in calibration and design, all ranges of level instruments may not indicate exactly the same; this operational difference is expected and is not to be used for deciding that conflicting RPV level indication exists. Multiple indications of level instruments pegged high is indication that the level is above the range and that it is known, also visual observation during refueling is indication of RPV water level.

If indeterminate RPV level is due to reference leg boil-off, it is an indicator of a potential loss of the Reactor Coolant System. Adequate core cooling would be rapidly assured using the guidance provided in the TRIP Procedures. If it can be determined that the cause is due to an instrument power or instrumentation failure, then it is not appropriate to classify the event as a potential loss of the Reactor Coolant System.

Operator attention should be given to the possibility that under depressurized conditions, there is the possibility that gases may come out of solution and result in distorted RPV level indications. Operators should be attentive to observe multiple level indications (particularly those which utilize separate reference legs) to ensure that actual RPV level is known and displayed. Unexplained and/or sudden changes in specific level indications may be a result of degassification of the coolant contained in the level instrumentation.

DEVIATION

None

REFERENCES

NUMARC NESP-007, FC EAL #4 and RC EAL #5
T-101, RPV Control, RC/L-1
T-112, Rapid Depressurization
T-117, Level/Power Control
T-116, RPV Flooding

3.0 Fission Product Barrier

3.3 Reactor Coolant System Barrier

RC.6 Emergency Director Judgment

EAL

Any condition in the judgment of the Emergency Director that indicates Loss or Potential Loss of the RCS barrier

MODE 1, 2, 3

BASIS

This EAL addresses any other factors that are to be used by the Emergency Director in determining whether the RCS barrier is lost or potentially lost. In addition, the inability to monitor the barrier should also be incorporated in this EAL as a factor in Emergency Director judgment that the barrier may be considered lost or potentially lost. (See also IC, "Prolonged Loss of All Offsite Power and Prolonged Loss of All Onsite AC Power", for additional information.)

DEVIATION

None

REFERENCES

NUMARC NESP-007, RCS EAL #6

3.0 Fission Product Barrier

3.4 Primary Containment Barrier

PC.1 Drywell Pressure

EAL

<u>LOSS</u>

Rapid, unexplained drop in Drywell Pressure following initial rise

<u>OR</u>

Drywell pressure response not consistent with LOCA conditions

<u>POTENTIAL LOSS</u>

Drywell Pressure > ***49 psig*** and rising

<u>OR</u>

Drywell Hydrogen > ***6%*** <u>AND</u> Drywell Oxygen > ***5%***

MODE 1, 2, 3

BASIS

Rapid unexplained loss of pressure (i.e., not attributable to drywell spray or condensation effects) following an initial pressure rise indicates a loss of containment integrity. Drywell pressure should increase as a result of mass and energy release into containment from a LOCA. Thus, drywell pressure not increasing under these conditions indicates a loss of containment integrity. The ***49 psig*** for potential loss of containment is based on the containment drywell design pressure and is equal to the peak pressure expected from a DBA LOCA.

The specified value of 6% hydrogen concentration is the minimum which can support a deflagration. Likewise, the minimum concentration of oxygen required to support a deflagration is 5%. Combustion of hydrogen in the deflagration concentration range creates a traveling flame causing a rapid rise in primary containment pressure. A deflagration may result in a peak primary containment pressure high enough to rupture the primary containment or damage the drywell-to-torus boundary.

DEVIATION

None

REFERENCES

NUMARC NESP-007, PC EAL #1
ON-110, Loss of Primary Containment
T-101, RPV Control
T-102, Primary Containment Control w/Bases
T-103, Secondary Containment Control

3.0 Fission Product Barrier

3.4 Primary Containment Barrier

PC.2 Containment Isolation Valve After Containment Isolation

EAL

LOSS Failure of both valves in any one line to close **AND** downstream pathway to the environment exists **OR** Intentional venting per T-200 is required **OR** Unisolable primary system leakage outside drywell as indicated by T-103, **Temperature Action Level** is exceeded in ONE area requiring a SCRAM **OR** Unisolable primary system leakage outside drywell as indicated by T-103, **Radiation Action Level** is exceeded in ONE area requiring a SCRAM POTENTIAL LOSS Not Applicable

MODE 1, 2, 3

BASIS

This EAL is intended to cover containment isolation failures allowing a direct flow path to the environment such as failure of both MSIVs to close with open valves downstream to the turbine or to the condenser. In addition, the presence of area radiation or temperature alarms indicating unisolable primary system leakage outside the drywell are covered. Also, an intentional venting of primary containment per TRIPS to the secondary containment and/or the environment is considered a loss of containment.

Loss of containment based on primary system leakage outside the drywell is determined from T-103 area temperatures or radiation levels. TRIP guidance stipulates that when the Temperature or Radiation Action Level limits have been exceeded for one area, that the reactor be manually SCRAMmed.

There are two ways that the temperatures in the Secondary Containment can reach these levels; i.e., primary leak into secondary and a fire within the secondary containment. As the temperatures rise above normal conditions, the plant staff will isolate the containment and all systems, except those required for shutdown and cooling, at the Temperature Action Level Isolation levels. If the temperatures continue to rise to the Temperature Action Levels it is indicative that an unisolable leak has occurred. If the radiation levels rise above the Radiation Action Levels, it also indicates that an unisolable leak has occurred.

DEVIATION

None

REFERENCES

NUMARC NESP-007, RCS EAL #1, PC EAL #2
T-103 Secondary Containment Control
T-104, Radioactivity Release Control
T-200, Primary Containment Venting

3.0 Fission Product Barrier

3.4 Primary Containment Barrier

PC.3 Significant Radioactive Inventory in Containment

EAL

<u>LOSS</u> Not Applicable <u>POTENTIAL LOSS</u> Drywell Rad Monitor reading > ***6x10^5 R/hr***

MODE 1, 2, 3

BASIS

A containment high range radiation monitor 9RI-8(9)103A,B,C,D reading 6x10^5 R/hr indicates significant fuel damage, well in excess of that required for the loss of the RCS and Fuel Clad. As stated in Section 3.8 of NUMARC/NESP-007, a major release of radioactivity requiring offsite protective actions from core damage is not possible unless a major failure of fuel cladding allows radioactive material to be released from the core into the reactor coolant. Regardless of whether containment is challenged, this amount of activity in containment, if released, could have such severe consequences that it is prudent to treat this as a potential loss of containment, such that a General Emergency declaration is warranted. NUREG-1228, "Source Estimations During Incident Response to Severe Nuclear Power Plant Accidents," indicates that such conditions do not exist when the amount of clad damage is less than 20%.

The reading was calculated assuming an instantaneous release of the Reactor Coolant volume into the Primary Containment (direct reading not shine) where the value corresponds to a release of approximately 20% of the gap region. This calculation is as follows:

Using Curve 3 [1%] of ERP-C-1410

Time after Shutdown = 1 hour (more conservative due to lower value for EAL)

1% fuel clad damage
the dose rate = 30,000 R/hr

Extrapolating to 20%
(30,000 R/hr/1%)(20) = 600,000 R/hr

There is no "Loss" EAL associated with this item.

DEVIATION

None

REFERENCES

NUMARC NESP-007, FC EAL #3, RC EAL #3 and PC EAL #3
NUREG 1228 - Source Term Estimation During Incident Response to Severe Nuclear Power Plant Accidents
ERP-C-1410

3.0 Fission Product Barrier

3.4 Primary Containment Barrier

PC.4 Reactor Vessel Water Level

EAL

<u>LOSS</u> Not Applicable <u>POTENTIAL LOSS</u> RPV level cannot be restored above ***-200 "*** within the time limit of the "SAFE" region of the Maximum Core Uncovery Time Limit Curve (T-116, RF-1)

MODE 1, 2, 3

BASIS

In this EAL, the ***-200 "*** water level corresponds to the level which is used in the TRIPS to indicate challenge of core cooling. This is the minimum value to assure core cooling without further degradation of the clad.

The conditions in this potential loss EAL represent imminent melt sequences which, if not corrected, could lead to vessel failure and increased potential for containment failure. In conjunction with the level EALs in the Fuel and RCS barrier columns, this EAL will result in the declaration of a General Emergency on loss of two barriers and the potential loss of a third. If the TRIPS have been ineffective in restoring reactor vessel level within the maximum core uncovery time limit, there is not a "success" path.

Severe accident analysis (e.g., NUREG-1150) have concluded that function restoration procedures can arrest core degradation with the reactor vessel in a significant fraction of the core damage scenarios, and the likelihood of containment failure is very small in these events. Given this, it is appropriate to provide a reasonable period to allow TRIPS to arrest the core melt sequence. Whether or not the procedures will be effective should be apparent within the time provided by the maximum core uncovery time limit. The Emergency Director should make the declaration as soon as it is determined that the procedures have been, or will be, ineffective.

There is no "Loss" EAL associated with this item.

DEVIATION

None

REFERENCES

NUMARC NESP-007, FC EAL #2 , RC EAL #4
T-101, RPV Control
T-111, Level Restoration/Steam Cooling, LR-11
T-112, Rapid Depressurization
T-117, Level/Power Control
T-116, RPV Flooding

3.0 Fission Product Barrier

3.4 Primary Containment Barrier

PC.5 Other Indications

EAL

<u>LOSS</u> Not Applicable <u>POTENTIAL LOSS</u> RPV level cannot be determined **<u>AND</u>** RPV Flooding cannot be established as indicated by inability to maintain 5 ADS/SRVs open with RPV pressure at least 60 psig above Torus pressure per T-116

MODE 1, 2, 3

BASIS

The decision to enter RPV Flooding is made when RPV water level cannot be determined. This judgment consists of evaluating all plant indications which can influence the ability to maintain adequate core cooling. Entry to RPV flooding requires rapid RPV depressurization. The minimum RPV Flooding Pressure is defined as the lowest differential pressure between the RPV and the Torus at which steam flow through the SRVs will be sufficient to remove all of the generated decay heat. Operation at the minimum reactor flooding pressure requires that a sufficient amount of water reach the core to carry away decay heat by boiling, which in turn requires that RPV water level increase. So RPV flooding not established requires containment flooding. This represents a potential loss of containment due to the potential need to vent containment in order to facilitate flooding. Additionally, it represents a potential inability to remove decay heat which may also lead to containment failure.

Inability to determine Reactor Pressure Vessel level may be due to reference leg boil-off, instrument power failure, or conflicting information on uncontrolled indication oscillations. TRIP procedure guidance will require the flooding of the Reactor Pressure Vessel, thus ensuring core submergence. Based on differences in calibration and design, all ranges of level instruments may not indicate exactly the same; this operational difference is expected and is not to be used for deciding that conflicting RPV level indication exists. Level indication pegged high is indication that the level is above the range and that it is known, also visual observation during refueling is indication of RPV water level.

If it can be determined that the loss of ability to monitor RPV level is due to an instrument power or instrumentation failure, then it is not appropriate to classify the event as a potential loss of the Primary Containment.

The minimum RPV flooding pressure will ensure that adequate core cooling exists independent of RPV level indication. Failure to establish the differential pressure between the RPV and the Torus in a timely manor can jeopardize the ability of the reactor coolant system to dissipate the decay heat generated.

Ample time must be allotted for determining the failure of ECCS systems to pressurize the RPV. Control Room indications such as RPV level (used for trending), RPV Pressure, ECCS injection flow rates, Containment parameters, and injection system operability should all be used to gauge the effectiveness of the RPV Flood.

If the loss of level indication was caused by reference leg flashing, then level indicators can still be utilized to monitor the trend in RPV level. Actual RPV level will never be higher than indicated level.

In the event that the loss of level indication is only a result of degassification of the coolant contained in the level instrumentation piping, then it is anticipated that flooding pressure can be obtained.

RPV water level below the top of active fuel for a sustained period of time represents an early indicator that significant core damage is in progress while providing sufficient time to initiate public protective actions. For events starting from power operation, some core melting can be expected. Even under these conditions vessel failure and containment failure with resultant release to the public would not be expected for some time.

DEVIATION

None

REFERENCES

NUMARC NESP-007, FC EAL #4 , RCS EAL #5 and PC EAL #5
T-101, RPV Control
T-111, Level Restoration/Steam Cooling, LR-11
T-112, Rapid Depressurization
T-117, Level/Power Control
T-116, RPV Flooding

3.0 Fission Product Barrier

3.4 Primary Containment Barrier

PC.6 Emergency Director Judgment

EAL

Any condition in the opinion of the Emergency Director that indicates Loss or Potential Loss of the Primary Containment barrier

MODE 1, 2, 3

BASIS

This EAL addresses any other factors that are to be used by the Emergency Director in determining whether the Containment Barrier is lost or potentially lost. In addition, the inability to monitor the barrier should also be incorporated in this EAL as a factor in Emergency Director judgment that the barrier may be considered lost or potentially lost. (See also IC, "Prolonged Loss of All Offsite Power and Prolonged Loss of All Onsite AC Power", for additional information.)

DEVIATION

None

REFERENCES

NUMARC NESP-007, PC EAL #6

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3.5 Fission Product Barrier Status Table

Barrier / Parameter	Fuel Clad Loss	Fuel Clad Potential Loss	Reactor Coolant System Loss	Reactor Coolant System Potential Loss	Primary Containment Loss	Primary Containment Potential Loss
Reactor Coolant Activity	Reactor Coolant activity > ***300 μCi/gm*** Dose Equivalent Iodine 131	N/A	N/A	N/A	N/A	N/A
RPV Level	RPV level < ***-200 "***	RPV level < ***-172 "***	RPV level < ***-172 "***	N/A	N/A	RPV level cannot be restored above ***-200 "*** within the time limit of the "SAFE" region of the Maximum Core Uncovery Time Limit Curve (T-116, RF-1)
RPV Level Unknown	N/A	N/A	N/A	RPV level cannot be determined	N/A	RPV level cannot be determined **AND** RPV Flooding cannot be established as indicated by inability to maintain 5 ADS/SRVs open with RPV pressure at least 60 psig above Torus pressure per T-116
RCS Leak Rate	N/A	N/A	N/A	RCS leakage ***>50 gpm***	N/A	N/A
Drywell Pressure	N/A	N/A	Drywell Pressure ***> 2.0 psig*** **AND** Indication of a leak inside drywell	N/A	Rapid, unexplained drop in Drywell Pressure following initial rise **OR** Drywell pressure response not consistent with LOCA conditions	Drywell Pressure ***> 49 psig*** and rising **OR** Drywell Hydrogen ***> 6%*** **AND** Drywell Oxygen ***> 5%***
Drywell Radiation	Drywell Rad Monitor reading ***> $8x10^4$ R/hr***	N/A	Drywell Rad Monitor reading ***> 15 R/hr***	N/A	N/A	Drywell Rad Monitor reading ***> $6x10^5$ R/hr***

3.5 Fission Product Barrier Status Table

Barrier / Parameter	Fuel Clad Loss	Fuel Clad Potential Loss	Reactor Coolant System Loss	Reactor Coolant System Potential Loss	Primary Containment Loss	Primary Containment Potential Loss
Containment Isolation	N/A	N/A	N/A	Unisolable primary system leakage outside drywell as indicated by T-103, **Temperature Action Level** is exceeded in ONE area requiring a SCRAM **OR** Unisolable primary system leakage outside drywell as indicated by T-103, **Radiation Action Level** is exceeded in ONE area requiring a SCRAM	Failure of both valves in any one line to close **AND** downstream pathway to the environment exists **OR** Intentional venting per T-200 is required **OR** Unisolable primary system leakage outside drywell as indicated by T-103, **Temperature Action Level** is exceeded in ONE area requiring a SCRAM **OR** Unisolable primary system leakage outside drywell as indicated by a T-103, **Radiation Action Level** is exceeded in ONE area requiring a SCRAM	N/A
Emergency Director Judgment	Any condition in the judgment of the Emergency Director that indicates Loss or Potential Loss of the FUEL CLAD barrier		Any condition in the judgment of the Emergency Director that indicates Loss or Potential Loss of the RCS barrier		Any condition in the judgment of the Emergency Director that indicates Loss or Potential Loss of the Primary Containment barrier	

In the table below, circle all of the appropriate X's in each applicable row for each Loss or Potential Loss of Fission Product Barrier as determined by the table above.

Classify the event as identified in the table heading if all X's in a column under that heading are circled.

Fission Product Barrier Status	Unusual Event		ALERT				SITE AREA EMERGENCY								GENERAL EMERGENCY			
Fuel Clad - Loss			X				X		X		X				X	X		X
Fuel Clad - Potential Loss				X				X		X		X					X	
Reactor Coolant System - Loss					X		X			X			X		X	X	X	
Reactor Coolant System-Potential Loss						X		X	X					X				X
Primary Containment - Loss	X										X	X	X	X	X		X	X
Primary Containment - Potential Loss		X														X		

4.0 Secondary Containment Bypass

4.1 Main Steam Line

UNUSUAL EVENT - 4.1.1

IC Fuel Clad Degradation

EAL

Main Steam Line HiHi Radiation (10xNFPB)

MODE 1, 2, 3

BASIS

Main Steam Line High-High Radiation alarm (2(3)-252,A,B,C,D and 2(3)-251,A,B,C,D) > 10 times normal full power background may be indicative of minor fuel cladding degradation and warrants the declaration of an Unusual Event. This level is set to preclude most spurious events including resin intrusion.

The main steam line high-high radiation condition requires a manual Main Steam Isolation Valve closure and a reactor scram. This transient may result in the introduction of fission product gases (previously contained in the gap area) to be suddenly released into the coolant due to the rapid down power transient and subsequent collapse of voids in the coolant.

This level of steam line activity is indicative of the release of gap activity to the coolant however, this level is not indication of a major failure of the fuel clad. The mechanics that caused main steam line radiation to increase to this level indicate there is a degradation of fuel clad integrity.

This event will escalate to an Alert based on the breach in the main steam line together with a failure of the MSIVs to isolate the main steam lines per Fission Product Barrier Table.

DEVIATION

The MODE applicability [1,2,3] is a deviation from NUMARC [all] in that, the SJAE Radiation Monitor and Main Steam Line Radiation Monitors will only be a valid indication of Fuel Clad Degradation in those MODE's. At Peach Bottom, there are no other monitors which can be an indicator of Fuel Clad Degradation. Degradation in cold shutdown or refueling will be first indicated by ventilation release monitors and covered in Effluent Release section.

REFERENCES

NUMARC NESP-007, SU4.1
T-099, Post Scram Recovery
T-101, RPV Control

4.0 Secondary Containment Bypass

4.1 Main Steam Line

ALERT - 4.1.2

IC RCS Leak Rate

EAL

Indication of a Main Steam Line Break: Hi Steam Flow Annunciator **AND** Hi Steam Tunnel Temperature Annunciator **OR** Direct report of steam release

MODE 1, 2, 3

BASIS

Design basis accident analyses of a Main Steam Line Break outside of secondary containment shows that even if MSIV closure occurs within design limits, dose consequences offsite from a "puff" release would be in excess of 10 millirem.

Hi Steam Flow Annunciator and Hi Steam Tunnel Temperature Annunciator are both indicators of a Main Steam Line Break. Both parameters will cause an isolation of the MSIV's. Should both valves in any one line fail to isolate, this event would be considered a loss of Primary Containment and a potential loss of the RCS per the Fission Product Barrier Table and appropriately classified as a Site Area Emergency.

Direct report of steam release is meant to provide an alternate means of classification if the Hi Steam Flow Annunciator or the Hi Steam Tunnel Temperature Annunciator fails to operate and the visual observation of conditions indicates a Main Steam Line Break in the judgment of the Emergency Director. This is not meant to cause a declaration based on leaks such as valve packing leaks where the consequences offsite would be negligible.

DEVIATION

None

REFERENCES

NUMARC NESP-007, RC.1
T-101, RPV Control
NUMARC Questions and Answers, June 1993, "Fission Product Barriers #7"

5.0 Radioactivity Release

5.1 Effluent Release and Dose

UNUSUAL EVENT - 5.1.1.a

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

EAL

A valid reading on one or more of the following radiation monitors that exceeds **TWO TIMES** the HiHi alarm setpoint value for ***> 60 minutes:***

Main Stack, Vent Stack, Radwaste Discharge, Service Water Discharge

<u>AND</u>

Calculated maximum offsite dose rate using computer dose model exceeds ***0.114 mRem/hr TPARD <u>OR</u> 0.342 mRem/hr child thyroid CDE*** based on a 60 minute average

Note: If the required dose projections cannot be completed within the 60 minute period, then the declaration must be made based on the valid sustained monitor reading.

MODE All

BASIS

The term "Unplanned", as used in this context, 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.

Unplanned releases in excess of 0.114 mRem/hr TPARD or 0.342 mRem/hr CDE that continue for > 60 minutes represent an uncontrolled situation and hence a potential degradation in the level of safety. The final integrated dose is very low and is not the primary concern. Rather it is the degradation in plant control implied by the fact that the release was not isolated within 60 minutes.

It is not intended that the release be averaged over 60 minutes, but exceed 0.114 mRem/hr TPARD or 0.342 mRem/hr CDE limits for 60 minutes. This EAL includes a 60 minute average for the dose projection with the release point radiation monitor above two times the HiHi alarm set point value for the entire 60 minutes. Also, it is intended that the event be declared as soon as it is determined that the release will exceed 0.114 mRem/hr TPARD or 0.342 mRem/hr CDE for greater than 60 minutes.

An indication or report is considered to be valid when it is verified by:

1. An instrument channel check
2. Indications on related or redundant instruments
3. By direct observation by plant personnel

Monitor indications are calculated based on the methodology of the site Offsite Dose Calculation Manual (ODCM). The HiHi alarm setpoints are set conservatively to indicate when a potential release may approach Technical Specification (ODCM) limits assuming multiple release points. Use of this conservative setpoint in establishing a monitor reading will not cause an inappropriate event classification since this EAL requires the magnitude of the monitor reading to be two times the setpoint, sustained for >60 minutes, and assessment by a dose projection indicating an offsite dose rate in excess of two times Technical Specification (ODCM) limits. In the unlikely event that a dose projection cannot be completed within the 60 minute period, the event will be declared based on the sustained monitor reading.

Total Protective Action Recommendation Dose (TPARD) is equal to Total Effective Dose Equivalent (TEDE) + 4 Day Deposition Dose. Committed Dose Equivalent (CDE) is equal to the thyroid exposure due to iodine. The computerized dose model provides projected TPARD and CDE.

The Total Protective Action Recommendation Dose (TPARD) is calculated by dividing the yearly allowable Technical Specification limit (500 mRem/yr.) by the number of hours per year (8760 hr/yr.), and then multiplying by a factor of 2 times Technical Specifications [ODCM].

TPARD = 2x(Tech Spec Limit)/(hours per year)
= 2(500 mRem/yr.)/(8760 hr/yr.)
= 0.114 mRem/hr

The Committed Dose Equivalent (CDE) is calculated by dividing the yearly allowable Technical Specification limit (1500 mRem/yr.) by the number of hours per year (8760 hr/yr.), and then multiplying by a factor of 2 times Technical Specifications [ODCM].

CDE = 2x(Tech Spec Limit)/(hours per year)
= 2(1500 mRem/yr.)/(8760 hr/yr.)
= 0.342 mRem/hr

This event will be escalated to an Alert when effluents increase.

DEVIATION

None

REFERENCES

NUMARC NESP-007, AU1.1
Offsite Dose Calculation Manual
NUMARC Questions and Answers, June 1993, "Abnormal Rad Levels/Radiological Effluents #9"

5.0 Radioactivity Release

5.1 Effluent Release and Dose

UNUSUAL EVENT - 5.1.1.b

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

EAL

Confirmed sample analyses for gaseous or liquid releases indicates concentrations or release rates exceeding **TWO TIMES** Tech Specs (Liquid Release ODCM. 3.8.B.1 and Gaseous Release ODCM 3.8.C.1.b) for **> *60 minutes***

MODE All

BASIS

Releases in excess of two times technical specifications that continue for > 60 minutes represent an uncontrolled situation and hence a potential degradation in the level of safety. The final integrated dose is very low and is not the primary concern. Rather it is the degradation in plant control implied by the fact that the release was not isolated within 60 minutes.

It is not intended that the release be averaged over 60 minutes, but exceed two times technical specifications limits for 60 minutes. Further, it is intended that the event be declared as soon as it is determined that the release will exceed two times technical specifications for greater than 60 minutes.

An indication or report is considered to be valid when it is verified by:
1. An instrument channel check
2. Indications on related or redundant instruments
3. By direct observation by plant personnel

The calculation called for in this EAL should also be conducted whenever a liquid release occurs for which a radioactive discharge permit wasn't prepared or that exceeds the conditions on the permit (e.g. minimum dilution, alarm setpoints, etc).

This event will be escalated to an Alert when effluents increase.

DEVIATION

None

REFERENCES

NUMARC NESP-007 AU1.2
Offsite Dose Calculation Manual
T-104, Radioactivity Release Control

5.0 Radioactivity Release

5.1 Effluent Release and Dose

ALERT - 5.1.2.a

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

EAL

A valid reading on one or more of the following radiation monitors that exceeds **TWO HUNDRED TIMES** the HiHi alarm setpoint value for ***> 15 minutes:*** Main Stack, Vent Stack, Radwaste Discharge, Service Water Discharge <u>AND</u> Calculated maximum offsite dose rate exceeds **11.4 mRem/hr TPARD <u>OR</u> 34.2 mRem/hr child thyroid CDE** based on a 15 minute average Note: If the required dose projections cannot be completed within the 15 minute period, then the declaration must be made based on the valid sustained monitor reading.

MODE All

BASIS

Releases in excess of 11.4 mRem/hr TPARD or 34.2 mRem/hr CDE that continue for > 15 minutes represent an uncontrolled situation and hence a potential degradation in the level of safety. The primary concern is the final integrated dose [100 times greater than the Unusual Event] and the degradation in plant control implied by the fact that the release was not isolated within 15 minutes.

This EAL includes a 15 minute average for the dose projection with the release point radiation monitor above two hundred times the HiHi alarm set point value for the entire 15 minutes. Also, it is intended that the event be declared as soon as it is determined that the release will exceed 11.4 mRem/hr TPARD or 34.2 mRem/hr CDE for greater than 15 minutes.

An indication or report is considered to be valid when it is verified by:

1. An instrument channel check
2. Indications on related or redundant instruments
3. By direct observation by plant personnel

Monitor indications are calculated based on the methodology of the site Offsite Dose Calculation Manual (ODCM). The HiHi alarm setpoints are set conservatively to indicate when a potential release may approach Technical Specification (ODCM) limits assuming multiple release points. Use of this conservative setpoint in establishing a monitor reading will not cause an inappropriate event classification since this EAL requires the magnitude of the monitor reading to be two hundred times the setpoint, sustained for >15 minutes, and assessment by a dose projection indicating an offsite dose rate in excess of two hundred times Technical Specification (ODCM) limits. In the unlikely event that a dose projection cannot be completed within the 15 minute period, the event will be declared based on the sustained monitor reading.

Total Protective Action Recommendation Dose (TPARD) is equal to Total Effective Dose Equivalent (TEDE) + 4 Day Deposition Dose. Committed Dose Equivalent (CDE) is equal to the thyroid exposure due to iodine. The computerized dose model provides projected TPARD and CDE.

The Total Protective Action Recommendation Dose (TPARD) is calculated by dividing the yearly allowable Technical Specification limit (500 mRem/yr.) by the number of hours per year (8760 hr/yr.), and then multiplying by a factor of 200 times Technical Specifications [ODCM].

TPARD = 200x(Tech Spec Limit)/(hours per year)
= 200(500 mRem/yr.)/(8760 hr/yr.)
= 11.4 mRem/hr

The Committed Dose Equivalent (CDE) is calculated by dividing the yearly allowable Technical Specification limit (1500 mRem/yr.) by the number of hours per year (8760 hr/yr.), and then multiplying by a factor of 200 times Technical Specifications [ODCM].

CDE = 200x(Tech Spec Limit)/(hours per year)
= 200(1500 mRem/yr.)/(8760 hr/yr.)
= 34.2 mRem/hr

This event will be escalated to a Site Area Emergency when actual or projected doses are determined to exceed 10CFR20 annual average population exposure limits.

DEVIATION

None

REFERENCES

NUMARC NESP-007 AA1.1
Offsite Dose Calculation Manual
NUMARC Questions and Answers, June 1993, "Abnormal Rad Levels/Radiological Effluents #9"

5.0 Radioactivity Release

5.1 Effluent Release and Dose

ALERT - 5.1.2.b

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

EAL

Confirmed sample analyses for gaseous or liquid releases indicates concentrations or release rates exceeding **TWO HUNDRED TIMES** Tech Specs (Liquid Release ODCM. 3.8.B.1 and Gaseous Release ODCM 3.8.C.1.b) for ***> 15 minutes***

MODE All

BASIS

Releases in excess of two hundred times technical specifications that continue for > 15 minutes represent an uncontrolled situation and hence a potential degradation in the level of safety. The primary concern is the final integrated dose [100 times greater than the Unusual Event] and the degradation in plant control implied by the fact that the release was not isolated within 15 minutes.

It is not intended that the release be averaged over 15 minutes, but exceed two hundred times technical specifications limits for 15 minutes. Further, it is intended that the event be declared as soon as it is determined that the release will exceed two hundred times technical specifications for greater than 15 minutes.

An indication or report is considered to be valid when it is verified by:
1. An instrument channel check
2. Indications on related or redundant instruments
3. By direct observation by plant personnel

The calculation called for in this EAL should also be conducted whenever a liquid release occurs for which a radioactive discharge permit wasn't prepared or that exceeds the conditions on the permit (e.g. minimum dilution, alarm setpoints, etc).

This event will be escalated to higher classifications based on plant conditions.

DEVIATION

None

REFERENCES

NUMARC NESP-007 AA1.2
Offsite Dose Calculation Manual
T-104, Radioactivity Release Control

5.0 Radioactivity Release

5.1 Effluent Release and Dose

SITE AREA EMERGENCY - 5.1.3

IC 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

EAL

A valid reading on one or more of the following radiation monitors that exceeds or is expected to exceed the value shown for **> *15 minutes*** **AND** Dose Projections are not available:

Main Stack	5.84 μCi/cc	Vent Stack	2.08E-3 μCi/cc
Torus Vent	203 cpm		

Note: If the dose projections cannot be completed within the 15 minute period, then the declaration must be made based on the valid sustained monitor reading.

OR

Projected offsite dose using computer dose model exceeds ***100 mRem TPARD*** **OR** ***500 mRem*** **child thyroid CDE**

OR

Analysis of Field Survey results indicate site boundary whole body dose rate exceeds ***100 mRem/hr*** expected to continue for more than one hour, **OR** Analysis of Field Survey results indicate child thyroid dose commitment of ***500 mRem*** for one hour of inhalation

MODE All

BASIS

Valid means that a radiation monitor reading has been confirmed by the operators to be correct.

A monitor reading is considered to be valid when it is verified by:

1. An instrument check indicating the monitor has not failed;
2. Indications on related or redundant instrumentation; or,
3. Direct observation by plant personnel.

Total Protective Action Recommendation Dose (TPARD) is equal to Total Effective Dose Equivalent (TEDE) + 4 Day Deposition Dose. Committed Dose Equivalent (CDE) is equal to the thyroid exposure due to iodine. The computerized dose model provides projected TPARD and CDE.

An actual or projected dose of 100 mrem Total Protective Action Recommendation Dose (TPARD) is based on the 10 CFR 20 annual average population exposure limit. This value also provides a desirable gradient (one order of magnitude) between the Site Area Emergency and General Emergency classifications. The 500 mrem integrated child thyroid dose was established in consideration of the 1:5 ratio of the EPA Protective Action Guidelines for TPARD and Child Thyroid Committed Dose Equivalent (CDE). Actual meteorology is used, since it gives the most accurate dose projection.

Monitor indications are calculated using the computerized dose model with UFSAR source terms applicable to each monitored pathway in conjunction with annual average meteorology and a one hour release duration. The inputs are as follows:

	Main Stack	Vent Stack	Torus Vent
Stability Class	E	E	E
Wind Speed	11.4 mph	6.3 mph	6.3 mph
Wind Direction	45°	22°	22°
Accident	LOCA	LOCA	LOCA
Release Rate	5.84 μCi/cc	2.08E-3 μCi/cc	203 cpm

Child thyroid dose factors, rather than adult thyroid dose factors, are used for consistency with Pennsylvania Emergency Management Agency (PEMA) / Bureau of Radiation Protection (BRP).

This event will be escalated to a General Emergency when actual or projected doses exceed EPA Protective Action Guidelines per EAL Section 5.1.4.

DEVIATION

None

REFERENCES

NUMARC NESP-007, AS1.1, AS1.3 and AS1.4
EPA 400

5.0 Radioactivity Release

5.1 Effluent Release and Dose

GENERAL EMERGENCY - 5.1.4

IC Boundary Dose Resulting from an Actual or Imminent Release of Gaseous Radioactivity that Exceeds 1000 mR Whole Body or 5000 mR Child Thyroid for the Actual or Projected Duration of the Release Using Actual Meteorology

EAL

A valid reading on one or more of the following radiation monitors that exceeds or is expected to exceed the value shown for **> *15 minutes*** **AND** Dose Projections are not available:

Main Stack	58.4 μCi/cc	Vent Stack	2.08E-2 μCi/cc
Torus Vent	2000 cpm		

Note: If the required dose projections cannot be completed within the 15 minute period, then the declaration must be made based on the valid sustained monitor reading.

OR

Projected offsite dose using computer dose model exceeds ***1000 mRem TPARD*** **OR** ***5000 mRem*** **child thyroid CDE**

OR

Analysis of Field Survey results indicate site boundary whole body dose rate exceeds ***1000 mRem/hr*** expected to continue for more than one hour, **OR** Analysis of Field Survey results indicate child thyroid dose commitment of ***5000 mRem*** for one hour of inhalation

MODE All

BASIS

Valid means that a radiation monitor reading has been confirmed by the operators to be correct.

A monitor reading is considered to be valid when it is verified by:

1. An instrument check indicating the monitor has not failed;
2. Indications on related or redundant instrumentation; or,
3. Direct observation by plant personnel.

Total Protective Action Recommendation Dose (TPARD) is equal to Total Effective Dose Equivalent (TEDE) + 4 Day Deposition Dose. Committed Dose Equivalent (CDE) is equal to the thyroid exposure due to iodine. The computerized dose model provides projected TPARD and CDE.

The 1000 mR TPARD and the 5000 mR child thyroid integrated dose are based on the EPA protective action guidance. This is consistent with the emergency class description for a General Emergency. This level constitutes the upper level of the desirable gradient for the

Site Area Emergency. Actual meteorology is specifically identified in the initiating condition since it gives the most accurate dose assessment.

Monitor indications are calculated using the computerized dose model with UFSAR source terms applicable to each monitored pathway in conjunction with annual average meteorology and a one hour release duration. The inputs are as follows:

	Main Stack	Vent Stack	Torus Vent
Stability Class	E	E	E
Wind Speed	11.4 mph	6.3 mph	6.3
Wind Direction	45°	22°	22°
Accident	LOCA	LOCA	LOCA
Release Rate	58.4 μCi/cc	2.08E-2 μCi/cc	2.026E+3 cpm

Child thyroid dose factors, rather than adult thyroid dose factors, are used for consistency with Pennsylvania Emergency Management Agency (PEMA) / Bureau of Radiation Protection (BRP).

DEVIATION

None

REFERENCES

NUMARC NESP-007, AG1.1, AG1.3 and AG1.4
EPA-400

5.0 Radioactivity Release

5.2 In-Plant Radiation

UNUSUAL EVENT - 5.2.1

IC Unexpected Rise in Plant Radiation or Airborne Concentration

EAL

Valid Direct Area Radiation Monitor readings rise 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.

MODE All

BASIS

Valid means that a radiation monitor reading has been confirmed by the operators to be correct.

An area monitor reading is considered to be valid when it is verified by:
1. an instrument channel check indicating the monitor has not failed;
2. indications on related or redundant instrumentation; or
3. direct observation by plant personnel

This EAL addresses unplanned increases in in-plant radiation levels that represent a degradation in the control of radioactive material, and represents a potential degradation in the level of safety of the plant.

This event will be escalated to an Alert when radiation levels increase in areas required for the safe shutdown of the plant resulting in impeded access.

DEVIATION

None

REFERENCES

NUMARC NESP-007, AU2.4
T-103, Secondary Containment Control

5.0 Radioactivity Release

5.2 In-Plant Radiation

ALERT - 5.2.2.a

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

EAL

Valid radiation level readings **> *5000 mR/hr*** in areas requiring infrequent access to maintain plant safety functions as identified in procedure SE-1 or SE-10 **AND** Access is required for safe plant operation, but is impeded, due to radiation dose rates

MODE All

BASIS

Valid means that a radiation monitor reading has been confirmed by the operators to be correct.

An area monitor reading is considered to be valid when it is verified by:

1. An instrument check indicating the monitor has not failed;
2. Indications on related or redundant instrumentation; or,
3. Direct observation by plant personnel.

The single value of 5000 mR/hr was selected because it is based on radiation levels which result in exposure control measures intended to maintain doses within normal occupational exposure guidelines and limits (i.e., 10 CFR 20), and in doing so, will impede necessary access. Stay times for levels up to that value are, generally several minutes, enough time to enter an area and manually operate the equipment.

This EAL addresses increased radiation levels that impede necessary access to operating stations, or other areas containing equipment that must be operated manually, in order to maintain safe operation or perform a safe shutdown. These areas are identified in procedures SE-1 and SE-10. Use of these procedures will indicate the need to access the areas. It is this impaired ability to operate the plant that results in the actual or potential substantial degradation of the level of safety of the plant. The cause and/or magnitude of the increase in radiation levels is not a concern of this IC. The Emergency Director must consider the source or cause of the increased radiation levels and determine if any other IC may be involved. For example, a dose rate of 15 mR/hr in the control room or hi radiation monitor readings may also be indicative of high dose rates in the containment due to a LOCA. In this latter case, a SAE or GE may be indicated by the fission product barrier table.

This EAL could result in declaration of an Alert at one unit due to a radioactivity release or radiation shine resulting from a major accident at the other unit.

This EAL is not meant to apply to increases in drywell radiation monitors, as these are events which are addressed in the fission product barrier table. Nor is it intended to apply to anticipated temporary increases due to planned events (e.g., incore detector movement, radwaste container movement, depleted resin transfers, etc.)

This event will be escalated to a Site Area Emergency when loss of control of radioactive materials cause significant offsite doses.

DEVIATION

None

REFERENCES

NUMARC NESP-007, AA3.2
T-103, Secondary Containment Control
SE-1, Plant Shutdown from the Remote Shutdown Panel
SE-10, Plant Shutdown from the Alternative Shutdown Panels

5.0 Radioactivity Release

5.2 In-Plant Radiation

ALERT - 5.2.2.b

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

EAL

Valid Control Room <u>OR</u> Central Alarm Station radiation reading ***> 15 mR/hr***

MODE All

BASIS

Valid means that a radiation monitor reading has been confirmed by the operators to be correct.

An area monitor reading is considered to be valid when it is verified by:

1. An instrument check indicating the monitor has not failed;
2. Indications on related or redundant instrumentation; or,
3. Direct observation by plant personnel.

The EAL address radiation levels which would impede operation of systems required to maintain safe operations or to establish or maintain cold shutdown. Radiation levels could be indicated by ARM or radiological survey.

Plant normal and emergency procedures may be implemented without requiring any areas except the Control Room and Central Alarm Station to be continuously occupied. The Radwaste Control Room is not required to be continuously occupied in order to maintain plant safety functions since inputs to radwaste will be isolated with a secondary containment isolation and releases can only be performed manually.

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

This event will be escalated to a Site Area Emergency when loss of control of radioactive materials cause significant offsite doses.

DEVIATION

None

REFERENCES

NUMARC NESP-007 AA3.1

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6.0 Loss of Power

6.1 Loss of AC or DC Power

UNUSUAL EVENT - 6.1.1.a

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

EAL

The following conditions exist: Loss of Power to 2 and 3 Startup and Emergency Aux. Transformers and 343 Startup Transformer for ***>15 minutes*** **AND** At least ***Two*** Diesel Generators are supplying power to their respective 4 KV emergency busses

MODE All

BASIS

This EAL addresses the loss of offsite AC power supplying the station. Offsite power is fed through 2 and 3 Startup and Emergency Aux. Transformers and 343 Startup Transformer. Loss of offsite power will cause a reactor scram and a containment isolation. All four (4) emergency Diesel Generators will be available to carry the essential loads for each unit (the four Diesel Generators are shared between each unit). Balance of Plant systems that would assist in plant operations (i.e., condensate pumps, etc.) may be unavailable due the loss of power.

Prolonged loss of AC power reduces required redundancy and potentially degrades the level of safety of the plant by rendering the plant more vulnerable to a complete Loss of AC Power (Station Blackout). Fifteen minutes was selected as a threshold to exclude transient or momentary power losses.

Implementation of this EAL is based on the number of powered 4 KV buses per unit.

Escalation of this event to an Alert would be based on having a loss of all offsite AC power coincident with onsite AC power being reduced to a single power source in Modes 1, 2, and 3 or having a loss of all offsite and onsite AC power in Modes 4 or 5.

DEVIATION

None

REFERENCES

NUMARC NESP-007, SU1
SE-11, Station Blackout

6.0 Loss of Power

6.1 Loss of AC or DC Power

UNUSUAL EVENT - 6.1.1.b

IC Unplanned Loss of Required DC Power During Cold Shutdown or Refueling Mode for Greater than 15 Minutes

EAL

The following conditions exist:

Unplanned Loss of ALL safety related DC Power indicated by **< *107.5 VDC*** bus voltage indications for DC Panels 2(3)0D21, 22, 23, 24

AND

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

MODE 4, 5

BASIS

The purpose of this EAL is to recognize a loss of DC power compromising the ability to monitor and control the removal of decay heat during Cold Shutdown or Refueling operations. This EAL is intended to be anticipatory in as much as the operating crew may not have necessary indication and control of equipment needed to respond to the loss. The safety related 125 volt DC Distribution Panels are as follows:

Unit 2	Unit 3
20D21	30D21
20D22	30D22
20D23	30D23
20D24	30D24

107.45 VDC bus voltage is based on the minimum bus voltage necessary for the operation of safety related equipment. The value of 107.5 VDC will be used for human factors concerns. This voltage value incorporates a margin of at least 15 minutes of operation before the onset of inability to operate those loads. This voltage is near the minimum voltage selected when battery sizing is performed.

Unplanned is included in this IC and EAL to preclude the declaration of an emergency as a result of planned maintenance activities. Routinely, plants will perform maintenance on a Train related basis during shutdown periods. It is intended that the loss of the operating (operable) train is to be considered. If this loss results in the inability to maintain cold shutdown, the escalation to an Alert will occur.

DEVIATION

None

REFERENCES

NUMARC NESP-007, SU7
SE-13, Loss of a 125/250 VDC Safety Related Bus

6.0 Loss of Power

6.1 Loss of AC or DC Power

ALERT - 6.1.2.a

IC 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

EAL

The following conditions exist: Loss of Power to 2 and 3 Startup and Emergency Aux. Transformers and 343 Startup Transformer for ***>15 minutes*** AND Only ***One*** 4 KV emergency bus powered from a Single Onsite Power Source due to the Loss of: Three of Four Division Diesel Generators, D/G Output Breakers, or 4 KV Emergency Busses as indicated by bus voltage

MODE 1, 2, 3

BASIS

This EAL is intended to provide an escalation from "Loss of offsite Power for greater than 15 minutes." This condition is a degradation of the offsite and onsite power systems such that any additional failure would result in a station blackout. Fifteen (15) minutes has been selected to exclude transient or momentary power losses. However, an Alert should be declared in less than 15 minutes if it can be determined in less than 15 minutes that the power loss is not transient or momentary.

Depending on the 4 KV AC bus that remains energized there is a disparity in the systems that may be available. The ability to remove heat from the containment via Torus cooling may be lost due to the need to operate the remaining available RHR pump in other than Torus cooling (e.g., LPCI). As such there is a decrease in the systems available to remove heat transferred to the containment and there is an ongoing release of energy from the reactor to the containment (via SRVs, HPCI and/or RCIC operation). The ability to cool the nuclear fuel, remove decay heat, and control containment parameters is severely limited. Should equipment be unavailable prior to the loss of power, functions necessary to maintain the plant in a cold shutdown condition may be threatened.

Implementation of this EAL is based on the number of powered 4 KV buses per unit.

Escalation of this event would be based on the loss of the remaining Emergency Diesel Generator.

DEVIATION

None

REFERENCES

NUMARC NESP-007, SA5
SE-11, Station Blackout

6.0 Loss of Power

6.1 Loss of AC or DC Power

ALERT - 6.1.2.b

IC Loss of All Offsite Power and Loss of All Onsite AC Power to Essential Busses During Cold Shutdown Or Refueling Mode

EAL

The following conditions exist: Loss of Power to 2 and 3 Startup and Emergency Aux. Transformers and 343 Startup Transformer **AND** Failure to restore power to at least ***One*** 4 KV emergency bus ***within 15 minutes*** from the time of loss of both offsite and onsite AC power

MODE 4, 5, D

BASIS

Loss of all AC power compromises all plant safety systems requiring electric power including RHR, ECCS, Containment Heat Removal, Spent Fuel Heat Removal and the Ultimate Heat Sink. When in cold shutdown, refueling, or defueled mode, the event can be classified as an Alert, because of the significantly reduced decay heat, lower temperature and pressure, increasing the time to restore one of the emergency busses, relative to that specified for the Site Area Emergency EAL. Escalating to Site Area Emergency, if appropriate, is be Effluent Release/In-Plant Radiation, or Emergency Director Judgment.

Fifteen (15) minutes has been selected to exclude transient or momentary power losses. However, an Alert should be declared in less than 15 minutes if it can be determined in less than 15 minutes that the power loss is not transient or momentary.

Implementation of this EAL is based on the number of powered 4 KV buses per unit.

DEVIATION

None

REFERENCES

NUMARC NESP-007, SA1
SE-11, Station Blackout

6.0 Loss of Power

6.1 Loss of AC or DC Power

SITE AREA EMERGENCY - 6.1.3.a

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

EAL

The following conditions exist: Loss of Power to 2 and 3 Startup and Emergency Aux. Transformers and 343 Startup Transformer **AND** Failure to restore power to at least ***One*** 4 KV emergency bus ***within 15 minutes*** from the time of loss of both offsite and onsite AC

MODE 1, 2, 3

BASIS

Control Room annunciators would indicate that all offsite and onsite AC power feeds have been lost. Loss of all AC power compromises all plant safety systems requiring electric power including RHR, ECCS, Containment Heat Removal, Spent Fuel Heat Removal, High Pressure Service Water, and Emergency Service Water. Although instrumentation (supplied through instrument inverters) and DC power loads would be available, their operability would be limited to the amount of stored energy contained in their respective batteries. Instrumentation, communication equipment, and in-plant lighting and ventilation will be significantly hampered by the loss of all AC power.

Fifteen (15) minutes has been selected to exclude transient or momentary power losses. However, an Alert should be declared in less than 15 minutes if it can be determined in less than 15 minutes that the power loss is not transient or momentary.

Implementation of this EAL is based on the number of powered 4 KV buses per unit.

Escalation of this event would be based on the time that the Emergency Diesel Generator are unavailable.

DEVIATION

None

REFERENCES

NUMARC NESP-007, SS1
SE-11, Station Blackout

6.0 Loss of Power

6.1 Loss of AC or DC Power

SITE AREA EMERGENCY - 6.1.3.b

IC Loss of All Vital DC Power

EAL

Loss of ALL Safety Related DC Power indicated by **< *107.5 VDC*** on DC Panels 2(3)0D21, 22, 23, 24 for **> *15 minutes***

MODE 1, 2, 3

BASIS:

A loss of all DC power compromises the ability to monitor and control plant functions. 125 Volt DC system provides control power to engineered safety features valve actuation, diesel generator auxiliaries, plant alarm and indication circuits as well as the control power for the associated load group. If 125 Volt DC power is lost for an extended period of time (greater than 15 minutes) critical plant functions such as RPS Logic, Alternate Rod Insertion, Emergency Service Water Indication, 4KV Breaker Controls, HPCI, RCIC and RHR pump controls required to maintain safe plant conditions may not operate and core uncovery with subsequent reactor coolant system and primary containment failure might occur. The 125 volt DC Main Distribution Panel Busses are as follows:

Unit 2	Unit 3
20D21	30D21
20D22	30D22
20D23	30D23
20D24	30D24

Loss of all DC Power causes the loss of the following equipment:

- Alternate Rod Insertion
- ADS
- HPCI
- RCIC
- Normal EDG Control
- Normal Recirculation Pump Trip
- Containment Instrument Gas Compressors
- Other 4KV Circuit Breakers (e.g., RHR, CS, CRD)

Loss of ADS creates a loss of low pressure ECCS due to the inability to depressurize the reactor. In addition, loss of these buses will eventually lead to MSIV closure and reactor trip due to the loss of the Containment Instrument Gas Compressor as a result of suction valve closure. Subsequent to MSIV closure, much of the equipment noted above will be required for plant stabilization and shutdown.

A sustained loss of DC power will threaten the ability to remove heat from the reactor core, resulting in eventual fuel clad damage. The loss of DC power will also result in the loss of the ability to remove heat from the containment. SRVs will remain operable in the relief mode and the heat addition to the containment could result in a loss of the primary containment as a fission product release barrier.

107.45 VDC bus voltage is based on the minimum bus voltage necessary for the operation of safety related equipment. This EAL uses 107.5 VDC for human factors concerns. This voltage value incorporates a margin of at least 15 minutes of operation before the onset of inability to operate those loads. This voltage is near the minimum voltage selected when battery sizing is performed.

DEVIATION

None

REFERENCES

NUMARC NESP-007, SS3
T-101, RPV Control
T-102, Primary Containment Control
SE-11, Station Blackout

6.0 Loss of Power

6.1 Loss of AC or DC Power

GENERAL EMERGENCY - 6.1.4

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

EAL

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

Loss of Power to 2 and 3 Startup and Emergency Aux. Transformers and 343 Startup Transformer

AND

Failure of ALL Emergency Diesel Generators to supply power to 4 KV emergency busses

AND

At least one of the following conditions exist:

- Restoration of at least ***One*** 4 KV emergency bus within ***2 hours*** is ***NOT*** likely

  OR

- Reactor Water Level cannot be maintained **> *-172 "***

  OR

- Torus temperature is on the **"UNSAFE"** side of the Heat Capacity Temperature Limit (HCTL) curve (T-102, T/T-1)

MODE 1, 2, 3

BASIS

Loss of all AC power compromises all plant safety systems requiring electric power including RHR, ECCS, Containment Heat Removal and the Ultimate Heat Sink. Prolonged loss of all AC power will lead to loss of fuel clad, RCS, and containment. The two hours to restore AC power is based on the site blackout coping analysis as described below. Although this IC may be viewed as redundant to the Fission Product Barrier Degradation IC, its inclusion is necessary to better assure timely recognition and emergency response.

10 CFR 50.2 defines Station Blackout (SBO) as complete loss of AC power to essential and non-essential buses. SBO does not include loss of AC Power to busses fed by station batteries through inverters, nor does it assume a concurrent single failure or design basis accident. Successful SBO coping maintains the following key parameters within given acceptable limits:

1. Reactor water level > -172" (TAF)
2. Torus level low enough to prevent HPCI and/or RCIC steam exhaust line flooding
3. Reactor pressure >150 psig to maintain HPCI and RCIC operable
4. Containment pressure < 60 psig, design limit
5. Torus temperature < 200 degrees F, HPCI/RCIC lube oil temperature concern when suction aligned to Torus

6. Drywell temperature
   <200 degrees F indefinitely
   <250 degrees F 99 days
   <320 degrees F 18 hours
   <340 degrees F 3 hours

Successful extended SBO coping depends on ability to keep HPCI/RCIC available for injection, and ability to maintain RPV depressurized for low pressure injection should HPCI and RCIC become unavailable. Control power for HPCI, RCIC and SRVs is provided by 125V DC. The parameters listed above can be maintained as long as the batteries are intact. Two hours is the earliest the batteries would fail, and thus is the basis for the time limit in this EAL.

The significance of a station blackout relative to the loss of fission product release barriers is that all three barriers will eventually be lost due to the inability to remove heat from the fuel and the containment. Although the RCS will be intact the longest, eventually SRVs will operate in the relief mode due to RPV over-pressurization and if the containment has already failed then there is a direct bypass of the RCS boundary.

Implementation of this EAL is based on the number of powered 4 KV buses per unit.

DEVIATION

None

REFERENCES

NUMARC NESP-007, SG1
SE-11, Station Blackout
T-101, RPV Control
T-102, Primary Containment Control
T-104, Radioactivity Release Control

7.0 Internal Events

7.1 Technical Specification & Control Room Evacuation

UNUSUAL EVENT - 7.1.1

IC Inability to Reach Required Shutdown Within Technical Specification Limits

EAL

Inability to reach required shutdown mode within Tech. Spec. LCO required action completion time.

MODE 1, 2, 3

BASIS

Limiting Conditions of Operation (LCOs) require the plant to be brought to a required shutdown mode when the Technical Specification required configuration cannot be restored. Depending on the circumstances, this may or may not be an emergency or precursor to a more severe condition. In any case, the initiation of plant shutdown required by the site Technical Specifications requires a one hour report under 10 CFR 50.72 (b) Non-emergency events. The plant is within its safety envelope when being shut down within the allowable action statement time in the Technical Specifications. An immediate Notification of an Unusual Event is required when it is determined that the plant cannot be brought to the required operating mode within the allowable action statement time in the Technical Specifications. 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. Other required Technical Specification shutdowns that involve precursors to more serious events are addressed by other various EAL Sections.

DEVIATION

None

REFERENCES

NUMARC NESP-007, SU2
Technical Specifications

7.0 Internal Events

7.1 Technical Specification & Control Room Evacuation

ALERT - 7.1.2

IC Control Room Evacuation Has Been Initiated

EAL

Entry into SE-1 or SE-10 procedure for Control Room evacuation

MODE All

BASIS

Control Room evacuation requires establishment of plant control from outside the control room (e.g., local control and remote shutdown panel) and support from the Technical Support Center and/or other emergency facilities as necessary. Control Room evacuation represents a serious plant situation since the level of control is not as complete as it would be without evacuation. The establishment of system control outside of the Control Room will bypass many protective trips and interlocks. In addition, much of the instrumentation and assessment tools available in the Control Room will not be available.

This event will be escalated to a Site Area Emergency if control cannot be established within fifteen minutes.

DEVIATION

None

REFERENCES

NUMARC NESP-007, HA5
SE-10, Alternate Shutdown
SE-1 Plant Shutdown from the Remote Shutdown Panel

7.0 Internal Events

7.1 Technical Specification & Control Room Evacuation

SITE AREA EMERGENCY - 7.1.3

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

EAL

The following conditions exist: Control room evacuation has been initiated AND Control of the plant cannot be established per SE-1 or SE-10 within ***15 minutes***

MODE All

BASIS

Transfer of safety system control has not been performed in an expeditious manner but it is unknown if any damage has occurred to the fission product barriers. The 15 minute time limit for transfer of control is based on a reasonable time period for personnel to leave the control room, arrive at the remote shutdown area, and reestablish plant control to preclude core uncovery and/or core damage. During this transitional period the function of monitoring and/or controlling parameters necessary for plant safety may not be occurring and as a result there may be a threat to plant safety.

This event will be escalated based upon system malfunctions or damage consequences.

DEVIATION

None

REFERENCES

NUMARC NESP-007, HS2
SE-10, Alternate Shutdown
SE-1, Plant Shutdown from the Remote Shutdown Panel

7.0 Internal Events

7.2 Loss of Decay Heat Removal Capability

ALERT - 7.2.2

IC Inability to Maintain Plant in Cold Shutdown

EAL

The following conditions exist:

Unplanned Loss of <u>ALL</u> Tech Spec required systems available to provide Decay Heat Removal functions

<u>**AND**</u>

Uncontrolled Temperature rise that either:

- Exceeds ***212 °F***
  (Excluding a <15 minute rise >212° F with a heat removal function restored)

  <u>**OR**</u>

- Results in temperature rise approaching ***212 °F***
  (with <u>NO</u> heat removal function restored)

MODE 4, 5

BASIS

This EAL addresses complete loss of functions required for core cooling during refueling and cold shutdown modes. A loss of Technical Specifications components is paired with exceeding temperature limits to acknowledge additional plant capabilities to maintain plant cooling. Escalation to Site Area Emergency or General Emergency would be via Effluent Release/In-Plant Radiation or Emergency Director Judgment ICs.

The statement "Unplanned Loss of <u>ALL</u> Tech Spec required systems available to provide Decay Heat Removal functions" is intended to represent a complete loss of functions available, or an inadequate ability, to provide core cooling during the Cold Shutdown and Refueling Modes, including alternate decay heat removal methods. This EAL allows for actions taken in ON-125, "Loss of Shutdown Cooling - Procedure," to reestablish RHR in the Shutdown Cooling Mode or provide for alternate methods of decay heat removal, with the intent of maintaining RCS temperature below 212° F.

For loss of an in-service Decay Heat Removal system with other decay heat removal methods available, actions taken to provide for restoration of a decay heat removal function may require time to implement. If the event results in RCS temperature "momentarily" (for less than 15 minutes) rising above 212°F with heat removal capability restored, Emergency Director/Shift Management judgment will be required to determine whether heat removal systems are adequate to prevent boiling in the core and restoration of RCS temperature control. Momentary (not to exceed 15 minutes) unplanned excursions above 212° F, when alternate decay heat removal capabilities exist, should not be classified under this EAL.

"Uncontrolled" means that system temperature rise is not the result of planned actions by the plant staff.

The EAL guidance related to 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.

This EAL is concerned with the ability to keep the reactor core temperature less than 212 °F. The criteria of uncontrolled Reactor Coolant temperature rise > 212 °F is met as soon as it becomes known that sufficient cooling cannot be restored in time to maintain the temperature < 212 °F, regardless of the current temperature. The inability to establish alternate methods of decay heat removal indicates that either alternate methods are unavailable to cool the core in the RPV or when the steam is transferred to the Torus, Torus cooling is unavailable. Loss of Torus cooling will result in a continuing, uncontrolled increase in reactor coolant temperature.

Escalation to the Site Area Emergency is by EAL IC, "Loss of Water Level in the Reactor Vessel that has or will uncover Fuel in the Reactor Vessel," or by Effluent Release/In-Plant Radiation ICs.

DEVIATION

None

REFERENCES

NUMARC NESP-007, SA3
ON-125, Loss of Shutdown Cooling - Procedure
Technical Specifications

7.0 Internal Events

7.2 Loss of Decay Heat Removal Capability

SITE AREA EMERGENCY - 7.2.3

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

EAL

Loss of TORUS heat sink capabilities as evidenced by T-102 T/T legs directing a T-112 Emergency Blowdown

MODE 1, 2, 3

BASIS:

This EAL addresses complete loss of functions, including ultimate heat sink, required for hot shutdown with the reactor at pressure and temperature. Reactivity control is addressed in other EALs. The loss of heat removal function is indicated by T-102 T/T legs requiring an Emergency Blowdown which is directed when the Heat Capacity Temperature Limit (HCTL) curve is exceeded.

Under these conditions, there is an actual major failure of a system intended for protection of the public. Thus, declaration of a Site Area Emergency is warranted. Escalation to General Emergency would be via Effluent Release/In-Plant Radiation, Emergency Director Judgment, or Fission Product Barrier Degradation ICs.

DEVIATION

None

REFERENCES

NEI 97-03, SSA
T-102, Primary Containment Control, SP/L-8

7.0 Internal Events

7.3 Loss of Assessment / Communication Capability

UNUSUAL EVENT - 7.3.1.a

IC Unplanned Loss of Most or All Safety System Annunciation or Indication in The Control Room for Greater Than 15 Minutes

EAL

Unplanned loss of most or all safety system annunciators (Table 7-1) **OR** indicators (Table 7-2) for > **15 minutes** requiring increased surveillance to safely operate the unit(s).

MODE 1, 2, 3

BASIS

This EAL recognizes the difficulty associated in monitoring conditions without normal annunciators. In the opinion of the Shift Supervisor this loss of annunciators requires increased surveillance to safely operate the plant. It is not intended that a detailed count of instrumentation be performed, but that only a rough approximation be used to determine the severity of the loss. The Plant Monitoring System (PMS) is available to provide compensatory indication. Fifteen minutes is used as a threshold to exclude transient or momentary power losses. Unplanned loss of annunciators excludes scheduled maintenance and testing activities. Control Room panels with annunciators and direction for response are included in ON-123, Loss of Control Room Annunciators.

Table 7-1 indicates those system annunciator panels considered to be safety related:

Table 7-1 Safety System Annunciators

ECCS
Containment Isolation
Reactor Trip
Process Radiation Monitoring

Table 7-2 indicates those indications important for monitoring:

Table 7-2 Safety Function Indicators

Reactor Power
Decay Heat Removal
Containment Safety Functions

Reportability of Technical Specification imposed shutdowns, or the inability to comply with Technical Specification action statements is covered in EAL section, Technical Specifications.

This EAL is not applicable in cold shutdown or refueling modes due to the limited number of safety systems required for operation.

This event will be escalated to an Alert if a transient is in progress or if compensatory indications become unavailable.

DEVIATION

None

REFERENCES

NUMARC NESP-007, SU3
ON-123, Loss of Control Room Annunciators
AIT A0004447, EP Self Assessment on Salem Loss of Annunciators

7.0 Internal Events

7.3 Loss of Assessment / Communication Capability

UNUSUAL EVENT - 7.3.1.b

IC Unplanned Loss of All Onsite or Offsite Communications Capabilities

EAL

Loss of ALL Onsite communications (Table 7-3) affecting the ability to perform routine operations

OR

Loss of ALL Offsite communications (Table 7-3)

MODE All

BASIS

This EAL recognizes a loss of communication ability that significantly degrades the plant operations staff's ability to perform tasks necessary for plant operations or the ability to communicate with offsite authorities. This EAL is separated into two groups of communications, Onsite and Offsite. A complete loss of either group is so severe, that the Unusual Event declaration is warranted. Table 7-2 is identified as follows:

Table 7-3 Communications

	Onsite	Offsite
Site Phones (GTE System)	X	X
OMNI System	X	X
Plant Public Address	X	
Station Radio	X	
NRC (FTS-2000)		X
PA State Police Radio		X
Load Dispatcher Radio		X
PECO Dial Network		X

There is no escalation to an Alert for loss of communications, although there is escalation to higher classifications if other communications for plant assessment is lost.

DEVIATION

None

REFERENCES

NUMARC NESP-007, SU6
Nuclear Emergency Plan

7.0 Internal Events

7.3 Loss of Assessment / Communication Capability

ALERT - 7.3.2

IC 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

EAL

Unplanned loss of most or all safety system annunciators (Table 7-1) **OR** indicators (Table 7-2) for > 15 minutes requiring increased surveillance to safely operate the unit(s) **AND EITHER** A significant plant transient is in progress (Table 7-4) **OR** the plant monitoring system (PMS) is unavailable.

MODE 1, 2, 3

BASIS

This EAL recognizes the difficulty associated in monitoring conditions without normal annunciators. In the opinion of the Shift Supervisor this loss of annunciators requires increased surveillance to safely operate the plant. This EAL represents an increase in severity above 7.3.1.a in that the Plant Monitoring System (PMS) can not provide compensatory indication, or that a significant transient is in progress.

Table 7-1 indicates those system annunciator panels considered to be safety related:

Table 7-1 Safety System Annunciators
ECCS
Containment Isolation
Reactor Trip
Process Radiation Monitoring

Table 7-2 indicates those indications important for monitoring:

Table 7-2 Safety Function Indicators
Reactor Power
Decay Heat Removal
Containment Safety Functions

Table 7-4, significant plant transients include response to automatic or manually initiated actions including:

Table 7-4 Plant Transients
SCRAM
Recirc runbacks > 25% thermal power
Sustained power oscillations 25% peak to peak
Stuck open relief valves
ECCS injection

Fifteen minutes is used as a threshold to exclude transient or momentary power loses. Control Room panels with annunciators and direction for restoration is included in ON-123, Loss of Control Room Annunciators.

Reportability of Technical Specification imposed shutdowns, or the inability to comply with Technical Specification action statements is covered in EAL section, Technical Specifications.

This EAL is not applicable in cold shutdown or refueling modes due to the limited number of safety systems required for operation.

This event will be escalated to a Site Area Emergency if a transient is in progress, the Plant Monitoring System is unavailable and a loss of annunciators occurs.

DEVIATION

None

REFERENCES

NUMARC NESP-007, SA4
ON-123, Loss of Control Room Annunciators
T-101, Bases
BWROG EPG/SAG (RC/Q-6)

7.0 Internal Events

7.3 Loss of Assessment / Communication Capability

SITE AREA EMERGENCY - 7.3.3

IC Inability to Monitor a Significant Transient in Progress

EAL

Loss of safety system annunciators (Table 7-1) **AND** indicators (Table 7-2) **AND** PMS **AND** a significant plant transient is in progress. (Table 7-4)

MODE 1, 2, 3

BASIS

This EAL recognizes the difficulty associated in monitoring conditions without normal annunciators. In the opinion of the Shift Supervisor this loss of annunciators requires increased surveillance to safely operate the plant. This EAL represents an increase in severity above 7.3.2 in that the Plant Monitoring System can not provide compensatory indication, and that a significant transient is in progress.

Table 7-1 indicates those system annunciator panels considered to be safety related:

Table 7-1 Safety System Annunciators
ECCS
Containment Isolation
Reactor Trip
Process Radiation Monitoring

Table 7-2 indicates those indications important for monitoring:

Table 7-2 Safety Function Indicators
Reactor Power
Decay Heat Removal
Containment Safety Functions

Table 7-4 significant plant transients include response to automatic or manually initiated actions including:

Table 7-4 Plant Transients
SCRAM
Recirc runbacks >25% thermal power change
Sustained power oscillations 25% peak to peak
Stuck open relief valves
ECCS injection

Planned maintenance or testing activities are included in this EAL due to the significance of this event. Control Room panels with annunciators and the restoration is included in ON-123, Loss of Control Room Annunciators.

DEVIATION

None

REFERENCES

NUMARC NESP-007, SS6
ON-123, Loss of Control Room Annunciators
T-101, Bases
BWROG EPG/SAG (RC/Q-6)

8.0 External Events

8.1 Security Events

UNUSUAL EVENT - 8.1.1

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

EAL

Credible sabotage or bomb threat within the Protected Areas **OR** Credible intrusion and attack threat to the Protected Areas **OR** Attempted intrusion and attack to the Protected Areas **OR** Attempted sabotage discovered within the Protected Areas **OR** Hostage/Extortion situation that threatens normal plant operations

MODE All

BASIS

A security threat that is identified as being directed towards the station and represents a potential degradation in the level of safety of the plant. A security threat is satisfied if physical evidence supporting the threat exists, if information independent from the actual threat exists, or if a specific group claims responsibility for the threat. The Shift Management will declare an Unusual Event subsequent to consulting with the on shift Security representative to determine the credibility of the security event.

Security threats which meet the threshold for declaration of an Unusual Event are:

1. Credible sabotage or bomb threat within the Protected Areas
2. Credible intrusion and attack threat to the Protected Areas
3. Attempted intrusion and attack to the Protected Areas
4. Attempted sabotage discovered within the Protected Areas
5. Hostage/Extortion situation that threatens normal plant operations

Security events which do not represent a potential degradation in the level of safety of the plant are reported under 10 CFR 73.71 or 10 CFR 50.72 and will not cause an Unusual Event to be declared.

This event will be escalated to an Alert based upon a hostile intrusion or act within the Protected Areas.

DEVIATION

A bomb device discovered within Plant Protected Areas and outside the Plant Vital Areas is an Alert declaration as determined per the site Safeguards Contingency Plan and therefore is not included as an Unusual Event in the EAL scheme.

REFERENCES

NUMARC NESP-007, HU4.1 and HU4.2
Safeguards Contingency Plan
Physical Security Plan

8.0 External Events

8.1 Security Events

ALERT - 8.1.2

IC Security Event in a Plant Protected Area

EAL

Intrusion into plant protected areas by a hostile force **OR** Confirmed bomb, sabotage or sabotage device discovered in the Protected Areas

MODE All

BASIS

This class of security event represents an escalated threat to the level of safety of the plant. This event is satisfied if physical evidence supporting the hostile intrusion or attack exists. The Shift Management will declare an Alert subsequent to consulting with the on shift Security representative to determine the validity of the entry conditions.

Security threats which meet the threshold for declaration of an Alert are:

1. Intrusion into plant protected areas by a hostile force
2. Confirmed bomb, sabotage or sabotage device discovered within the Protected Areas

This event will be escalated to a Site Area Emergency based upon a hostile intrusion or act in plant Vital Areas.

DEVIATION

None

REFERENCES

NUMARC NESP-007, HA4.1 and HA4.2
Safeguards Contingency Plan
Physical Security Plan

8.0 External Events

8.1 Security Events

SITE AREA EMERGENCY - 8.1.3

IC Security Event in a Plant Vital Area

EAL

Intrusion into plant Vital area by a hostile force **OR** Confirmed bomb, sabotage or sabotage device discovered in a Vital Area

MODE All

BASIS

This class of security event represents an escalated threat to plant safety above that contained in an Alert in that a hostile intrusion or attack has progressed from the Protected Area to a Vital Area. The Vital Areas are within the Protected Area and are generally controlled by key card readers. These areas contain vital equipment which includes any equipment, system, device or material, the failure, destruction or release of could directly or indirectly endanger the public health and safety by exposure to radiation. Equipment or systems which would be required to function to protect health and safety following such failure, destruction or release are also considered vital.

Security threats which meet the threshold for declaration of a Site Area Emergency are:

1. Intrusion into plant Vital area by a hostile force
2. Confirmed bomb, sabotage or sabotage device discovered in a Vital Area

This event will be escalated to a General Emergency based upon the loss of physical control of the Control Room or Remote Shutdown Capability

DEVIATION

None

REFERENCES

NUMARC NESP-007, HS1.1 and HS1.2
Safeguards Contingency Plan
Physical Security Plan

8.0 External Events

8.1 Security Events

GENERAL EMERGENCY - 8.1.4

IC Security Event Resulting in Loss of Ability to Reach and Maintain Cold Shutdown

EAL

Loss of physical control of the control room due to security event **OR** Loss of physical control of all remote shutdown capability due to security event

MODE All

BASIS

This class of security event represents conditions under which a hostile force has taken physical control of areas required to reach and maintain cold shutdown. Loss of Remote Shutdown Capability would occur if the control function of the Remote Shutdown Panels was lost.

Security events which meet the threshold for declaration of a General Emergency are physical loss of the Control Room or the Remote and Alternate Shutdown Panels.

This situation leaves the plant in a very unstable condition with a high potential of multiple barrier failures.

DEVIATION

None

REFERENCES

NUMARC NESP-007, HG1.1 and HG1.2
Safeguards Contingency Plan
Physical Security Plan

8.0 External Events

8.2 Fire / Explosion and Toxic / Flammable Gases

UNUSUAL EVENT - 8.2.1.a

IC Fire Within Protected Area Boundary Not Extinguished Within 15 Minutes of Detection

EAL

Fire within ON-114 Plant Vital Structures (Table 8-1) which is not extinguished within ***15 minutes*** of control room notification or verification of a control room alarm

MODE All

BASIS

The purpose of this IC is to address the magnitude and extent of fires that may be potentially significant precursors to damage to safety systems. This excludes such items as fires within administration buildings, waste-basket fires, and other small fires of no safety consequence. This IC applies to buildings and areas contiguous to plant vital areas or other significant buildings or areas. The intent of this IC is not to include buildings (e.g., warehouses) or areas that are not contiguous or immediately adjacent to plant vital areas. Verification of the alarm in this context means those actions taken in the control room to determine that the control room alarm is not spurious.

This EAL addresses fires in Plant Vital Structures that house safety systems. These fires may be precursors to damage to safety systems contained in these structures. There are no areas/buildings contiguous to Plant Vital Structures which could effect a safety system in one of the listed Plant Vital Structures except for those already on the list. Therefore, no additional areas/buildings are considered for this EAL. Verification that a fire exists is by operator actions to confirm that fire alarms received in the Control Room are not spurious or by any verbal notification by plant personnel. Fifteen minutes has been established to allow plant staff to respond and control small fires or to verify that no fire exists. Table 8-1 Plant Vital Structures are as follows:

Table 8-1 Plant Vital Structures

Power Block
Diesel Generator Building
Emergency Pump Structure
Inner Screen Structure
Emergency Cooling Tower

This event will be escalated to an Alert if the fire damages redundant trains of plant safety systems required for the current operating condition.

DEVIATION

None

REFERENCES

NUMARC NESP-007, HU2

8.0 External Events

8.2 Fire / Explosion and Toxic / Flammable Gases

UNUSUAL EVENT - 8.2.1.b

IC Release of Toxic or Flammable Gasses Deemed Detrimental to Safe Operation of the Plant

EAL

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

MODE All

BASIS

This EAL addresses toxic/flammable gas releases within the Protected Area in concentrations high enough to affect health of plant personnel or the safe operation of the plant. This includes releases that originate both onsite and offsite. A toxic/flammable gas is considered to be any substance that is dangerous to life or limb by reason of inhalation or skin contact. A gas release is considered to be impeding normal plant operations if concentrations are high enough to restrict normal operator movements. It also includes areas where access is only possible with respiratory equipment, as this equipment restricts normal visibility and mobility. It should not be construed to include confined spaces that must be ventilated prior to entry or situation involving the Fire Brigade who are using respiratory equipment during the performance of their duties unless it also affects personnel not involved with the Fire Brigade.

An offsite event (such as a tanker truck accident or train derailment releasing toxic gases) may place the Protected Area within the evacuation area. This evacuation is determined from the DOT Evacuation Tables for Selected Hazardous Materials in the DOT Emergency Response Guide for Hazardous Materials.

DEVIATION

None

REFERENCES

NUMARC NESP-007, HU3.1 and HU3.2

8.0 External Events

8.2 Fire / Explosion and Toxic / Flammable Gases

UNUSUAL EVENT - 8.2.1.c

IC Natural and Destructive Phenomena Affecting the Protected Area

EAL

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

MODE All

BASIS

The protected area boundary is typically that part within the security isolation zone and is defined in the site security plan.

Only those explosions of sufficient force to damage permanent structures or equipment within the protected area 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. No attempt is made in this EAL to assess the actual magnitude of the damage. The occurrence of the explosion with reports of evidence of damage (e.g., deformation, scorching) is sufficient for declaration. The Emergency Director also needs to consider any security aspects of the explosion, if applicable.

Any security aspects of this event should be considered under EAL Section 8.1, Security Events.

This event will be escalated to an Alert if the explosion damages one or more redundant trains of plant safety systems required for the current operating condition.

DEVIATION

None

REFERENCES

NUMARC NESP-007, HU1.5

8.0 External Events

8.2 Fire / Explosion and Toxic / Flammable Gases

ALERT - 8.2.2.a

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

EAL

The following conditions exist:

Fire or explosion which potentially makes inoperable:

Two or More subsystems of a Safe Shutdown System (Table 8-2) **OR** *Two or More* Safe Shutdown Systems **OR** Plant Vital Structures containing Safe Shutdown Equipment

AND

Safe Shutdown System or Plant Vital Structure is required for the present Operational Mode

MODE All

BASIS

The primary concern of this EAL is the magnitude of the fire and the effects on Safe Shutdown Systems required for the present Operational Mode. A Safe Shutdown System is defined as any system required to maintain safe operation or to establish or maintain Cold Shutdown. A system being "inoperable" means that it is incapable of performing the design function. For example, the LPCI System is intended to maintain adequate core cooling by covering the core to at least 2/3 core height following a DBA LOCA. In order for the system to be unable to maintain its intended function, multiple loops would need to be disabled by the fire. In addition to indication of degraded system performance, potential inoperability may be determined by visual observation and other control room indications such as loss of indicating lights.

Table 8-2 Safe Shutdown Systems

Diesel Generators	4KV Safeguard Buses	ADS
HPCI	RCIC	RHR (All Modes)
Core Spray	HPSW	ESW
SBGTS	ECW	CAC/CAD
PCIS	Control Room Ventilation	

Safe Shutdown Analysis is consulted to determine systems required for the applicable mode.

Two examples of applying this methodology are as follows:

- Diesel Generators and 4 KV Safeguard Buses

  The fire disables multiple Diesel Generators or 4 KV Safeguard Buses so that the number of emergency power systems available would be decreased to below what would be required to mitigate an accident under the current operating conditions. For 100% power, this could be conservatively interpreted as at least two Diesel Generators or 4 KV Buses disabled.

- RHR - LPCI Mode

  The fire disables multiple loops of LPCI so that adequate core submergence could not be assured following a DBA LOCA. For 100% power, this could also be conservatively interpreted as at least two loops disabled.

The EAL includes the condition that the fire must make "TWO OR MORE" subsystems or "TWO OR MORE" systems inoperable. In those cases where it is believed that the fire may have caused damage to *Safety Systems*, then an Alert declaration is warranted, since the full extent of the damage may not be known. For Plant Vital Structure damage, classification is required under this EAL if the structure houses or otherwise supports *Safety Systems* required for the present Operational Mode.

Degraded system performance or observation of damage that could degrade system performance is used as the indicator that the safe shutdown system was actually affected or made inoperable. A report of damage should not be interpreted as mandating a lengthy damage assessment prior to classification. No attempt is made in this EAL to assess the actual magnitude of damage. The occurrence of the fire or explosion with reports of damage (e.g., deformation, scorching) is sufficient for declaration.

Fire is defined as combustion characterized by the generation of heat and smoke. Sources of smoke such as overheated electrical equipment and slipping drive belts, for example, do not constitute fires. Observation of a flame is preferred, but is NOT required if large quantities of smoke and heat are observed.

This event will be escalated to higher classifications based upon damage consequences covered under other various EAL Sections.

DEVIATION

None

REFERENCES

NUMARC NESP-007, HA2
PBAPS Safe Shutdown Analysis
NUMARC Questions and Answers, June 1993, "Hazards Question #7"

8.0 External Events

8.2 Fire / Explosion and Toxic / Flammable Gases

ALERT - 8.2.2.b

IC 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

EAL

Report or detection of toxic gases within Plant Vital Structures (Table 8-1) in concentrations that will be life threatening to plant personnel **OR** Report or detection of flammable gases within Plant Vital Structures (Table 8-1) in concentrations affecting the safe operation of the plant

MODE All

BASIS

This EAL recognizes that toxic/flammable gases have entered Plant Vital Structures and are affecting safe operation of the plant by impeding operator access to the safety systems that must be operated manually in these structures. The cause and/or magnitude of the gas concentrations is not a concern, but rather that access is required to an area and is impeded. Plant Vital Structures that must be accessed are as follows:

Table 8-1 Plant Vital Structures
Power Block
Diesel Generator Building
Emergency Pump Structure
Inner Screen Structure
Emergency Cooling Tower

The intent of this IC is not to include buildings (e.g., warehouses) or other areas that are not contiguous or immediately adjacent to plant Vital Areas. It is appropriate that increased monitoring be done to ascertain whether consequential damage has occurred. This event will be escalated to higher classifications based upon damage consequences covered under other various EAL Sections.

DEVIATION

None

REFERENCES

NUMARC NESP-007, HA3.1 and HA3.2

8.0 External Events

8.3 Man-Made Events

UNUSUAL EVENT - 8.3.1.a

IC Destructive Phenomena Affecting the Protected Area

EAL

Vehicle crash within protected area boundary that may potentially damage plant structures containing functions and systems required for safe shutdown of the plant.

MODE All

BASIS

This EAL is intended to address such items as plane, helicopter, or train crash that may potentially damage plant structures containing functions and systems required for safe shutdown of the plant. If the crash is confirmed to affect a plant vital area, the event may be escalated to Alert.

DEVIATION

None

REFERENCES

NUMARC NESP-007, HU1.4

8.0 External Events

8.3 Man-Made Events

UNUSUAL EVENT - 8.3.1.b

IC Destructive Phenomena Affecting the Protected Area

EAL

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

MODE All

BASIS

This EAL is intended to address main turbine rotating component failures of sufficient magnitude to cause observable damage to the turbine casing or to the seals of the turbine generator. Of major concern is the potential for leakage of combustible fluids (e.g., lubricating oils) and gases (e.g., hydrogen) to the plant environs. Actual fires and flammable gas build up are appropriately classified via other EALs. This EAL is consistent with the definition of an Unusual Event while maintaining the anticipatory nature desired and recognizing the risk to non-safety related equipment. Escalation of the emergency classification is based on potential damage done by missiles generated by the failure or by the radiological releases and would be classified by the radiological ICs or Fission Product Barrier ICs.

Turbine failure of sufficient magnitude to cause observable damage to the turbine casing or seals of the turbine generator increases the potential for leakage of combustible fluids and gases (Hydrogen cooling) to the Turbine Building. The damage should be readily observable and should not require equipment disassembly to locate.

DEVIATION

None

REFERENCES

NUMARC NESP-007, HU1.6

8.0 External Events

8.3 Man-Made Events

ALERT - 8.3.2

IC Destructive Phenomena Affecting the Plant Vital Area

EAL

Vehicle crash affecting Plant Vital Structures (Table 8-1) **OR** Turbine failure generated missiles result in any visible structural damage to or penetration of any Plant Vital Structures (Table 8-1)

MODE All

BASIS

This EAL address crashes of vehicles or missile impacts that have caused damage to Plant Vital Structures, and thus damage may be assumed to have occurred to safe shutdown systems. No attempt should be made to assess the magnitude of damage to Plant Vital Structures prior to classification. The evidence of damage is sufficient for declaration. A vehicle crash includes aircraft and large motor vehicles, such as a crane. Missile impacts including flying objects from offsite, onsite rotating equipment or turbine failure causing casing penetration. Table 8-1 Plant Vital Structures are as follows:

Table 8-1 Plant Vital Structures
Power Block
Diesel Generator Building
Emergency Pump Structure
Inner Screen Structure
Emergency Cooling Tower

This event will be escalated to higher classifications based upon damage consequences covered under other various EAL Sections.

DEVIATION

None

REFERENCES

NUMARC NESP-007, HA1.5 and HA1.6

8.0 External Events

8.4 Natural Events

UNUSUAL EVENT - 8.4.1.a

IC Natural and Destructive Phenomena Affecting the Protected Area

EAL

Earthquake **>*.01 g*** as determined by procedure SO 67.7.A

MODE All

BASIS

This EAL addresses a sensed earthquake. The magnitude of .01g is the lowest detectable earthquake measured on PBAPS seismic instrumentation per S0 67.7.A. An earthquake of this magnitude may be sufficient to cause minor damage to plant structures or equipment within the Protected Area. Damage is considered to be minor, as it would not affect physical or structural integrity. This event is not expected to affect the capabilities of plant safety functions.

This event will be escalated to an Alert if the earthquake reaches an Operating Basis Earthquake.

DEVIATION

None

REFERENCES

NUMARC NESP-007, HU1.1
SE-5, Earthquake and Bases
UFSAR, section 1.6

8.0 External Events

8.4 Natural Events

UNUSUAL EVENT - 8.4.1.b

IC Natural and Destructive Phenomena Affecting the Protected Area

EAL

Report by plant personnel of tornado striking within protected areas **OR** Wind speeds **> *75 mph*** as indicated on site Meteorological data for **> *15 minutes***

MODE All

BASIS

A tornado touching down within the Protected Areas or wind speeds > 75 mph within the owner controlled Area are of sufficient velocity to have the potential to cause damage to Plant Vital Structures. The value of 75 mph was selected to maintain consistency with plant value and to coincide with the Beaufort Scale for Hurricane wind speed winds of 73-136 mph. These conditions are indicative of unstable weather conditions and represent a potential degradation in the level of safety of the plant. Verification of a tornado will be by direct observation and reporting by station personnel. Verification of wind speeds > 75 mph will be via meteorological data in the control room. For purposes of this EAL, sustained is > 15 minutes.

This event will be escalated to an Alert if the tornado or high wind speeds strike Plant Vital Structures. If it is determined that the tornado or high wind speeds have caused a loss of shutdown cooling, then escalation will be by EAL IC, Loss of Decay Heat Removal Capability.

DEVIATION

None

REFERENCES

NUMARC NESP-007, HU1.2 and HU1.7

8.0 External Events

8.4 Natural Events

UNUSUAL EVENT - 8.4.1.c

IC Natural and Destructive Phenomena Affecting the Protected Area

EAL

Assessment by the control room that an event has occurred. (Natural and Destructive Phenomena Affecting the Protected Areas)

MODE All

BASIS

This EAL allows for the control room to determine that an event has occurred and take appropriate action based on personal assessment as opposed to verification (e.g., an earthquake is felt but does not register on any plant-specific instrumentation, etc.)

DEVIATION

None

REFERENCES

NUMARC NESP-007, HU1.3

8.0 External Events

8.4 Natural Events

UNUSUAL EVENT - 8.4.1.d

IC Natural and Destructive Phenomena Affecting the Protected Area

EAL

High River level **> *112'*** **<u>OR</u>** Low River level **< *98.5'***

MODE All

BASIS

High River level of greater than 112 feet on instrument LI-2(3)278A,B,C or LI-2(3)278A,B,C is indication of the river being in flood. By procedure, the units will be SCRAMmed and be brought to cold shutdown.

Low River level of less than 98.5 feet is indication of loss of Conowingo Pond and loss of circulation water pumps. Procedures require the unit to be SCRAMmed and brought to cold shutdown.

This event will be escalated to an Alert classification based continuation of the river situation.

DEVIATION

None

REFERENCES

NUMARC NESP-007, HU1.7
SE-4, Flood
SE-3, Loss of Conowingo Pond

8.0 External Events

8.4 Natural Events

ALERT - 8.4.2.a

IC Natural and Destructive Phenomena Affecting the Plant Vital Area

EAL

Earthquake **>*.05 g*** (Operating Basis Earthquake OBE) as determined by procedure SO 67.7.A

MODE All

BASIS

This EAL addresses an earthquake that exceeds the Operating Basis Earthquake level of .05g and is beyond design basis limits. An earthquake of this magnitude may be sufficient to cause damage to safety related systems and functions.

The Max Credible Earthquake for PBAPS is 0.12g per UFSAR section 1.6, therefore this EAL is conservative and warrants an Alert classification.

This event will be escalated to a higher emergency classification based upon damage consequences covered under other various EAL Sections.

DEVIATION

None

REFERENCES

NUMARC NESP-007, HA1.1
SE-5, Earthquake and Bases
UFSAR section 1.6

8.0 External Events

8.4 Natural Events

ALERT - 8.4.2.b

IC Natural and Destructive Phenomena Affecting the Plant Vital Area

EAL

Tornado or wind speeds **> *75 mph*** causing damage to Plant Vital Structures (Table 8-1)

MODE All

BASIS

This EAL is based on FSAR design basis. Wind loads of this magnitude can cause damage to safety functions.

This EAL addresses events where Plant Vital Structures have been struck with high winds, and thus damage may have occurred to safe shutdown systems. No attempt should be made to assess the magnitude of damage to Plant Vital Structures prior to classification. Table 8-1 Plant Vital Structures are as follows:

Table 8-1 Plant Vital Structures
Power Block
Diesel Generator Building
Emergency Pump Structure
Inner Screen Structure
Emergency Cooling Tower

This event will be escalated to a higher emergency classification based upon damage consequences covered under other various EAL Sections.

DEVIATION

None

REFERENCES

NUMARC NESP-007, HA1.2

8.0 External Events

8.4 Natural Events

ALERT - 8.4.2.c

IC Natural and Destructive Phenomena Affecting the Plant Vital Area

EAL

Report of any visible structural damage to any Plant Vital Structure (Table 8-1)

MODE All

BASIS

This EAL specifies the Plant Vital Structures which contain systems and functions required for safe shutdown of the plant. Table 8-1 Plant Vital Structures are as follows:

Table 8-1 Plant Vital Structures
Power Block
Diesel Generator Building
Emergency Pump Structure
Inner Screen Structure
Emergency Cooling Tower

Other site structures listed in the NUMARC document are not plant vital structures and are not required for safe shutdown. Those are: RWST, CST.

This event will be escalated to a higher emergency classification based upon damage consequences covered under other various EAL Sections.

DEVIATION

None

REFERENCES

NUMARC NESP-007, HA1.3

8.0 External Events

8.4 Natural Events

ALERT - 8.4.2.d

IC Natural and Destructive Phenomena Affecting the Plant Vital Area

EAL

High River level **> *116'*** OR Low River Level **< *92.5'***

MODE All

BASIS

High River level > 116 feet is indication of the river being in flood. This level is capable of causing flooding that can affect Plant Vital Structures. No attempt should be made to determine the magnitude of flooding. This is a long lead time event but this level is ground elevation of the reactor building and intake pump structure so classification as an Alert Event is appropriate. The evidence of flooding is sufficient for declaration.

Low River level < 92.5 feet is indication of loss of Conowingo Pond and loss of circulation water pumps. Procedures require the unit to be SCRAMmed and brought to cold shutdown and utilization of the ECW pump and Emergency Cooling Tower.

This event will be escalated to a higher emergency classification based upon damage consequences covered under other various EAL Sections.

DEVIATION

None

REFERENCES

NUMARC NESP-007, HA1.7
SE-4, Flood
SE-3, Loss of Conowingo Pond

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9.0 Other

9.1 General

UNUSUAL EVENT - 9.1.1

IC Other Conditions Existing Which in the Judgment of the Emergency Director Warrant Declaration of an Unusual Event

EAL

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

MODE All

BASIS

This EAL allows the Shift Management to declare an Unusual Event upon the determination that the level of safety of the plant has degraded. Where the degradation is associated with equipment or system malfunctions, the decision that it is degraded should be made upon functionality, not operability. A system, subsystem, train, component or device, though degraded in equipment condition or configuration, should be considered functional if it is capable of maintaining respective system parameters within acceptable design limits.

Releases of radioactive materials requiring offsite response or monitoring are not expected to occur at this level unless further degradation of safety systems occurs. However, if one does occur, it will be classified under "Radioactivity Releases."

DEVIATION

None

REFERENCES

NUMARC NESP-007, HU5

9.0 Other

9.1 General

ALERT - 9.1.2

IC Other Conditions Existing Which in the Judgment of the Emergency Director Warrant Declaration of an Alert

EAL

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.

MODE All

BASIS

This EAL allows the Shift Management to declare an Alert upon the determination that the level of safety of the plant has substantially degraded but is not explicitly addressed by other EALs. This includes a determination by Shift Management that the TSC and OSC should be activated and command and control functions should be transferred for the event to be effectively mitigated. Transfer of command and control functions is used as an initiator since an event significant to warrant transfer is a substantial reduction in the level of safety of the plant. Other examples are:

Internal flooding affects the operability of plant safety systems required to establish or maintain cold shutdown.

Releases that are expected will be limited to a small fraction of the EPA Protective Action Guidelines and will be classified under "Radioactivity Releases."

DEVIATION

None

REFERENCES

NUMARC NESP-007, HA6

9.0 Other

9.1 General

SITE AREA EMERGENCY - 9.1.3

IC Other Conditions Existing Which in the Judgment of the Emergency Director Warrant Declaration of Site Area Emergency

EAL

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

MODE All

BASIS

This EAL allows the Shift Management to declare a Site Area Emergency upon the determination of an actual or likely major failure of plant functions needed for protection of the public, but is not explicitly addressed by other EALs.

Releases are not expected to result in exposure levels which exceed the EPA Protective Action Guidelines except within the site boundary and will be classified under "Radioactivity Releases."

DEVIATION

None

REFERENCES

NUMARC NESP-007, HS3

9.0 Other

9.1 General

GENERAL EMERGENCY - 9.1.4

IC Other Conditions Existing Which in the Judgment of the Emergency Director Warrant Declaration of General Emergency

EAL

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

MODE All

BASIS

This EAL allows the Shift Management to declare a General Emergency upon the determination of an actual or imminent substantial core degradation or melting with the potential for loss of containment integrity, but is not explicitly addressed by other EALs.

Releases may exceed the EPA Protective Action Guidelines for more than the immediate site area and will be classified under "Radioactivity Releases."

DEVIATION

None

REFERENCES

NUMARC NESP-007, HG2

PROCEDURE INDEX REPORT:

FAC	DOC TYPE	PROC TYPE	PROCEDURE NUMBER	CURR REV NBR	TITLE	EFFECTIVE DATE	RESP GROUP	SYSTEM NBR
PB	PROC	ERP	ERP-C-1000	0005	EMERGENCY OPERATIONS FACILITY (EOF) ACTIVATION/DECACTIVATION	04/21/99	PWE	
PB	PROC	ERP	ERP-C-1000-1	0002	EOF ACTIVATION CHECKLIST	04/21/99	PWE	
PB	PROC	ERP	ERP-C-1000-2	0003	EOF DEACTIVATION CHECKLIST	04/21/99	PWE	
PB	PROC	ERP	ERP-C-1000-3	0000	EOF BUSINESS HOURS FIRST RESPONDER CHECKLIST	04/21/99	PWE	
PB	PROC	ERP	ERP-C-1000-4	0000	EOF AFTER HOURS FIRST RESPONDER CHECKLIST	04/21/99	PWE	
PB	PROC	ERP	ERP-C-1100	0003	EOF STAFF AUGMENTATION- CANCELLED - REPLACED BY ERP-C-1250	09/14/94	PWE	
PB	PROC	ERP	ERP-C-1200	0009	EMERGENCY REPSONSE MANAGER	04/03/00	PWE	
PB	PROC	ERP	ERP-C-1200-1	0000	EMERGENCY RESPONSE MANAGER TURNOVER/BRIEFING FORM	09/14/94	PWE	
PB	PROC	ERP	ERP-C-1200-2 EXH	0000	PROTECTIVE ACTION RECOMMENDATION WORKSHEET CANCELLED REPLACED BY ERP-C-1200	10/24/95	PWE	
PB	PROC	ERP	ERP-C-1200-3	0000	ERM PAR DELIVERY CHECKLIST	04/03/00	PWE	
PB	PROC	ERP	ERP-C-1210	0002	ASSISTANT EMERGENCY RESPONSE MANAGER (AERM) CANCELLED - REPLACED BY ERP-C-1200	10/24/95	PWE	
PB	PROC	ERP	ERP-C-1250	0003	EMERGENCY PREPAREDNESS COORDINATOR/EOF	11/02/98	PWE	
PB	PROC	ERP	ERP-C-1250-1	0000	EMERGENCY POWER INSTRUCTIONS	09/14/94	PWE	
PB	PROC	ERP	ERP-C-1250-2	0001	EMERGENCY PREPAREDNESS COORDINATOR INSTRUCTIONS FOR ASPEN BACKUP NOTIFICATION SYSTEM	04/02/98	PWE	
PB	PROC	ERP	ERP-C-1250-3	0000	EMERGENCY PREPAREDNESS COORDINATOR INSTRUCTIONS TO STOP STAFFING	09/14/94	PWE	
PB	PROC	ERP	ERP-C-1250-4	0000	EMERGENCY PREPAREDNESS COORDINATOR INSTUCTIONS FOR SYSTEM RESET	09/14/94	PWE	
PB	PROC	ERP	ERP-C-1300	0010	EMERGENCY OPERATIONS FACILITY (EOF) DOSE ASSESSMENT TEAM LEADER	08/31/00	PWE	
PB	PROC	ERP	ERP-C-1300-1	0003	DOSE ASSESSMENT TEAM LEADER INITIAL ACTIONS	04/04/00	PWE	
PB	PROC	ERP	ERP-C-1300-2	0000	DOSE ASSESSMENT TURNOVER LIST	09/23/94	PWE	
PB	PROC	ERP	ERP-C-1300-3	0003	PROTECTIVE ACTION RECOMMENDATION WORKSHEET	11/02/98	PWE	
PB	PROC	ERP	ERP-C-1300-4	0000	OFFSITE SAMPLE ANALYSIS REQUESTS	09/23/94	PWE	
PB	PROC	ERP	ERP-C-1300-5	0001	DETERMINATION OF PROTECTIVE ACTION RECOMMENDATIONS (PARS)	11/02/98	PWE	
PB	PROC	ERP	ERP-C-1300-6	0001	DOSE ASSESSMENT GROUP INITIAL ACTIONS	04/10/98	PWE	
PB	PROC	ERP	ERP-C-1300-7	0000	OBTAINING EPDS MET/RAD DATA	03/26/97	PWE	
PB	PROC	ERP	ERP-C-1300-8	0000	USE OF MODE A/MODE B OF CDM	03/26/97	PWE	
PB	PROC	ERP	ERP-C-1300-9	0001	OBTAINING MET DATA FROM NATIONAL WEATHER SERVICE	09/12/97	PWE	
PB	PROC	ERP	ERP-C-1310	0003	EMERGENCY OPERATIONS FACILITY (EOF) DOSE ASSESSMENT GROUP - CANCELLED - REPLACED BY ERP-C-1300	03/26/97	PWE	
PB	PROC	ERP	ERP-C-1310-1	0000	DOSE ASSESSMENT GROUP LEADER INITIAL ACTIONS CANCELLED - REPLACED BY ERP-C-1300	03/26/97	PWE	
PB	PROC	ERP	ERP-C-1310-2	0000	OBTAINING MET DATA FROM NATIONAL WEATHER SERVICE CANCELLED - REPLACED BY ERP-C-1300	03/24/97	PWE	
PB	PROC	ERP	ERP-C-1310-3	0000	OBTAINING EPDS MET/RAD DATA - CANCELLED - NO REPLACED BY ERP-C-1300	03/26/97	PWE	
PB	PROC	ERP	ERP-C-1310-4	0000	USE OF MODE A/MODE B OF CDM CANCELLED - REPLACED BY ERP-C-1300	03/26/97	PWE	
PB	PROC	ERP	ERP-C-1320	0007	EMERGENCY OPERATIONS FACILITY (EOF) FIELD SURVEY GROUP LEADER	08/31/00	PWE	
PB	PROC	ERP	ERP-C-1320-1	0002	FIELD SURVEY GROUP LEADER INITIAL ACTIONS	04/10/98	PWE	
PB	PROC	ERP	ERP-C-1320-2	0001	FIELD SURVEY GROUP LEADER TURNOVER SHEET	03/26/97	PWE	
PB	PROC	ERP	ERP-C-1320-3	0002	FIELD SURVEY GROUP LEADER DATA SHEET	08/31/00	PWE	
PB	PROC	ERP	ERP-C-1400	0004	ENGINEERING SUPPORT TEAM	11/02/98	PWE	
PB	PROC	ERP	ERP-C-1400-1	0002	ENGINEERING SUPPORT TEAM CHECKLIST	11/02/98	PWE	
PB	PROC	ERP	ERP-C-1410	0002	CORE DAMAGE ASSESSMENT	09/09/98	PWE	
PB	PROC	ERP	ERP-C-1410-1	0000	RADIOLOGICAL DATA	09/14/94	PWE	
PB	PROC	ERP	ERP-C-1410-2	0001	HYDROGEN CONCENTRATION DATA	09/09/98	PWE	
PB	PROC	ERP	ERP-C-1410-3	0001	CONTAINMENT RADIATION MONITOR DATA	09/09/98	PWE	
PB	PROC	ERP	ERP-C-1410-4	0000	METAL WATER REACTION - CANCELLED NO REPLACEMENT	09/09/98	PWE	

FAC	DOC TYPE	PROC TYPE	PROCEDURE NUMBER	CURR REV NBR	TITLE	EFFECTIVE DATE	RESP GROUP	SYSTEM NBR
PB	PROC	ERP	ERP-C-1410-5	0001	PERCENT OF FUEL INVENTORY AIRBORNE IN THE CONTAINMENT VS. APPROXIMATE SOURCE AND DAMAGE ESTIMATE	09/09/98	PWE	
PB	PROC	ERP	ERP-C-1410-6	0001	PROCEDURES FOR ESTIMATING FUEL DAMAGE BASED ON MEASURED I-131 AND XE-133 CONCENTRATIONS	09/09/98	PWE	
PB	PROC	ERP	ERP-C-1500	0006	LOGISTICS SUPPORT TEAM	04/14/00	PWE	
PB	PROC	ERP	ERP-C-1500-1	0001	MESSAGE AND INFORMATION INSTRUCTIONS	10/24/95	PWE	
PB	PROC	ERP	ERP-C-1500-2	0001	HELICOPTER LANDING INFORMATION	10/24/95	PWE	
PB	PROC	ERP	ERP-C-1900	0004	RECOVERY PHASE IMPLEMENTATION	11/02/98	PWE	
PB	PROC	ERP	ERP-C-1900-1	0000	RECOVERY PHASE IMPLEMENTATION FLOW CHART	06/28/93	PWE	
PB	PROC	ERP	ERP-C-1900-2	0002	PEACH BOTTOM ATOMIC POWER STATION RECOVERY ACCEPTANCE CHECKLIST	04/02/98	PWE	
PB	PROC	ERP	ERP-C-1900-3	0002	LIMERICK GENERATING STATION RECOVERY ACCEPTANCE CHECKLIST	04/02/98	PWE	
PB	PROC	ERP	ERP-C-1900-4	0002	RECOVERY PLAN OUTLINE	04/02/98	PWE	
PB	PROC	ERP	ERP-C-1900-5	0002	ASSESSMENT CONSIDERATIONS	12/28/99	PWE	
PB	PROC	ERP	ERP-101	0022	CLASSIFICATION OF EMERGENCIES	08/15/00	PWE	
PB	PROC	ERP	ERP-101 BASES	0000	PBAPS EAL TECHNICAL BASIS MANUAL TABLE OF CONTNENTS	09/22/00	PWE	
PB	PROC	ERP	ERP-110	0012	EMERGENCY NOTIFICATIONS	08/06/98	PWE	
PB	PROC	ERP	ERP-110 APP 1	0055	EMERGENCY NOTIFICATION TELEPHONE LIST	08/07/00	PWE	
PB	PROC	ERP	ERP-110 APP 2	0024	EMERGENCY CLASSIFICATION NOTIFICATION TELEPHONE LIST FOR A SITE EMERGENCY OR GENERAL EMERGENCY CANCELLED - REPLACED BY ERP-110 APPENDIX 1	07/21/93	PWE	
PB	PROC	ERP	ERP-120	0002	PARTIAL PLANT EVACUATION CANCELLED - REPLACED BY ERP-130 & GP-15	08/10/92	PWE	
PB	PROC	ERP	ERP-130	0014	SITE EVACUATION	02/16/00	PWE	
PB	PROC	ERP	ERP-140	0019	EMERGENCY RESPONSE ORGANIZATION (ERO) CALL OUT	03/04/99	PWE	
PB	PROC	ERP	ERP-140 APP 1	0019	AUTOMATED ERO ACTIVATION	08/06/98	PWE	
PB	PROC	ERP	ERP-140 APP 2	0022	ASPEN EMERGENCY MESSAGE CANCELLED - REPLACED BY ERP-110 APP 1	08/06/98	PWE	
PB	PROC	ERP	ERP-140 APP 3	0022	DOSE ASSESSMENT TEAM CANCELLED - REPLACED BY PIMS PRINTOUTS ISSUED MONTHLY PER RT/ERP-2	08/20/92		
PB	PROC	ERP	ERP-140 APP 4	0015	CHEMISTRY SAMPLING & ANALYSIS TEAM CANCELLED - REPLACED BY PIMS PRINTOUTS ISSUED MONTHLY PER RT/ERP-2	08/20/92		
PB	PROC	ERP	ERP-140 APP 5	0014	DAMAGE REPAIR TEAM CANCELLED - REPLACED BY PIMS PRINTOUTS ISSUED MONTHLY PER RT/ERP-2	08/20/92		
PB	PROC	ERP	ERP-140 APP 6	0013	SECURITY TEAM CANCELLED - REPLACED BY PIMS PRINTOUTS ISSUED MONTHLY PER RT/ERP-2	08/20/92		
PB	PROC	ERP	ERP-140 APP 7	0017	PERSONNEL SAFETY TEAM CANCELLED - REPLACED BY PIMS PRINTOUTS ISSUED MONTHLY PER RT/ERP-2	08/20/92		
PB	PROC	ERP	ERP-140 APP 8	0009	COMPANY CONSULTANTS AND CONTRACTORS CANCELLED - INCLUDED IN EMERGENCY TELEPHONE DIRECTORY	08/20/92		
PB	PROC	ERP	ERP-140 APP 9	0011	NEARBY PUBLIC AND INDUSTRIAL USERS OF DOWNSTREAM WATER CANCELLED - INCLUDED IN EMERGENCY TELEPHONE DIRECTORY	08/20/92		
PB	PROC	ERP	ERP-200	0016	EMERGENCY DIRECTOR (ED)	07/10/00	PWE	
PB	PROC	ERP	ERP-200 APP 1	0003	EMERGENCY DIRECTOR CHECKLIST (MCR)	07/10/00	PWE	
PB	PROC	ERP	ERP-200 APP 2	0004	EMERGENCY DIRECTOR CHECKLIST (TSC)	07/10/00	PWE	
PB	PROC	ERP	ERP-200 APP 3	0004	EVENT NOTIFICATION FORM	07/10/00	PWE	
PB	PROC	ERP	ERP-200 APP 4	0004	STATION PUBLIC ADDRESS ANNONCEMENTS	07/10/00	PWE	
PB	PROC	ERP	ERP-200 APP 5	0003	PAR DEVELOPMENT AND ISSUANCE	07/10/00	PWE	
PB	PROC	ERP	ERP-200 APP 6	0001	DOSE ASSESSMENT DATA SHEET	07/10/00		
PB	PROC	ERP	ERP-200 APP 7	0000	TURNOVER/BREIFING FORM	07/10/00	PWE	
PB	PROC	ERP	ERP-205	0008	EMERGENCY PREPAREDNESS COORDINATOR/TSC	06/20/00	PWE	
PB	PROC	ERP	ERP-206	0007	SUPPORT SERVICES GROUP	03/03/00	PWE	
PB	PROC	ERP	ERP-210	0000	TRIP TABLE COMMUNICATOR (TSC)	09/12/97	PWE	

FAC	DOC TYPE	PROC TYPE	PROCEDURE NUMBER	CURR REV NBR	TITLE	EFFECTIVE DATE	RESP GROUP	SYSTEM NBR
PB	PROC	ERP	ERP-220	0006	OPERATIONS GROUP	10/05/95	PWE	
PB	PROC	ERP	ERP-230	0016	OPERATIONS SUPPORT CENTER (OSC) ACTIVATION	10/07/98	PWE	
PB	PROC	ERP	ERP-230 APP 1	0001	PERSONNEL EXPOSURE LOG OPERATIONS SUPPORT CENTER (OSC) CANCELLED - NO REPLACEMENT	11/28/95	PWE	
PB	PROC	ERP	ERP-250	0011	TECHNICAL SUPPORT CENTER (TSC) ACTIVATION CANCELLED - NO REPLACEMENT	10/14/93		
PB	PROC	ERP	ERP-300	0007	DOSE ASSESSMENT TEAM LEADER (DATL) CANCELLED - NO REPLACEMENT	09/23/94	PWE	
PB	PROC	ERP	ERP-301	0004	DOSE ASSESSMENT COORDINATOR (DAC)	08/29/00	PWE	
PB	PROC	ERP	ERP-305	0004	DOSE ASSESSMENT GROUP LEADER (DAGL) CANCELLED - NO REPLACEMENT	03/12/93		
PB	PROC	ERP	ERP-306	0000	LIMERICK RESPONSE FOR SHIFT DOSE ASSESSMENT PERSONNEL (SDAP)	06/30/00	PWE	
PB	PROC	ERP	ERP-310	0007	DOSE ASSESSMENT GROUP CANCELLED - NO REPLACEMENT	09/23/94	PWE	
PB	PROC	ERP	ERP-315	0014	OPERATION OF THE DOSE ASSESSMENT COMPUTER	04/24/00	PWE	
PB	PROC	ERP	ERP-318	0001	LIQUID RELEASE DOSE CALCULATIONS AT DOWNSTREAM WATER INTAKE FACILITIES CANCELLED - REPLACED BY ERP-360	06/18/93		
PB	PROC	ERP	ERP-319	0001	LIQUID RELEASE DOSE CALCULATIONS FOR FISH INGESTION CANCELLED - REPLACED BY ERP-360	06/18/93		
PB	PROC	ERP	ERP-325	0005	SHIFT DOSE ASSESSMENT PERSONNEL	08/25/98	PWE	
PB	PROC	ERP	ERP-325 APP 1	0000	CANCELLED - REPLACED BY MESOREM PROGRAM	03/03/95	PWE	
PB	PROC	ERP	ERP-330	0009	FIELD SURVEY GROUP LEADER (FSGL) CANCELLED - NO REPLACEMENT	09/23/94	PWE	
PB	PROC	ERP	ERP-340	0006	FIELD SURVEY GROUP	03/19/97	PWE	
PB	PROC	ERP	ERP-340 APP 1	0005	FIELD SURVEY DATA SHEET	08/29/00	PWE	
PB	PROC	ERP	ERP-360	0000	RADIOACTIVE LIQUID RELEASE CANCELLED - REPLACED BY ERP-315	06/23/94		
PB	PROC	ERP	ERP-400	0006	CHEMISTRY TEAM LEADER (CTL)	01/20/00	PWE	
PB	PROC	ERP	ERP-410	0009	CHEMISTRY GROUP	04/30/98	PWE	
PB	PROC	ERP	ERP-410 APP 1	0000	CHEMISTRY SAMPLE CHECK-OFF LIST CANCELLED - REPLACED BY ERP-410	12/11/96	PWE	
PB	PROC	ERP	ERP-410 APP 2	0000	CHEMISTRY SAMPLE AND ANALYSIS LOG SHEETCANCELLED - REPLACED BY ERP-410	12/11/96	PWE	
PB	PROC	ERP	ERP-500	0010	SECURITY TEAM LEADER (STL)	04/24/00	PWE	
PB	PROC	ERP	ERP-510	0009	PERSONNEL ACCOUNTABILITY CANCELLED - NO REPLACEMENT	11/28/95	PWE	
PB	PROC	ERP	ERP-520	0005	SECURITY GROUP LEADERS	11/28/95	PWE	
PB	PROC	ERP	ERP-520 APP 1	0000	UNIT 1 PERSONNEL LOG CANCELLED - NO REPLACEMENT	11/28/95	PWE	
PB	PROC	ERP	ERP-600	0013	HEALTH PHYSICS TEAM LEADER (HPTL)	07/07/99	PWE	
PB	PROC	ERP	ERP-610	0004	FIRST AID/SEARCH AND RESCUE GROUP CANCELLED - NO REPLACEMENT	02/05/93		
PB	PROC	ERP	ERP-620	0011	HEALTH PHYSICS GROUP (HPG)	09/04/98	PWE	
PB	PROC	ERP	ERP-620 APP 1	0000	HABITABILITY STATUS LOG SHEET	11/05/93	PWE	101
PB	PROC	ERP	ERP-620 APP 2	0000	ARM STATUS LOG	11/05/93	PWE	100
PB	PROC	ERP	ERP-620 APP 3	0002	HEALTH PHYSICS BRIEFING GUIDE	09/04/98	PWE	
PB	PROC	ERP	ERP-620 APP 4	0000	ACCESS BRIEFING GUIDE CANCELLED - NO REPLACEMENT	05/08/96	PWE	
PB	PROC	ERP	ERP-630	0003	DOSIMETRY, BIOASSAY, AND RESPIRATORY PROTECTION GROUP CANCELLED - NO REPLACEMENT	03/18/93		
PB	PROC	ERP	ERP-640	0006	VEHICLE AND EVACUEE CONTROL GROUP	05/28/97	PWE	
PB	PROC	ERP	ERP-640 APP 1	0000	CONTAMINATED VEHICLE SURVEY FORM CANCELLED - NO REPLACEMENT	05/28/97	PWE	
PB	PROC	ERP	ERP-640 APP 2	0000	UNCONTAMINATED VEHICLE FORM CANCELLED - NO REPLACEMENT	05/28/97	PWE	
PB	PROC	ERP	ERP-650	0006	TRANSPORT OF CONTAMINATED INJURY OFF-SITE	11/27/96	PWE	
PB	PROC	ERP	ERP-660	0007	ENTRY FOR EMERGENCY REPAIR AND OPERATIONS CANCELLED - REPLACED BY ERP-620	07/11/94		
PB	PROC	ERP	ERP-670	0004	EMERGENCY RADIATION EXPOSURE GUIDELINES AND CONTROLS	12/11/96	PWE	
PB	PROC	ERP	ERP-680	0007	CONTROL OF THYROID BLOCKING POTASSIUM IODIDE (KI) TABLETS	09/22/00	PWE	
PB	PROC	ERP	ERP-680 APP 1	0001	POTASSIUM IODIDE WORKSHEET	02/20/97	PWE	

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PB	PROC	ERP	ERP-680 APP 2	0000	POTASSIUM IODIDE CONSENT FORM	11/30/94	PWE	
PB	PROC	ERP	ERP-680 APP 3	0001	INSTRUCTION AND RECORD SHEET FOR PERSONS RECEIVING KI	02/20/97	PWE	
PB	PROC	ERP	ERP-680 APP 4	0001	KI AUTHORIZATION	02/20/97	PWE	
PB	PROC	ERP	ERP-700	0010	TECHNICAL SUPPORT TEAM	09/22/00	PWE	
PB	PROC	ERP	ERP-710	0008	TECHNICAL SUPPORT GROUP CANCELLED - REPLACED BY ERP-700	11/02/98	PWE	
PB	PROC	ERP	ERP-800	0006	OPERATIONS SUPPORT CENTER DIRECTOR (OSC DIRECTOR)	10/07/98	PWE	
PB	PROC	ERP	ERP-810	0011	MAINTENANCE TEAM	07/07/99	PWE	

** END OF REPORT **