ATTACHMENT 1

RESPONSE TO NRC REQUEST FOR ADDITIONAL INFORMATION

REGARDING THE PROPOSED CHANGE TO THE

EMERGENCY ACTION LEVEL SCHEME

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RESPONSE TO NRC REQUEST FOR ADDITIONAL INFORMATION REGARDING THE PROPOSED CHANGE TO THE EMERGENCY ACTION LEVEL SCHEME

By letter dated May 6, 2011, Nine Mile Point Nuclear Station, LLC (NMPNS) requested NRC approval to implement revised Emergency Action Levels (EALs) at Nine Mile Point Unit 1 (NMP1) and Nine Mile Point Unit 2 (NMP2) in accordance with 10 CFR 50, Appendix E, Section IV(B)(1). Specifically, the EAL scheme used at both NMP1 and NMP2 would change from NUMARC/NESP-007, Methodology for Development of Emergency Action Levels, Revision 2, to the scheme delineated in NEI 99-01, Methodology for Development of Emergency Action Levels, Revision 5. This attachment provides supplemental information in response to the request for additional information documented in the NRC's letter dated November 17, 2011. Each individual NRC question is repeated (in italics), followed by the NMPNS response.

1. *Definitions: Please provide site-specific definitions for the following:*
   - *Containment Closure, and*
   - *Vital Area.*

   Additionally, NMP2 definition of "Hostile Action" contains the wording "take FIREs" vice the wording "take HOSTAGEs" endorsed in NEI 99-01 (Revision 5). Please provide further justification for difference or revise consistent with endorsed guidance.

   Response RAI-1

   Neither NMP1 nor NMP2 have site specific definitions for the terms Containment Closure or Vital Area that meet the intent of NEI 99-01 generic definitions. The definitions used in the NMP1 and NMP2 EAL Bases documents are as provided in NEI 99-01 Revision 5.

   The NMP2 EAL Bases definition of Hostile Action has been revised to read:

   "An act toward NMP2 or its personnel that includes the use of violent force to destroy equipment, take HOSTAGEs, and/or intimidate the licensee to achieve an end. This includes attack by air, land, or water using guns, explosives, PROJECTILEs, vehicles, or other devices used to deliver destructive force. Other acts that satisfy the overall intent may be included."

2. *RA2.2 (NMP2): NMP1 EAL RA2.2 has procedural guidance listed to determine that the fuel is uncovered. For NMP2, please clarify methods of determining that the fuel is uncovered (e.g., video cameras) and revise the EAL accordingly.*

   Response RAI-2

   The NMP2 EAL Bases for RA2.2 is revised to include a corresponding statement and reference similar to the NMP1 EAL Bases for RA2.2. The following sentence is added to the NMP2 EAL Bases for RA2.2:

   "N2-SOP-39, Refuel Floor Events, provides appropriate instructions to report a visual observation of irradiated fuel uncovery (ref. 4)."

3. *RU2.1 (NMP2): Please clarify whether the following radiation monitors are applicable to this EAL and revise accordingly:*
    - *2HVR*RE14A, and*
    - *2HVR*RE14B.*

Response RAI-3

2HVR*RE14A and 2HVR*RE14B are not area radiation monitors intended to be used to indicate loss of spent fuel pool inventory as specified in RU2.1. These are the above refuel floor reactor building ventilation exhaust monitors. These are ventilation monitors that provide indication of a radioactivity release from the fuel, confirming that damage has occurred.

4. *RU2.2: Please clarify whether installed plant radiation monitors have the range to declare this EAL.*

Response RAI-4

The installed plant radiation monitors have the range to declare the EAL for a rise in unplanned radiation levels to the upper limit of the area radiation monitors scale. For instances when area radiation readings rise by a factor of 1,000 over normal levels and are beyond the scale of the installed instrumentation, levels would be verified by using portable instrumentation.

NMP1 installed area radiation monitors have a range of 0.01 mR/hr - 100 mR/hr or 0.1 mR/hr - 1000 mR/hr. The Lower range is used in very low dose rate areas, such as the Control Room.

NMP2 installed area radiation monitors have a range of 0.01 mR/hr - 1000 mR/hr or 0.1 mR/hr - 10,000 mR/hr. The Lower range is used in very low dose rate areas, such as the Control Room.

5. *HA 1.1: Please clarify the following:*

   a. *Timeliness for confirmation by the James A. Fitzpatrick Nuclear Power Plant (JAFNPP) seismic instrumentation; and*

   b. *Whether corroboration by the Nation Earthquake Information Center (NEIC) is used for declaration of this EAL. The plant-specific basis has information related to corroboration by the NEIC; however, this information was removed from the EAL and described as a deviation from the endorsed guidance.*

   Response RAI-5

   a. NMP1 procedure N1-SOP-28, Seismic Event, and NMP2 procedure N2-SOP-90, Natural Events, both prescribe immediate operator action to contact the JAFNPP Control Room for confirmatory indication of seismic activity onsite following confirmation of station seismic instrumentation alarms. Both NMP1 and NMP2 Control Rooms have direct communication capability with the JAFNPP Control Room (RECS).

   b. The NMP1 and NMP2 EAL Bases for HA1.1 has been revised to delete reference to NEIC as a confirmatory indicator for onsite seismic activity.

6. *HA1.2 and HU1.2 (NMP1): NMP2 basis provides that winds can be measured up to 100 miles-per-hour (mph). Please clarify whether NMP1 instrumentation can measure winds of 90 mph as listed in the EAL.*

   Response RAI-6

   The NMP1 EAL Bases for HA1.2 and HU1.2 have been revised to add a statement that wind speed can be measured up to 100 mph. The NMP1 Control Room uses the same meteorological instrumentation as NMP2.

7. *HA 1.5 and HU1.5: Please clarify how and where the site-specific water level is read.*

   Response RAI-7

   The NMP2 lake level is determined by reading a meter in the control room, 2SWP-LI502.

   The NMP1 lake level is determined by a level indicator in the NMP1 screen house, PI-73-32. A pressure transmitter, PS-74-08, located in the NMP1 screen house inputs to an annunciator in the control room, H2-1-3, should lake level lower to 238.8'.

8. *HU2.1: The standard EAL scheme required by 10 CFR 50.47(b)(4), and endorsed by the NRC (NEI 99-01, Revision 5), has the timing expectation well defined. Please explain the deviation from endorsed guidance or revise accordingly.*

<u>Response RAI-8</u>

The NMP1 and NMP2 EAL Bases for HU2.1 have been revised to reflect the generic NEI 99-01 Revision 5 wording regarding the classification timing expectation.

9. *HA3.1: Please confirm that the areas as listed in Table H-1 are required to be entered for safe operation or safe shutdown/cooldown per endorsed guidance. If access to the area is unnecessary to operate the said equipment, then the table does not need the area listed. Additionally, the EAL wording is not in accordance with the endorsed guidance. Please revise accordingly or justify how this EAL, as written, meets the intent of the guidance.*

<u>Response RAI-9</u>

HA3.1 is based on NEI 99-01, IC HA3 which states: *"Access to a VITAL AREA is prohibited due to toxic, corrosive, asphyxiant or flammable gases which jeopardize operation of operable equipment required to maintain safe operations or safely shutdown the reactor."*

The areas listed in Table H-1are areas that are, or can be, required to be entered for: normal operation, should there be a need to perform EOP actions based on another event, or if the station is otherwise challenged. Without further clarifying guidance upon which to base conditions under which access may or may not be required, we have determined that the areas listed provide the most conservative approach under all conditions.

Additionally, the NMP1 and NMP2 EAL Bases HA3.1 wording have been revised to read:

"Access to ANY Table H-1 area is prohibited due to toxic, corrosive, asphyxiant or flammable gases which jeopardize operation of systems required to maintain safe operations or safely shutdown the reactor (Note 5)."

10. *HS5.1: The endorsed guidance (NEI 99-01, Revision 5) does not have a timing note associated with this EAL. Please justify why Note 4 is applicable to this EAL or revise accordingly.*

<u>Response RAI-10</u>

The timing note and reference to the note from the NMP1 and NMP2 EAL Bases for HS5.1 were deleted to align with the endorsed guidance in NEI 99-01, Revision 5.

11. *CA1.1 (NMP2): Please clarify whether or not a typo exists for the electrical bus designation and revise accordingly.*

<u>Response RAI-11</u>

The typo has been corrected by moving the bus designator asterisk to the proper location.

12. *CU6.1 and SU6.1: Please explain how the "UHF radios" are acceptable for contacting the NRC in the required timeframe or revise table accordingly. Additionally, please clarify whether the Control Room satellite phones, identified to meet this EAL criteria, are described in the NMPNS Emergency Plan.*

Response RAI-12

The UHF radios were deleted from Tables S-3 and C-5.

The Control Room satellite phones are not specifically described in the NMPNS Emergency Plan. However, procedure EPIP-EPP-17, Emergency Communications, identifies the satellite phones in both the Control Room and Emergency Operating Facility as a backup to the Federal Telephone System and RECS for notification of offsite agencies.

Satellite phones are periodically tested in accordance with Procedure EPMP-EPP-02.

13. *SU1.1 (NMP2): Please justify why this EAL is missing the 15-minute timing information per endorsed guidance or revise accordingly.*

Response RAI-13

The NMP2 electrical distribution configuration precludes restoration of offsite power sources within 15 minutes in all instances, once lost. Therefore, no time component is allocated for this EAL threshold.

The reference to the fifteen-minute interval has been deleted from the NMP2 EAL bases.

14. *SA3.1: Please clarify the processes in place to prevent additional information related to reactor scram setpoints from being changed under another process (e.g., plant modifications, etc.) that may cause this technical bases document to be inadvertently made inaccurate.*

Response RAI-14

Reactor scram setpoints are controlled by other processes. The table of RPS scram parameters and setpoints from the NMP1 and NMP2 EAL Bases for SA3.1 were deleted to align with the endorsed guidance in NEI 99-01, Revision 5.

15. *SU7.2 (NMP2): Please justify the difference between unit thresholds listed (NMP1: $\geq$ 15 minutes / NMP2: > 15 minutes) based on endorsed guidance or revise accordingly.*

Response RAI-15

The typo has been corrected by revising the EAL to reflect "$\geq$" vs. ">" consistent with NMP1 SU7.2.

ATTACHMENT 2

MARKED-UP VERSION OF THE UPDATED REVISION TO NINE MILE POINT UNIT 1 AND UNIT 2 EMERGENCY ACTION LEVELS

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D-01 THROUGH D-02

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EPIP-EP-002, "ATTACHMENT 1 EAL MATRIX UNIT 2" PAGES 1 & 2.

WITHIN THIS PACKAGE

D-03 THROUGH D-04

ATTACHMENT 3

MARKED-UP VERSION OF THE UPDATED REVISION TO NINE MILE POINT UNIT 1 AND UNIT 2 EMERGENCY ACTION LEVEL BASIS DOCUMENTS

NINE MILE POINT NUCLEAR STATION

EMERGENCY PLAN MAINTENANCE PROCEDURE

EPMP-EPP-0101

REVISION 00 (Draft RAI 1-9-12)

UNIT 1 EMERGENCY CLASSIFICATION TECHNICAL BASES

TECHNICAL SPECIFICATION REQUIRED

Approved by: ______________________________ ____________
J. Kaminski Director Emergency Planning Date

THIS IS A COMPLETE REVISION

Effective Date:______________

PERIODIC REVIEW DUE DATE:__________

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ABBEVIATIONS / ACRONYMS

AC	Alternating Current
APRM	Average Power Range Meter
ATWS	Anticipated Transient Without Scram
BLDG	Building
BWR	Boiling Water Reactor
CDE	Committed Dose Equivalent
CFR	Code of Federal Regulations
DC	Direct Current
EAL	Emergency Action Level
EC	Emergency Condenser
ECCS	Emergency Core Cooling System
ED	Emergency Director
EL.	Elevation
EOF	Emergency Operations Facility
EOP	Emergency Operating Procedure
EPA	Environmental Protection Agency
EPMP	Emergency Plan Maintenance Procedure
EPRI	Electric Power Research Institute
EPIP	Emergency Plan Implementing Procedure
FBI	Federal Bureau of Investigation
GE	General Emergency
HCTL	Heat Capacity Temperature Limit
HOO	Headquarters (NRC) Operations Officer
IC	Initiating Condition
ISFSI	INDEPENDENT SPENT FUEL STORAGE INSTALLATION
JAFNPP	James A. FitzPatrick Nuclear Power Plant
LCO	Limiting Condition of Operation
LOCA	Loss of Coolant Accident
mR	milliRoentgen
MSCRWL	Minimum Steam Cooling RPV Water Level
MSIV	Main Steam Isolation Valve
MSL	Main Steam Line
NEI	Nuclear Energy Institute
NESP	National Environmental Studies Project
NRC	Nuclear Regulatory Commission
NORAD	North American Aerospace Defense Command
NUMARC	Nuclear Management and Resources Council
OBE	Operating Basis Earthquake

ACRONYMS & ABBREVIATIONS (continued)

OCA	Owner Controlled Area
ODCM	Off-site Dose Calculation Manual
OGESM	Offgas Effluent Stack Monitor
ORO	Off-site Response Organization
PAG	Protective Action Guideline
PC	Primary Containment
PSIG	Pounds per Square Inch Gauge
R	Roentgen
RB	Reactor Building
RCS	Reactor Coolant System
Rem	Roentgen Equivalent Man
RPS	Reactor Protection System
RPV	Reactor Pressure Vessel
RW	Raw Water
RWCU	Reactor Water Cleanup
SAE	Site Area Emergency
SBO	Station Blackout
SPDS	Safety Parameter Display System
TEDE	Total Effective Dose Equivalent
TSC	Technical Support Center
UE	Unusual Event
USAR	Updated Safety Analysis Report

1.0 PURPOSE

This document provides an explanation and rationale for each Emergency Action Level (EAL) included in the EAL Upgrade Project for Nine Mile Point Nuclear Station Unit 1 (NMP1). It should be used to facilitate review of the NMP1 EALs and provide historical documentation for future reference. Decision-makers responsible for implementation of EPIP-EPP-01, "Classification of Emergency Conditions at Unit 1," and the Emergency Action Level Matrices, may use this document as a technical reference in support of EAL interpretation. This information may assist the Emergency Director in making classifications, particularly those involving judgment or multiple events. The basis information may also be useful in training, for explaining event classifications to offsite officials, and facilitates regulatory review and approval of the classification scheme.

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

2.0 DISCUSSION

2.1 Background

EALs are the plant-specific indications, conditions or instrument readings that are utilized to classify emergency conditions defined in the Nine Mile Point Site Emergency Plan.

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

NEI 99-01 (NUMARC/NESP-007) Revision 4 was subsequently issued for industry implementation. Enhancements over earlier revisions included:

- Consolidating the system malfunction initiating conditions and example emergency action levels which address conditions that may be postulated to occur during plant shutdown conditions.

- Initiating conditions and example emergency action levels that fully address conditions that may be postulated to occur at permanently Defueled Stations and INDEPENDENT SPENT FUEL STORAGE INSTALLATIONs (ISFSIs).
- Simplifying the fission product barrier EAL threshold for a Site Area Emergency.

Subsequently, Revision 5 of NEI 99-01 has been issued which incorporates resolutions to numerous implementation issues including the NRC EAL FAQs. Using NEI 99-01 Revision 5 Final , February 2008 (ADAMS Accession Number ML080450149), NMP1 conducted an EAL implementation upgrade project that produced the EALs discussed herein.

2.2 Fission Product Barriers

Many of the EALs derived from the NEI methodology are fission product barrier based. That is, the conditions that define the EALs are based upon loss or potential loss of one or more of the three fission product barriers. "Loss" and "Potential Loss" signify the relative damage and threat of damage to the barrier. "Loss" means the barrier no longer assures containment of radioactive materials; "potential loss" implies an increased probability of barrier loss and decreased certainty of maintaining the barrier.

The primary fission product barriers are:

A. Fuel Clad (FC): Zirconium tubes which house the ceramic uranium oxide pellets along with the end plugs which are welded into each end of the fuel rods comprise the FC barrier.

B. Reactor Coolant System (RCS): The reactor vessel shell, vessel head, CRD housings, vessel nozzles and penetrations, and all primary systems directly connected to the RPV up to the outermost Primary Containment isolation valve comprise the RCS barrier.

C. Containment (PC): The drywell, the torus, their respective interconnecting paths, and other connections up to and including the outermost containment isolation valves comprise the Primary Containment barrier.

2.3 Emergency Classification Based on Fission Product Barrier Degradation

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

Unusual Event:

Any loss or any potential loss of Containment

Alert:

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

Site Area Emergency:

Loss or potential loss of any two barriers

General Emergency:

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

2.4 EAL Relationship to EOPs

Where possible, the EALs have been made consistent with and utilize the conditions defined in the NMP1 Emergency Operating Procedures (EOPs). While the symptoms that drive operator actions specified in the EOPs are not indicative of all possible conditions which warrant emergency classification, they define the symptoms, independent of initiating events, for which reactor plant safety and/or fission product barrier integrity are threatened. When these symptoms are clearly representative of one of the NEI Initiating Conditions, they have been utilized as an EAL. This permits rapid classification of emergency situations based on plant conditions without the need for additional evaluation or event diagnosis. Although some of the EALs presented here are based on conditions defined in the EOPs, classification of emergencies using these EALs is not dependent upon EOP entry or execution. The EALs can be utilized independently or in conjunction with the EOPs.

2.5 Symptom-Based vs. Event-Based Approach

To the extent possible, the EALs are symptom-based. That is, the action level threshold is defined by values of key plant operating parameters that identify emergency or potential emergency conditions. This approach is appropriate because it allows the full scope of variations in the types of events to be classified as emergencies. However, a purely symptom-based approach is not sufficient to address all events for which emergency classification is appropriate. Particular events to which no predetermined symptoms can be ascribed have also been utilized as EALs since they may be indicative of potentially more serious conditions not yet fully realized.

2.6 EAL Organization

The NMP1 EAL scheme includes the following features:

- Division of the EAL set into three broad groups:
  - EALs applicable under all plant operating modes – This group would be reviewed by the EAL-user any time emergency classification is considered.
  - EALs applicable only under hot operating modes – This group would only be reviewed by the EAL-user when the plant is in Hot Shutdown or Power Operation mode.
  - EALs applicable only under cold operating modes – This group would only be reviewed by the EAL-user when the plant is in Cold Shutdown, Refuel or Defueled mode.

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

- Within each of the above three groups, assignment of EALs to categories/subcategories – Category and subcategory titles are selected to represent conditions that are operationally significant to the EAL-user. Subcategories are used as necessary to further divide the EALs of a category into logical sets of possible emergency classification thresholds. The NMP1 EAL categories/subcategories and their relationship to NEI 99-01 Rev. 5 Recognition Categories are listed below.

EAL Groups, Categories and Subcategories

EAL Group/Category	EAL Subcategory
Any Operating Mode:	
R – Abnormal Radiation Levels / Radiological Effluents	1 – Offsite Rad Conditions 2 – Onsite Rad Conditions & Spent Fuel Events 3 – CR/CAS Rad
H – **H**azards and Other Conditions Affecting Plant Safety	1 – Natural or Destructive Phenomena 2 – FIRE or EXPLOSION 3 – Hazardous Gas 4 – Security 5 – Control Room Evacuation 6 – Judgment
E – ISFSI	None
Cold Conditions:	
C – **C**old Shutdown / Refueling System Malfunction	1 – Loss of AC Power 2 – Loss of DC Power 3 – RPV Water Level 4 – RCS Temperature 5 – Inadvertent Criticality 6 – Communications
Hot Conditions:	
S – **S**ystem Malfunction	1 – Loss of AC Power 2 – Loss of DC Power 3 – Criticality & RPS Failure 4 – Inability to Reach or Maintain Shutdown Conditions 5 – Instrumentation 6 – Communications 7 – Fuel Clad Degradation 8 – RCS Leakage
F – **F**ission Product Barrier Degradation	None

The primary tool for determining the emergency classification level is the EAL Classification Matrix. The user of the EAL Classification Matrix may (but is not required to) consult the EAL Technical Bases Document in order to obtain additional information concerning the EALs under classification consideration. The user should consult Sections 2.7 and 2.8, and Attachments 1 and 2 of this document for such information.

2.7 Technical Bases Information

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

<u>Category Letter & Title</u>

<u>Subcategory Number & Title</u>

<u>Initiating Condition (IC)</u>

Site-specific description of the generic IC given in NEI 99-01 Rev. 5.

<u>EAL Identifier (enclosed in rectangle)</u>

Each EAL is assigned a unique identifier to support accurate communication of the emergency classification to onsite and offsite personnel. Four characters define each EAL identifier:

1. First character (letter): Corresponds to the EAL category as described above (R, H, E, C, S or F)
2. Second character (letter): The emergency classification (G, S, A or U)

   G = General Emergency

   S = Site Area Emergency

   A = Alert

   U = Unusual Event
3. Third character (number): Subcategory number within the given category. Subcategories are sequentially numbered beginning with the number one (1). If

a category does not have a subcategory, this character is assigned the number one (1).

4. Fourth character (number): The numerical sequence of the EAL within the EAL subcategory. If the subcategory has only one EAL, it is given the number one (1).

Classification (enclosed in rectangle):

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

EAL (enclosed in rectangle)

Wording enclosed in the rectangle appears as it is displayed in the EAL Classification Matrix. Selected terms are highlighted for emphasis:

- Bold, uppercase print is assigned to: "**ANY**," EAL identifiers, and logic terms such as **AND**, **OR**, **EITHER**, etc. (When used as conjunctions, the words "and" and "or" are not highlighted.)
- Bold, mixed case print is assigned to: "**all**," "**only**," "**both**," table titles and column headings, numbers following the word "**ANY**," and negative terms (e.g., "**not**," "**cannot**," etc.)
- Uppercase print is assigned to acronyms, abbreviations, and terms defined in Section 4.0.

Mode Applicability

One or more of the following plant operating conditions comprise the mode to which each EAL is applicable: 1 - Power Operation, 2 - Hot Shutdown, 3 - Cold Shutdown, 4 - Refuel, D - Defueled, or All. (See Section 2.8 for operating mode definitions.)

Basis:

A Plant-Specific basis section provides NMP1-relevant information concerning the EAL. This is followed by a Generic basis section that provides a description of the rationale for the EAL as provided in NEI 99-01 Rev. 5.

NMP1 Basis Reference(s):

Site-specific source documentation from which the EAL is derived

2.8 Operating Mode Applicability (Technical Specifications Definitions 1.1)

1 Power Operation

- Reactor mode switch is in startup or run position.
- Reactor is critical or criticality is possible due to control rod withdrawal.

2 Hot Shutdown

- The reactor mode switch is in the shutdown position.
- No core alterations leading to an addition of reactivity are being performed.
- Reactor coolant temperature is greater than 212°F.

3 Cold Shutdown

- The reactor mode switch is in the shutdown position or refuel position.
- No Core alterations leading to an addition of reactivity are being performed. .
- Reactor coolant temperature is less than or equal to 212°F.

4 Refuel

- The reactor mode switch is in the refuel position.
- The reactor coolant temperature is less than 212°F.
- Fuel may be loaded or unloaded.
- No more than one operable control rod may be withdrawn.

D Defueled

No fuel is in the reactor. (Note: this is equivalent to the technical specification defined condition "Major Maintenance")

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

2.9 Validation of Indications, Reports and Conditions

All emergency classifications shall be based upon VALID indications, reports or conditions. An indication, report, or condition, is considered to be VALID when it is verified by (1) an instrument channel check, or (2) indications on related or redundant indicators, or (3) by direct observation by plant personnel, such that doubt related to the indicator's operability, the condition's existence, or the report's accuracy is removed. Implicit in this definition is the need for timely assessment.

2.10 Planned vs. UNPLANNED Events

Planned evolutions involve preplanning to address the limitations imposed by the condition, the performance of required surveillance testing, and the implementation of specific controls prior to knowingly entering the condition in accordance with the specific requirements of the site's Technical Specifications. Activities which cause the site to operate beyond that allowed by the site's Technical Specifications, planned or UNPLANNED, may result in an EAL threshold being met or exceeded. Planned evolutions to test, manipulate, repair, perform maintenance or modifications to systems and equipment that result in an EAL value being met or exceeded are not subject to classification and activation requirements as long as the evolution proceeds as planned and is within the operational limitations imposed by the specific operating license. However, these conditions may be subject to the reporting requirements of 10 CFR 50.72.

2.11 Classifying Transient Events

For some events, the condition may be corrected before a declaration has been made. The key consideration in this situation is to determine whether or not further plant damage occurred while the corrective actions were being taken. In some situations, this can be readily determined, in other situations, further analyses may be necessary (e.g., coolant radiochemistry following an ATWS event, plant structural examination following an earthquake, etc.). Classify the event as indicated and terminate the emergency once assessment shows that there were no consequences from the event and other termination criteria are met.

Existing guidance for classifying transient events addresses the period of time of event recognition and classification (15 minutes). However, in cases when EAL declaration criteria may be met momentarily during the normal expected response of the plant, declaration requirements should not be considered to be met when the conditions are a part of the designed plant response, or result from appropriate Operator actions.

There may be cases in which a plant condition that exceeded an EAL was not recognized at the time of occurrence but is identified well after the condition has occurred (e.g., as a result of routine log or record review), and the condition no longer exists. In these cases, an emergency should not be declared. Reporting requirements of 10 CFR 50.72 are applicable and the guidance of NUREG-1022, Event Reporting Guidelines 10 CFR 50.72 and 50.73, should be applied.

2.12 Multiple Simultaneous Events and IMMINENT EAL Thresholds

When multiple simultaneous events occur, the emergency classification level is based on the highest EAL reached. For example, two Alerts remain in the Alert category. Or, an Alert and a Site Area Emergency is a Site Area Emergency. Further guidance is provided in RIS 2007-02, Clarification of NRC Guidance for Emergency Notifications During Quickly Changing Events.

Since NMP1 is at a multi-unit site, emergency classification level upgrading must also consider the effects of a loss of a common system on more than one unit (e.g., potential for radioactive release from more than one core).

Although the majority of the EALs provide very specific thresholds, the Emergency Director (ED) must remain alert to events or conditions that lead to the conclusion that exceeding the EAL threshold is IMMINENT. If, in the judgment of the ED, an IMMINENT situation is at hand, the classification should be made as if the threshold has been exceeded. While this is particularly prudent at the higher emergency classes (the early classification may permit more effective implementation of protective measures), it is nonetheless applicable to all emergency classes.

2.13 Emergency Classification Level Downgrading

Another important aspect of usable EAL guidance is the consideration of what to do when the risk posed by an emergency is clearly decreasing. A combination approach involving recovery from General Emergencies and some Site Area Emergencies and termination from Unusual Events, Alerts, and certain Site Area Emergencies causing no long term plant damage appears to be the best choice. Downgrading to lower emergency classification levels adds notifications but may have merit under certain circumstances.

3.0 REFERENCES

3.1 Developmental

3.1.1 NEI 99-01 Rev. 5 Final, Methodology for Development of Emergency Action Levels, February 2008, ADAMS Accession Number ML080450149

3.1.2 NRC Regulatory Issue Summary (RIS) 2003-18, Supplement 2, Use of Nuclear Energy Institute (NEI) 99-01, Methodology for Development of Emergency Action Levels Revision 4, Dated January 2003 (December 12, 2005)

3.1.3 RIS 2007-02 Clarification of NRC Guidance for Emergency Notifications During Quickly Changing Events

3.1.4 Nine Mile Point Site Emergency Plan

3.2 Implementing

3.2.1 EPIP-EPP-01 Classification of Emergency Conditions at Unit 1

3.2.2 EAL Comparison Matrix

3.3 Commitments

None

4.0 DEFINITIONS (ref. 3.1.1 except as noted)

AFFECTING SAFE SHUTDOWN

Event in progress has adversely affected functions that are necessary to bring the plant to and maintain it in the applicable hot or cold shutdown condition. Plant condition applicability is determined by Technical Specification LCOs in effect.

Example 1: Event causes damage that results in entry into an LCO that requires the plant to be placed in hot shutdown. Hot shutdown is achievable, but cold shutdown is not. This event is not "AFFECTING SAFE SHUTDOWN."

Example 2: Event causes damage that results in entry into an LCO that requires the plant to be placed in cold shutdown. Hot shutdown is achievable, but cold shutdown is not. This event is "AFFECTING SAFE SHUTDOWN."

AIRLINER/LARGE AIRCRAFT

Any size or type of aircraft with the potential for causing significant damage to the plant (refer to the Security Plan for a more detailed definition).

BOMB

Refers to an explosive device suspected of having sufficient force to damage plant systems or structures.

CIVIL DISTURBANCE

A group of people violently protesting station operations or activities at the site.

CONFINEMENT BOUNDARY

The barrier(s) between areas containing radioactive substances and the environment.

CONTAINMENT CLOSURE

The procedurally defined actions taken to secure containment (primary or secondary) and its associated structures, systems, and components as a functional barrier to fission product release under existing plant conditions.

EXPLOSION

A rapid, violent, unconfined combustion, or catastrophic failure of pressurized/energized equipment that imparts energy of sufficient force to potentially damage permanent structures, systems, or components.

EXTORTION

An attempt to cause an action at the station by threat of force.

FIRE

Combustion characterized by heat and light. Sources of smoke such as slipping drive belts or overheated electrical equipment do not constitute FIREs. Observation of flame is preferred but is not required if large quantities of smoke and heat are observed.

HOSTAGE

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

HOSTILE ACTION

An act toward NMP1 or its personnel that includes the use of violent force to destroy equipment, take HOSTAGEs, and/or intimidate the licensee to achieve an end. This includes attack by air, land, or water using guns, explosives, PROJECTILEs, vehicles, or other devices used to deliver destructive force. Other acts that satisfy the overall intent may be included.

HOSTILE ACTION should not be construed to include acts of civil disobedience or felonious acts that are not part of a concerted attack on NMP1. Non-terrorism-based EALs should be used to address such activities, (e.g., violent acts between individuals in the owner controlled area).

HOSTILE FORCE

One or more individuals who are engaged in a determined assault, overtly or by stealth and deception, equipped with suitable weapons capable of killing, maiming, or causing destruction.

IMMINENT

Mitigation actions have been ineffective, additional actions are not expected to be successful, and trended information indicates that the event or condition will occur. Where IMMINENT timeframes are specified, they shall apply.

INTACT

The RCS should be considered INTACT when the RCS pressure boundary is in its normal condition for the cold shutdown mode of operation (e.g., no freeze seals or nozzle dams).

INTRUSION

The act of entering without authorization. Discovery of a BOMB in a specified area is indication of INTRUSION into that area by a HOSTILE FORCE.

INDEPENDENT SPENT FUEL STORAGE INSTALLATION (ISFSI)

A complex that is designed and constructed for the interim storage of spent nuclear fuel and other radioactive materials associated with spent fuel storage.

NORMAL LEVELS

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

NORMAL PLANT OPERATIONS

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

PROJECTILE

An object directed toward NMP1 that could cause concern for its continued operability, reliability, or personnel safety.

PROTECTED AREA

The area which normally encompasses all controlled areas within the security PROTECTED AREA fence. NMP1 and NMP2 share a common PROTECTED AREA border. NMP1 and NMP2 PROTECTED AREA boundaries are illustrated in USAR Figure 1.2-1.

SABOTAGE

Deliberate damage, mis-alignment, or mis-operation of plant equipment with the intent to render the equipment inoperable. Equipment found tampered with or damaged due to malicious mischief may not meet the definition of SABOTAGE until this determination is made by security supervision.

SAFETY-RELATED STRUCTUREs, SYSTEMs and COMPONENTs (as defined in 10CFR50.2)

Those structures, systems and components that are relied upon to remain functional during and following design basis events to assure:

(1) The integrity of the reactor coolant pressure boundary;

(2) The capability to shut down the reactor and *maintain* it in a safe shutdown condition;

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

SECURITY CONDITION

Any security event as listed in the approved security contingency plan that constitutes a threat/compromise to site security, threat/risk to site personnel, or a potential degradation to the level of safety of the plant. A SECURITY CONDITION does not involve a HOSTILE ACTION.

SITE BOUNDARY

Per NMP2 ODCM Figure D 1.0-1, the line around the Nine Mile Point Nuclear Station beyond which the land is not owned, leased or otherwise controlled by the owners and operators of Nine Mile Point Nuclear Station and James A. Fitzpatrick Nuclear Power Plant.

STRIKE ACTION

Work stoppage within the PROTECTED AREA by a body of workers to enforce compliance with demands made on NMP1. The STRIKE ACTION must threaten to interrupt NORMAL PLANT OPERATIONS.

UNISOLABLE

A breach or leak that cannot be promptly isolated.

UNPLANNED

A parameter change or an event, the reasons for which may be known or unknown, that is not the result of an intended evolution or expected plant response to a transient.

VALID

An indication, report, or condition, is considered to be VALID when it is verified by (1) an instrument channel check, or (2) indications on related or redundant indicators, or (3) by direct observation by plant personnel, such that doubt related to the indicator's operability, the condition's existence, or the report's accuracy is removed. Implicit in this definition is the need for timely assessment.

VISIBLE DAMAGE

Damage to equipment or structure that is readily observable without measurements, testing, or analysis. Damage is sufficient to cause concern regarding the continued operability or reliability of affected SAFETY-RELATED STRUCTURE, SYSTEM or COMPONENT. Example damage includes: deformation due to heat or impact, denting, penetration, rupture, cracking, paint blistering. Surface blemishes (e.g., paint chipping, scratches) should not be included.

VITAL AREA

Any areas, normally within the NMP1 PROTECTED AREA, that contains equipment, systems, components, or material, the failure, destruction, or release of which could directly or indirectly endanger the public health and safety by exposure to radiation.

5.0 NMP1-TO-NEI 99-01 EAL CROSSREFERENCE

This cross-reference is provided to facilitate association and location of a NMP1 EAL within the NEI 99-01 IC/EAL identification scheme. Further information regarding the development of the NMP1 EALs based on the NEI guidance can be found in the EAL Comparison Matrix.

NMP1	NEI 99-01	
EAL	IC	Example EAL
RG1.2	AG1	2
RG1.3	AG1	4
RS1.1	AS1	1
RS1.2	AS1	2
RS1.3	AS1	4
RA1.1	AA1	1
RA1.2	AA1	2
RA1.3	AA1	3
RU1.1	AU1	1
RU1.2	AU1	2
RU1.3	AU1	3
RA2.1	AA2	2
RA2.2	AA2	1
RU2.1	AU2	1
RU2.2	AU2	2
RA3.1	AA3	1
HA1.1	HA1	1
HA1.2	HA1	2

NMP1	NEI 99-01	
EAL	**IC**	**Example EAL**
HA1.3	HA1	3
HA1.4	HA1	4
HA1.5	HA1	6
HA1.6	HA1	5
HU1.1	HU1	1
HU1.2	HU1	2
HU1.3	HU1	3
HU1.4	HU1	4
HU1.5	HU1	5
HA2.1	HA2	1
HU2.1	HU2	1
HU2.2	HU2	2
HA3.1	HA3	1
HU3.1	HU3	1
HU3.2	HU3	2
HG4.1	HG1	1
HG4.2	HG1	2
HS4.1	HS4	1
HA4.1	HA4	1, 2
HU4.1	HU4	1, 2, 3
HS5.1	HS2	1
HA5.1	HA5	1
HG6.1	HG2	1
HS6.1	HS3	1

NMP1	NEI 99-01	
EAL	IC	Example EAL
HA6.1	HA6	1
HU6.1	HU5	1
EU1.1	E-HU1	1
CA1.1	CA3	1
CU1.1	CU3	1
CU2.1	CU7	1
CG3.1	CG1	1
CG3.2	CG1	2
CS3.1	CS1	1
CS3.2	CS1	2
CS3.3	CS1	3
CA3.1	CA1	1, 2
CU3.1	CU1	1
CU3.2	CU2	1
CU3.3	CU2	2
CA4.1	CA4	1, 2
CU4.1	CU4	1
CU4.2	CU4	2
CU5.1	CU8	1
CU6.1	CU6	1, 2
SG1.1	SG1	1
SS1.1	SS1	1
SA1.1	SA5	1
SU1.1	SU1	1

NMP1	NEI 99-01	
EAL	IC	Example EAL
SS2.1	SS3	1
SG3.1	SG2	1
SS3.1	SS2	1
SA3.1	SA2	1
SU3.1	SU8	1
SU4.1	SU2	1
SS5.1	SS6	1
SA5.1	SA4	1
SU5.1	SU3	1
SU6.1	SU6	1, 2
SU7.1	SU4	2
SU7.2	SU4	1
SU8.1	SU5	1, 2
FG1.1	FG1	1
FS1.1	FS1	1
FA1.1	FA1	1
FU1.1	FU1	1

6.0 ATTACHMENTS

6.1 Attachment 1, Emergency Action Level Technical Bases

6.2 Attachment 2, Fission Product Barrier Loss / Potential Loss Matrix and Basis

Category R – Abnormal Rad Release / Rad Effluent

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

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

At lower levels, abnormal radioactivity releases may be indicative of a failure of containment systems or precursors to more significant releases. At higher release rates, offsite radiological conditions may result which require offsite protective actions. Elevated area radiation levels in plant may also be indicative of the failure of containment systems or preclude access to plant vital equipment necessary to ensure plant safety.

Events of this category pertain to the following subcategories:

1. Offsite Rad Conditions

Direct indication of effluent radiation monitoring systems provides a rapid assessment mechanism to determine releases in excess of classifiable limits. Projected offsite doses, actual offsite field measurements or measured release rates via sampling indicate doses or dose rates above classifiable limits.

2. Onsite Rad Conditions & Spent Fuel Events

Sustained general area radiation levels in excess of those indicating loss of control of radioactive materials or those levels which may preclude access to vital plant areas also warrant emergency classification.

3. CR/CAS Rad

Sustained general area radiation levels which may preclude access to areas requiring continuous occupancy also warrant emergency classification.

Category: R – Abnormal Rad Release / Rad Effluent

Subcategory: 1 – Offsite Rad Conditions

Initiating Condition: Offsite dose resulting from an actual or IMMINENT release of gaseous radioactivity > 1,000 mRem TEDE or 5,000 mRem thyroid CDE for the actual or projected duration of the release using actual meteorology

EAL:

RG1.2	**General Emergency**
Dose assessment using actual meteorology indicates doses > 1,000 mRem TEDE or 5,000 mRem thyroid CDE at or beyond the SITE BOUNDARY	

Mode Applicability:

All

Basis:

Plant-Specific

The 1,000 mRem TEDE dose is set at 100% of the EPA PAG, while the 5,000 mRem thyroid CDE was established in consideration of the 1:5 ratio of the EPA PAG for TEDE and thyroid CDE.

Dose assessment is performed using EPIP-EPP-08 "Off-site Dose Assessment and Protective Action Recommendation" (ref. 1).

The SITE BOUNDARY is the line beyond which the land is not owned, leased, nor otherwise controlled by Constellation (ref. 2).

Generic

This EAL addresses radioactivity releases that result in doses at or beyond the SITE BOUNDARY that exceed the EPA Protective Action Guides (PAGs). Public protective actions will be necessary. Releases of this magnitude are associated with the failure of plant systems needed for the protection of the public and likely involve fuel damage.

Since dose assessment is based on actual meteorology, whereas the monitor reading EAL is not, the results from these assessments may indicate that the classification is not warranted, or may indicate that a higher classification is warranted. For this reason, emergency implementing procedures should call for the timely performance of dose assessments using actual meteorology and release information. If the results of these dose assessments are available when the classification is made (e.g., initiated at a lower classification level), the dose assessment results override the monitor reading EAL.

NMP1 Basis Reference(s):

1. EPIP-EPP-08 Off-site Dose Assessment and Protective Action Recommendation
2. NMP1 ODCM Figure 5.1.3-1
3. NEI 99-01 IC AG1

Category: R – Abnormal Rad Release / Rad Effluent

Subcategory: 1 – Offsite Rad Conditions

Initiating Condition: Offsite dose resulting from an actual or IMMINENT release of gaseous radioactivity > 1,000 mRem TEDE or 5,000 mRem thyroid CDE for the actual or projected duration of the release using actual meteorology

EAL:

RG1.3 General Emergency

Field survey results indicate closed window dose rates > 1,000 mRem/hr expected to continue for ≥ 60 min. at or beyond the SITE BOUNDARY (Note 1)

OR

Analyses of field survey samples indicate thyroid CDE > 5,000 mRem for 1 hr of inhalation at or beyond the SITE BOUNDARY (Note 1)

Note 1: The ED should **not** wait until the applicable time has elapsed, but should declare the event as soon as it is determined that the condition will likely exceed the applicable time

Mode Applicability:

All

Basis:

Plant-Specific

Real time field surveys and sample analysis is performed by offsite field monitoring teams per EPIP-EPP-07, "Downwind Radiological Monitoring ," (ref. 1) and assessed for radiological dose consequences per EPIP-EPP-08, "Off-site Dose Assessment and Protective Action Recommendation " (ref. 2).

The SITE BOUNDARY is the line beyond which the land is not owned, leased, nor otherwise controlled by Constellation (ref. 3).

Generic

This EAL addresses radioactivity releases that result in doses at or beyond the SITE BOUNDARY that exceed the EPA Protective Action Guides (PAGs). Public protective actions will be necessary. Releases of this magnitude are associated with the failure of plant systems needed for the protection of the public and likely involve fuel damage.

Since dose assessment is based on actual meteorology, whereas the monitor reading EAL is not, the results from these assessments may indicate that the classification is not warranted. For this reason, emergency implementing procedures should call for the timely performance of dose assessments using actual meteorology and release information. If the results of these dose assessments are available when the classification is made (e.g., initiated at a lower classification level), the dose assessment results override the monitor reading EAL.

NMP1 Basis Reference(s):

1. EPIP-EPP-07 Downwind Radiological Monitoring
2. EPIP-EPP-08 Off-site Dose Assessment and Protective Action Recommendation
3. NMP1 ODCM Figure 5.1.3-1
4. NEI 99-01 IC AG1

Category: R – Abnormal Rad Release / Rad Effluent

Subcategory: 1 – Offsite Rad Conditions

Initiating Condition: Offsite dose resulting from an actual or IMMINENT release of gaseous radioactivity exceeds 100 mRem TEDE or 500 mRem thyroid CDE for the actual or projected duration of the release using actual meteorology

EAL:

RS1.1 Site Area Emergency

ANY monitor reading > Table R-1 "SAE" column for ≥ 15 min. (Note 1)

- Do **not** delay declaration awaiting dose assessment results
- If dose assessment results are available, declaration should be based on dose assessment instead of radiation monitor values (see EAL RS1.2)

Note 1: The ED should **not** wait until the applicable time has elapsed, but should declare the event as soon as it is determined that the condition will likely exceed the applicable time

Table R-1 Effluent Monitor Classification Thresholds

Monitor	GE	SAE	Alert	UE
Gaseous				
Stack (RN10A/B)	N/A	N/A	3.0E4 cps	300 cps
EC Vent	N/A	300 mRem/hr	30 mRem/hr	10 mRem/hr
Liquid				
SW Effluent	N/A	N/A	90,000 cpm	900 cpm
RW Discharge	N/A	N/A	200 x batch	2 x batch

Mode Applicability:

All

Basis:

Plant-Specific

The SAE value for the EC Vent is based on the boundary dose resulting from an actual or IMMINENT release of gaseous radioactivity that exceeds 100 mRem whole body or 500 mRem child thyroid for the actual or projected duration of the release (ref. 1). The 100 mRem integrated dose is based on 10% of the PAG level for TEDE whole body dose. The 500 mRem integrated child thyroid dose was established in consideration of the 1:5 ratio of the EPA Protective Action Guidelines for whole body thyroid.

Liquid effluent radiation monitors are not addressed in Table R-1 at the Site Area Emergency and General Emergency levels because the dose assessment code used to calculate these Table R-1 readings only considers a release through the Radwaste/RB Vent or the Main Stack.

Certain gaseous effluent radiation monitor readings are "N/A" at the Site Area Emergency and General Emergency levels because the dose rate corresponding to the EPA PAGs would generate a reading that is beyond the upper range of the monitors.

A radiation monitor reading is VALID when a release path is established. If the release path to the environment has been isolated, the radiation monitor reading is not VALID for classification.

Generic

This EAL addresses radioactivity releases that result in doses at or beyond the SITE BOUNDARY that exceed 10% of the EPA Protective Action Guides (PAGs). Releases of this magnitude are associated with the failure of plant systems needed for the protection of the public.

The site specific monitor is the only monitor in potential release pathways that will still be on scale.

Since dose assessment is based on actual meteorology, whereas the monitor reading EAL is not, the results from these assessments may indicate that the classification is not warranted, or may indicate that a higher classification is warranted. For this reason, emergency implementing procedures should call for the timely performance of dose assessments using actual meteorology and release information. If the results of these dose assessments are available when the classification is made (e.g., initiated at a lower classification level), the dose assessment results override the monitor reading EAL.

NMP1 Basis Reference(s):

1. Calculation 1H21C003, Rev 0
2. NEI 99-01 IC AS1

Category: R – Abnormal Rad Release / Rad Effluent

Subcategory: 1 – Offsite Rad Conditions

Initiating Condition: Offsite dose resulting from an actual or IMMINENT release of gaseous radioactivity exceeds 100 mRem TEDE or 500 mRem thyroid CDE for the actual or projected duration of the release using actual meteorology

EAL:

RS1.2	**Site Area Emergency**
Dose assessment using actual meteorology indicates doses > 100 mRem TEDE or 500 mRem thyroid CDE at or beyond the SITE BOUNDARY	

Mode Applicability:

All

Basis:

Plant-Specific

The 100 mRem TEDE dose is set at 10% of the EPA PAG, while the 500 mRem thyroid CDE was established in consideration of the 1:5 ratio of the EPA PAG for TEDE and thyroid CDE (ref. 1).

The SITE BOUNDARY is the line beyond which the land is not owned, leased, nor otherwise controlled by Constellation (ref. 2).

Generic

This EAL addresses radioactivity releases that result in doses at or beyond the SITE BOUNDARY that exceed 10% of the EPA Protective Action Guides (PAGs). Releases of this magnitude are associated with the failure of plant systems needed for the protection of the public.

Since dose assessment is based on actual meteorology, whereas the monitor reading EAL is not, the results from these assessments may indicate that the classification is not warranted, or may indicate that a higher classification is warranted. For this reason, emergency implementing procedures should call for the timely performance of dose assessments using actual meteorology and release information. If the results of these dose assessments are available when the classification is made (e.g., initiated at a lower classification level), the dose assessment results override the monitor reading EAL.

NMP1 Basis Reference(s):

1. EPIP-EPP-08 Offsite Dose Assessment and Protective Action Recommendation
2. NMP1 ODCM Figure 5.1.3-1
3. NEI 99-01 IC AS1

Category: R – Abnormal Rad Release / Rad Effluent

Subcategory: 1 – Offsite Rad Conditions

Initiating Condition: Offsite dose resulting from an actual or IMMINENT release of gaseous radioactivity exceeds 100 mRem TEDE or 500 mRem thyroid CDE for the actual or projected duration of the release using actual meteorology

EAL:

RS1.3 Site Area Emergency

Field survey results indicate closed window dose rates > 100 mRem/hr expected to continue for ≥ 60 min. at or beyond the SITE BOUNDARY (Note 1)

OR

Analyses of field survey samples indicate thyroid CDE > 500 mRem for 1 hr of inhalation at or beyond the SITE BOUNDARY (Note 1)

Note 1: The ED should **not** wait until the applicable time has elapsed, but should declare the event as soon as it is determined that the condition will likely exceed the applicable time

Mode Applicability:

All

Basis:

Plant-Specific

Real time field surveys and sample analysis is performed by offsite field monitoring teams per EPIP-EPP-07 "Downwind Radiological Monitoring" (ref. 1) and assessed for radiological dose consequences per EPIP-EPP-08 "Off-site Dose Assessment and Protective Action Recommendation" (ref. 2).

The SITE BOUNDARY is the line beyond which the land is not owned, leased, nor otherwise controlled by Constellation (ref. 3).

Generic

This EAL addresses radioactivity releases that result in doses at or beyond the SITE BOUNDARY that exceed 10% of the EPA Protective Action Guides (PAGs). Releases of this magnitude are associated with the failure of plant systems needed for the protection of the public.

Since dose assessment is based on actual meteorology, whereas the monitor reading EAL is not, the results from these assessments may indicate that the classification is not warranted, or may indicate that a higher classification is warranted. For this reason, emergency implementing procedures should call for the timely performance of dose assessments using actual meteorology and release information. If the results of these dose assessments are available when the classification is made (e.g., initiated at a lower classification level), the dose assessment results override the monitor reading EAL.

NMP1 Basis Reference(s):

1. EPIP-EPP-07 Downwind Radiological Monitoring
2. EPIP-EPP-08 Off-site Dose Assessment and Protective Action Recommendation
3. NMP1 ODCM Figure 5.1.3-1
4. NEI 99-01 IC AS1

Category: R – Abnormal Rad Release / Rad Effluent

Subcategory: 1 – Offsite Rad Conditions

Initiating Condition: **ANY** release of gaseous or liquid radioactivity to the environment > 200 times the ODCM for 15 minutes or longer

EAL:

RA1.1 Alert

ANY gaseous monitor reading > Table R-1 "Alert" column for ≥ 15 min. (Note 2)

Note 2: The ED should **not** wait until the applicable time has elapsed, but should declare the event as soon as it is determined that the release duration has exceeded, or will likely exceed, the applicable time. In the absence of data to the contrary, assume that the release duration has exceeded the applicable time if an ongoing release is detected and the release start time is unknown.

Table R-1 Effluent Monitor Classification Thresholds				
Monitor	**GE**	**SAE**	**Alert**	**UE**
Gaseous				
Stack (RN10A/B)	N/A	N/A	3.0E4 cps	300 cps
EC Vent	N/A	300 mRem/hr	30 mRem/hr	10 mRem/hr
Liquid				
SW Effluent	N/A	N/A	90,000 cpm	900 cpm
RW Discharge	N/A	N/A	200 x batch	2 x batch

Mode Applicability:

All

Basis:

Plant-Specific

The EC Vent monitor value in Table R-1 for the Alert level is not based on 200 times the ODCM value (ref. 1). The method used to determine ODCM values differs from the method used to determine the Table R-1 SAE levels and, if applied to the EC Vent monitor, would yield a value greater than the SAE level. Instead, a value of 30 mRem/hr has been selected because it provides a graded escalation between the UE level and the SAE level (ref. 2).

The Reactor Building Vent Monitors are not included in this EAL because the Reactor Building ventilation discharges to the main stack. Radioactivity release from the Reactor Building would, therefore, be assessed by the main stack monitor.

A radiation monitor reading is VALID when a release path is established. If the release path to the environment has been isolated, the radiation monitor reading is not VALID for classification.

Generic

The Emergency Director should not wait until the applicable time has elapsed, but should declare the event as soon as it is determined that the condition will likely exceed the applicable time.

This EAL addresses an actual or substantial potential decrease in the level of safety of the plant as indicated by a radiological release that exceeds regulatory commitments for an extended period of time.

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

The 200 x ODCM limit multiples are specified only to distinguish between non-emergency conditions. While these multiples obviously correspond to an off-site dose or dose rate, the emphasis in classifying these events is the degradation in the level of safety of the Releases should not be prorated or averaged. For example, a release exceeding 600x ODCM for 5 minutes does not meet the threshold.

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

This EAL is intended for sites that have established effluent monitoring on non-routine release pathways for which a discharge permit would not normally be prepared.

NMP1 Basis Reference(s):

1. NMP1 Offsite Dose Calculation Manual (ODCM)
2. Calculation 1H21C003, Rev 0
3. N1-ARP-H1 Annunciator H1-1-8
4. N1-OP-50B Process Radiation Monitoring System
5. NEI 99-01 IC AA1

Category: R – Abnormal Rad Release / Rad Effluent

Subcategory: 1 – Offsite Rad Conditions

Initiating Condition: **ANY** release of gaseous or liquid radioactivity to the environment > 200 times the ODCM for 15 minutes or longer

EAL:

RA1.2 Alert

ANY liquid monitor > Table R-1 "Alert" column for ≥ 15 min. (Note 2)

Note 2: The ED should **not** wait until the applicable time has elapsed, but should declare the event as soon as it is determined that the release duration has exceeded, or will likely exceed, the applicable time. In the absence of data to the contrary, assume that the release duration has exceeded the applicable time if an ongoing release is detected and the release start time is unknown.

Table R-1 Effluent Monitor Classification Thresholds

Monitor	GE	SAE	Alert	UE
Gaseous				
Stack (RN10A/B)	N/A	N/A	3.0E4 cps	300 cps
EC Vent	N/A	300 mRem/hr	30 mRem/hr	10 mRem/hr
Liquid				
SW Effluent	N/A	N/A	90,000 cpm	900 cpm
RW Discharge	N/A	N/A	200 x batch	2 x batch

Mode Applicability:

All

Basis:

Plant-Specific

The Containment Spray Raw Water Monitors (RW) are not included in this EAL because these monitors detect radiation in the discharge from their respective processes. The monitors are located upstream of the Service Water monitor. Therefore, the Service Water radiation monitor adequately detects offsite radioactivity releases from these systems.

A radiation monitor reading is VALID when a release path is established. If the release path to the environment has been isolated, the radiation monitor reading is not VALID for classification.

Generic

The Emergency Director should not wait until the applicable time has elapsed, but should declare the event as soon as it is determined that the condition will likely exceed the applicable time.

This EAL addresses an actual or substantial potential decrease in the level of safety of the plant as indicated by a radiological release that exceeds regulatory commitments for an extended period of time.

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

The 200 x ODCM limit multiples are specified only to distinguish between non-emergency conditions. While these multiples obviously correspond to an off-site dose or dose rate, the emphasis in classifying these events is the degradation in the level of safety of the plant, not the magnitude of the associated dose or dose rate.

Releases should not be prorated or averaged. For example, a release exceeding 600x ODCM for 5 minutes does not meet the threshold.

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

This EAL addresses radioactivity releases, that for whatever reason, cause effluent radiation monitor readings to exceed the threshold identified in the EAL established by the radioactivity discharge permit. This value may be associated with a planned batch release, or a continuous release path.

NMP1 Basis Reference(s):

1. NMP1 Offsite Dose Calculation Manual (ODCM)
2. N1-ARP-H1 Annunciator H1-4-5
3. N1-OP-50B Process Radiation Monitoring System
4. N1-CSP-Q215, Service Water Alarm Setpoint Determination, Attachment 2
5. N1-CSP-Q308, Attachment 2
6. NEI 99-01 IC AA1

Category: R – Abnormal Rad Release / Rad Effluent

Subcategory: 1 – Offsite Rad Conditions

Initiating Condition: **ANY** release of gaseous or liquid radioactivity to the environment > 200 times the ODCM for 15 minutes or longer

EAL:

RA1.3	**Alert**
Confirmed sample analyses for gaseous or liquid releases indicate concentrations or release rates > 200 x ODCM limits for ≥ 15 min. (Note 2)	

Note 2: The ED should **not** wait until the applicable time has elapsed, but should declare the event as soon as it is determined that the release duration has exceeded, or will likely exceed, the applicable time. In the absence of data to the contrary, assume that the release duration has exceeded the applicable time if an ongoing release is detected and the release start time is unknown.

Mode Applicability:

All

Basis:

Plant-Specific

Releases in excess of two hundred times the site Offsite Dose Calculation Manual (ODCM) (ref. 1) instantaneous limits that continue for 15 minutes or longer represent an uncontrolled situation and hence, a potential significant degradation in the level of safety. The final integrated dose (which is very low in the Alert emergency class) is not the primary concern here; it is the degradation in plant control implied by the fact that the release was not isolated within 15 minutes. Therefore, it is not intended that the release be averaged over 15 minutes. For example, a release of 400 times the ODCM limit for 7.5 minutes does not exceed this initiating condition. Further, the ED should not wait until 15 minutes has elapsed, but should declare the event as soon as it is determined that the release duration has or will likely exceed 15 minutes.

Generic

The Emergency Director should not wait until the applicable time has elapsed, but should declare the event as soon as it is determined that the condition will likely exceed the applicable time.

This EAL addresses an actual or substantial potential decrease in the level of safety of the plant as indicated by a radiological release that exceeds regulatory commitments for an extended period of time.

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

The 200 x ODCM limit are specified only to distinguish between non-emergency conditions. While these multiples obviously correspond to an off-site dose or dose rate, the emphasis in classifying these events is the degradation in the level of safety of the plant, not the magnitude of the associated dose or dose rate.

Releases should not be prorated or averaged. For example, a release exceeding 600 x ODCM for 5 minutes does not meet the threshold.

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

This EAL addresses uncontrolled releases that are detected by sample analyses, particularly on unmonitored pathways, e.g., spills of radioactive liquids into storm drains, heat exchanger leakage.

NMP1 Basis Reference(s):

1. NMP1 Offsite Dose Calculation Manual (ODCM)
2. NEI 99-01 IC AA1

Category: R – Abnormal Rad Release / Rad Effluent

Subcategory: 1 – Offsite Rad Conditions

Initiating Condition: **ANY** release of gaseous or liquid radioactivity to the environment > 2 times the ODCM for 60 minutes or longer

EAL:

RU1.1	**Unusual Event**
ANY gaseous monitor reading > Table R-1 "UE" column for ≥ 60 min. (Note 2)	

Note 2: The ED should **not** wait until the applicable time has elapsed, but should declare the event as soon as it is determined that the release duration has exceeded, or will likely exceed, the applicable time. In the absence of data to the contrary, assume that the release duration has exceeded the applicable time if an ongoing release is detected and the release start time is unknown.

Table R-1 Effluent Monitor Classification Thresholds				
Monitor	**GE**	**SAE**	**Alert**	**UE**
Gaseous				
Stack (RN10A/B)	N/A	N/A	3.0E4 cps	300 cps
EC Vent	N/A	300 mRem/hr	30 mRem/hr	10 mRem/hr
Liquid				
SW Effluent	N/A	N/A	90,000 cpm	900 cpm
RW Discharge	N/A	N/A	200 x batch	2 x batch

Mode Applicability:

All

Basis:

Plant-Specific

The Emergency Condenser (EC) Vent monitor value shown for the Table R-1 UE level is two times the high alarm setpoint; the main Stack (OGESM, RN 10A/B) monitor value is two times the high-high alarm setpoint. The alarm setpoints for the listed monitors are conservatively set to ensure ODCM radioactivity release limits are not exceeded (ref. 1).

The Reactor Building Vent Monitors are not included in this EAL because the Reactor Building ventilation discharges to the main stack. Radioactivity release from the Reactor Building would, therefore, be assessed by the main stack monitor.

A radiation monitor reading is VALID when a release path is established. If the release path to the environment has been isolated, the radiation monitor reading is not VALID for classification.

Generic

The Emergency Director should not wait until the applicable time has elapsed, but should declare the event as soon as it is determined that the condition will likely exceed the applicable time.

This EAL addresses a potential decrease in the level of safety of the plant as indicated by a radiological release that exceeds regulatory commitments for an extended period of time.

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

The 2 x ODCM limit multiples are specified only to distinguish between non-emergency conditions. While these multiples obviously correspond to an off-site dose or dose rate, the emphasis in classifying these events is the degradation in the level of safety of the plant, not the magnitude of the associated dose or dose rate.

Releases should not be prorated or averaged. For example, a release exceeding 4x ODCM for 30 minutes does not meet the threshold.

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

This EAL addresses radioactivity releases, that for whatever reason, cause effluent radiation monitor readings to exceed the threshold identified in the IC.

This EAL is intended for sites that have established effluent monitoring on non-routine release pathways for which a discharge permit would not normally be prepared.

NMP1 Basis Reference(s):

1. NMP1 Offsite Dose Calculation Manual (ODCM)
2. Calculation 1H21C003, Rev 0
3. N1-ARP-H1 Annunciator H1-1-8
4. N1-OP-50B Process Radiation Monitoring System
5. NEI 99-01 IC AU1

Category: R – Abnormal Rad Release / Rad Effluent

Subcategory: 1 – Offsite Rad Conditions

Initiating Condition: **ANY** release of gaseous or liquid radioactivity to the environment > 2 times the ODCM for 60 minutes or longer

EAL:

RU1.2	**Unusual Event**
ANY liquid monitor reading > Table R-1 "UE" column for ≥ 60 min. (Note 2)	

Note 2: The ED should **not** wait until the applicable time has elapsed, but should declare the event as soon as it is determined that the release duration has exceeded, or will likely exceed, the applicable time. In the absence of data to the contrary, assume that the release duration has exceeded the applicable time if an ongoing release is detected and the release start time is unknown.

Table R-1 Effluent Monitor Classification Thresholds

Monitor	GE	SAE	Alert	UE
Gaseous				
Stack (RN10A/B)	N/A	N/A	3.0E4 cps	300 cps
EC Vent	N/A	300 mRem/hr	30 mRem/hr	10 mRem/hr
Liquid				
SW Effluent	N/A	N/A	90,000 cpm	900 cpm
RW Discharge	N/A	N/A	200 x batch	2 x batch

Mode Applicability:

All

Basis:

Plant-Specific

The Service Water effluent monitor value shown for the UE level is two times the high alarm setpoint. The alarm setpoint is conservatively set to ensure ODCM radioactivity release limits are not exceeded (ref. 1).

The Containment Spray Raw Water Monitors (RW) are not included in this EAL because these monitors detect radiation in the discharge from their respective processes. The monitors are located upstream of the Service Water monitor. Therefore, the Service Water radiation monitor adequately detects offsite radioactivity releases from these systems.

A radiation monitor reading is VALID when a release path is established. If the release path to the environment has been isolated, the radiation monitor reading is not VALID for classification.

Generic

The Emergency Director should not wait until the applicable time has elapsed, but should declare the event as soon as it is determined that the condition will likely exceed the applicable time.

This IC addresses a potential decrease in the level of safety of the plant as indicated by a radiological release that exceeds regulatory commitments for an extended period of time.

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

The 2 x ODCM limit multiples are specified only to distinguish between non-emergency conditions. While these multiples obviously correspond to an off-site dose or dose rate, the emphasis in classifying these events is the degradation in the level of safety of the plant, not the magnitude of the associated dose or dose rate.

Releases should not be prorated or averaged. For example, a release exceeding 4x ODCM for 30 minutes does not meet the threshold.

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

This EAL addresses radioactivity releases, that for whatever reason, cause effluent radiation monitor readings to exceed the threshold identified in the EAL established by the radioactivity discharge permit. This value may be associated with a planned batch release, or a continuous release path.

NMP1 Basis Reference(s):

1. NMP1 Offsite Dose Calculation Manual (ODCM)
2. N1-ARP-H1 Annunciator H1-4-5
3. N1-OP-50B Process Radiation Monitoring System
4. N1-CSP-Q215, Service Water Alarm Setpoint Determination, Attachment 2
5. N1-CSP-Q308, Attachment 2
6. NEI 99-01 IC AU1

Category: R – Abnormal Rad Release / Rad Effluent

Subcategory: 1 – Offsite Rad Conditions

Initiating Condition: **ANY** release of gaseous or liquid radioactivity to the environment > 2 times the ODCM for 60 minutes or longer

EAL:

RU1.3	**Unusual Event**
Confirmed sample analyses for gaseous or liquid releases indicate concentrations or release rates > 2 x ODCM limits for ≥ 60 min. (Note 2)	

Note 2: The ED should **not** wait until the applicable time has elapsed, but should declare the event as soon as it is determined that the release duration has exceeded, or will likely exceed, the applicable time. In the absence of data to the contrary, assume that the release duration has exceeded the applicable time if an ongoing release is detected and the release start time is unknown.

Mode Applicability:

All

Basis:

Plant-Specific

Releases in excess of two times the site Offsite Dose Calculation Manual (ODCM) (ref. 1) instantaneous limits that continue for 60 minutes or longer represent an uncontrolled situation and hence, a potential degradation in the level of safety. The final integrated dose (which is very low in the Unusual Event emergency class) is not the primary concern here; it is the degradation in plant control implied by the fact that the release was not isolated within 60 minutes. Therefore, it is not intended that the release be averaged over 60 minutes. For example, a release of 4 times the ODCM limit for 30 minutes does not exceed this initiating condition. Further, the ED should not wait until 60 minutes has elapsed, but should declare the event as soon as it is determined that the release duration has or will likely exceed 60 minutes.

Generic

The Emergency Director should not wait until the applicable time has elapsed, but should declare the event as soon as it is determined that the condition will likely exceed the applicable time.

This EAL addresses a potential decrease in the level of safety of the plant as indicated by a radiological release that exceeds regulatory commitments for an extended period of time.

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

The 2 x ODCM limit multiples are specified only to distinguish between non-emergency conditions. While these multiples obviously correspond to an off-site dose or dose rate, the emphasis in classifying these events is the degradation in the level of safety of the plant, not the magnitude of the associated dose or dose rate.

Releases should not be prorated or averaged. For example, a release exceeding 4x ODCM for 30 minutes does not meet the threshold.

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

This EAL addresses uncontrolled releases that are detected by sample analyses, particularly on unmonitored pathways, e.g., spills of radioactive liquids into storm drains, heat exchanger leakage in lake water systems, etc.

NMP1 Basis Reference(s):

1. NMP1 Off-Site Dose Calculation Manual (ODCM)
2. NEI 99-01 IC AU1

Category: R – Abnormal Rad Release / Rad Effluent

Subcategory: 2 – Onsite Rad Conditions & Spent Fuel Events

Initiating Condition: Damage to irradiated fuel or loss of water level that has resulted or will result in the uncovering of irradiated fuel outside the Reactor Vessel

EAL:

RA2.1 Alert

Alarm on **ANY** of the following radiation monitors due to damage to irradiated fuel or loss of water level:

- ARM 18 (West end of shield wall)
- ARM 25 (Rx building - east wall)
- ARM 29 (Refuel bridge (LOW RANGE))
- Refuel Bridge (HIGH RANGE)
- Reactor Building Vent Radiation Monitor

Mode Applicability:

All

Basis:

Plant-Specific

This EAL is defined by the specific areas where irradiated fuel is located such as the refueling cavity, RPV or Spent Fuel Pool.

The bases for the area radiation high include a spent fuel handling accident and are, therefore, appropriate for this EAL (ref. 1).

Elevated readings on ventilation monitors may also be indication of a radioactivity release from the fuel, confirming that damage has occurred. However, elevated background at the monitor due to water level lowering may mask elevated ventilation exhaust airborne activity and needs to be considered. The Reactor Building Ventilation Radiation Monitoring System is used to monitor gross gamma radiation levels within the Reactor Building Ventilation System exhaust. Two detectors located in the exhaust plenum upstream of the Ventilation System isolation valves are utilized to signal off normal radiation levels and to automatically isolate the Reactor Building Ventilation System by closing the isolation valves and tripping the normal ventilating fans (ref. 2).

However, while radiation monitors may detect a rise in dose rate due to a drop in the water level, it might not be a reliable indication of whether or not the fuel is covered. For example, the monitor could in fact be properly responding to a known event involving transfer or relocation of a source stored in or near the fuel pool or responding to a planned evolution such as removal of the reactor head. Interpretation of these EAL thresholds requires some understanding of the actual radiological conditions present in the vicinity of the monitors.

Generic

This EAL addresses increases in radiation dose rates within plant buildings, and may be a precursor to a radioactivity release to the environment. These events represent a loss of control over radioactive material and represent an actual or substantial potential degradation in the level of safety of the plant.

This EAL addresses radiation monitor indications of fuel uncovery and/or fuel damage.

Increased ventilation monitor readings may be indication of a radioactivity release from the fuel, confirming that damage has occurred. Increased background at the ventilation monitor due to water level decrease may mask increased ventilation exhaust airborne activity and needs to be considered.

While a radiation monitor could detect an increase in dose rate due to a drop in the water level, it might not be a reliable indication of whether or not the fuel is covered.

Escalation of this emergency classification level, if appropriate, would be based on RS1.1, RS1.2, RS1.3, RG1.2 or RG1.3.

NMP1 Basis Reference(s):

1. N1-OP-50A ARM System Attachment 2
2. N1-OP-50B Process Radiation Monitoring System
3. NEI 99-01 IC AA2

Category: R – Abnormal Rad Release / Rad Effluent

Subcategory: 2 – Onsite Rad Conditions & Spent Fuel Events

Initiating Condition: Damage to irradiated fuel or loss of water level that has resulted or will result in the uncovering of irradiated fuel outside the Reactor Vessel

EAL:

RA2.2	**Alert**
A water level drop in a reactor refueling pathway that will result in irradiated fuel becoming uncovered	

Mode Applicability:

All

Basis:

Plant-Specific

The reactor cavity and Spent Fuel Pool (SFP) comprise the reactor refueling pathway (ref. 1).

Allowing level to decrease could result in spent fuel being uncovered, reducing spent fuel decay heat removal and creating an extremely hazardous radiation environment.

There is no indication that water level in the spent fuel pool has dropped to the level of the fuel other than by visual observation by personnel on the refueling floor. N1-SOP-6.1, Loss of SFP/Rx Cavity Level/Decay Heat Removal, provides appropriate instructions to report a visual observation of irradiated fuel uncovery (ref. 2).

Generic

This event represents a loss of control over radioactive material and represents an actual or substantial potential degradation in the level of safety of the plant.

Escalation of this emergency classification level, if appropriate, would be based on RS1.1, RS1.2, RS1.3, RG1.2 or RG1.3.

NMP1 Basis Reference(s):

1. UFSAR Section X.J.2.1
2. N1-SOP-6.1 Loss of SFP/Rx Cavity Level/Decay Heat Removal
3. NEI 99-01 IC AA2

Category: R – Abnormal Rad Release / Rad Effluent

Subcategory: 2 – Onsite Rad Conditions & Spent Fuel Events

Initiating Condition: UNPLANNED rise in plant radiation levels

EAL:

RU2.1 Unusual Event

UNPLANNED water level drop in a reactor refueling pathway as indicated by inability to restore and maintain SFP level > low water level alarm (Note 3)

AND

Area radiation monitor reading rise on **ANY** of the following:

- ARM 18 (West end of shield wall)
- ARM 25 (Rx building - east wall)
- ARM 29 (Refuel bridge (LOW RANGE))
- Refuel Bridge (HIGH RANGE)

Note 3: If loss of water level in the refueling pathway occurs while in Mode 3, 4 or D, consider classification under EALs CU3.1, CU3.2 or CU3.3

Mode Applicability:

All

Basis:

Plant-Specific

The reactor cavity and Spent Fuel Pool (SFP) comprise the reactor refueling pathway (ref. 1).

The SFP is equipped with level switch LSE-(1S77)54-26C that actuates a low level alarm at El 338' 0" (ref. 2).

The definition of "... inability to restore and maintain SFP level >..." allows the operator to visually observe the low water level condition, if possible, and to attempt water level restoration instructions as long as water level remains above the top of irradiated fuel. Water level restoration instructions are performed in accordance with procedure N1-SOP-6.1, Loss of SFP/Rx Cavity Level/Decay Heat Removal (ref. 3).

The listed Area radiation monitors are located in the proximity of where spent fuel may be located and have been selected to be indicative of a decrease in radiation shielding due to decreasing refueling pathway water level (ref. 4). While a radiation monitor could detect a rise in dose due to a drop in the water level, it might not be a reliable indication, in and of itself, of whether or not the fuel is uncovered. For example, the reading on an area radiation monitor located on the refueling bridge may rise due to planned evolutions such as head lift, or even a fuel assembly being raised in the manipulator mast. Elevated radiation monitor indications will need to be combined with another indicator (or personnel report) of water loss.

This event escalates to an Alert if irradiated fuel outside the RPV is uncovered.

Generic

This EAL addresses increased radiation levels as a result of water level decreases above irradiated fuel or events that have resulted, or may result, in UNPLANNED increases in radiation dose rates within plant buildings. These radiation increases represent a loss of control over radioactive material and represent a potential degradation in the level of safety of the plant.

The refueling pathway is a combination of cavities, tubes, canals and pools. While a radiation monitor could detect an increase in dose rate due to a drop in the water level, it might not be a reliable indication of whether or not the fuel is covered.

For refueling events where the water level drops below the RPV flange classification would be via EAL CU3.1, CU3.2 or CU3.3. This event escalates to an Alert per EAL RA2.1 if irradiated fuel outside the reactor vessel is uncovered. For events involving irradiated fuel in the reactor vessel, escalation would be via the Fission Product Barrier Table for events in operating modes 1 and 2.

NMP1 Basis Reference(s):

1. UFSAR Section X.J.2.1
2. N1-ARP-L1 Annunciator L1-3-5
3. N1-SOP-6.1 Loss of SFP/Rx Cavity Level/Decay Heat Removal
4. N1-OP-50A ARM System Attachment 2
5. NEI 99-01 IC AU2

Category: R – Radioactivity Release / Area Radiation

Subcategory: 2 – Onsite Rad Conditions & Spent Fuel Events

Initiating Condition: UNPLANNED rise in plant radiation levels

EAL:

RU2.2	**Unusual Event**
UNPLANNED area radiation readings rise by a factor of 1,000 over NORMAL LEVELS	

Mode Applicability:

All

Basis:

Plant-Specific

Assessment of this EAL may be made with survey readings using portable instruments as well as installed radiation monitors.

Generic

This EAL addresses increased radiation levels as a result of water level decreases above irradiated fuel or events that have resulted, or may result, in UNPLANNED increases in radiation dose rates within plant buildings. These radiation increases represent a loss of control over radioactive material and represent a potential degradation in the level of safety of the plant.

This EAL addresses increases in plant radiation levels that represent a loss of control of radioactive material resulting in a potential degradation in the level of safety of the plant.

This EAL excludes radiation level increases that result from planned activities such as use of radiographic sources and movement of radioactive waste materials. A specific list of ARMs is not required as it would restrict the applicability of the threshold. The intent is to identify loss of control of radioactive material in any monitored area.

NMP1 Basis Reference(s):

1. NEI 99-01 IC AU2

Category: R – Abnormal Rad Release / Rad Effluent

Subcategory: 3 – CR/CAS Rad

Initiating Condition: Rise in radiation levels within the facility that impedes operation of systems required to maintain plant safety functions

EAL:

RA3.1 Alert

Dose rates > 15 mRem/hr in **EITHER** of the following areas requiring continuous occupancy to maintain plant safety functions:

Control Room

OR

CAS

Mode Applicability:

All

Basis:

Plant-Specific

The Control Room and Central Alarm Station (CAS) must be continuously occupied in all plant operating modes at NMP1.

Area radiation monitor (ARM) #3 monitors radiation levels in the vicinity of the main Control Room. This ARM alarms at 1 mR/hr giving personnel sufficient warning of changing levels (ref. 1). There is no area radiation monitoring system at NMP1 for the CAS. Abnormal radiation levels may be initially detected by routine radiological surveys (ref. 1).

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

This EAL could result in declaration of an Alert at NMP1 due to a radioactivity release or radiation shine resulting from a major accident at NMP2 or JAFNPP. Such a declaration would be appropriate if the increase impairs safe plant operation.

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

Generic

This EAL addresses increased radiation levels that: impact continued operation in areas requiring continuous occupancy to maintain safe operation or to perform a safe shutdown.

The cause and/or magnitude of the increase in radiation levels is not a concern of this EAL. The Emergency Director must consider the source or cause of the increased radiation levels and determine if any other EAL may be involved.

Areas requiring continuous occupancy include the Control Room and any other control stations that are staffed continuously, such as the security alarm station CAS .

NMP1 Basis Reference(s):

1. N1-OP-50A ARM System
2. NEI 99-01 IC AA3

Category H – Hazards and Other Conditions Affecting Plant Safety

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

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

The events of this category pertain to the following subcategories:

1. Natural or Destructive Phenomena

Natural events include hurricanes, earthquakes or tornados that have potential to cause plant structure or equipment damage of sufficient magnitude to threaten personnel or plant safety. Non-naturally occurring events that can cause damage to plant facilities and include aircraft crashes, missile impacts, etc.

2. FIRE or EXPLOSION

FIREs can pose significant hazards to personnel and reactor safety. Appropriate for classification are FIREs within the site PROTECTED AREA or which may affect operability of equipment needed for safe shutdown

3. Hazardous Gas

Non-naturally occurring events that can cause damage to plant facilities and include toxic, asphyxiant, corrosive or flammable gas leaks.

4. Security

Unauthorized entry attempts into the PROTECTED AREA, BOMB threats, SABOTAGE attempts, and actual security compromises threatening loss of physical control of the plant.

5. Control Room Evacuation

Events that are indicative of loss of Control Room habitability. If the Control Room must be evacuated, additional support for monitoring and controlling plant functions is necessary through the emergency response facilities.

6. Judgment

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

Category: H – Hazards and Other Conditions Affecting Plant Safety

Subcategory: Natural or Destructive Phenomena

Initiating Condition: Natural or destructive phenomena affecting VITAL AREAs

EAL:

HA1.1 Alert

NMP-2 seismic instrumentation indicates > 0.075 g

AND

Earthquake confirmed by **ANY** of the following:

- Earthquake felt in plant
- JAFNPP seismic instrumentation
- Control Room indication of degraded performance of systems required for the safe shutdown of the plant

Mode Applicability:

All

Basis:

Plant-Specific

This EAL is based on the NMP2 USAR design basis operating earthquake of 0.075g (ref. 1, 2). Seismic events of this magnitude can cause damage to plant safety functions.

The method of detection relies on actuation of the NMP2 seismic monitor OBE alarm confirmed by one or more indications such as shift operators on duty in the Control Room determining that the ground motion was felt, indication received from JAFNPP instrumentation~~,~~ or degraded system performance ~~or corroborated by the NEIC~~.

The NMP1 design basis operating earthquake is 0.11g. However, due to the seismic instrumentation available at NMP1, determination of seismic activity levels beyond the Seismic Event value of 0.01 g will require evaluation of data recorded by the Seismic Monitoring Recorders. Since this could cause unnecessary delay in classification, action is taken at the lower NMP2 level which is indicated in real time by the NMP2 seismic instrumentation.

NMP1 and NMP2 share a common PROTECTED AREA border. Consideration should be given to the opposite unit when classifying under this EAL.

Generic

These EALs escalate from HU1.1 in that the occurrence of the event has resulted in VISIBLE DAMAGE to plant structures or areas containing equipment necessary for a safe shutdown, or has caused damage to the safety systems in those structures evidenced by control room indications of degraded system response or performance. The occurrence of VISIBLE DAMAGE and/or degraded system response is intended to discriminate against lesser events. The initial report should not be interpreted as mandating a lengthy damage assessment prior to classification. No attempt is made in this EAL to assess the actual magnitude of the damage. The significance here is not that a particular system or structure was damaged, but rather, that the event was of sufficient magnitude to cause this degradation.

Escalation of this emergency classification level, if appropriate, would be based on System Malfunction EALs.

Seismic events of this magnitude can result in a VITAL AREA being subjected to forces beyond design limits, and thus damage may be assumed to have occurred to plant safety systems.

NMP1 Basis Reference(s):

1. NMP2 USAR Section 3.7A.1.1
2. N2-SOP-90 Natural Events
3. NMP2 USAR Section 2.1.1.1
4. N1-SOP-28 Seismic Event
5. USAR Section I.B.13 Characteristics – Structural Design
6. NEI 99-01 IC HA1

Category: H – Hazards and Other Conditions Affecting Plant Safety

Subcategory: 1 – Natural or Destructive Phenomena

Initiating Condition: Natural or destructive phenomena affecting VITAL AREAs

EAL:

HA1.2 Alert

Tornado striking

OR

Sustained high winds > 90 mph
resulting in **EITHER**:

VISIBLE DAMAGE to **ANY** SAFETY-RELATED STRUCTURE, SYSTEM or COMPONENT within **ANY** Table H-1 area

OR

Control Room indication of degraded performance of **ANY** SAFETY-RELATED STRUCTURE, SYSTEM or COMPONENT within **ANY** Table H-1 area

Table H-1 Safe Shutdown Areas

• Reactor Building (including Primary Containment)
• Control Room
• Screenhouse
• Turbine Building
    • Battery Rooms
    • Battery Board Rooms
    • Cable Spreading Room
    • Main Steam Isolation Valve Room
    • Diesel Generator Engine and Board Rooms
• Security
    • Central Alarm Station
    • Secondary Alarm Station
    • Security Uninterruptible Power Supply Room

Mode Applicability:

All

Basis:

Plant-Specific

All Category 1 structures are designed for a wind velocity of 125 mph. NMP2 design wind speed is 90 mph. The more limiting wind speed, therefore, has been selected for NMP1. (ref. 1, 3)

Weather conditions are monitored at three locations:

- The 200 foot high Primary OR Main Meteorological Tower located 0.6 miles west-southwest of NMP2
- The 90 foot Backup Tower located east of JAFNPP
- The 30 foot Inland Tower located at the Oswego County Airport near Fulton

Meteorological parameters such as wind speed are sent to the Control Rooms and Technical Support Centers (TSC) at NMP1, NMP2, JAFNPP and the Emergency Operations Facility (EOF). Data from sensors mounted on these towers are sent to both digital and analog systems for display, processing and storage. Wind speed and wind direction, as well as wind speed deviation and differential temperatures are monitored in NMP1 Control Room and recorded on strip chart recorders on the G panel. (ref. 2)

Wind speed can be measured up to 100 mph.

Weather information may be obtained from (ref. 4):

- National Weather Service: Buffalo 716-565-9001 or 800-462-7751; or Binghamton 607-729-7629
- Accu-Weather: 815-235-8650 or 814-237-5803

The PROTECTED AREA Boundary is depicted in NMP2 USAR Figure 1.2-1, Plot Plan (ref. 5).

This threshold addresses events that may have resulted in a Safe Shutdown Area being subjected to forces beyond design limits and thus damage may be assumed to have occurred to plant safety systems. Safe Shutdown Areas are areas that house equipment the operation of which may be needed to ensure the reactor safely reaches and is maintained in cold shutdown. Safe Shutdown Areas include structures that contain the equipment of concern. The Alert classification is appropriate if relevant plant parameters indicate that the performance of safety systems in the affected Safe Shutdown Areas has been degraded. No attempt should be made to fully inventory the actual magnitude of the damage or quantify the degradation of safety system performance prior to declaration of an Alert under this threshold.

Table H-1 Safe Shutdown Areas include all structures containing Category I equipment and systems needed for safe shutdown (ref. 6).

NMP1 and NMP2 share a common PROTECTED AREA border. Consideration should be given to the opposite unit when classifying under this EAL.

Generic

This EAL escalates from HU1.2 in that the occurrence of the event has resulted in VISIBLE DAMAGE to plant structures or areas containing equipment necessary for a safe shutdown, or has caused damage to the safety systems in those structures evidenced by control room indications of degraded system response or performance. The occurrence of VISIBLE DAMAGE and/or degraded system response is intended to discriminate against lesser events. The initial report should not be interpreted as mandating a lengthy damage assessment prior to classification. No attempt is made in this EAL to assess the actual magnitude of the damage. The significance here is not that a particular system or structure was damaged, but rather, that the event was of sufficient magnitude to cause this degradation.

Escalation of this emergency classification level, if appropriate, would be based on System Malfunction EALs.

This EAL is based on a tornado striking (touching down) or high winds that have caused VISIBLE DAMAGE to structures containing functions or systems required for safe shutdown of the plant.

NMP1 Basis Reference(s):

1. USAR Section VI.C.1.1 Wind and Snow Loadings
2. N1-OP-64 Meteorological Monitoring
3. NMP2 USAR Section 3.3.1.1
4. N2-SOP-64 High Winds
5. NMP2 USAR Figure 1.2-1
6. USAR Section X
7. NEI 99-01 IC HA1

Category: H – Hazards and Other Conditions Affecting Plant Safety

Subcategory: 1 – Natural or Destructive Phenomena

Initiating Condition: Natural or destructive phenomena affecting VITAL AREAs

EAL:

HA1.3 Alert

Internal flooding resulting in **EITHER**:

An electrical shock hazard that precludes access to operate or monitor **ANY** SAFETY-RELATED STRUCTURE, SYSTEM or COMPONENT within **ANY** Table H-1 area

OR

Control Room indication of degraded performance of **ANY** SAFETY-RELATED STRUCTURE, SYSTEM or COMPONENT within **ANY** Table H-1 area

Table H-1 Safe Shutdown Areas

• Reactor Building (including Primary Containment)
• Control Room
• Screenhouse
• Turbine Building
    • Battery Rooms
    • Battery Board Rooms
    • Cable Spreading Room
    • Main Steam Isolation Valve Room
    • Diesel Generator Engine and Board Rooms
• Security
    • Central Alarm Station
    • Secondary Alarm Station
    • Security Uninterruptible Power Supply Room

Mode Applicability:

All

Basis:

<u>Plant-Specific</u>

This threshold addresses the affect of flooding caused by internal events such as component failures, Circulating, Component Cooling or Service Water line ruptures, equipment misalignment, FIRE suppression system actuation, and outage activity mishaps. The internal flooding areas contain systems that are:

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

Table H-1 Safe Shutdown Areas include all structures containing Category I equipment and systems needed for safe shutdown (ref. 1).

Generic

Escalation of this emergency classification level, if appropriate, would be based on System Malfunction EALs.

This EAL addresses the effect of internal flooding caused by events such as component failures, equipment misalignment, or outage activity mishaps. It is based on the degraded performance of systems, or has created industrial safety hazards (e.g., electrical shock) that preclude necessary access to operate or monitor safety equipment. The inability to access, operate or monitor safety equipment represents an actual or substantial potential degradation of the level of safety of the plant.

Flooding as used in this EAL describes a condition where water is entering the room faster than installed equipment is capable of removal, resulting in a rise of water level within the room. Classification of this EAL should not be delayed while corrective actions are being taken to isolate the water source.

NMP1 Basis Reference(s):

1. USAR Section X
2. NEI 99-01 IC HA1

Category: H – Hazards and Other Conditions Affecting Plant Safety

Subcategory: 1 – Natural or Destructive Phenomena

Initiating Condition: Natural or destructive phenomena affecting VITAL AREAs

EAL:

HA1.4 Alert

Turbine failure-generated PROJECTILEs
resulting in **EITHER**:

VISIBLE DAMAGE to or penetration of **ANY** SAFETY-RELATED STRUCTURE, SYSTEM or COMPONENT within **ANY** Table H-1 area

OR

Control Room indication of degraded performance of **ANY** SAFETY-RELATED STRUCTURE, SYSTEM or COMPONENT within **ANY** Table H-1 area

Table H-1 Safe Shutdown Areas

• Reactor Building (including Primary Containment)
• Control Room
• Screenhouse
• Turbine Building
  • Battery Rooms
  • Battery Board Rooms
  • Cable Spreading Room
  • Main Steam Isolation Valve Room
  • Diesel Generator Engine and Board Rooms
• Security
  • Central Alarm Station
  • Secondary Alarm Station
  • Security Uninterruptible Power Supply Room

Mode Applicability:

All

Basis:

Plant-Specific

The turbine generator stores large amounts of rotational kinetic energy in its rotor. In the unlikely event of a major mechanical failure, this energy may be transformed into both rotational and translational energy of rotor fragments. These fragments may impact the surrounding stationary parts. If the energy-absorbing capability of these stationary turbine generator parts is insufficient, external PROJECTILEs will be released. These ejected PROJECTILEs may impact various plant structures, including those housing safety related equipment.

Table H-1 Safe Shutdown Areas include all structures containing Category I equipment and systems needed for safe shutdown (ref. 1).

Generic

This EAL escalates from HU1.4 in that the occurrence of the event has resulted in VISIBLE DAMAGE to plant structures or areas containing equipment necessary for a safe shutdown, or has caused damage to the safety systems in those structures evidenced by control room indications of degraded system response or performance. The occurrence of VISIBLE DAMAGE and/or degraded system response is intended to discriminate against lesser events. The initial report should not be interpreted as mandating a lengthy damage assessment prior to classification. No attempt is made in this EAL to assess the actual magnitude of the damage. The significance here is not that a particular system or structure was damaged, but rather, that the event was of sufficient magnitude to cause this degradation.

Escalation of this emergency classification level, if appropriate, would be based on System Malfunction EALs.

This EAL addresses the threat to safety related equipment imposed by PROJECTILEs generated by main turbine rotating component failures. Therefore, this EAL is consistent with the definition of an Alert in that the potential exists for actual or substantial potential degradation of the level of safety of the plant.

NMP1 Basis Reference(s):

1. USAR Section X
2. NEI 99-01 IC HA1

Category: H – Hazards and Other Conditions Affecting Plant Safety

Subcategory: 1 – Natural or Destructive Phenomena

Initiating Condition: Natural or destructive phenomena affecting VITAL AREAs

EAL:

HA1.5 **Alert**
Lake water level > 254 ft
OR
Intake water level < 236 ft

Mode Applicability:

All

Basis:

Plant-Specific

This threshold covers high and low water level conditions that may have resulted in a plant VITAL AREA being subjected to levels beyond design limits, and thus damage may be assumed to have occurred to plant safety systems.

The high lake level is based upon the maximum probable flood level (ref. 1).

The low intake water level corresponds to the minimum level before damage may occur to the service water pumps (ref. 2-6).

Generic

This EAL addresses other site specific phenomena that result in VISIBLE DAMAGE to VITAL AREAs or results in indication of damage to SAFETY STRUCTURES, SYSTEMS, or COMPONENTS containing functions and systems required for safe shutdown of the plant that can also be precursors of more serious events.

NMP1 Basis Reference(s):

1. USAR Section III-F Screenhouse, Intake and Discharge Tunnels
2. USAR Section X-F Service Water System
3. N1-SOP-18.1 Service Water Failure/Low Intake Level
4. S13.1-100-F003
5. S14-93-F003
6. S16.9NPSHAM002
7. NEI 99-01 IC HA1

Category: H – Hazards and Other Conditions Affecting Plant Safety

Subcategory: 1 – Natural or Destructive Phenomena

Initiating Condition: Natural or destructive phenomena affecting VITAL AREAs

EAL:

HA1.6 Alert

Vehicle crash
resulting in **EITHER**:

VISIBLE DAMAGE to **ANY** SAFETY-RELATED STRUCTURE, SYSTEM or COMPONENT within **ANY** Table H-1 area

OR

Control Room indication of degraded performance of **ANY** SAFETY-RELATED STRUCTURE, SYSTEM or COMPONENT within **ANY** Table H-1 area

Table H-1 Safe Shutdown Areas

• Reactor Building (including Primary Containment)
• Control Room
• Screenhouse
• Turbine Building
    • Battery Rooms
    • Battery Board Rooms
    • Cable Spreading Room
    • Main Steam Isolation Valve Room
    • Diesel Generator Engine and Board Rooms
• Security
    • Central Alarm Station
    • Secondary Alarm Station
    • Security Uninterruptible Power Supply Room

Mode Applicability:

All

Basis:

<u>Plant-Specific</u>

This EAL is intended to address crashes of vehicle types large enough to cause significant damage to plant structures containing functions and systems required for safe shutdown of the plant. Vehicle types include automobiles, aircraft, trucks, cranes, forklifts, waterborne craft, etc.

Table H-1 Safe Shutdown Areas include all Class I Structures and structures containing Class I equipment and systems needed for safe shutdown (ref. 1).

Generic

The occurrence of VISIBLE DAMAGE and/or degraded system response is intended to discriminate against lesser events. The initial report should not be interpreted as mandating a lengthy damage assessment prior to classification. No attempt is made in this EAL to assess the actual magnitude of the damage. The significance here is not that a particular system or structure was damaged, but rather, that the event was of sufficient magnitude to cause this degradation.

Escalation of this emergency classification level, if appropriate, would be based on System Malfunction EALs.

This EAL addresses vehicle crashes within the PROTECTED AREA that results in VISIBLE DAMAGE to VITAL AREAs or indication of damage to SAFETY STRUCTURES, SYSTEMS, or COMPONENTS containing functions and systems required for safe shutdown of the plant.

NMP1 Basis Reference(s):

1. USAR Section X
2. NEI 99-01 IC HA1

Category: H – Hazards and Other Conditions Affecting Plant Safety

Subcategory: 1 – Natural or Destructive Phenomena

Initiating Condition: Natural or destructive phenomena affecting the PROTECTED AREA

EAL:

HU1.1 Unusual Event

Seismic event identified by **ANY two** of the following:

- Annunciator H2-1-6 SEISMIC DETECTION EQUIPMENT EVENT indicates seismic event detected
- Confirmation of earthquake received on NMP-2 or JAFNPP seismic instrumentation
- Earthquake felt in plant

Mode Applicability:

All

Basis:

Plant-Specific

The NMP1seismic instrumentation actuates at 0.01 g causing Annunciator H2-1-6 to be received. This annunciator provides the most direct indication in the Control Room that a seismic event has occurred. Other methods are indication received from NMP-2 or JAFNPP instrumentation.

Evaluation of the magnitude of the event will require evaluation of data recorded by the Seismic Monitoring Recorders.

NMP1 and NMP2 share a common PROTECTED AREA border. Consideration should be given to the opposite unit when classifying under this EAL.

Generic

This EAL is categorized on the basis of the occurrence of an event of sufficient magnitude to be of concern to plant operators.

Damage may be caused to some portions of the site, but should not affect ability of safety functions to operate.

As defined in the EPRI-sponsored Guidelines for Nuclear Plant Response to an Earthquake, dated October 1989, a "felt earthquake" is: An earthquake of sufficient intensity such that: (a) the vibratory ground motion is felt at the nuclear plant site and recognized as an earthquake based on a consensus of control room operators on duty at the time, and (b) for plants with operable seismic instrumentation, the seismic switches of the plant are activated.

NMP-1 Basis Reference(s):

1. N1-ARP-H2 annunciator H2-1-6
2. N1-SOP-28 Seismic Event
3. USAR Section I.B.13 Characteristics
4. NEI 99-01 IC HU1

Category: H – Hazards and Other Conditions Affecting Plant Safety

Subcategory: 1 – Natural or Destructive Phenomena

Initiating Condition: Natural or destructive phenomena affecting the PROTECTED AREA

EAL:

HU1.2 Unusual Event

Tornado striking within PROTECTED AREA boundary

OR

Sustained high winds > 90 mph

Mode Applicability:

All

Basis:

Plant-Specific

All Category 1 structures are designed for a wind velocity of 125 mph. This EAL is declared on a site-wide basis. NMP2 design wind speed is 90 mph. The more limiting wind speed, therefore, has been selected for NMP1. (ref. 1, 3)

Weather conditions are monitored at three locations:

- The 200 foot high Primary OR Main Meteorological Tower located 0.6 miles west-southwest of NMP2
- The 90 foot Backup Tower located east of JAFNPP
- The 30 foot Inland Tower located at the Oswego County Airport near Fulton

Meteorological parameters such as wind speed are sent to the Control Rooms and Technical Support Centers (TSC) at NMP1, NMP2, JAFNPP and the Emergency Operations Facility (EOF). Data from sensors mounted on these towers are sent to both digital and analog systems for display, processing and storage. Wind speed and wind direction, as well as wind speed deviation and differential temperatures are monitored in NMP1 Control Room and recorded on strip chart recorders on the G panel. (ref. 2)

Wind speed can be measured up to 100 mph.

Weather information may be obtained from (ref. 4):

- National Weather Service: Buffalo 716-565-9001 or 800-462-7751; or Binghamton 607-729-7629
- Accu-Weather: 815-235-8650 or 814-237-5803

NMP1 and NMP2 share a common PROTECTED AREA border. Consideration should be given to the opposite unit when classifying under this EAL.

Generic

This EAL is categorized on the basis of the occurrence of an event of sufficient magnitude to be of concern to plant operators.

This EAL is based on a tornado striking (touching down) or high winds within the PROTECTED AREA.

Escalation of this emergency classification level, if appropriate, would be based on VISIBLE DAMAGE, or by other in plant conditions, via EAL HA1.2.

NMP1 Basis Reference(s):

1. USAR Section VI.C.1.1 Wind and Snow Loadings
2. N1-OP-64 Meteorological Monitoring
3. NMP2 USAR Section 3.3.1.1
4. N1-SOP-64 High Winds
5. NEI 99-01 IC HU1

Category: H – Hazards and Other Conditions Affecting Plant Safety

Subcategory: 1 – Natural or Destructive Phenomena

Initiating Condition: Natural or destructive phenomena affecting the PROTECTED AREA

EAL:

HU1.3	**Unusual Event**
Internal flooding that has the potential to affect **ANY** SAFETY-RELATED STRUCTURE, SYSTEM or COMPONENT required by Technical Specifications for the current operating mode in **ANY** Table H-1 area	

Table H-1 Safe Shutdown Areas

- Reactor Building (including Primary Containment)
- Control Room
- Screenhouse
- Turbine Building
  - Battery Rooms
  - Battery Board Rooms
  - Cable Spreading Room
  - Main Steam Isolation Valve Room
  - Diesel Generator Engine and Board Rooms
- Security
  - Central Alarm Station
  - Secondary Alarm Station
  - Security Uninterruptible Power Supply Room

Mode Applicability:

All

Basis:

<u>Plant-Specific</u>

Plant structures evaluated for impact of internal flooding in the NMP-1 Internal Flooding Hazards Analysis (ref. 1) are:

- Reactor Building
- Turbine Building
- Screenhouse

Table H-1 Safe Shutdown Areas include all structures containing Category I equipment and systems needed for safe shutdown (ref. 2).

Generic

This EAL is categorized on the basis of the occurrence of an event of sufficient magnitude to be of concern to plant operators.

This EAL addresses the effect of internal flooding caused by events such as component failures, equipment misalignment, or outage activity mishaps.

Escalation of this emergency classification level, if appropriate, would be based VISIBLE DAMAGE via EAL HA1.3, or by other plant conditions.

NMP1 Basis Reference(s):

1. Calculation S0-FLOOD-F001 Internal Flooding Hazard Analysis
2. USAR Section X
3. NEI 99-01 IC HU1

Category: H – Hazards and Other Conditions Affecting Plant Safety

Subcategory: 1 – Natural or Destructive Phenomena

Initiating Condition: Natural or destructive phenomena affecting the PROTECTED AREA

EAL:

HU1.4 Unusual Event

Turbine failure resulting in **ANY** of the following:

- Casing penetration
- Damage to turbine seals
- Damage to generator seals

Mode Applicability:

All

Basis:

Plant-Specific

The turbine generator stores large amounts of rotational kinetic energy in its rotor. In the unlikely event of a major mechanical failure, this energy may be transformed into both rotational and translational energy of rotor fragments. These fragments may impact the surrounding stationary parts. If the energy-absorbing capability of these stationary turbine generator parts is insufficient, external PROJECTILEs will be released. These ejected PROJECTILEs may impact various plant structures, including those housing safety related equipment.

In the event of PROJECTILE ejection, the probability of a strike on a plant region is a function of the energy and direction of an ejected PROJECTILE and of the orientation of the turbine with respect to the plant region.

Failure of turbine or generator seals may be indicated by a loss of seal oil pressure or loss of condenser vacuum (ref. 3, 4, 5).

Generic

These EALs are categorized on the basis of the occurrence of an event of sufficient magnitude to be of concern to plant operators.

This EAL addresses main turbine rotating component failures of sufficient magnitude to cause observable damage to the turbine casing or to the seals of the turbine generator. Generator seal damage observed after generator purge does not meet the intent of this EAL because it did not impact normal operation of the plant.

Of major concern is the potential for leakage of combustible fluids (lubricating oils) and gases (hydrogen cooling) to the plant environs. Actual FIRES and flammable gas build up are appropriately classified via EAL HU2.1 and EAL HU3.1.

This EAL is consistent with the definition of a UE while maintaining the anticipatory nature desired and recognizing the risk to non-safety related equipment.

Escalation of this emergency classification level, if appropriate, would be to EAL HA1.4 based on damage done by PROJECTILES generated by the failure or in conjunction with a steam generator tube rupture. These latter events would be classified by the Category R EALs or Category F EALs.

NMP1 Basis Reference(s):

1. N1-OP-31 Tandem Compound Reheat Turbine
2. N1-SOP-31.1 Turbine Trip
3. N1-ARP-A1 3-4 Condenser Vacuum Below 24" Hg
4. N1-ARP-A1 4-1 Generator H_2 Seal Oil Pressure Low
5. N1-SOP-25.1 Unplanned Loss of Condenser Vacuum
6. NEI 99-01 IC HU1

Category: H – Hazards and Other Conditions Affecting Plant Safety

Subcategory: 1 – Natural or Destructive Phenomena

Initiating Condition: Natural or destructive phenomena affecting the PROTECTED AREA

EAL:

HU1.5	**Unusual Event**
Lake water level > 248.2 ft	
OR	
Intake water level < 238.8 ft	

Mode Applicability:

All

Basis:

Plant-Specific

This threshold addresses high and low bay water level conditions that could be a precursor of more serious events (ref. 1, 2).

The high lake level is based upon the maximum attainable uncontrolled lake water level as specified in the NMP 2 USAR. Dams on the St. Lawrence River, under the authority of the International St. Lawrence River Board of Control, are now used to regulate the lake level. The low limit is set for el 74.37 m (244 ft) on April 1 and is maintained at or above that elevation during the entire navigation season (April 1 to November 30). The upper limit of the lake level is el 75.59 m (248.2 ft) (ref. 3).

The low level is based on intake forebay level and corresponds to the minimum intake water level for operability of Emergency Service Water, Emergency Diesel Generator cooling water, Containment Spray Raw Water and Diesel and Electric FIRE Pump (ref. 4-9).

During planned evolutions such as intake water gate manipulation for reverse flow operations in which continuous monitoring of the intake level is being accomplished, entry into this EAL would not be warranted unless UNPLANNED /unexpected conditions and/or indications occur.

Generic

This EAL is categorized on the basis of the occurrence of an event of sufficient magnitude to be of concern to plant operators.

This EAL addresses other site specific phenomena that can also be precursors of more serious events.

NMP1 Basis Reference(s):

1. USAR Section III-F Screenhouse, Intake and Discharge Tunnels
2. USAR Section X-F Service Water System
3. NMP 2 USAR Section 2.4.11.2
4. N1-ARP-H2 Annunciator H2-1-3
5. N1-SOP-18.1 Service Water Failure/Low Intake Level
6. S13.1-100F003
7. S14-93F003
8. S16.9NPSHAM002
9. Calc No. S14-93-F007
10. NEI 99-01 IC HU1

Category: H – Hazards and Other Conditions Affecting Plant Safety

Subcategory: 2 – FIRE or EXPLOSION

Initiating Condition: FIRE or EXPLOSION affecting the operability of plant safety systems required to establish or maintain safe shutdown

EAL:

HA2.1 Alert

FIRE or EXPLOSION
resulting in **EITHER**:

VISIBLE DAMAGE to **ANY** SAFETY-RELATED STRUCTURE, SYSTEM or COMPONENT within **ANY** Table H-1 area

OR

Control Room indication of degraded performance of **ANY** SAFETY-RELATED STRUCTURE, SYSTEM or COMPONENT within **ANY** Table H-1 area

Table H-1 Safe Shutdown Areas

- Reactor Building (including Primary Containment)
- Control Room
- Screenhouse
- Turbine Building
  - Battery Rooms
  - Battery Board Rooms
  - Cable Spreading Room
  - Main Steam Isolation Valve Room
  - Diesel Generator Engine and Board Rooms
- Security
  - Central Alarm Station
  - Secondary Alarm Station
  - Security Uninterruptible Power Supply Room

Mode Applicability:

All

Basis:

Plant-Specific

Table H-1 Safe Shutdown Areas include all structures containing Category I equipment and systems needed for safe shutdown (ref. 1).

Generic

VISIBLE DAMAGE is used to identify the magnitude of the FIRE or EXPLOSION and to discriminate against minor FIREs and EXPLOSIONs.

The reference to structures containing safety systems or components is included to discriminate against FIREs or EXPLOSIONs in areas having a low probability of affecting safe operation. The significance here is not that a safety system was degraded but the fact that the FIRE or EXPLOSION was large enough to cause damage to these systems.

The use of VISIBLE DAMAGE should not be interpreted as mandating a lengthy damage assessment prior to classification. The declaration of an Alert and the activation of the Technical Support Center will provide the Emergency Director with the resources needed to perform detailed damage assessments.

The Emergency Director also needs to consider any security aspects of the EXPLOSION.

Escalation of this emergency classification level, if appropriate, will be based on EALs in Category S, Category F or Category R.

NMP1 Basis Reference(s):

1. USAR Section X
2. NEI 99-01 IC HA2

Category: H – Hazards and Other Conditions Affecting Plant Safety

Subcategory: 2 – FIRE or EXPLOSION

Initiating Condition: FIRE within the PROTECTED AREA **not** extinguished within 15 min. of detection or EXPLOSION within the PROTECTED AREA

EAL:

HU2.1 Unusual Event

FIRE **not** extinguished within 15 min. of Control Room notification or verification of a Control Room FIRE alarm in **ANY** Table H-1 area, RadWaste Solidification and Storage Bldg, or Security West Bldg (Note 4)

Note 4: The ED should **not** wait until the applicable time has elapsed, but should declare the event as soon as it is determined that the condition has exceeded, or will likely exceed, the applicable time.

Table H-1 Safe Shutdown Areas

- Reactor Building (including Primary Containment)
- Control Room
- Screenhouse
- Turbine Building
  - Battery Rooms
  - Battery Board Rooms
  - Cable Spreading Room
  - Main Steam Isolation Valve Room
  - Diesel Generator Engine and Board Rooms
- Security
  - Central Alarm Station
  - Secondary Alarm Station
  - Security Uninterruptible Power Supply Room

Mode Applicability:

All

Basis:

Plant-Specific

Table H-1 Safe Shutdown Areas include all structures containing Category I equipment and systems needed for safe shutdown (ref. 1). The RadWaste Solidification and Storage Bldg. and Security West Bldg. are included because they are immediately adjacent to one or more Table H-1 areas and a FIRE within either of those buildings may potentially impact safe shutdown equipment should the FIRE not be controlled (ref. 1).

Generic

This EAL addresses the magnitude and extent of FIREs that may be potentially significant precursors of damage to safety systems. It addresses the FIRE, and not the degradation in performance of affected systems that may result.

As used here, detection is visual observation and either report by plant personnel or sensor alarm indication.

The 15 minute time period begins with a credible notification that a FIRE is occurring, or indication of a FIRE detection system alarm/actuation. Verification of a FIRE detection system alarm/actuation includes actions that can be taken within the control room or other nearby site specific location to ensure that it is not spurious. An alarm is assumed to be an indication of a FIRE unless it is disproved within the 15 minute period by personnel dispatched to the scene. In other words, a personnel report from the scene may be used to disprove a sensor alarm if received within 15 minutes of the alarm, but shall not be required to verify the alarm.~~The purpose of this threshold is to address the magnitude and extent of FIREs that may be potentially significant precursors to damage to safety systems. As used here, notification is visual observation and either report by plant personnel or sensor alarm indication. The 15-minute period to extinguish the FIRE begins with a credible notification that a FIRE is occurring or indication of a VALID FIRE detection system alarm. Determination of a VALID FIRE detection system alarm includes actions that can be taken within the Control Room or at nearby Fire Panels to determine that the alarm is not spurious. These actions include the use of direct or indirect indications such as redundant alarms or instrumentation readings associated with the area to ensure the alarm is not spurious and is an indication of a FIRE. An alarm verified in this manner is assumed to be an indication of a FIRE unless personnel dispatched to the scene disprove the alarm within the 15-minute period. The report, however, shall not be required to verify the alarm. If the alarm cannot be verified by redundant Control Room or nearby Fire Panel indications, notification from the field that a FIRE exists would be required to start both the 15-minute classification and FIRE extinguishment clocks.~~

The intent of this 15 minute duration is to size the FIRE and to discriminate against small FIREs that are readily extinguished (e.g., smoldering waste paper basket).

NMP1 Basis Reference(s):

1. USAR Section X
2. NEI 99-01 IC HU2

Category: H – Hazards and Other Conditions Affecting Plant Safety

Subcategory: 2 – FIRE or EXPLOSION

Initiating Condition: FIRE within the PROTECTED AREA **not** extinguished within 15 min. of detection or EXPLOSION within the PROTECTED AREA

EAL:

HU2.2 **Unusual Event**
EXPLOSION of sufficient force to damage permanent structures or equipment within the PROTECTED AREA

Mode Applicability:

All

Basis:

Plant-Specific

While some EXPLOSIONs may also result in FIREs that exceed EAL HU2.1, no FIRE is necessary to declare an emergency in the event of an EXPLOSION. If a FIRE also occurs as a result or with an EXPLOSION, declare the Unusual Event based on the EXPLOSION and monitor the progress of the FIRE for potential escalation due to FIRE damage.

NMP1 and NMP2 share a common PROTECTED AREA border. NMP1 and NMP2 PROTECTED AREA boundaries are illustrated in NMP-2 USAR Figure 1.2-1 (ref. 1).

Generic

This EAL addresses the magnitude and extent of EXPLOSIONs that may be potentially significant precursors of damage to safety systems. It addresses the EXPLOSION, and not the degradation in performance of affected systems that may result.

This EAL addresses only those EXPLOSIONs of sufficient force to damage permanent structures or equipment within the PROTECTED AREA.

No attempt is made to assess the actual magnitude of the damage. The occurrence of the EXPLOSION is sufficient for declaration.

The Emergency director also needs to consider any security aspects of the EXPLOSION, if applicable.

Escalation of this emergency classification level, if appropriate, would be based on EAL HA2.1.

NMP1 Basis Reference(s):

1. NMP-2 USAR Figure 1.2-1

2. NEI 99-01 IC HU2

Category: H – Hazards and Other Conditions Affecting Plant Safety

Subcategory: 3 – Hazardous Gas

Initiating Condition: Access to a VITAL AREA is prohibited due to toxic, corrosive, asphyxiant or flammable gases which jeopardize operation of operable equipment required to maintain safe operations or safely shutdown the reactor

EAL:

HA3.1 **Alert**
Access to **ANY** Table H-1 area is prohibited due to toxic, corrosive, asphyxiant or flammable gases which jeopardize operation of systems required to maintain safe operations or safely shutdown the reactor~~**ANY** SAFETY-RELATED STRUCTURE, SYSTEM or COMPONENT~~ (Note 5)

Note 5: If the equipment in the stated area was already inoperable, or out of service, before the event occurred, then EAL HA3.1 should **not** be declared as it will have **no** adverse impact on the ability of the plant to safely operate or safely shutdown beyond that already allowed by Technical Specifications at the time of the event.

Table H-1 Safe Shutdown Areas
• Reactor Building (including Primary Containment)
• Control Room
• Screenhouse
• Turbine Building
• Battery Rooms
• Battery Board Rooms
• Cable Spreading Room
• Main Steam Isolation Valve Room
• Diesel Generator Engine and Board Rooms
• Security
• Central Alarm Station
• Secondary Alarm Station
• Security Uninterruptible Power Supply Room

Mode Applicability:

All

Basis:

Plant-Specific

Table H-1 Safe Shutdown Areas include all Class I Structures and structures containing Class I equipment and systems needed for safe shutdown (ref. 1).

For areas that contain no safety-related structure, system or component that would potentially be required to be operated or for which the structure, system or component was already out of service or inoperable before the event, this EAL would not be applicable.

For purposes of this EAL, any gas (CO_2 included) is considered toxic when oxygen concentrations in the affected areas have been or could be expected to be reduced to <19.5% or toxicity of the gas will be injurious to persons inhaling it. For discharges of Halon, NMP's systems are designed for discharge concentration from 5% up to 6.5%. In accordance with NFPA 12 A, Halon 1301 Fire Extinguishing Systems, exposures to levels of up to 7% produce little if any noticeable effect (ref. 2).

Generic

Gases in a Safe Shutdown Area can affect the ability to safely operate or safely shutdown the reactor.

The fact that SCBA may be worn does not eliminate the need to declare the event.

Declaration should not be delayed for confirmation from atmospheric testing if the atmosphere poses an immediate threat to life and health or an immediate threat of severe exposure to gases. This could be based upon documented analysis, indication of personal ill effects from exposure, or operating experience with the hazards.

If the equipment in the stated area was already inoperable, or out of service, before the event occurred, then this EAL should not be declared as it will have no adverse impact on the ability of the plant to safely operate or safely shutdown beyond that already allowed by Technical Specifications at the time of the event.

An asphyxiant is a gas capable of reducing the level of oxygen in the body to dangerous levels. Most commonly, asphyxiants work by merely displacing air in an enclosed environment. This reduces the concentration of oxygen below the normal level of around 19%, which can lead to breathing difficulties, unconsciousness or even death.

An uncontrolled release of flammable gasses within a facility structure has the potential to affect safe operation of the plant by limiting either operator or equipment operations due to the potential for ignition and resulting equipment damage/personnel injury. Flammable gasses, such as hydrogen and acetylene, are routinely used to maintain plant systems (hydrogen) or to repair equipment/components (acetylene - used in welding). This EAL assumes concentrations of flammable gasses which can ignite/support combustion.

Escalation of this emergency classification level, if appropriate, will be based on EALs in Category S, Category F or Category R.

NMP1 Basis Reference(s):

1. USAR Section X
2. NFPA 12 A Halon 1301 Fire Extinguishing Systems
3. NEI 99-01 IC HA3

Category: H – Hazards and Other Conditions Affecting Plant Safety

Subcategory: 3 – Hazardous Gas

Initiating Condition: Release of toxic, corrosive, asphyxiant or flammable gases deemed detrimental to NORMAL PLANT OPERATIONS

EAL:

HU3.1	**Unusual Event**
Toxic, corrosive, asphyxiant or flammable gases in amounts that have or could adversely affect NORMAL PLANT OPERATIONS	

Mode Applicability:

All

Basis:

Plant-Specific

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

For purposes of this EAL, any gas (CO_2 included) is considered toxic when oxygen concentrations in the affected areas have been or could be expected to be reduced to <19.5% or toxicity of the gas will be injurious to persons inhaling it. For discharges of Halon, NMP's systems are designed for discharge concentration from 5% up to 6.5%. In accordance with NFPA 12 A, Halon 1301 Fire Extinguishing Systems, exposures to levels of up to 7% produce little if any noticeable effect (ref. 1).

Generic

This EAL is based on the release of toxic, corrosive, asphyxiant or flammable gases of sufficient quantity to affect NORMAL PLANT OPERATIONS.

The fact that SCBA may be worn does not eliminate the need to declare the event.

This EAL is not intended to require significant assessment or quantification. It assumes an uncontrolled process that has the potential to affect plant operations. This would preclude small or incidental releases, or releases that do not impact structures needed for plant operation.

An asphyxiant is a gas capable of reducing the level of oxygen in the body to dangerous levels. Most commonly, asphyxiants work by merely displacing air in an enclosed environment. This reduces the concentration of oxygen below the normal level of around 19%, which can lead to breathing difficulties, unconsciousness or even death.

Escalation of this emergency classification level, if appropriate, would be based on EAL HA3.1.

NMP1 Basis Reference(s):

1. NFPA 12 A Halon 1301 Fire Extinguishing Systems
2. NEI 99-01 IC HU3

Category: H – Hazards and Other Conditions Affecting Plant Safety

Subcategory: 3 – Hazardous Gas

Initiating Condition: Release of toxic, corrosive, asphyxiant or flammable gases deemed detrimental to NORMAL PLANT OPERATIONS

EAL:

HU3.2	**Unusual Event**
Recommendation by local, county or state officials to evacuate or shelter site personnel based on an offsite event	

Mode Applicability:

All

Basis:

Plant-Specific

A recommendation by offsite officials that a potential evacuation of site personnel may be required based on an offsite event assumes that the plant lies within an evacuation area established by offsite officials due to a release of toxic, corrosive, asphyxiant or flammable gas. In this case, it can be assumed that an actual or potential release of such hazardous gas is anticipated to enter the PROTECTED AREA in amounts that could affect the health of plant personnel or NORMAL PLANT OPERATIONS.

Generic

Escalation of this emergency classification level, if appropriate, would be based on EAL HA3.1.

NMP1 Basis Reference(s):

1. NEI 99-01 IC HU3

Category: H – Hazards and Other Conditions Affecting Plant Safety

Subcategory: 4 – Security

Initiating Condition: HOSTILE ACTION resulting in loss of physical control of the facility

EAL:

HG4.1	**General Emergency**
A HOSTILE ACTION has occurred such that plant personnel are unable to operate equipment required to maintain safety functions	

Mode Applicability:

All

Basis:

Plant-Specific

Safety functions include:

- Reactivity control – ability to shut down the reactor and keep it shutdown
- RPV water level control – ability to cool the core
- Decay heat removal – ability to maintain a heat sink

Generic

This EAL encompasses conditions under which a HOSTILE ACTION has resulted in a loss of physical control of VITAL AREAs (containing vital equipment or controls of vital equipment) required to maintain safety functions and control of that equipment cannot be transferred to and operated from another location.

If control of the plant equipment necessary to maintain safety functions can be transferred to another location, then the threshold is not met.

NMP1 Basis Reference(s):

1. NEI 99-01 IC HG1

Category: H – Hazards and Other Conditions Affecting Plant Safety

Subcategory: 4 – Security

Initiating Condition: HOSTILE ACTION resulting in loss of physical control of the facility

EAL:

HG4.2	**General Emergency**
A HOSTILE ACTION has caused failure of Spent Fuel Cooling systems **AND** IMMINENT fuel damage is likely	

Mode Applicability:

All

Basis:

Plant-Specific

None

Generic

This EAL addresses failure of spent fuel cooling systems as a result of HOSTILE ACTION if IMMINENT fuel damage is likely.

NMP1 Basis Reference(s):

1. NEI 99-01 IC HG1

Category: H – Hazards and Other Conditions Affecting Plant Safety

Subcategory: 4 – Security

Initiating Condition: HOSTILE ACTION within the PROTECTED AREA

EAL:

HS4.1	**Site Area Emergency**
A HOSTILE ACTION is occurring or has occurred within the PROTECTED AREA as reported by the Security Site Supervisor	

Mode Applicability:

All

Basis:

Plant-Specific

None

Generic

This condition represents an escalated threat to plant safety above that contained in the Alert in that a HOSTILE FORCE has progressed from the Owner Controlled Area to the PROTECTED AREA.

This EAL addresses the contingency for a very rapid progression of events, such as that experienced on September 11, 2001. It is not premised solely on the potential for a radiological release. Rather the issue includes the need for rapid assistance due to the possibility for significant and indeterminate damage from additional air, land or water attack elements.

The fact that the site is under serious attack with minimal time available for further preparation or additional assistance to arrive requires Offsite Response Organization (ORO) readiness and preparation for the implementation of protective measures.

This EAL addresses the potential for a very rapid progression of events due to a HOSTILE ACTION. It is not intended to address incidents that are accidental events or acts of civil disobedience, such as small aircraft impact, hunters, or physical disputes between employees within the PROTECTED AREA. Those events are adequately addressed by other EALs.

Escalation of this emergency classification level, if appropriate, would be based on actual plant status after impact or progression of attack.

NMP1 Basis Reference(s):

1. NMP Site Security Plan
2. NEI 99-01 IC HS4

Category: H – Hazards and Other Conditions Affecting Plant Safety

Subcategory: 4 – Security

Initiating Condition: HOSTILE ACTION within the Owner Controlled Area or airborne attack threat

EAL:

HA4.1	**Alert**
A HOSTILE ACTION is occurring or has occurred within the Owner Controlled Area as reported by the Security Site Supervisor **OR** A validated notification from NRC of an AIRLINER attack threat within 30 min. of the site	

Mode Applicability:

All

Basis:

Plant-Specific

None

Generic

Note: Timely and accurate communication between the Security Site Supervisor and the Control Room is crucial for the implementation of effective Security EALs.

This EAL addresses the contingency for a very rapid progression of events, such as that experienced on September 11, 2001. They are not premised solely on the potential for a radiological release. Rather the issue includes the need for rapid assistance due to the possibility for significant and indeterminate damage from additional air, land or water attack elements.

The fact that the site is under serious attack or is an identified attack target with minimal time available for further preparation or additional assistance to arrive requires a heightened state of readiness and implementation of protective measures that can be effective (such as on-site evacuation, dispersal or sheltering).

First Condition

This condition addresses the potential for a very rapid progression of events due to a HOSTILE ACTION. It is not intended to address incidents that are accidental events or acts of civil disobedience, such as small aircraft impact, hunters, or physical disputes between employees within the Owner Controlled Area. Those events are adequately addressed by other EALs.

Note that this condition is applicable for any HOSTILE ACTION occurring, or that has occurred, in the Owner Controlled Area.

Second Condition

This condition addresses the immediacy of an expected threat arrival or impact on the site within a relatively short time.

The intent of this condition is to ensure that notifications for the AIRLINER attack threat are made in a timely manner and that Offsite Response Organizations (OROs) and plant personnel are at a state of heightened awareness regarding the credible threat. AIRLINER is meant to be a LARGE AIRCRAFT with the potential for causing significant damage to the plant.

This condition is met when a plant receives information regarding an AIRLINER attack threat from NRC and the AIRLINER is within 30 minutes of the plant. Only the plant to which the specific threat is made need declare the Alert.

The NRC Headquarters Operations Officer (HOO) will communicate to the licensee if the threat involves an AIRLINER (AIRLINER is meant to be a LARGE AIRCRAFT with the potential for causing significant damage to the plant). The status and size of the plane may be provided by NORAD through the NRC.

NMP1 Basis Reference(s):

1. NMP Site Security Plan
2. NEI 99-01 IC HA4

Category: H – Hazards and Other Conditions Affecting Plant Safety

Subcategory: 4 – Security

Initiating Condition: Confirmed SECURITY CONDITION or threat which indicates a potential degradation in the level of safety of the plant

EAL:

HU4.1 Unusual Event

A SECURITY CONDITION that does **not** involve a HOSTILE ACTION as reported by the Security Site Supervisor

OR

A credible site-specific security threat notification

OR

A validated notification from NRC providing information of an aircraft threat

Mode Applicability:

All

Basis:

Plant-Specific

If the Security Site Supervisor determines that a threat notification is credible, the Security Site Supervisor will notify the Operations Shift Manager that a "Credible Threat" condition exists for NMP. Generally, NMP Security Procedures address standard practices for determining credibility. The three main criteria for determining credibility are: technical feasibility, operational feasibility, and resolve. For NMP1, a validated notification delivered by the FBI, the NRC or similar agency is treated as credible.

Generic

Note: Timely and accurate communication between the Security Site Supervisor and the Control Room is crucial for the implementation of effective Security EALs.

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 in some cases under 10 CFR 50.72. Security events assessed as HOSTILE ACTIONs are classifiable under EAL HA4.1, EAL HS4.1 and EAL HG4.1.

A higher initial classification could be made based upon the nature and timing of the security threat and potential consequences. The licensee shall consider upgrading the emergency response status and emergency classification level in accordance with the NMP Site Security Plan.

First Condition

Reference is made to security shift supervision because these individuals are the designated personnel on-site qualified and trained to confirm that a security event is occurring or has occurred. Training on security event classification confirmation is closely controlled due to the strict secrecy controls placed on the NMP Site Security Plan.

This threshold is based on the NMP Site Security Plan. The NMP Site Security Plan is based on guidance provided by NEI 03-12.

Second Condition

This threshold is included to ensure that appropriate notifications for the security threat are made in a timely manner. This includes information of a credible threat. Only the plant to which the specific threat is made need declare the Unusual Event.

The determination of "credible" is made through use of information found in the NMP Site Security Plan.

Third Condition

The intent of this EAL is to ensure that notifications for the aircraft threat are made in a timely manner and that Offsite Response Organizations and plant personnel are at a state of heightened awareness regarding the credible threat. It is not the intent of this EAL to replace existing non-hostile related EALs involving aircraft.

This EAL is met when a plant receives information regarding an aircraft threat from NRC. Validation is performed by calling the NRC or by other approved methods of authentication. Only the plant to which the specific threat is made need declare the Unusual Event.

The NRC Headquarters Operations Officer (HOO) will communicate to the licensee if the threat involves an AIRLINER (AIRLINER is meant to be a LARGE AIRCRAFT with the potential for causing significant damage to the plant). The status and size of the plane may be provided by NORAD through the NRC.

Escalation to Alert emergency classification level via EAL HA4.1 would be appropriate if the threat involves an AIRLINER within 30 minutes of the plant.

NMP1 Basis Reference(s):

1. NMP Site Security Plan
2. NEI 99-01 IC HU4

Category: H – Hazards and Other Conditions Affecting Plant Safety

Subcategory: 5 – Control Room Evacuation

Initiating Condition: Control Room evacuation has been initiated and plant control **cannot** be established

EAL:

HS5.1 Site Area Emergency
Control Room evacuation has been initiated
AND
Control of the plant **cannot** be established within 15 min. ~~(Note 4)~~

~~Note 4: The ED should **not** wait until the applicable time has elapsed, but should declare the event as soon as it is determined that the condition has exceeded, or will likely exceed, the applicable time.~~

Mode Applicability:

All

Basis:

Plant-Specific

N1-SOP-21.2, Control Room Evacuation, provides specific instructions for evacuating the Control Room/Building and establishing plant control in alternate locations.

Generic

The intent of this EAL is to capture those events where control of the plant cannot be reestablished in a timely manner. In this case, expeditious transfer of control of safety systems has not occurred (although fission product barrier damage may not yet be indicated).

The intent of the EAL is to establish control of important plant equipment and knowledge of important plant parameters in a timely manner. Primary emphasis should be placed on those components and instruments that supply protection for and information about safety functions. Typically, these safety functions are reactivity control (ability to reach and maintain reactor shutdown), reactor water level (ability to cool the core), and decay heat removal (ability to maintain a heat sink).

The determination of whether or not control is established at the remote shutdown panel is based on Emergency Director (ED) judgment. The Emergency Director is expected to make a reasonable, informed judgment within the site specific time for transfer that the licensee has control of the plant from the remote shutdown panel.

Escalation of this emergency classification level, if appropriate, would be by EALs in Category F or Category R.

NMP1 Basis Reference(s):

1. N1-SOP-21.2 Control Room Evacuation

2. NEI 99-01 IC HS2

Category: H – Hazards and Other Conditions Affecting Plant Safety

Subcategory: 5 – Control Room Evacuation

Initiating Condition: Control Room evacuation has been initiated

EAL:

HA5.1	**Alert**
Control Room evacuation has been initiated	

Mode Applicability:

All

Basis:

Plant-Specific

N1-SOP-21.2, Control Room Evacuation, provides specific instructions for evacuating the Control Room/Building and establishing plant control in alternate locations.

Generic

With the control room evacuated, additional support, monitoring and direction through the Technical Support Center and/or other emergency response facilities may be necessary.

Inability to establish plant control from outside the control room will escalate this event to a Site Area Emergency.

NMP1 Basis Reference(s):

1. N1-SOP-21.2 Control Room Evacuation
2. NEI 99-01 IC HA5

Category: H – Hazards and Other Conditions Affecting Plant Safety

Subcategory: 6 – Judgment

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

EAL:

HG6.1	**General Emergency**
Other conditions exist which in the judgment of the Emergency Director indicate that events are in progress or have occurred which involve actual or IMMINENT substantial core degradation or melting with potential for loss of containment integrity or HOSTILE ACTION that results in an actual loss of physical control of the facility. Releases can be reasonably expected to exceed EPA Protective Action Guideline exposure levels (1,000 mRem TEDE or 5,000 mRem thyroid CDE) offsite for more than the immediate site area	

Mode Applicability:

All

Basis:

Plant-Specific

None

Generic

This EAL addresses unanticipated conditions not addressed explicitly elsewhere but that warrant declaration of an emergency because conditions exist which are believed by the Emergency Director to fall under the emergency classification level description for General Emergency.

NMP1 Basis Reference(s):

1. NEI 99-01 IC HG2

Category: H – Hazards and Other Conditions Affecting Plant Safety

Subcategory: 6 – Judgment

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

EAL:

HS6.1	**Site Area Emergency**
Other conditions exist which in the judgment of the Emergency Director indicate that events are in progress or have occurred which involve actual or likely major failures of plant functions needed for protection of the public or HOSTILE ACTION that results in intentional damage or malicious acts; (1) toward site personnel or equipment that could lead to the likely failure of or; (2) that prevent effective access to equipment needed for the protection of the public. **ANY** releases are **not** expected to result in exposure levels which exceed EPA Protective Action Guideline exposure levels (1,000 mRem TEDE or 5,000 mRem thyroid CDE) beyond the SITE BOUNDARY	

Mode Applicability:

All

Basis:

<u>Plant-Specific</u>

None

<u>Generic</u>

This EAL addresses unanticipated conditions not addressed explicitly elsewhere but that warrant declaration of an emergency because conditions exist which are believed by the Emergency Director to fall under the emergency classification level description for Site Area Emergency.

NMP1 Basis Reference(s):

1. NEI 99-01 IC HS3

Category: H – Hazards and Other Conditions Affecting Plant Safety

Subcategory: 6 – Judgment

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

EAL:

HA6.1	**Alert**
Other conditions exist which in the judgment of the Emergency Director indicate that events are in progress or have occurred which involve an actual or potential substantial degradation of the level of safety of the plant or a security event that involves probable life threatening risk to site personnel or damage to site equipment because of HOSTILE ACTION. **ANY** releases are expected to be limited to small fractions of the EPA Protective Action Guideline exposure levels (1,000 mRem TEDE ord 5,000 mRem thyroid CDE)	

Mode Applicability:

All

Basis:

Plant-Specific

None

Generic

This EAL addresses unanticipated conditions not addressed explicitly elsewhere but that warrant declaration of an emergency because conditions exist which are believed by the Emergency Director to fall under the Alert emergency classification level.

NMP1 Basis Reference(s):

1. NEI 99-01 IC HA6

Category: H – Hazards and Other Conditions Affecting Plant Safety

Subcategory: 6 – Judgment

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

EAL:

HU6.1	**Unusual Event**
Other conditions exist which in the judgment of the Emergency Director indicate that events are in progress or have occurred which indicate a potential degradation of the level of safety of the plant or indicate a security threat to facility protection has been initiated. **No** releases of radioactive material requiring offsite response or monitoring are expected unless further degradation of safety systems occurs	

Mode Applicability:

All

Basis:

Plant-Specific

None

Generic

This EAL addresses unanticipated conditions not addressed explicitly elsewhere but that warrant declaration of an emergency because conditions exist which are believed by the Emergency Director to fall under the UE emergency classification level.

NMP1 Basis Reference(s):

1. NEI 99-01 IC HU5

Category E – INDEPENDENT SPENT FUEL STORAGE INSTALLATION (ISFSI)

EAL Group: Not Applicable (the EAL in this category is applicable independent of plant operating mode)

An INDEPENDENT SPENT FUEL STORAGE INSTALLATION (ISFSI) is a complex that is designed and constructed for the interim storage of spent nuclear fuel and other radioactive materials associated with spent fuel storage. A significant amount of the radioactive material contained within a cask/canister must escape its packaging and enter the biosphere for there to be a significant environmental effect resulting from an accident involving the dry storage of spent nuclear fuel. Formal offsite planning is not required because the postulated worst-case accident involving an ISFSI has insignificant consequences to the public health and safety.

A Notification of Unusual Event is declared on the basis of the occurrence of an event of sufficient magnitude that a loaded cask CONFINEMENT BOUNDARY is damaged or violated. This includes classification based on a loaded fuel storage cask/canister CONFINEMENT BOUNDARY loss leading to the degradation of the fuel during storage or posing an operational safety problem with respect to its removal from storage.

A hostile security event that leads to a potential loss in the level of safety of the ISFSI is a classifiable event under Security category EAL HA4.1.

Minor surface damage that does not affect storage cask/canister boundary is excluded from the scope of these EALs.

Category: E – ISFSI

Subcategory: Not Applicable

Initiating Condition: Damage to a loaded cask CONFINEMENT BOUNDARY

EAL:

EU1.1	**Unusual Event**
Damage to a loaded cask CONFINEMENT BOUNDARY as indicated by measured dose rates > then **ANY** of the following: • 400 mRem/hr at 3 feet from the HSM surface • 100 mRem/hr outside HSM door on centerline • 20 mRem/hr end shield wall exterior	

Mode Applicability:

All

Basis:

Plant-Specific

The NMP site ISFSI utilizes the NUHOMS Horizontal Modular Storage System.

This EAL addresses any condition which indicates a loss of a cask CONFINEMENT BOUNDARY and thus a potential degradation in the level of safety of the ISFSI. The cask CONFINEMENT BOUNDARY is the NUHOMS 61BT Dry Shielded Canister (DSC). The DSC is the pressure-retaining component of the storage system (ref. 1). Each loaded DSC is housed within a Horizontal Storage Module (HSM). Indication of a loss of CONFINEMENT BOUNDARY is any increase in external HSM radiation levels in excess of Technical Specification limits (ref. 2).

Generic

An UE in this EAL is categorized on the basis of the occurrence of an event of sufficient magnitude that a loaded cask CONFINEMENT BOUNDARY is damaged or violated. This includes classification based on a loaded fuel storage cask CONFINEMENT BOUNDARY loss leading to the degradation of the fuel during storage or posing an operational safety problem with respect to its removal from storage.

NMP1 Basis Reference(s):

1. CDP No. N1-07-092/N2-07-070 Nine Mile Point Nuclear Station - Conceptual Design, Independent Spent Fuel Storage Installation

2. Transnuclear, Inc. Standardized NUHOMS Horizontal Modular Storage System Certificate of Compliance No. 1004, Attachment A Technical Specifications Section 1.2.7 HSM Dose Rates with a Loaded 24P, 52B or 61BT DSC
3. NEI 99-01 IC E-HU1

Category C – Cold Shutdown / Refueling System Malfunction

EAL Group: Cold Conditions (RCS temperature ≤ 212°F); EALs in this category are applicable only in one or more cold operating modes.

Category C EALs are directly associated with cold shutdown or refueling system safety functions. Given the variability of plant configurations (e.g., systems out-of-service for maintenance, containment open, reduced AC power redundancy, time since shutdown) during these periods, the consequences of any given initiating event can vary greatly. For example, a loss of decay heat removal capability that occurs at the end of an extended outage has less significance than a similar loss occurring during the first week after shutdown. Compounding these events is the likelihood that instrumentation necessary for assessment may also be inoperable. The cold shutdown and refueling system malfunction EALs are based on performance capability to the extent possible with consideration given to RCS integrity, CONTAINMENT CLOSURE, and fuel clad integrity for the applicable operating modes (3 - Cold Shutdown, 4 - Refuel, D – Defueled).

The events of this category pertain to the following subcategories:

1. Loss of AC Power

Loss of emergency plant electrical power can compromise plant safety system operability including decay heat removal and emergency core cooling systems which may be necessary to ensure fission product barrier integrity. This category includes loss of onsite and offsite power sources for the 4.16 kV emergency buses.

2. Loss of DC Power

Loss of emergency plant electrical power can compromise plant safety system operability including decay heat removal and emergency core cooling systems which may be necessary to ensure fission product barrier integrity. This category includes loss of power to the 125 VDC buses.

3. RPV Water Level

RPV water level is a measure of inventory available to ensure adequate core cooling and, therefore, maintain fuel clad integrity. The RPV provides a volume for the coolant that covers the reactor core. The RPV and associated pressure piping (reactor coolant system) together provide a barrier to limit the release of radioactive material should the reactor fuel clad integrity fail.

4. RCS Temperature

Uncontrolled or inadvertent temperature or pressure increases are indicative of a potential loss of safety functions.

5. Inadvertent Criticality

Inadvertent criticalities pose potential personnel safety hazards as well being indicative of losses of reactivity control.

6. Communications

Certain events that degrade plant operator ability to effectively communicate with essential personnel within or external to the plant warrant emergency classification.

Category: C – Cold Shutdown / Refueling System Malfunction

Subcategory: 1 – Loss of AC Power

Initiating Condition: Loss of **all** offsite and **all** onsite AC power to 4.16 kV emergency buses for ≥ 15 min.

EAL:

CA1.1	**Alert**
Loss of **all** offsite and **all** onsite AC power, Table C-1, to 4.16 kV emergency buses for ≥ 15 min. (Note 4)	

Note 4: The ED should **not** wait until the applicable time has elapsed, but should declare the event as soon as it is determined that the condition has exceeded, or will likely exceed, the applicable time.

Table C-1 AC Power Sources	
Onsite	• DG 102 • DG 103
Offsite	• T-101N • T-101S • T-10 backfed from offsite through T-1 or T-2 (**only** if already aligned)

Mode Applicability:

3 - Cold Shutdown, 4 - Refuel, D - Defueled

Basis:

Plant-Specific

NMP1 4.16 kV emergency buses are buses PB102 and PB103, which feed all Station redundant safety-related loads.. There are three offsite power sources available to these buses (ref. 1, 2):

- Offsite 115 kV through transformer 101N. This is the normal power supply to PB102.
- Offsite 115 kV through transformer 101S. This is the normal power supply to PB103.

- Offsite 345 kV through transformer T-1 or T-2 backfed through transformer T-10.

Based on operational experience, if the backfeed is not already aligned, this cannot be considered available/capable of supplying the bus due to the time it will take to align it. In any case, if this cannot be accomplished within 15 minutes, it is not available and an Unusual Event must be declared.

There are two onsite AC power sources:

- DG 102 for PB102
- DG 103 for PB 103

The fifteen-minute interval was selected as a threshold to exclude transient power losses. If multiple sources fail to energize the unit safety-related buses within 15 minutes, an Unusual Event is declared under this EAL. The subsequent loss of the single remaining power source escalates the event to an Alert under EAL CA1.1.

This EAL is the cold condition equivalent of the hot condition loss of all AC power EAL SS1.1.

<u>Generic</u>

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.

The event can be classified as an Alert when in cold shutdown, refuel, or defueled mode because of the significantly reduced decay heat and 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 by EALs in Category R.

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

NMP1 Basis Reference(s):

1. N1-OP-30 4.16 kV, 600V, and 480V House Service
2. USAR section IX Electrical Systems
3. NEI 99-01 IC CA3

Category: C – Cold Shutdown / Refueling System Malfunction

Subcategory: 1 – Loss of AC Power

Initiating Condition: AC power capability to 4.16 kV emergency buses reduced to a single power source for ≥ 15 min. such that **ANY** additional single failure would result in a complete loss of all 4.16 kV emergency bus power

EAL:

CU1.1	**Unusual Event**

AC power capability to 4.16 kV emergency buses reduced to a single power source, Table C-1, for ≥ 15 min. (Note 4)

AND

ANY additional single power source failure will result in a loss of **all** 4.16 kV emergency bus power

Note 4: The ED should **not** wait until the applicable time has elapsed, but should declare the event as soon as it is determined that the condition has exceeded, or will likely exceed, the applicable time.

Table C-1 AC Power Sources	
Onsite	• DG 102 • DG 103
Offsite	• T-101N • T-101S • T-10 backfed from offsite through T-1 or T-2 (**only** if already aligned)

Mode Applicability:

3 - Cold Shutdown, 4 - Refuel, D - Defueled

Basis:

<u>Plant-Specific</u>

NMP1 4.16 kV emergency buses are buses PB102 and PB103, which feed all Station redundant safety-related loads.. There are three offsite power sources available to these buses (ref. 1, 2):

- Offsite 115 kV through transformer 101N. This is the normal power supply to PB102.
- Offsite 115 kV through transformer 101S. This is the normal power supply to PB103.
- Offsite 345 kV through transformer T-1 or T-2 backfed through transformer T-10.

Based on operational experience, if the backfeed is not already aligned, this cannot be considered available/capable of supplying the bus due to the time it will take to align it. In any case, if this cannot be accomplished within 15 minutes, it is not available and an Unusual Event must be declared.

There are two onsite AC power sources:

- DG 102 for PB102
- DG 103 for PB 103

The fifteen-minute interval was selected as a threshold to exclude transient power losses. If multiple sources fail to energize the unit safety-related buses within 15 minutes, an Unusual Event is declared under this EAL. The subsequent loss of the single remaining power source escalates the event to an Alert under EAL CA1.1.

<u>Generic</u>

The condition indicated by this EAL is the degradation of the off-site and on-site AC power systems such that any additional single failure would result in a complete loss of 4.16 kV emergency bus AC power. This condition could occur due to a loss of off-site power with a concurrent failure of one emergency generator to supply power to its emergency bus. The subsequent loss of this single power source would escalate the event to an Alert in accordance with EAL CA1.1.

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

NMP1 Basis Reference(s):

1. N1-OP-30 4.16 kV, 600V, and 480V House Service
2. USAR section IX Electrical Systems
3. NEI 99-01 IC CU3

Category: C – Cold Shutdown / Refueling System Malfunction

Subcategory: 2 – Loss of DC Power

Initiating Condition: Loss of **required** DC power for ≥ 15 min.

EAL:

CU2.1	**Unusual Event**
< 106 VDC on **required** 125 VDC buses (Battery board 11, Battery board 12) for ≥ 15 min. (Note 4)	

Note 4: The ED should **not** wait until the applicable time has elapsed, but should declare the event as soon as it is determined that the condition has exceeded, or will likely exceed, the applicable time.

Mode Applicability:

3 - Cold Shutdown, 4 - Refuel

Basis:

Plant-Specific

A Safety Related (SR) system and a Quality Related (QR) system comprise the 125 VDC Power System. The two SR 125 VDC systems (Battery board 11 and Battery board 12) each consist of: one battery, two Static Chargers in parallel, and a DC distribution board. The one QR 125 VDC system consists of: a battery, one Static Charger, and one battery board. This EAL addresses only the Safety Related battery boards (ref. 1, 2).

106 VDC is the minimum voltage for battery operability (ref.3).

This EAL is the cold condition equivalent of the hot condition loss of DC power EAL SS2.1.

Generic

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.

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

NMP1 Basis Reference(s):

1. N1-OP-47A 125 VDC Power System
2. USAR section IX Electrical Systems
3. NMP1 Technical Specification 3.6.3
4. NEI 99-01 IC CU7

Category: C – Cold Shutdown / Refueling System Malfunction

Subcategory: 3 – RPV Water Level

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

EAL:

CG3.1	**General Emergency**
RPV water level < -84 in. for ≥ 30 min. (Note 4)	
AND	
ANY Containment Challenge Indication, Table C-3	

Note 4: The ED should **not** wait until the applicable time has elapsed, but should declare the event as soon as it is determined that the condition has exceeded, or will likely exceed, the applicable time.

Table C-3 Containment Challenge Indications
• CONTAINMENT CLOSURE **not** established • Explosive mixture exists inside Primary Containment (H_2 ≥ 6% and O_2 ≥ 5%) • UNPLANNED rise in Primary Containment pressure • RB area radiation > 8 R/hr

Mode Applicability:

3 - Cold Shutdown, 4 - Refuel

Basis:

<u>Plant-Specific</u>

When RPV water level drops the top of active fuel, an indicated RPV water level of -84 in. (rounded from 84 7/16"), core uncovery starts to occur (ref. 1, 2).

Four conditions are associated with a challenge to Primary Containment integrity:

- CONTAINMENT CLOSURE is the procedurally defined actions taken to secure containment (primary or secondary) and its associated structures, systems, and components as a functional barrier to fission product release under existing plant conditions. This definition is less restrictive than Technical Specification criteria governing Primary and Secondary Containment integrity. If the Technical

Specification criteria are met, therefore, CONTAINMENT CLOSURE has been established. (ref. 3, 4)

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

  The specified values for this threshold are the minimum global deflagration concentration limits (6% hydrogen and 5% oxygen), and readily recognizable because 6% hydrogen is well above the EOP flowchart entry condition. The minimum global deflagration hydrogen/oxygen concentrations (6%/5%, respectively) require intentional Primary Containment venting, which is defined to be a loss of the Primary Containment barrier. (ref. 2, 5)

  If the hydrogen or oxygen monitor is unavailable, sampling and analysis may determine gas concentrations. The validity of sample results must be judged based upon plant conditions, since drawing and analyzing samples may take some time. If sample results cannot be relied upon and hydrogen concentrations cannot be determined by any other means, the concentrations must be considered "unknown." The monitors should not be considered "unavailable" until an attempt has been made to place them in service. (ref. 2)

- Any UNPLANNED rise in Primary Containment pressure in the Cold Shutdown or Refuel mode indicates CONTAINMENT CLOSURE cannot be assured and the Primary Containment cannot be relied upon as a barrier to fission product release.

- RB (Reactor Building) area radiation monitors should provide indication of increased release that may be indicative of a challenge to CONTAINMENT CLOSURE. The EOP Maximum Safe Operating level is 8 R/hr and is indicative of problems in the secondary containment that are spreading. The locations into which the primary system discharge is of concern correspond to the areas addressed in Detail S of N1-EOP-5 (ref. 6).

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

Generic

This EAL represents the inability to restore and maintain RPV water level to above the top of active fuel with containment challenged. Fuel damage is probable if RPV water level cannot be restored, as available decay heat will cause boiling, further reducing the RPV water level. With the Containment breached or challenged then the potential for unmonitored fission product release to the environment is high. This represents a direct path for radioactive inventory to be released to the environment. This is consistent with the definition of a GE. The GE is declared on the occurrence of the loss or IMMINENT loss of function of all three barriers.

A number of variables can have a significant impact on heat removal capability challenging the fuel clad barrier. Examples include: initial vessel level and shutdown heat removal system design.

Analysis indicates that core damage may occur within an hour following continued core uncovery therefore, 30 minutes was conservatively chosen.

If CONTAINMENT CLOSURE is re-established prior to exceeding the 30 minute core uncovery time limit then escalation to General Emergency would not occur.

NMP1 Basis Reference(s):

1. N2-EOP-2 RPV Control
2. NER-1M-095, NMP1 Emergency Operating Procedures (EOP) Basis Document
3. NIP-OUT-01 Shutdown Safety
4. NMP1 Technical Specifications Definitions 1.11 and 1.12
5. N1-EOP-4.2 Hydrogen Control
6. N1-EOP-5 Secondary Containment Control
7. NEI 99-01 IC CG1

Category: C – Cold Shutdown / Refueling System Malfunction

Subcategory: 3 – RPV Water Level

Initiating Condition: Loss of Reactor Vessel inventory affecting fuel clad integrity with Containment challenged

EAL:

CG3.2 General Emergency

RPV water level **cannot** be monitored with core uncovery indicated by **ANY** of the following for ≥ 30 min. (Note 4):

- **ANY** UNPLANNED RPV leakage indication, Table C-2
- Erratic Source Range Monitor indication

AND

ANY Containment Challenge Indication, Table C-3

Note 4: The ED should **not** wait until the applicable time has elapsed, but should declare the event as soon as it is determined that the condition has exceeded, or will likely exceed, the applicable time

Table C-2 RPV Leakage Indications
• Drywell equipment drain tank level rise
• Drywell floor drain tank level rise
• Reactor building equipment sump level rise
• Reactor Building floor drain sump level rise
• Torus water level rise
• UNPLANNED rise in RPV make-up rate
• Observation of UNISOLABLE RCS leakage

Table C-3 Containment Challenge Indications
• CONTAINMENT CLOSURE **not** established
• Explosive mixture exists inside Primary Containment (H_2 ≥ 6% and O_2 ≥ 5%)
• UNPLANNED rise in Primary Containment pressure
• RB area radiation > 8 R/hr

Mode Applicability:

3 - Cold Shutdown, 4 - Refuel

Basis:

Plant-Specific

If RPV water level monitoring capability is unavailable, all RPV water level indication would be unavailable and, the RPV inventory loss must be detected by Table C-2, RPV Leakage Indications. Level increases must be evaluated against other potential sources of leakage such as cooling water sources inside the drywell to ensure they are indicative of RPV leakage. Drywell equipment and floor drain tank level rise is the normal method of monitoring and calculating leakage from the RPV. A Reactor Building equipment or floor drain sump level rise may also be indicative of RPV inventory losses external to the Primary Containment from systems connected to the RPV. A rise in torus water level could be indicative of valve misalignment or leakage. If the make-up rate to the RPV unexplainably rises above the pre-established rate, a loss of RPV inventory may be occurring even if the source of the leakage cannot be immediately identified. Visual observation of leakage from systems connected to the RCS in areas outside the Primary Containment that cannot be isolated could be indicative of a loss of RPV inventory. (ref. 1through 6)

Four channels of log count rate meters are available in the Control Room to detect erratic source range monitor indications (ref. 7): SRM Channels 11, 12, 13, and 14 located on the E console.

Post-TMI studies indicated that the installed nuclear instrumentation will operate erratically when the core is uncovered and that source range monitors can be used as a tool for making such determinations. Figure C-2 shows the response of the source range monitor during the first few hours of the TMI-2 accident. The instrument reported an increasing signal about 30 minutes into the accident. At this time, the reactor coolant pumps were running and the core was adequately cooled as indicated by the core outlet thermocouples. Hence, the increasing signal was the result of an increasing two-phase void fraction in the reactor core and vessel downcomer and the reduced shielding that the two-phase mixture provides to the source range monitor.

Four conditions are associated with a challenge to Primary Containment integrity:

- CONTAINMENT CLOSURE is the procedurally defined actions taken to secure containment (primary or secondary) and its associated structures, systems, and components as a functional barrier to fission product release under existing plant conditions. This definition is less restrictive than Technical Specification criteria governing Primary and Secondary Containment integrity. If the Technical Specification criteria are met, therefore, CONTAINMENT CLOSURE has been established. (ref. 10, 11)
- Explosive (deflagration) mixtures in the Primary Containment are assumed to be elevated concentrations of hydrogen and oxygen. BWR industry evaluation of hydrogen generation for development of EOPs/SAGs indicates that any hydrogen concentration above minimum detectable is not to be expected within the short term. Post-LOCA hydrogen generation primarily caused by radiolysis is a slowly evolving, long-term condition. Hydrogen concentrations that rapidly develop are most likely caused by metal-water reaction. A metal-water reaction is indicative of an accident more severe than accidents considered in the plant design basis and would be indicative, therefore, of a potential threat to Primary Containment integrity. Hydrogen concentration of approximately 6% is considered the global deflagration concentration limit.

  The specified values for this threshold are the minimum global deflagration concentration limits (6% hydrogen and 5% oxygen), and readily recognizable because 6% hydrogen is well above the EOP flowchart entry condition. The minimum global deflagration hydrogen/oxygen concentrations (6%/5%, respectively) require intentional Primary Containment venting, which is defined to be a loss of the Primary Containment barrier. (ref. 9, 12)

  If the hydrogen or oxygen monitor is unavailable, sampling and analysis may determine gas concentrations. The validity of sample results must be judged based upon plant conditions, since drawing and analyzing samples may take some time. If sample results cannot be relied upon and hydrogen concentrations cannot be determined by any other means, the concentrations must be considered "unknown."

The monitors should not be considered "unavailable" until an attempt has been made to place them in service. (ref. 9)

- Any UNPLANNED rise in Primary Containment pressure in the Cold Shutdown or Refuel mode indicates CONTAINMENT CLOSURE cannot be assured and the Primary Containment cannot be relied upon as a barrier to fission product release.
- RB (Reactor Building) area radiation monitors should provide indication of increased release that may be indicative of a challenge to CONTAINMENT CLOSURE. The EOP Maximum Safe Operating level is 8 R/hr and is indicative of problems in the secondary containment that are spreading. The locations into which the primary system discharge is of concern correspond to the areas addressed in Detail S of N1-EOP-5 (ref. 13).

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

Generic

This EAL represents the inability to restore and maintain RPV water level to above the top of active fuel with containment challenged. Fuel damage is probable if RPV water level cannot be restored, as available decay heat will cause boiling, further reducing the RPV water level. With the Containment breached or challenged then the potential for unmonitored fission product release to the environment is high. This represents a direct path for radioactive inventory to be released to the environment. This is consistent with the definition of a GE. The GE is declared on the occurrence of the loss or IMMINENT loss of function of all three barriers.

A number of variables can have a significant impact on heat removal capability challenging the fuel clad barrier. Examples include: initial RPV water level and shutdown heat removal system design.

Analysis indicates that core damage may occur within an hour following continued core uncovery therefore, 30 minutes was conservatively chosen.

If CONTAINMENT CLOSURE is re-established prior to exceeding the 30 minute core uncovery time limit then escalation to General Emergency would not occur.

Sump and tank level increases must be evaluated against other potential sources of leakage such as cooling water sources inside the containment to ensure they are indicative of RCS leakage.

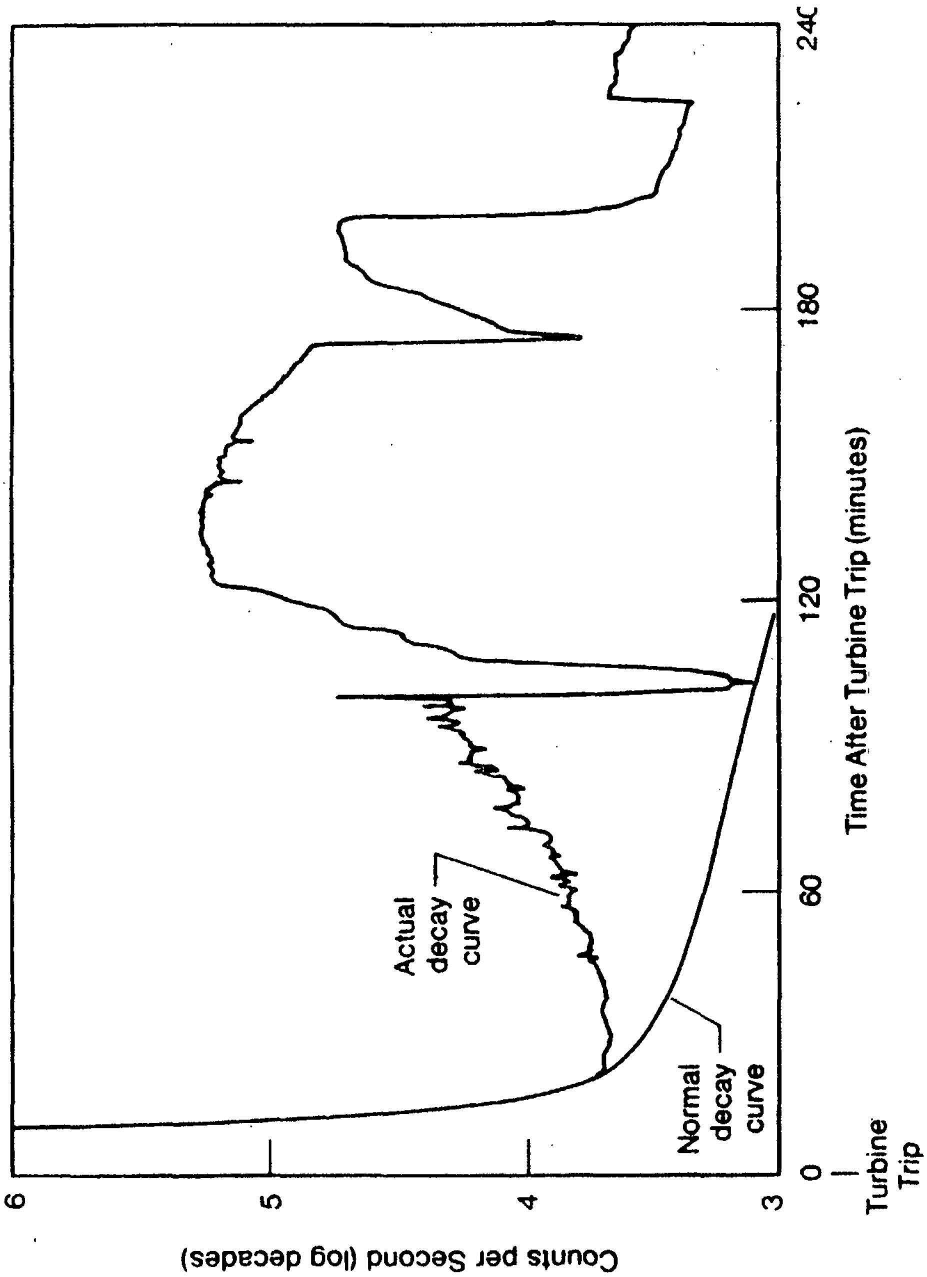
As water level in the RPV lowers, the dose rate above the core will increase. The dose rate due to this core shine should result in site specific monitor indication and possible alarm.

Post-TMI studies indicated that the installed nuclear instrumentation will operate erratically when the core is uncovered and that this should be used as a tool for making such determinations.

NMP1 Basis Reference(s):

1. S-ODP-OPS-0110 Containment Leakage Evaluation
2. USAR 1.4 Primary Coolant Leakage
3. Annunciator H2-1-1 DRYWELL FLOOR DRAIN LEVEL-HIGH
4. Annunciator H2-4-7 DRYWELL WATER LEAK DETECTION SYS
5. Annunciator H2-2-1 R BLDG FL DR SUMPS 11-16 AREA WTR LVL LEVEL HIGH
6. Annunciator H2-2-2 R BUILDING EQUIP DRAIN LEVEL-HIGH
7. N1-OP-38A Source Range Monitor
8. N1-EOP-2 RPV Control
9. NER-1M-095, NMP1 Emergency Operating Procedures (EOP) Basis Document
10. NIP-OUT-01 Shutdown Safety
11. NMP1 Technical Specifications Definitions 1.11 and 1.12
12. N1-EOP-4.2 Hydrogen Control
13. N1-EOP-5 Secondary Containment Control
14. NEI 99-01 IC CG1

Figure C-2: Response of the TMI-2 Source Range Measurement During the First Six Hours of the Accident



Category: C – Cold Shutdown / Refueling System Malfunction

Subcategory: 3 – RPV Water Level

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

EAL:

CS3.1	**Site Area Emergency**
With CONTAINMENT CLOSURE **not** established, RPV water level < -1 in.	

Mode Applicability:

3 - Cold Shutdown, 4 - Refuel

Basis:

Plant-Specific

When RPV water level decreases to -1 in., water level is six inches below the Core Spray initiation setpoint (ref. 1).

The inability to restore and maintain level after reaching this setpoint infers a failure of the RCS barrier and Potential Loss of the Fuel Clad barrier.

CONTAINMENT CLOSURE is the procedurally defined actions taken to secure containment (primary or secondary) and its associated structures, systems, and components as a functional barrier to fission product release under existing plant conditions. This definition is less restrictive than Technical Specification criteria governing Primary and Secondary Containment integrity. If the Technical Specification criteria are met, therefore, CONTAINMENT CLOSURE has been established. (ref. 3)

Generic

Under the conditions specified by this EAL, continued decrease in /RPV water level is indicative of a loss of inventory control. Inventory loss may be due to an RCS breach, pressure boundary leakage, or continued boiling in the RPV. Thus, declaration of a Site Area Emergency is warranted.

Escalation to a General Emergency is via EAL CG3.1, EAL CG3.2, RG1.2 or RG1.3.

NMP1 Basis Reference(s):

1. NMP1 Technical Specification 3.6.2 Table 3.6.2.d
2. NIP-OUT-01 Shutdown Safety
3. NMP1 Technical Specification Definitions 1.11 and 1.12
4. NEI 99-01 IC CS1

Category: C – Cold Shutdown / Refueling System Malfunction

Subcategory: 3 – RPV Water Level

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

EAL:

CS3.2	**Site Area Emergency**
With CONTAINMENT CLOSURE established, RPV water level < -84 in.	

Mode Applicability:

3 - Cold Shutdown, 4 - Refuel

Basis:

Plant-Specific

When RPV water level drops the top of active fuel, an indicated RPV water level of -84 in. (rounded from 84 7/16"), core uncovery starts to occur (ref. 1, 2).

CONTAINMENT CLOSURE is the procedurally defined actions taken to secure containment (primary or secondary) and its associated structures, systems, and components as a functional barrier to fission product release under existing plant conditions. This definition is less restrictive than Technical Specification criteria governing Primary and Secondary Containment integrity. If the Technical Specification criteria are met, therefore, CONTAINMENT CLOSURE has been established. (ref. 3, 4)

Generic

Under the conditions specified by this EAL, continued decrease in RCS level is indicative of a loss of inventory control. Inventory loss may be due to an RCS breach, pressure boundary leakage, or continued boiling in the RPV. Thus, declaration of a Site Area Emergency is warranted.

Escalation to a General Emergency is via EAL CG3.1, EAL CG3.2, RG1.2 or RG1.3.

NMP1 Basis Reference(s):

1. N1-EOP-2 RPV Control
2. NER-1M-095, NMP1 Emergency Operating Procedures (EOP) Basis Document
3. NIP-OUT-01 Shutdown Safety
4. NMP1 Technical Specification Definitions 1.11 and 1.12
5. NEI 99-01 IC CS1

Category: C – Cold Shutdown / Refueling System Malfunction

Subcategory: 3 – RPV Water Level

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

EAL:

CS3.3 Site Area Emergency

RPV water level **cannot** be monitored for ≥ 30 min. (Note 4) with a loss of RPV inventory as indicated by **ANY** of the following:

- **ANY** UNPLANNED RPV leakage indication, Table C-2
- Erratic Source Range Monitor indication

Note 4: The ED should **not** wait until the applicable time has elapsed, but should declare the event as soon as it is determined that the condition has exceeded, or will likely exceed, the applicable time.

Table C-2 RPV Leakage Indications
• Drywell equipment drain tank level rise
• Drywell floor drain tank level rise
• Reactor building equipment sump level rise
• Reactor Building floor drain sump level rise
• Torus water level rise
• UNPLANNED rise in RPV make-up rate
• Observation of UNISOLABLE RCS leakage

Mode Applicability:

3 - Cold Shutdown, 4 - Refuel

Basis:

Plant-Specific

If RPV water level monitoring capability is unavailable, all RPV water level indication would be unavailable and, the RPV inventory loss must be detected by Table C-2, RPV Leakage Indications. Level increases must be evaluated against other potential sources of leakage such as cooling water sources inside the drywell to ensure they are indicative of RPV leakage. Drywell equipment and floor drain tank level rise is the normal method of monitoring and calculating leakage from the RPV. A Reactor Building equipment or floor drain sump level rise may also be indicative of RPV inventory losses external to the Primary Containment from systems connected to the RPV. A rise in torus water level could be indicative of valve misalignment or leakage. If the make-up rate to the RPV unexplainably rises above the pre-established rate, a loss of RPV inventory may be occurring even if the source of the leakage cannot be immediately identified. Visual observation of leakage from systems connected to the RCS in areas outside the Primary Containment that cannot be isolated could be indicative of a loss of RPV inventory. (ref. 1through 6)

Four channels of log count rate meters are available in the Control Room to detect erratic source range monitor indications (ref. 7): SRM Channels 11, 12, 13, and 14 located on the E console.

Post-TMI studies indicated that the installed nuclear instrumentation will operate erratically when the core is uncovered and that source range monitors can be used as a tool for making such determinations. Figure C-2 shows the response of the source range monitor during the first few hours of the TMI-2 accident. The instrument reported an increasing signal about 30 minutes into the accident. At this time, the reactor coolant pumps were running and the core was adequately cooled as indicated by the core outlet thermocouples. Hence, the increasing signal was the result of an increasing two-phase void fraction in the reactor core and vessel downcomer and the reduced shielding that the two-phase mixture provides to the source range monitor.

Generic

Under the conditions specified by this EAL, continued decrease in /RPV water level is indicative of a loss of inventory control. Inventory loss may be due to an RCS breach, pressure boundary leakage, or continued boiling in the RPV. Thus, declaration of a Site Area Emergency is warranted.

Escalation to a General Emergency is via EAL CG3.1, EAL CG3.2, RG1.2 or RG1.3.

The 30-minute duration allows sufficient time for actions to be performed to recover inventory control equipment.

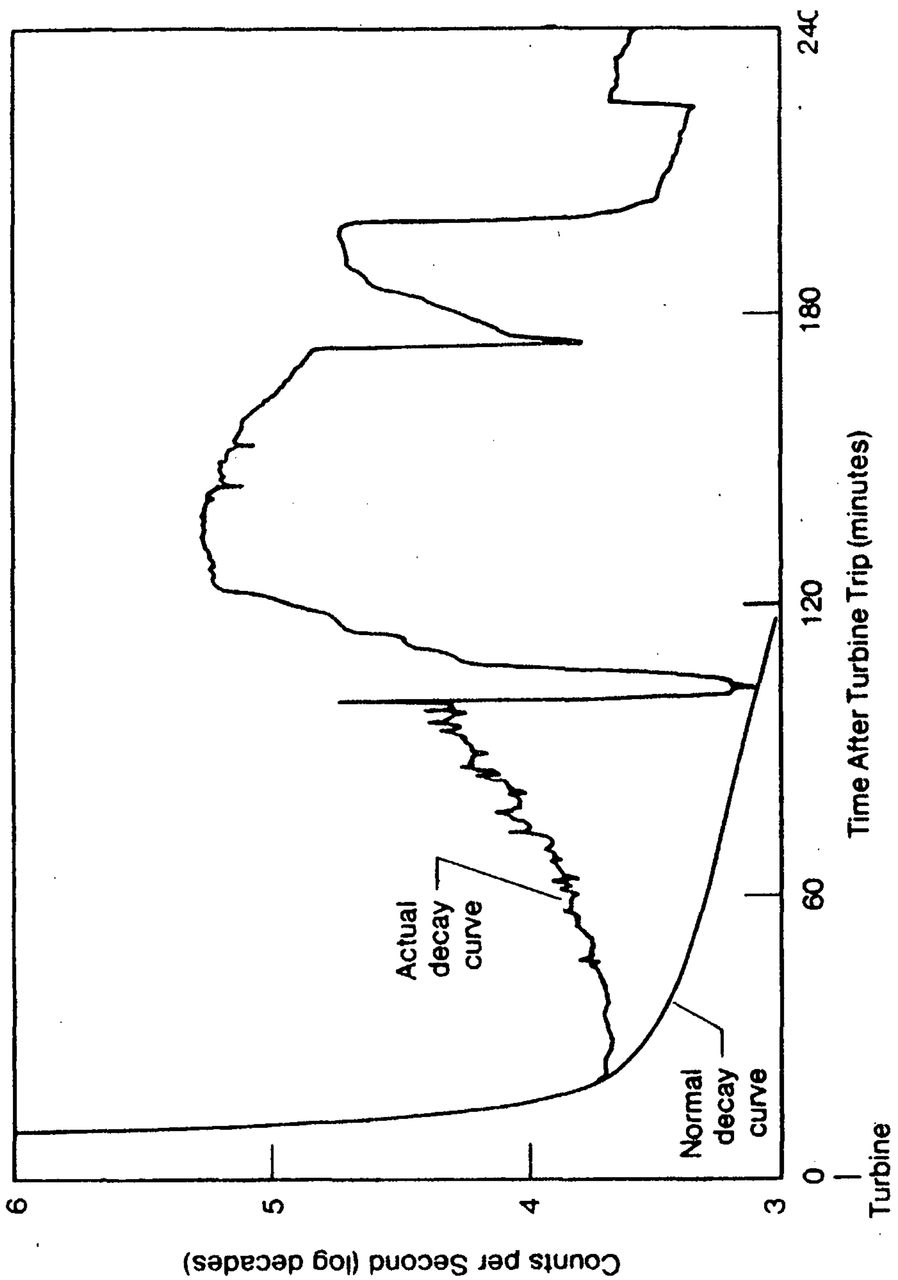
As water level in the RPV lowers, the dose rate above the core will increase. The dose rate due to this core shine should result in site specific monitor indication and possible alarm.

NMP1 Basis Reference(s):

1. S-ODP-OPS-0110 Containment Leakage Evaluation

2. USAR 1.4 Primary Coolant Leakage
3. Annunciator H2-1-1 DRYWELL FLOOR DRAIN LEVEL-HIGH
4. Annunciator H2-4-7 DRYWELL WATER LEAK DETECTION SYS
5. Annunciator H2-2-1 R BLDG FL DR SUMPS 11-16 AREA WTR LVL LEVEL HIGH
6. Annunciator H2-2-2 R BUILDING EQUIP DRAIN LEVEL-HIGH
7. N1-OP-38A Source Range Monitor
8. NEI 99-01 IC CS1

Figure C-2: Response of the TMI-2 Source Range Measurement During the First Six Hours of the Accident



Category: C – Cold Shutdown / Refueling System Malfunction

Subcategory: 3 – RPV Water Level

Initiating Condition: Loss of RPV inventory

EAL:

CA3.1 Alert
RPV water level < +5 in.
OR
RPV water level **cannot** be monitored for ≥ 15 min. with **ANY** UNPLANNED RPV leakage indication, Table C-2 (Note 4)

Note 4: The ED should **not** wait until the applicable time has elapsed, but should declare the event as soon as it is determined that the condition has exceeded, or will likely exceed, the applicable time.

Table C-2 RPV Leakage Indications
• Drywell equipment drain tank level rise
• Drywell floor drain tank level rise
• Reactor building equipment sump level rise
• Reactor Building floor drain sump level rise
• Torus water level rise
• UNPLANNED rise in RPV make-up rate
• Observation of UNISOLABLE RCS leakage

Mode Applicability:

3 - Cold Shutdown, 4 - Refuel

Basis:

Plant-Specific

The threshold RPV water level of +5 in. is the Core Spray initiation setpoint (ref. 1). RPV water level is normally monitored using the instruments in Figure C-1 (ref. 2).

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

In the second condition of this EAL, all RPV water level indication is unavailable and the RPV inventory loss must be detected by the leakage indications listed in Table C-2. Level increases must be evaluated against other potential sources of leakage such as cooling water sources inside the drywell to ensure they are indicative of RPV leakage. Drywell equipment and floor drain tank level rise is the normal method of monitoring and calculating leakage from the RPV (ref. 3 through 8). A Reactor Building equipment or floor drain sump level rise may also be indicative of RPV inventory losses external to the Primary Containment from systems connected to the RPV. A rise in torus water level could be indicative of valve misalignment or leakage. If the make-up rate to the RPV unexplainably rises above the pre-established rate, a loss of RPV inventory may be occurring even if the source of the leakage cannot be immediately identified. Visual observation of leakage from systems connected to the RCS in areas outside the Primary Containment that cannot be isolated could be indicative of a loss of RPV inventory.

The 15-minute interval for the loss of level indication was chosen because it is half of the Site Area Emergency EAL duration. The interval allows this EAL to be an effective precursor to the Site Area Emergency EAL CS3.1. Therefore this EAL meets the definition for an Alert emergency.

Generic

This EAL serves as a precursor to a loss of ability to adequately cool the fuel. The magnitude of this loss of water indicates that makeup systems have not been effective and may not be capable of preventing further RPV water level decrease and potential core uncovery. This condition will result in a minimum emergency classification level of an Alert.

The inability to restore and maintain level after reaching this setpoint would be indicative of a failure of the RCS barrier.

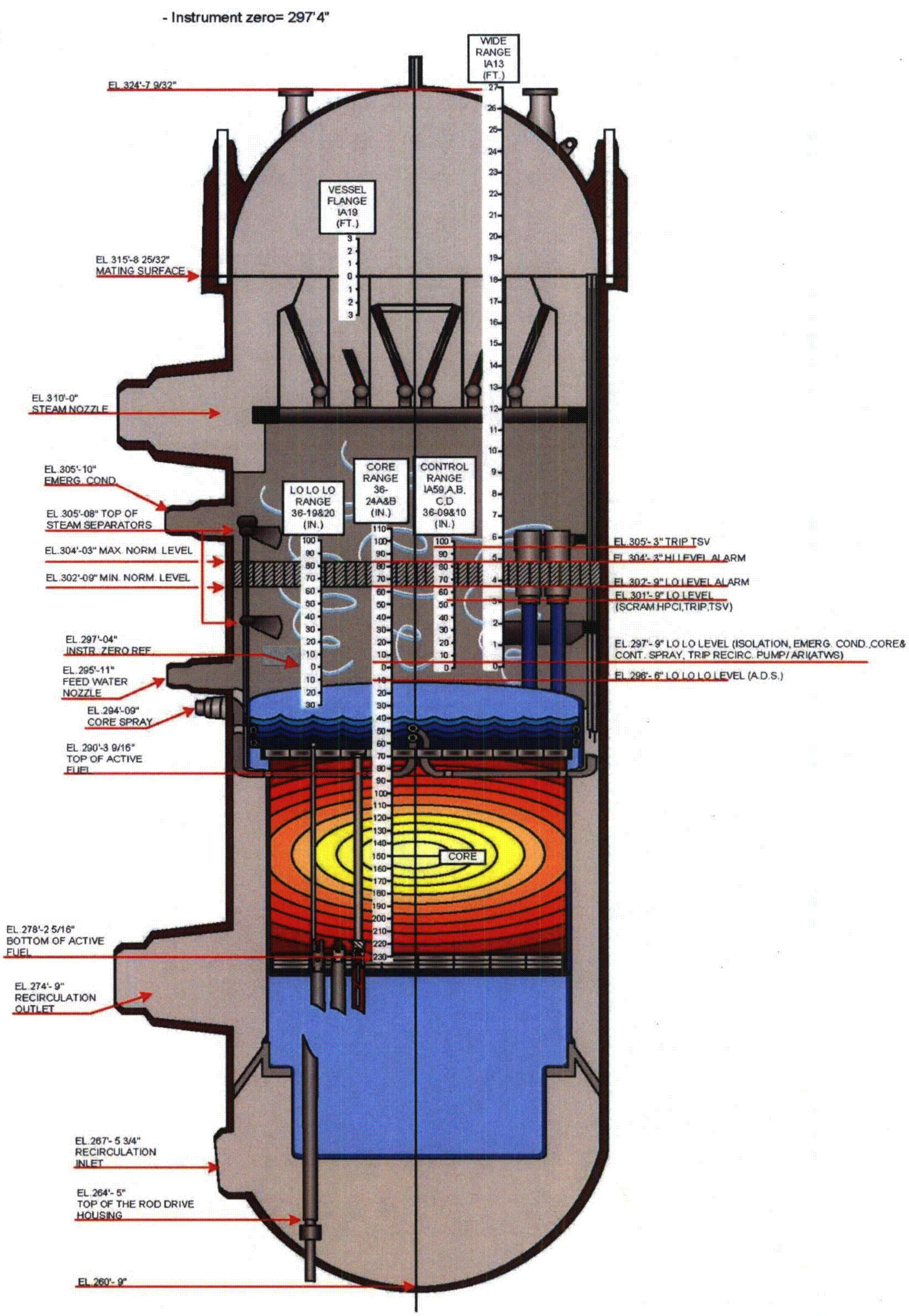
If RPV water level continues to lower then escalation to Site Area Emergency will be via EAL CS3.1, EAL CS3.2 or EAL CS3.3.

NMP1 Basis Reference(s):

1. NMP1 Technical Specification 3.6.2 Table 3.6.2.d
2. P&ID C-18015-C, Reactor Vessel Instrumentation
3. S-ODP-OPS-0110 Containment Leakage Evaluation
4. USAR 1.4 Primary Coolant Leakage
5. Annunciator H2-1-1 DRYWELL FLOOR DRAIN LEVEL-HIGH
6. Annunciator H2-4-7 DRYWELL WATER LEAK DETECTION SYS

7. Annunciator H2-2-1 R BLDG FL DR SUMPS 11-16 AREA WTR LVL LEVEL HIGH
8. Annunciator H2-2-2 R BUILDING EQUIP DRAIN LEVEL-HIGH
9. NEI 99-01 IC CA1

Figure C-1 RPV Water Levels



Category: C – Cold Shutdown / Refueling System Malfunction

Subcategory: 3 – RPV Water Level

Initiating Condition: RCS leakage

EAL:

CU3.1	**Unusual Event**
RCS leakage results in the inability to maintain or restore RPV water level > +53 in. for ≥ 15 min. (Note 4)	

Note 4: The ED should **not** wait until the applicable time has elapsed, but should declare the event as soon as it is determined that the condition has exceeded, or will likely exceed, the applicable time

Mode Applicability:

3 - Cold Shutdown

Basis:

Plant-Specific

Figure C-1 illustrates the elevations of the RPV water level instrument ranges (ref. 1).

+53 in. is the RPV low water level scram setpoint (ref. 5).

RPV water level is monitored from the bottom of active fuel to the top of the RPV head to ensure adequate coverage for expected and postulated conditions of RPV water level. RPV water level measurement is derived by the differential pressure that exists between a reference leg and variable leg. All instruments are referenced to a benchmark at 439 in. above the inside bottom head of the reactor vessel with the exception of the Suppressed or Flange Level range. This benchmark corresponds to 84 in. (rounded from 84 7/16") above the top of active fuel and is the 0 in. reference indication on the RPV water level instruments (except the suppressed or flange level). RPV water level monitoring is subdivided into five ranges identified as:

- Control range provides indication and control signals for normal plant operation and protection system actuation.
- Lo-lo-lo range provides indication and control signals for transient conditions below the normal operating band and emergency equipment actuation.

- Wide range provides indication for transient conditions above normal operating band.
- Suppressed or flange level provides indication relative to the reactor vessel flange.
- Fuel Zone provides indication for long term accident conditions where RPV water level cannot be restored.

In preparation for refueling operations, the Wide and Suppressed RPV water level instruments are modified to provide continuous level indication from within the RPV to the reactor cavity.

Generic

This EAL is considered to be a potential degradation of the level of safety of the plant. The inability to maintain or restore level is indicative of loss of RCS inventory.

Relief valve normal operation should be excluded from this EAL. However, a relief valve that operates and fails to close per design should be considered applicable to this EAL if the relief valve cannot be isolated.

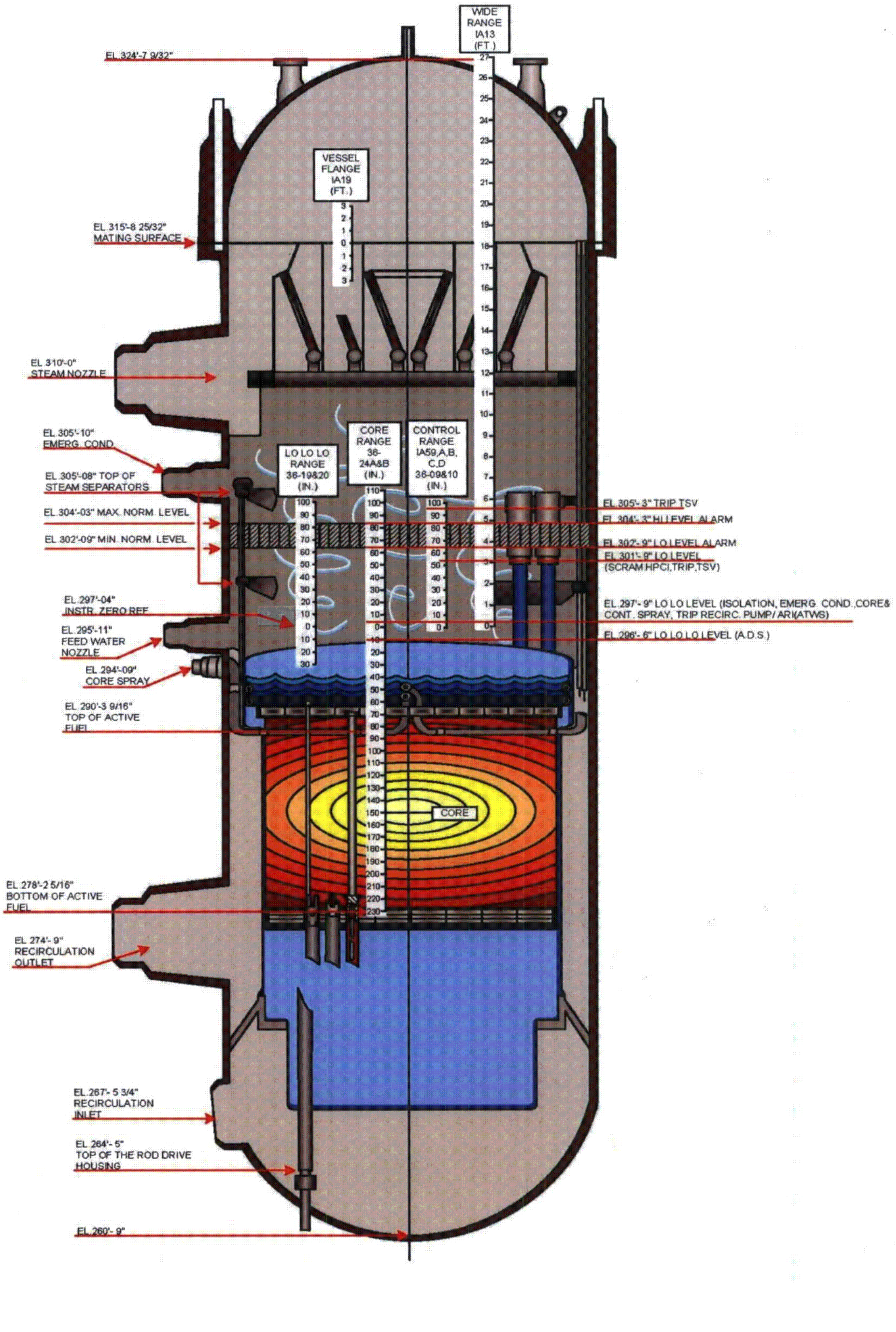
Prolonged loss of RCS inventory may result in escalation to the Alert emergency classification level via either EAL CA2.1 or EAL CA3.1.

NMP1 Basis Reference(s):

1. P&ID C-18015-C, Reactor Vessel Instrumentation
2. N1-OP-58 RPV Level Backfill Injection System
3. USAR Section VIII.A.2.1, Reactor Water Level
4. USAR Section VIII.C.2.1.1, Reactor Water Level
5. NMP1 Technical Specification 3.6.2 Table 3.6.2.a
6. NEI 99-01 IC CU1

Figure C-1 RPV Water Levels

- Instrument zero= 297'4"



Category: C – Cold Shutdown / Refueling System Malfunction

Subcategory: 3 – RPV Water Level

Initiating Condition: RCS leakage

EAL:

CU3.2 Unusual Event

UNPLANNED RPV water level drop below **EITHER** of the following for ≥ 15 min. (Note 4):

- 0 ft Flange Level (RPV flange)
- RPV water level band (when the RPV water level band is established below the RPV flange)

Note 4: The ED should **not** wait until the applicable time has elapsed, but should declare the event as soon as it is determined that the condition has exceeded, or will likely exceed, the applicable time

Mode Applicability:

4 - Refuel

Basis:

Plant-Specific

Figure C-1 illustrates the elevations of the RPV water level instrument ranges (ref. 1).

The RPV flange mating surface is at 0 ft on the Flange Level instrument or EL 315'-8 25/32" (ref. 1).

RPV water level is monitored from the bottom of active fuel to the top of the RPV head to ensure adequate coverage for expected and postulated conditions of RPV water level. RPV water level measurement is derived by the differential pressure that exists between a reference leg and variable leg. All instruments are referenced to a benchmark at 439 in. above the inside bottom head of the reactor vessel with the exception of the Suppressed or Flange Level range. This benchmark corresponds to 84 in. (rounded from 84 7/16") above the top of active fuel and is the 0 in. reference indication on the RPV water level instruments (except the suppressed or flange level). RPV water level monitoring is subdivided into five ranges identified as:

- Control range provides indication and control signals for normal plant operation and protection system actuation.

- Lo-lo-lo range provides indication and control signals for transient conditions below the normal operating band and emergency equipment actuation.
- Wide range provides indication for transient conditions above normal operating band.
- Suppressed or flange level provides indication relative to the reactor vessel flange.
- Fuel Zone provides indication for long term accident conditions where RPV water level cannot be restored.

In preparation for refueling operations, the Wide and Suppressed RPV water level instruments are modified to provide continuous level indication from within the RPV to the reactor cavity.

Generic

This EAL is a precursor of more serious conditions and considered to be a potential degradation of the level of safety of the plant.

Refueling evolutions that decrease RPV water level below the RPV flange are carefully planned and procedurally controlled. An UNPLANNED event that results in water level decreasing below the RPV flange, or below the planned RPV water level for the given evolution (if the planned RPV water level is already below the RPV flange), warrants declaration of a UE due to the reduced RCS inventory that is available to keep the core covered.

The allowance of 15 minutes was chosen because it is reasonable to assume that level can be restored within this time frame using one or more of the redundant means of refill that should be available. If level cannot be restored in this time frame then it may indicate a more serious condition exists.

Continued loss of RCS Inventory will result in escalation to the Alert emergency classification level via either EAL CA2.1 or EAL CA3.1.

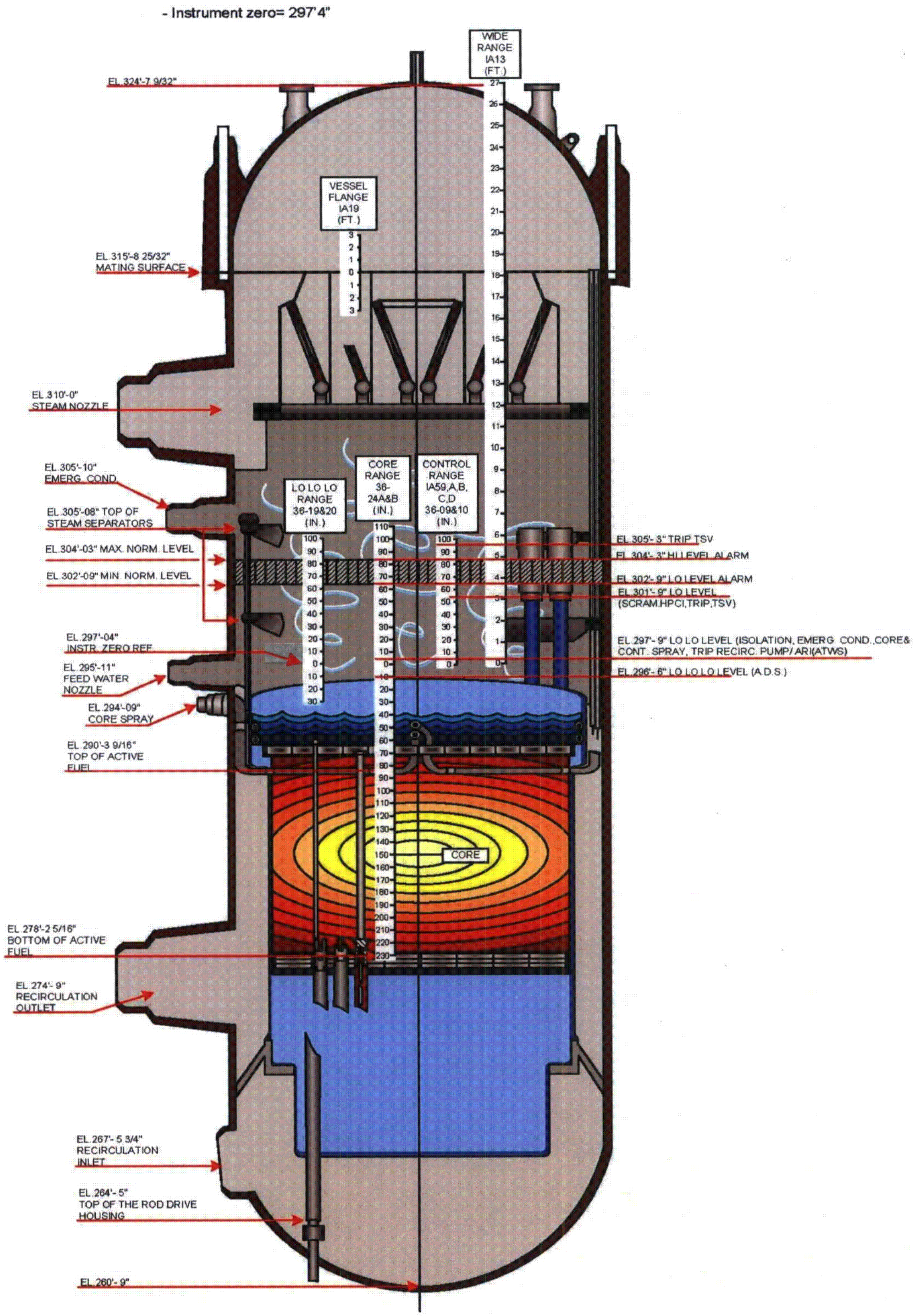
This EAL involves a decrease in RPV water level below the top of the RPV flange that continues for 15 minutes due to an UNPLANNED event. This EAL is not applicable to decreases in flooded reactor cavity level, which is addressed by EAL RU2.1, until such time as the level decreases to the level of the vessel flange.

NMP1 Basis Reference(s):

1. P&ID C-18015-C, Reactor Vessel Instrumentation
2. N1-OP-58 RPV Level Backfill Injection System
3. USAR Section VIII.A.2.1, Reactor Water Level
4. USAR Section VIII.C.2.1.1, Reactor Water Level
5. NMP1 Technical Specification 3.6.2 Table 3.6.2.a

6. NEI 99-01 IC CU2

Figure C-1 RPV Water Levels



Category: C – Cold Shutdown / Refueling System Malfunction

Subcategory: 3 – RPV Water Level

Initiating Condition: RCS leakage

EAL:

CU3.3 **Unusual Event**
RPV water level **cannot** be monitored with a loss of RPV inventory as indicated by **ANY** UNPLANNED RPV leakage indication, Table C-2

Table C-2 RPV Leakage Indications
• Drywell equipment drain tank level rise • Drywell floor drain tank level rise • Reactor building equipment sump level rise • Reactor Building floor drain sump level rise • Torus water level rise • UNPLANNED rise in RPV make-up rate • Observation of UNISOLABLE RCS leakage

Mode Applicability:

4 - Refuel

Basis:

Plant-Specific

In this EAL, all RPV water level indication is unavailable and the RPV inventory loss must be detected by the leakage indications listed in Table C-2, RPV Leakage Indications. Level increases must be evaluated against other potential sources of leakage such as cooling water sources inside the drywell to ensure they are indicative of RPV leakage. Drywell equipment and floor drain tank level rise is the normal method of monitoring and calculating leakage from the RPV (ref. 1, 2). A Reactor Building equipment or floor drain sump level rise may also be indicative of RPV inventory losses external to the Primary Containment from systems connected to the RPV. A rise in torus water level could be indicative of valve misalignment or leakage. If the make-up rate to the RPV unexplainably rises above the pre-established rate, a loss of RPV inventory may be occurring even if the source of the leakage cannot be immediately identified. Visual observation of leakage from systems connected to the RCS in areas outside the Primary Containment that cannot be isolated could be indicative of a loss of RPV inventory.

Generic

This EAL is a precursor of more serious conditions and considered to be a potential degradation of the level of safety of the plant.

Refueling evolutions that decrease RCS water level below the Reactor Vessel flange are carefully planned and procedurally controlled. An UNPLANNED event that results in water level decreasing below the Reactor Vessel flange, or below the planned RCS water level for the given evolution (if the planned RCS water level is already below the Reactor Vessel flange), warrants declaration of a UE due to the reduced RCS inventory that is available to keep the core covered.

Continued loss of RCS Inventory will result in escalation to the Alert emergency classification level via either EAL CA3.1 or EAL CA4.1.

This EAL addresses conditions in the refueling mode when normal means of core temperature indication and RPV water level indication may not be available. Redundant means of RPV water level indication will normally be installed (including the ability to monitor level visually) to assure that the ability to monitor level will not be interrupted. However, if all level indication were to be lost during a loss of RPV inventory event, the operators would need to determine that RPV inventory loss was occurring by observing sump and tank level changes. Sump and tank level increases must be evaluated against other potential sources of leakage such as cooling water sources inside the containment to ensure they are indicative of RCS leakage.

NMP1 Basis Reference(s):

1. S-ODP-OPS-0110 Containment Leakage Evaluation
2. USAR 1.4 Primary Coolant Leakage
3. Annunciator H2-1-1 DRYWELL FLOOR DRAIN LEVEL-HIGH
4. Annunciator H2-4-7 DRYWELL WATER LEAK DETECTION SYS
5. Annunciator H2-2-1 R BLDG FL DR SUMPS 11-16 AREA WTR LVL LEVEL HIGH
6. Annunciator H2-2-2 R BUILDING EQUIP DRAIN LEVEL-HIGH
7. NEI 99-01 IC CU2

Category: C – Cold Shutdown / Refueling System Malfunction

Subcategory: 4 – RCS Temperature

Initiating Condition: Inability to maintain plant in cold shutdown

EAL:

CA4.1 Alert

An UNPLANNED event results in **EITHER**:

RCS temperature > 212°F for > Table C-4 duration

OR

RPV pressure increase > 10 psi due to an UNPLANNED loss of decay heat removal capability

Table C-4 RCS Reheat Duration Thresholds		
RCS Status	CONTAINMENT CLOSURE Status	Duration
INTACT	N/A	60 min.*
Not INTACT	Established	20 min.*
	Not established	0 min.

* If an RCS heat removal system is in operation within this time frame and RCS temperature is being reduced, the EAL is **not** applicable.

Mode Applicability:

3 - Cold Shutdown, 4 – Refuel

Basis:

Plant-Specific

Several instruments are capable of providing indication of RCS temperature with respect to the Technical Specification cold shutdown temperature limit (212°F). These include (ref. 2, 3):

- Recirc operating – Recirc pump temperature points:
  - 11-RRP - A427
  - 12-RRP - A431
  - 13-RRP - A435

- 14-RRP - A439
- 15-RRP - A443

- Shutdown cooling operating – Temperature Recorder 38-146

If Rx Recirc or Shutdown Cooling pumps are not in operation and reactor coolant temperature is greater than or equal to 212°F, RCS temperature can be obtained by converting the RPV pressure to temperature using the saturated steam tables.

The RCS should be considered INTACT when the RCS pressure boundary is in its normal condition for the cold shutdown mode of operation (e.g., no freeze seals or nozzle dams).

CONTAINMENT CLOSURE is the procedurally defined actions taken to secure containment (primary or secondary) and its associated structures, systems, and components as a functional barrier to fission product release under existing plant conditions. This definition is less restrictive than Technical Specification criteria governing Primary and Secondary Containment integrity. If the Technical Specification criteria are met, therefore, CONTAINMENT CLOSURE has been established. (ref. 4, 5)

The pressure rise of greater than 10 psig infers an RCS temperature in excess of the Technical Specification cold shutdown limit (212°F) for which this EAL would otherwise permit up to sixty minutes to restore RCS cooling before declaration of an Alert (RCS INTACT). This EAL therefore covers situations in which it is determined that, due to high decay heat loads, the time provided to reestablish temperature control should be less than sixty minutes (as indicated by significant RCS re-pressurization).

Wide range pressure indication (0-1600 psig) is capable of measuring pressure changes of 10 psig.

If RCS temperature exceeds 212°F, an operating mode change occurs. Although the event may have originated in cold conditions, the emergency classification shall be based on the operating mode that existed at the time the event occurred (prior to any protective system or operator action initiated in response to the condition). For events that occur in Cold Shutdown or Refuel, escalation is via EALs that have Cold Shutdown or Refuel for mode applicability, even if Hot Shutdown (or a higher mode) is entered during any subsequent heat-up. In particular, the fission product barrier EALs are applicable only to events that initiate in Hot Shutdown or higher.

Escalation to a Site Area Emergency would be under EAL CS3.1 should boiling result in significant RPV water level loss leading to core uncovery.

Generic

The RCS Reheat Duration Thresholds table addresses complete loss of functions required for core cooling for greater than 60 minutes during refuel and cold shutdown modes when RCS integrity is established. The 60 minute time frame should allow sufficient time to restore cooling without there being a substantial degradation in plant safety.

The RCS Reheat Duration Thresholds table also addresses the complete loss of functions required for core cooling for greater than 20 minutes during Refuel and cold shutdown modes when CONTAINMENT CLOSURE is established but RCS integrity is not established or RCS inventory is reduced. The allowed 20 minute time frame was included to allow operator action to restore the heat removal function, if possible.

Finally, complete loss of functions required for core cooling during Refuel and cold shutdown modes when neither CONTAINMENT CLOSURE nor RCS integrity are established is addressed. No delay time is allowed because the evaporated reactor coolant that may be released into the Containment during this heatup condition could also be directly released to the environment.

The note (*) indicates that this EAL is not applicable if actions are successful in restoring an RCS heat removal system to operation and RCS temperature is being reduced within the specified time frame.

The 10 psig pressure increase addresses situations where, due to high decay heat loads, the time provided to restore temperature control, should be less than 60 minutes. The RPV pressure setpoint was chosen because it is the lowest pressure that the site can read on installed Control Board instrumentation that is equal to or greater than 10 psig.

Escalation to Site Area Emergency would be via EAL CS3.1 should boiling result in significant RPV water level loss leading to core uncovery.

A loss of Technical Specification components alone is not intended to constitute an Alert. The same is true of a momentary UNPLANNED excursion above the Technical Specification cold shutdown temperature limit when the heat removal function is available.

The Emergency Director must remain alert to events or conditions that lead to the conclusion that exceeding the EAL is IMMINENT. If, in the judgment of the Emergency Director, an IMMINENT situation is at hand, the classification should be made as if the threshold has been exceeded.

NMP1 Basis Reference(s):

1. NMP1 Technical Specifications Definitions 1.1
2. N1-OP-43C Plant Shutdown
3. N1-OP-4 Shutdown Cooling System
4. NIP-OUT-01 Shutdown Safety

5. NMP1 Technical Specifications Definitions 1.11 and 1.12
6. NEI 99-01 IC CA4

Category: C – Cold Shutdown / Refueling System Malfunction

Subcategory: 4 – RCS Temperature

Initiating Condition: UNPLANNED loss of decay heat removal capability

EAL:

CU4.1	**Unusual Event**
UNPLANNED event results in RCS temperature > 212°F	

Mode Applicability:

3 - Cold Shutdown, 4 - Refuel

Basis:

Plant-Specific

Several instruments are capable of providing indication of RCS temperature with respect to the Technical Specification cold shutdown temperature limit (212°F). These include (ref. 2, 3):

- Recirc operating – Recirc pump temperature points:
  - 11-RRP - A427
  - 12-RRP - A431
  - 13-RRP - A435
  - 14-RRP - A439
  - 15-RRP - A443
- Shutdown cooling operating – Temperature Recorder 38-146

If Rx Recirc or Shutdown Cooling pumps are not in operation and reactor coolant temperature is greater than or equal to 212°F, RCS temperature can be obtained by converting the RPV pressure to temperature using the saturated steam tables.

If RCS temperature exceeds 212°F, an operating mode change occurs. Although the event may have originated in cold conditions, the emergency classification shall be based on the operating mode that existed at the time the event occurred (prior to any protective system or operator action initiated in response to the condition). For events that occur in Cold Shutdown or Refuel, escalation is via EALs that have Cold Shutdown or Refuel for mode applicability, even if Hot Shutdown (or a higher mode) is entered during any subsequent heat-up. In particular, the fission product barrier EALs are applicable only to events that initiate in Hot Shutdown or higher.

Generic

This EAL is a precursor of more serious conditions and, as a result, is considered to be a potential degradation of the level of safety of the plant. In cold shutdown the ability to remove decay heat relies primarily on forced cooling flow. Operation of the systems that provide this forced cooling may be jeopardized due to the unlikely loss of electrical power or RCS inventory. Since the RCS usually remains INTACT in the cold shutdown mode a large inventory of water is available to keep the core covered.

During refueling the level in the RPV will normally be maintained above the RPV flange. Refueling evolutions that decrease water level below the RPV flange are carefully planned and procedurally controlled. Loss of forced decay heat removal at reduced inventory may result in more rapid increases in RCS/RPV temperatures depending on the time since shutdown.

Normal means of core temperature indication and RPV water level indication may not be available in the Refuel mode. Redundant means of RPV water level indication are therefore procedurally installed to assure that the ability to monitor level will not be interrupted. Escalation to Alert would be via EAL CA3.1 based on an inventory loss or EAL CA4.1 based on exceeding its temperature duration or pressure criteria.

NMP1 Basis Reference(s):

1. NMP1 Technical Specifications Definitions 1.1
2. N1-OP-43C Plant Shutdown
3. N1-OP-4 Shutdown Cooling System
4. NEI 99-01 IC CU4

Category: C – Cold Shutdown / Refueling System Malfunction

Subcategory: 4 – RCS Temperature

Initiating Condition: UNPLANNED loss of decay heat removal capability

EAL:

CU4.2	**Unusual Event**
Loss of **all** RCS temperature and RPV water level indication for ≥ 15 min. (Note 4)	

Note 4: The ED should **not** wait until the applicable time has elapsed, but should declare the event as soon as it is determined that the condition has exceeded, or will likely exceed, the applicable time

Mode Applicability:

3 - Cold Shutdown, 4 - Refuel

Basis:

Plant-Specific

Several instruments are capable of providing indication of RCS temperature with respect to the Technical Specification cold shutdown temperature limit (212°F). These include (ref. 2, 3):

- Recirc operating – Recirc pump temperature points:
  - 11-RRP - A427
  - 12-RRP - A431
  - 13-RRP - A435
  - 14-RRP - A439
  - 15-RRP - A443
- Shutdown cooling operating – Temperature Recorder 38-146

If Rx Recirc or Shutdown Cooling pumps are not in operation and reactor coolant temperature is greater than or equal to 212°F, RCS temperature can be obtained by converting the RPV pressure to temperature using the saturated steam tables.

RPV water level is monitored from the bottom of active fuel to the top of the RPV head to ensure adequate coverage for expected and postulated conditions of RPV water level. RPV water level measurement is derived by the differential pressure that exists between a reference leg and variable leg. All instruments are referenced to a benchmark at 439 in.

above the inside bottom head of the reactor vessel with the exception of the Suppressed or Flange Level range. This benchmark corresponds to 84 in. (rounded from 84 7/16") above the top of active fuel and is the 0 in. reference indication on the RPV water level instruments (except the suppressed or flange level). RPV water level monitoring is subdivided into five ranges identified as:

- Control range provides indication and control signals for normal plant operation and protection system actuation.
- Lo-lo-lo range provides indication and control signals for transient conditions below the normal operating band and emergency equipment actuation.
- Wide range provides indication for transient conditions above normal operating band.
- Suppressed or flange level provides indication relative to the reactor vessel flange.
- Fuel Zone provides indication for long term accident conditions where RPV water level cannot be restored.

In preparation for refueling operations, the Wide and Suppressed RPV water level instruments are modified to provide continuous level indication from within the RPV to the reactor cavity.

Although the event may have originated in cold conditions, the emergency classification shall be based on the operating mode that existed at the time the event occurred (prior to any protective system or operator action initiated in response to the condition). For events that occur in Cold Shutdown or Refuel, escalation is via EALs that have Cold Shutdown or Refuel for mode applicability, even if Hot Shutdown (or a higher mode) is entered during any subsequent heat-up. In particular, the fission product barrier EALs are applicable only to events that initiate in Hot Shutdown or higher.

<u>Generic</u>

This EAL is a precursor of more serious conditions and, as a result, is considered to be a potential degradation of the level of safety of the plant. In cold shutdown the ability to remove decay heat relies primarily on forced cooling flow. Operation of the systems that provide this forced cooling may be jeopardized due to the unlikely loss of electrical power or RPV inventory. Since the RCS usually remains INTACT in the cold shutdown mode a large inventory of water is available to keep the core covered.

During refueling the level in the RPV will normally be maintained above the RPV flange. Refueling evolutions that decrease water level below the RPV flange are carefully planned and procedurally controlled. Loss of forced decay heat removal at reduced inventory may result in more rapid increases in RPV temperatures depending on the time since shutdown.

Normal means of core temperature indication and RPV water level indication may not be available in the refueling mode. Redundant means of RPV water level indication are therefore procedurally installed to assure that the ability to monitor level will not be interrupted. However, if all level and temperature indication were to be lost in either the cold shutdown of refueling modes, this EAL would result in declaration of a UE if both temperature and level indication cannot be restored within 15 minutes from the loss of both means of indication. Escalation to Alert would be via EAL CA3.1 based on an inventory loss or EAL CA4.1 based on exceeding its temperature criteria.

NMP1 Basis Reference(s):

1. NMP1 Technical Specifications Definitions 1.1
2. N1-OP-43C Plant Shutdown
3. N1-OP-4 Shutdown Cooling System
4. NEI 99-01 IC CU4

Category: C – Cold Shutdown / Refueling System Malfunction

Subcategory: 5 – Inadvertent Criticality

Initiating Condition: Inadvertent criticality

EAL:

CU5.1	**Unusual Event**
An UNPLANNED sustained positive period observed on nuclear instrumentation	

Mode Applicability:

3 - Cold Shutdown, 4 - Refuel

Basis:

Plant-Specific

The term "sustained" is used to allow exclusion of expected short-term positive periods from planned fuel bundle or control rod movements during core alteration. These short-term positive periods are the result of the rise in neutron population due to subcritical multiplication.

Generic

This EAL addresses criticality events that occur in Cold Shutdown or Refueling modes such as fuel mis-loading events and inadvertent dilution events. This EAL indicates a potential degradation of the level of safety of the plant, warranting a UE classification.

Escalation would be by Emergency Director judgment.

NMP1 Basis Reference(s):

1. NEI 99-01 IC CU8

Category: C – Cold Shutdown / Refueling System Malfunction

Subcategory: 6 – Communications

Initiating Condition: Loss of **all** onsite or offsite communications capabilities

EAL:

CU6.1 Unusual Event

Loss of **all** Table C-5 onsite (internal) communication methods affecting the ability to perform routine operations

OR

Loss of **all** Table C-5 offsite (external) communication methods affecting the ability to perform offsite notifications

Table C-5 Communications Systems		
System	**Onsite** (internal)	**Offsite** (external)
PBX (normal dial telephones)	X	X
Gaitronics	X	
Hand-Held Portable Radio (station radio)	X	
Control Room installed satellite phones (non portable)		X
ENS		X
RECS		X
~~UHF radios~~		~~X~~

Mode Applicability:

3 - Cold Shutdown, 4 - Refuel, D - Defueled

Basis:

Plant-Specific

Onsite/offsite communications systems are listed in Table C-2 (ref. 1, 2).

This EAL is the cold condition equivalent of the hot condition EAL SU6.1.

Generic

The purpose of this EAL is to recognize a loss of communications capability that either defeats the plant operations staff ability to perform routine tasks necessary for plant operations or the ability to communicate issues with off-site authorities. The loss of off-site communications ability is expected to be significantly more comprehensive than the condition addressed by 10 CFR 50.72.

The availability of one method of ordinary off-site communications is sufficient to inform federal, state, and local authorities of plant issues. This EAL is intended to be used only when extraordinary means (e.g., relaying of information from radio transmissions, individuals being sent to off-site locations, etc.) are being utilized to make communications possible.

NMP1 Basis Reference(s):

1. USAR 2.4.5 Lighting and Communication
2. Nine Mile Point Nuclear Station Site Emergency Plan, Section 7.2
3. NEI 99-01 IC CU6

Category S – System Malfunction

EAL Group: Hot Conditions (RCS temperature > 212°F); EALs in this category are applicable only in one or more hot operating modes.

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

The events of this category pertain to the following subcategories:

1. Loss of AC Power

Loss of emergency plant electrical power can compromise plant safety system operability including decay heat removal and emergency core cooling systems which may be necessary to ensure fission product barrier integrity. This category includes loss of onsite and offsite power sources for the 4.16 kV safeguard buses.

2. Loss of DC Power

Loss of emergency plant electrical power can compromise plant safety system operability including decay heat removal and emergency core cooling systems which may be necessary to ensure fission product barrier integrity. This category includes loss of power to the 125 VDC buses.

3. Criticality & RPS Failure

Inadvertent criticalities pose potential personnel safety hazards as well being indicative of losses of reactivity control.

Events related to failure of the Reactor Protection System (RPS) to initiate and complete reactor trips. In the plant licensing basis, postulated failures of the RPS to complete a reactor trip comprise a specific set of analyzed events referred to as Anticipated Transient Without Scram (ATWS) events. For EAL classification however, ATWS is intended to mean any trip failure event that does not achieve reactor shutdown. If RPS actuation fails to assure reactor shutdown, positive control of reactivity is at risk and could cause a threat to fuel clad, RCS and Containment integrity.

4. Inability to Reach or Maintain Shutdown Conditions

System malfunctions may lead to failure of the plant to be brought to the required plant operating condition required by technical specifications if a limiting condition for operation (LCO) is not met.

5. Instrumentation

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

6. Communications

Certain events that degrade plant operator ability to effectively communicate with essential personnel within or external to the plant warrant emergency classification.

7. Fuel Clad Degradation

During normal operation, reactor coolant fission product activity is very low. Small concentrations of fission products in the coolant are primarily from the fission of tramp uranium in the fuel clad or minor perforations in the clad itself. Any significant increase from these base-line levels (~5% clad failures) is indicative of fuel failures and is covered under Category F, Fission Product Barrier Degradation. However, lesser amounts of clad damage may result in coolant activity exceeding Technical Specification limits. These fission products will be circulated with the reactor coolant and can be detected by coolant sampling and/or the Letdown radiation monitor.

8. RCS Leakage

The Reactor Vessel provides a volume for the coolant that covers the reactor core. The Reactor Vessel and associated pressure piping (reactor coolant system) together provide a barrier to limit the release of radioactive material should the reactor fuel clad integrity fail.

Excessive RCS leakage greater than Technical Specification limits are utilized to indicate potential pipe cracks that may propagate to an extent threatening fuel clad, RCS and Containment integrity.

Category: S –System Malfunction

Subcategory: 1 – Loss of AC Power

Initiating Condition: Prolonged loss of **all** offsite and **all** onsite AC power to 4.16 kV emergency buses

EAL:

SG1.1 General Emergency

Loss of **all** offsite and **all** onsite AC power, Table S-1, to 4.16 kV emergency buses

AND EITHER:

Restoration of at least one 4.16 kV emergency bus within 4 hours is **not** likely

OR

RPV water level **cannot** be restored and maintained above -84 in. or RPV water level **cannot** be determined

Table S-1 AC Power Sources	
Onsite	• DG 102 • DG 103
Offsite	• T-101N • T-101S • T-10 backfed from offsite through T-1 or T-2 (**only** if already aligned)

Mode Applicability:

1 - Power Operation, 2 - Hot Shutdown

Basis:

Plant-Specific

NMP1 4.16 kV emergency buses are buses PB102 and PB103, which feed all Station redundant safety-related loads.. There are three offsite power sources available to these buses (ref. 1):

- Offsite 115 kV through transformer 101N. This is the normal power supply to PB102.

- Offsite 115 kV through transformer 101S. This is the normal power supply to PB103.
- Offsite 345 kV through transformer T-1 or T-2 backfed through transformer T-10.

Based on operational experience, if the backfeed is not already aligned, this cannot be considered available/capable of supplying the bus due to the time it will take to align it.

There are two onsite AC power sources:

- DG 102 for PB102
- DG 103 for PB 103

Consideration should be given to operable loads necessary to remove decay heat or provide RPV makeup capability when evaluating loss of all AC power to emergency buses. Even though an emergency bus may be energized, if necessary loads (i.e., loads that if lost would inhibit decay heat removal capability or RPV makeup capability) are not operable on the energized bus then the bus should not be considered operable.

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

An RPV water level instrument reading of -84 in. (rounded from 84 7/16") indicates RPV water level is at the top of active fuel. When RPV water level is at or above the top of active fuel, the core is completely submerged. Core submergence is the most desirable means of core cooling. When RPV water level is below the top of active fuel, the uncovered portion of the core must be cooled by less reliable means (i.e., steam cooling or spray cooling). If core uncovery is threatened, the EOPs specify alternate, more extreme, RPV water level control measures in order to restore and maintain adequate core cooling (ref. 3). Since core uncovery begins if RPV water level drops to -84 in., the level is indicative of a challenge to core cooling and the Fuel Clad barrier.

Consistent with the EOP definition of "cannot be restored and maintained," the determination that RPV water level cannot be restored and maintained above the top of active fuel may be made at, before, or after RPV water level actually decreases to this point.

<u>Generic</u>

Loss of all AC power to emergency busses 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 to emergency busses will lead to loss of fuel clad, RCS, and containment, thus warranting declaration of a General Emergency.

This EAL is specified to assure that in the unlikely event of a prolonged loss of all AC power to emergency 4.16 kV buses, timely recognition of the seriousness of the event occurs and that declaration of a General Emergency occurs as early as is appropriate, based on a reasonable assessment of the event trajectory.

The likelihood of restoring at least one emergency bus should be based on a realistic appraïsal of the situation since a delay in an upgrade decision based on only a chance of mitigating the event could result in a loss of valuable time in preparing and implementing public protective actions.

In addition, under these conditions, fission product barrier monitoring capability may be degraded.

NMP1 Basis Reference(s):

1. N1-OP-30 4.16 KV, 600V, and 480V House Service
2. NER-1M-025 SBO Evaluation
3. NER-1M-095-R02 NMP1 EOP and SAP Basis Document
4. NEI 99-01 IC SG1

Category: S – System Malfunction

Subcategory: 1 – Loss of AC Power

Initiating Condition: Loss of **all** offsite and **all** onsite AC power to 4.16 kV emergency buses for ≥ 15 min.

EAL:

SS1.1 **Site Area Emergency**
Loss of **all** offsite and **all** onsite AC power, Table S-1, to 4.16 kV emergency buses for ≥ 15 min. (Note 4)

Note 4: The ED should **not** wait until the applicable time has elapsed, but should declare the event as soon as it is determined that the condition has exceeded, or will likely exceed, the applicable time.

Table S-1 AC Power Sources	
Onsite	• DG 102 • DG 103
Offsite	• T-101N • T-101S • T-10 backfed from offsite through T-1 or T-2 (**only** if already aligned)

Mode Applicability:

1 - Power Operation, 2 - Hot Shutdown

Basis:

Plant-Specific

NMP1 4.16 kV emergency buses are buses PB102 and PB103, which feed all Station redundant safety-related loads. There are three offsite power sources available to these buses (ref. 1, 2):

- Offsite 115 kV through transformer 101N. This is the normal power supply to PB102.
- Offsite 115 kV through transformer 101S. This is the normal power supply to PB103.

- Offsite 345 kV through transformer T-1 or T-2 backfed through transformer T-10.

Based on operational experience, if the backfeed is not already aligned, this cannot be considered available/capable of supplying the bus due to the time it will take to align it. In any case, if this cannot be accomplished within 15 minutes, it is not available and an Unusual Event must be declared.

There are two onsite AC power sources:

- DG 102 for PB102
- DG 103 for PB 103

Consideration should be given to operable loads necessary to remove decay heat or provide RPV makeup capability when evaluating loss of all AC power to vital buses. Even though an essential bus may be energized, if necessary loads (i.e., loads that if lost would inhibit decay heat removal capability or RPV makeup capability) are not operable on the energized bus then the bus should not be considered operable.

The fifteen-minute interval was selected as a threshold to exclude transient power losses.

Generic

Loss of all AC power to emergency busses 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 to emergency 4.16 kV buses will lead to loss of Fuel Clad, RCS, and Containment, thus this event can escalate to a General Emergency.

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

Escalation to General Emergency is via EALs in Category F or EAL SG1.1.

NMP1 Basis Reference(s):

1. N1-OP-30 4.16 kV, 600V, and 480V House Service
2. USAR section IX Electrical Systems
3. NEI 99-01 IC SS1

Category: S – System Malfunction

Subcategory: 1 – Loss of AC Power

Initiating Condition: AC power capability to 4.16 kV emergency buses reduced to a single power source for ≥15 min. such that **ANY** additional single failure would result in a complete loss of all 4.16 kV emergency bus power

EAL:

SA1.1 Alert

AC power capability to 4.16 kV emergency buses reduced to a single power source, Table S-1, for ≥ 15 min. (Note 4)

AND

ANY additional single power source failure will result in a loss of **all** 4.16 kV emergency bus power

Note 4: The ED should **not** wait until the applicable time has elapsed, but should declare the event as soon as it is determined that the condition has exceeded, or will likely exceed, the applicable time.

Table S-1 AC Power Sources	
Onsite	• DG 102 • DG 103
Offsite	• T-101N • T-101S • T-10 backfed from offsite through T-1 or T-2 (**only** if already aligned)

Mode Applicability:

1 - Power Operation, 2 - Hot Shutdown

Basis:

Plant-Specific

NMP1 4.16 kV emergency buses are buses PB102 and PB103, which feed all Station redundant safety-related loads.. There are three offsite power sources available to these buses:

- Offsite 115 kV through transformer 101N. This is the normal power supply to PB102.
- Offsite 115 kV through transformer 101S. This is the normal power supply to PB103.
- Offsite 345 kV through transformer T-1 or T-2 backfed through transformer T-10.

Based on operational experience, if the backfeed is not already aligned, this cannot be considered available/capable of supplying the bus due to the time it will take to align it. In any case, if this cannot be accomplished within 15 minutes, it is not available and an Unusual Event must be declared.

There are two onsite AC power sources:

- DG 102 for PB102
- DG 103 for PB 103

The fifteen-minute interval was selected as a threshold to exclude transient power losses. If the capability for multiple sources to energize the unit emergency buses within 15 minutes is not restored, an Alert is declared under this EAL. The subsequent loss of the single remaining power source escalates the event to a Site Area Emergency under EAL SS1.1.

Generic

The condition indicated by this EAL is the degradation of the off-site and on-site AC power systems such that any additional single failure would result in a complete loss of 4.16 kV emergency bus AC power. This condition could occur due to a loss of off-site power with a concurrent failure of all but one emergency generator to supply power to its emergency busses. Another related condition could be the loss of all off-site power and loss of on-site emergency generators with only one train of 4.16 kV emergency buses being backfed from the unit main generator, or the loss of on-site emergency generators with only one train of 4.16 kV emergency buses being backfed from off-site power. The subsequent loss of this single power source would escalate the event to a Site Area Emergency in accordance with EAL SS1.1.

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

NMP1 Basis Reference(s):

1. N1-OP-30 4.16 kV, 600V, and 480V House Service
2. USAR section IX Electrical Systems
3. NEI 99-01 IC SA5

Category: S – System Malfunction

Subcategory: 1 – Loss of AC Power

Initiating Condition: Loss of **all** offsite AC power to 4.16 kV emergency buses for ≥ 15 min.

EAL:

SU1.1	**Unusual Event**
Loss of **all** offsite AC power, Table S-1, to 4.16 kV emergency buses for ≥ 15 min. (Note 4)	

Note 4: The ED should **not** wait until the applicable time has elapsed, but should declare the event as soon as it is determined that the condition has exceeded, or will likely exceed, the applicable time.

Table S-1 AC Power Sources	
Onsite	• DG102 • DG 103
Offsite	• T-101N • T-101S • T-10 backfed from offsite through T-1 or T-2 (**only** if already aligned)

Mode Applicability:

1 - Power Operation, 2 - Hot Shutdown

Basis:

Plant-Specific

NMP1 4.16 kV emergency buses are buses PB102 and PB103, which feed all Station redundant safety-related loads.. There are three offsite power sources available to these buses (ref. 1, 2):

- Offsite 115 kV through transformer 101N. This is the normal power supply to PB102.
- Offsite 115 kV through transformer 101S. This is the normal power supply to PB103.

- Offsite 345 kV through transformer T-1 or T-2 backfed through transformer T-10.

Based on operational experience, if the backfeed is not already aligned, this cannot be considered available/capable of supplying the bus due to the time it will take to align it. In any case, if this cannot be accomplished within 15 minutes, it is not available and an Unusual Event must be declared.

There are two onsite AC power sources:

- DG 102 for PB102
- DG 103 for PB 103

The fifteen-minute interval was selected as a threshold to exclude transient power losses.

Generic

Prolonged loss of off-site 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 to emergency busses.

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

NMP1 Basis Reference(s):

1. N1-OP-30 4.16 kV, 600V, and 480V House Service
2. USAR section IX Electrical Systems
3. NEI 99-01 IC SU1

Category: S – System Malfunction

Subcategory: 2 – Loss of DC Power

Initiating Condition: Loss of **all** emergency DC power for ≥ 15 min.

EAL:

SS2.1	**Site Area Emergency**
< 106 VDC on **both** Battery Board 11 and Battery Board 12 for ≥ 15 min. (Note 4)	

Note 4: The ED should **not** wait until the applicable time has elapsed, but should declare the event as soon as it is determined that the condition has exceeded, or will likely exceed, the applicable time.

Mode Applicability:

1 - Power Operation, 2 - Hot Shutdown

Basis:

Plant-Specific

A Safety Related (SR) system and a Quality Related (QR) system comprise the 125 VDC Power System. The two SR 125 VDC systems (Battery board 11 and Battery board 12) each consist of: one battery, two Static Chargers in parallel, and a DC distribution board. The one QR 125 VDC system consists of: a battery, one Static Charger, and one battery board. This EAL addresses only the Safety Related battery boards (ref. 1, 2).

106 VDC is the minimum voltage for battery operability (ref.3).

This EAL is the hot condition equivalent of the cold condition loss of DC power EAL CU2.1.

Generic

Loss of all DC power compromises ability to monitor and control plant safety functions. Prolonged loss of all DC power will cause core uncovering and loss of containment integrity when there is significant decay heat and sensible heat in the reactor system.

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

Escalation to a General Emergency would occur by EALs in Category R and Category F.

NMP1 Basis Reference(s):

1. N1-OP-47A 125 VDC Power System
2. USAR section IX Electrical Systems
3. NMP1 Technical Specification 3.6.3

4. NEI 99-01 IC SS3

Category: S – System Malfunction

Subcategory: 3 – Criticality & RPS Failure

Initiating Condition: Automatic scram and **all** manual actions fail to shut down the reactor and indication of an extreme challenge to the ability to cool the core exists

EAL:

SG3.1 General Emergency

An automatic scram fails to shut down the reactor as indicated by reactor power > 6%

AND

All manual actions fail to shut down the reactor as indicated by reactor power > 6%

AND EITHER of the following exist or have occurred:

RPV water level **cannot** be restored and maintained above -109 in. or RPV water level **cannot** be determined

OR

Torus temperature and RPV pressure **cannot** be maintained below the Heat Capacity Temperature Limit (N1-EOP-4 Figure M)

Mode Applicability:

1 - Power Operation

Basis:

Plant-Specific

This EAL addresses the following:

- Any automatic reactor scram signal followed by a manual scram that fails to shut down the reactor to an extent the reactor is producing energy in excess of the heat load for which the safety systems were designed (EAL SS3.1), and
- Indications that either core cooling is extremely challenged or heat removal is extremely challenged.

Reactor shutdown achieved by use of the alternate control rod insertion methods of N1-EOP-3.1 is also credited as a successful manual scram provided reactor power can be reduced below the APRM downscale trip setpoint before indications of an extreme challenge to either core cooling or heat removal exist (ref. 3).

The APRM downscale trip setpoint (6%) is a minimum reading on the power range scale that indicates power production (ref. 1, 2). It also approximates the decay heat which the shutdown systems were designed to remove and is indicative of a condition requiring immediate response to prevent subsequent core damage. At or below the APRM downscale trip setpoint, plant response will be similar to that observed during a normal shutdown. Nuclear instrumentation (APRM/IRM) indications or other reactor parameters (e.g., number of open SRVs, number of open main turbine bypass valves, main steam flow, RPV pressure and suppression pool temperature trend, etc.) can be used to determine if reactor power is greater than 6% power (ref. 2).

The combination of failure of both front line and backup protection systems to function in response to a plant transient, along with the continued production of heat, poses a direct threat to the Fuel Clad and RCS barriers.

By definition, an operating mode change occurs when the Mode Switch is moved from the startup or run position to the shutdown position. The plant operating mode that existed at the time the event occurs (i.e., Power Operation), however, requires emergency classification of at least an Alert. The operating mode change associated with movement of the Mode Switch, by itself, does not justify failure to declare an emergency for ATWS events.

Indication that core cooling is extremely challenged is manifested by:

- RPV water level cannot be restored and maintained above -109 in. (ref. 1, 2). The Minimum Steam Cooling RPV Water Level (MSCRWL) is the lowest RPV water level at which the covered portion of the reactor core will generate sufficient steam to preclude any clad temperature in the uncovered portion of the core from exceeding 1500°F. Consistent with the EOP definition of "cannot be restored and maintained," the determination that RPV water level cannot be restored and maintained above the MSCRWL may be made at, before, or after RPV water level actually decreases to this point.

  When RPV water level cannot be determined, EOPs require RPV flooding strategies. RPV water level indication provides the primary means of knowing if adequate core cooling is being maintained. When all means of determining RPV water level are unavailable, the fuel clad barrier is threatened and reliance on alternate means of assuring adequate core cooling must be attempted. The instructions in N1-EOP-7 specify these means, which include emergency depressurization of the RPV and injection into the RPV at a rate needed to flood to the elevation of the main steam lines or hold RPV pressure above the Minimum Steam Cooling Pressure (in ATWS events) (ref. 4).

- The HCTL is the highest torus water temperature from which emergency RPV depressurization will not raise (ref. 2):
  - Torus temperature above the design value (205°F), or
  - Torus pressure above Primary Containment Pressure Limit before the rate of energy transfer from the RPV to the containment is greater than the capacity of the containment vent.

The HCTL is a function of RPV pressure and torus water level. It is utilized to preclude failure of the containment and equipment in the containment necessary for the safe shutdown of the plant. Plant parameters in excess of the HCTL could be a precursor of Primary Containment failure. (ref. 2)

The HCTL is given in N1-EOP-4 Figure M. This threshold is met when RPV BLOW DOWN is required in N1-EOP-4, Step TT-5 (ref. 5). This condition addresses loss of functions required for hot shutdown with the reactor at pressure and temperature.

<u>Generic</u>

Under these conditions, the reactor is producing more heat than the maximum decay heat load for which the safety systems are designed and efforts to bring the reactor subcritical are unsuccessful.

The reactor should be considered shutdown when it producing less heat than the maximum decay heat load for which the safety systems are designed (6% power). In the event either of these challenges exists at a time that the reactor has not been brought below the power associated with the safety system design a core melt sequence exists. In this situation, core degradation can occur rapidly. For this reason, the General Emergency declaration is intended to be anticipatory of the fission product barrier table declaration to permit maximum off-site intervention time.

NMP1 Basis Reference(s):

1. N1-EOP-3 Failure to Scram
2. NER-1M-095 NMP1 EOP/SAP Basis Document
3. N1-EOP-3.1 Alternate Rod Insertion

4. N1-EOP-7 RPV Flooding
5. N1-EOP-4 Primary Containment Control
6. NEI 99-01 IC SG2

Category: S – System Malfunction

Subcategory: 3 – Criticality & RPS Failure

Initiating Condition: Automatic scram fails to shut down the reactor and manual actions taken from the reactor control console are **not** successful in shutting down the reactor

EAL:

SS3.1 Site Area Emergency

An automatic scram failed to shut down the reactor as indicated by reactor power > 6%

AND

Manual actions taken at the reactor control console (mode switch in shutdown, manual scram push buttons and ARI) failed to shut down the reactor as indicated by reactor power > 6%

Mode Applicability:

1 - Power Operation

Basis:

Plant-Specific

This EAL addresses any automatic reactor scram signal followed by a manual scram that failed to shut down the reactor to an extent the reactor is producing energy in excess of the heat load for which the safety systems were designed.

The first condition of this EAL identifies the need to cease critical reactor operations by actuation of the automatic Reactor Protection System (RPS) scram function. A reactor scram is automatically initiated by the Reactor Protection System (RPS) when certain continuously monitored parameters exceed predetermined setpoints. A reactor scram may be the result of manual or automatic action in response to any of the following conditions (ref. 1):

Parameter	Setpoint
High level in the Scram Dump Volume	45 gal
MSIV closure	10%
IRM neutron flux	96% of range
APRM neutron flux	TS 2.1.2.a

Parameter	Setpoint
High reactor pressure	1080 psig
Low reactor water level	+53 inches
High drywell pressure	3.5 psig
TSV closure	10%
Generator load rejection	TCV fast closure

Following a successful reactor scram, rapid insertion of the control rods occurs. Nuclear power promptly drops to a fraction of the original power level and then decays to a level several decades less with a negative period. The reactor power drop continues until reactor power reaches the point at which the influence of source neutrons on reactor power starts to be observable. A predictable post-scram response from an automatic reactor scram signal should therefore consist of a prompt drop in reactor power as sensed by the nuclear instrumentation and a lowering of power into the source range. A successful scram has therefore occurred when there is sufficient rod insertion from the trip of RPS to bring the reactor power to or below the APRM downscale trip setpoint of 6%. For the purposes of this EAL, a successful automatic initiation of ARI that reduces reactor power to or below 6% is a not a successful automatic scram. If automatic actuation of ARI has occurred and caused reactor shutdown, the automatic RPS scram must have failed. ARI is a backup means of inserting control rods in the unlikely event that an automatic RPS scram signal exists but the reactor continues to generate significant power. (ref. 2, 3)

For the purposes of emergency classification at the Site Area emergency level, successful manual scram actions are those which can be quickly performed from the reactor control console (i.e., Mode Switch, manual scram pushbuttons, and manual ARI actuation). Reactor shutdown achieved by use of the alternate control rod insertion methods of N1-EOP-3.1 does not constitute a successful manual scram (ref. 4).

The APRM downscale trip setpoint (6%) is a minimum reading on the power range scale that indicates power production (ref. 3). It also approximates the decay heat which the shutdown systems were designed to remove and is indicative of a condition requiring immediate response to prevent subsequent core damage. At or below the APRM downscale trip setpoint, plant response will be similar to that observed during a normal shutdown. Nuclear instrumentation (APRM/IRM) indications or other reactor parameters (e.g., number of open SRVs, number of open main turbine bypass valves, main steam flow, RPV pressure and wetwell temperature trend, etc.) can be used to determine if reactor power is greater than 6% power.

By definition, an operating mode change occurs when the Mode Switch is moved from the startup or run position to the shutdown position. The plant operating mode that existed at the time the event occurs (i.e., Power Operation), however, requires emergency classification of at least an Alert. The operating mode change associated with movement of the Mode Switch, by itself, does not justify failure to declare an emergency for ATWS events.

Escalation of this event to a General Emergency would be under EAL SG3.1 or Emergency Director judgment.

<u>Generic</u>

Under these conditions, the reactor is producing more heat than the maximum decay heat load for which the safety systems are designed and efforts to bring the reactor subcritical are unsuccessful. A Site Area Emergency is warranted because conditions exist that lead to IMMINENT loss or potential loss of both fuel clad and RCS.

The reactor should be considered shutdown when it producing less heat than the maximum decay heat load for which the safety systems are designed (6% power).

Manual scram actions taken at the reactor control console are any set of actions by the reactor operator(s) at which causes or should cause control rods to be rapidly inserted into the core and shuts down the reactor.

Manual scram actions are not considered successful if action away from the reactor control console is required to scram the reactor. This EAL is still applicable even if actions taken away from the reactor control console are successful in shutting the reactor down because the design limits of the fuel may have been exceeded or because of the gross failure of the Reactor Protection System to shutdown the plant.

Escalation of this event to a General Emergency would be due to a prolonged condition leading to an extreme challenge to either core-cooling or heat removal.

NMP1 Basis Reference(s):

1. Technical Specifications Table 3.6.2.a
2. N1-EOP-2 RPV Control
3. N1-EOP-3 Failure to Scram
4. N1-EOP-3.1 Alternate Rod Insertion

5. NEI 99-01 IC SS2

Category: S – System Malfunction

Subcategory: 3 – Criticality & RPS Failure

Initiating Condition: Automatic scram failed to shut down the reactor and the manual actions taken from the reactor control console are successful in shutting down the reactor

EAL:

SA3.1 Alert

An automatic scram failed to shut down the reactor

AND

Manual actions taken at the reactor control console (mode switch in shutdown, manual scram push buttons or ARI) successfully shut down the reactor as indicated by reactor power ≤ 6%

Mode Applicability:

1 - Power Operation

Basis:

Plant-Specific

The first condition of this EAL identifies the need to cease critical reactor operations by actuation of the automatic Reactor Protection System (RPS) scram function. A reactor scram is automatically initiated by the Reactor Protection System (RPS) when certain continuously monitored parameters exceed predetermined setpoints. A reactor scram may be the result of manual or automatic action in response to ~~any of the following~~various conditions (ref. 1):

~~Parameter~~	~~Setpoint~~
~~High level in the Scram Dump Volume~~	~~45 gal~~
~~MSIV closure~~	~~10%~~
~~IRM neutron flux~~	~~96% of range~~
~~APRM neutron flux~~	~~TS 2.1.2.a~~
~~High reactor pressure~~	~~1080 psig~~
~~Low reactor water level~~	~~+53 inches~~
~~High drywell pressure~~	~~3.5 psig~~

~~Parameter~~	~~Setpoint~~
~~TSV closure~~	~~10%~~
~~Generator load rejection~~	~~TCV fast closure~~

Following a successful reactor scram, rapid insertion of the control rods occurs. Nuclear power promptly drops to a fraction of the original power level and then decays to a level several decades less with a negative period. The reactor power drop continues until reactor power reaches the point at which the influence of source neutrons on reactor power starts to be observable. A predictable post-scram response from an automatic reactor scram signal should therefore consist of a prompt drop in reactor power as sensed by the nuclear instrumentation and a lowering of power into the source range. A successful scram has therefore occurred when there is sufficient rod insertion from the trip of RPS to bring the reactor power to or below the APRM downscale trip setpoint of 6%. For the purposes of this EAL, a successful automatic initiation of ARI that reduces reactor power to or below 6% is a not a successful automatic scram. If automatic actuation of ARI has occurred and caused reactor shutdown, the automatic RPS scram must have failed. ARI is a backup means of inserting control rods in the unlikely event that an automatic RPS scram signal exists but the reactor continues to generate significant power. (ref. 2, 3)

For the purposes of emergency classification at the Alert level, successful manual scram actions are those which can be quickly performed from the reactor control console (i.e., mode switch, manual scram pushbuttons, and manual ARI actuation). Reactor shutdown achieved by use of the alternate control rod insertion methods of N1-EOP-3.1 does not constitute a successful manual scram (ref. 4).

Following any automatic RPS scram signal EOPs prescribe insertion of redundant manual scram signals to back up the automatic RPS scram function and ensure reactor shutdown is achieved. Even if the first subsequent manual scram signal inserts all control rods to the full-in position immediately after the initial failure of the automatic scram, the lowest level of classification that must be declared is an Alert.

If the operator determines the reactor must be scrammed before one of the RPS setpoints is reached, procedures require that the Mode Switch first be placed in the shutdown position. Although manipulation of the Mode Switch is a manual action, the RPS logic trains are actuated as with an automatic RPS-initiated scram. If reactor power remains above the APRM downscale trip setpoint after the Mode Switch is placed in shutdown, RPS has failed and, as a minimum, an Alert emergency declaration is required. If subsequent actuation of the reactor scram pushbuttons and manual initiation of ARI do not reduce reactor power to or below the APRM downscale trip setpoint, a Site Area Emergency declaration is required under EAL SS3.1.

In the event that the operator identifies a reactor scram is IMMINENT and initiates a successful manual reactor scram before the automatic scram setpoint is reached, no declaration is required. The successful manual scram of the reactor before it reaches its automatic scram setpoint or reactor scram signals caused by instrumentation channel failures do not lead to a potential fission product barrier loss. If manual reactor scram actions fail to reduce reactor power to or below 6%, the event escalates to the Site Area Emergency under EAL SS3.1.

By procedure, operator actions include the initiation of an immediate manual scram following receipt of an automatic scram signal. If there are no clear indications that the automatic scram failed (such as a time delay following indications that a scram setpoint was exceeded), it may be difficult to determine if the reactor was shut down because of automatic scram or manual actions. If a subsequent review of the scram actuation indications reveals that the automatic scram did not cause the reactor to be shut down, consideration should be given to evaluating the fuel for potential damage and the reporting requirements of 50.72 should be considered for the transient event.

By definition, an operating mode change occurs when the Mode Switch is moved from the startup or run position to the shutdown position. The plant operating mode that existed at the time the event occurs (i.e., Power Operation), however, requires emergency classification of at least an Alert. The operating mode change associated with movement of the Mode Switch, by itself, does not justify failure to declare an emergency for ATWS events.

Generic

The reactor should be considered shutdown when it producing less heat than the maximum decay heat load for which the safety systems are designed (6% power).

Manual scram actions taken at the reactor control console are any set of actions by the reactor operator(s) which causes or should cause control rods to be rapidly inserted into the core and shuts down the reactor.

This condition indicates failure of the automatic protection system to scram the reactor. This condition is more than a potential degradation of a safety system in that a front line automatic protection system did not function in response to a scram signal. Thus the plant safety has been compromised because of the failure of RPS to automatically shut down the plant. An Alert is indicated because conditions may exist that lead to potential loss of fuel clad barrier or RCS barrier and because of the failure of the Reactor Protection System to automatically shut down the plant.

If manual actions taken at the reactor control console fail to shut down the reactor, the event would escalate to a Site Area Emergency.

NMP1 Basis Reference(s):

1. Technical Specifications Table 3.6.2.a
2. N1-EOP-2 RPV Control
3. N1-EOP-3 Failure to Scram
4. N1-EOP-3.1 Alternate Rod Insertion
5. NEI 99-01 IC SA2

Category: S – System Malfunction

Subcategory: 3 – Criticality & RPS Failure

Initiating Condition: Inadvertent criticality

EAL:

SU3.1	**Unusual Event**
An UNPLANNED sustained positive period observed on nuclear instrumentation	

Mode Applicability:

2 - Hot Shutdown

Basis:

Plant-Specific

The term "sustained" is used to allow exclusion of expected short-term positive periods from planned fuel bundle or control rod movements during core alteration. These short-term positive periods are the result of the rise in neutron population due to subcritical multiplication.

Generic

This EAL addresses inadvertent criticality events. While the primary concern of this EAL is criticality This EAL addresses inadvertent criticality events. This EAL indicates a potential degradation of the level of safety of the plant, warranting a UE classification. This EAL excludes inadvertent criticalities that occur during planned reactivity changes associated with reactor startups (e.g., criticality earlier than estimated).

Escalation would be by EALs in Category F, as appropriate to the operating mode at the time of the event.

NMP1 Basis Reference(s):

1. NEI 99-01 IC SU8

Category: S – System Malfunction

Subcategory: 4 – Inability to Reach or Maintain Shutdown Conditions

Initiating Condition: Inability to reach required shutdown within Technical Specification limits

EAL:

SU4.1	**Unusual Event**
Plant is **not** brought to required operating mode within Technical Specifications LCO required action completion time	

Mode Applicability:

1 - Power Operation, 2 - Hot Shutdown

Basis:

Plant-Specific

Limiting Conditions for Operation (LCOs) specify minimum requirements for ensuring safe operation of the unit. The actions associated with an LCO state conditions that typically describe the ways in which the requirements of the LCO can fail to be met. Specified with each stated condition are required action completion times. (ref. 1)

Generic

Limiting Conditions of Operation (LCOs) require the plant to be brought to a required operating 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 four 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 required action completion time in the Technical Specifications. An immediate UE is required when the plant is not brought to the required operating mode within the allowable required action completion time in the Technical Specifications. Declaration of a UE is based on the time at which the LCO-specified required action completion time period elapses under the site Technical Specifications and is not related to how long a condition may have existed.

NMP1 Basis Reference(s):

1. Nine Mile Point Unit 1Technical Specifications
2. NEI 99-01 IC SU2

Category: S – System Malfunction

Subcategory: 5 – Instrumentation

Initiating Condition: Inability to monitor a significant transient in progress

EAL:

SS5.1 Site Area Emergency

Loss of > approximately 75% of annunciation or indication on Control Room panels L, K, H, F and G for ≥ 15 min. (Note 4)

AND

A significant transient is in progress, Table S-2

AND

Compensatory indications are unavailable (Plant Process Computer, SPDS)

Note 4: The ED should **not** wait until the applicable time has elapsed, but should declare the event as soon as it is determined that the condition has exceeded, or will likely exceed, the applicable time

Table S-2 Significant Transients
• Turbine runback > 25% thermal reactor power • Electric load rejection > 25% full electrical load • Reactor scram • ECCS injection • Thermal power oscillations > 10%

Mode Applicability:

1 - Power Operation, 2 - Hot Shutdown

Basis:

<u>Plant-Specific</u>

Plant computer and SPDS are considered compensatory indication (ref. 1).

Significant transients are listed in Table S-2.

<u>Generic</u>

This EAL is intended to recognize the threat to plant safety associated with the complete loss of capability of the control room staff to monitor plant response to a significant transient.

"Planned" and "UNPLANNED" actions are not differentiated since the loss of instrumentation of this magnitude is of such significance during a transient that the cause of the loss is not an ameliorating factor.

Quantification is arbitrary, however, it is estimated that if approximately 75% of the safety system annunciators or indicators are lost, there is an increased risk that a degraded plant condition could go undetected. It is not intended that plant personnel perform a detailed count of the instrumentation lost but use the value as a judgment threshold for determining the severity of the plant conditions. It is also not intended that the Shift Manager be tasked with making a judgment decision as to whether additional personnel are required to provide increased monitoring of system operation.

It is further recognized that most plant designs provide redundant safety system indication powered from separate uninterruptible power supplies. While failure of a large portion of annunciators is more likely than a failure of a large portion of indications, the concern is included in this EAL due to difficulty associated with assessment of plant conditions. The loss of specific, or several, safety system indicators should remain a function of that specific system or component operability status. This will be addressed by the specific Technical Specification. The initiation of a Technical Specification imposed plant shutdown related to the instrument loss will be reported via 10 CFR 50.72. If the shutdown is not in compliance with the Technical Specification action, the NOUE is based on EAL SU4.1

A Site Area Emergency is considered to exist if the control room staff cannot monitor safety functions needed for protection of the public while a significant transient is in progress.

Annunciators for this EAL are limited to include those identified in the Abnormal Operating Procedures, in the Emergency Operating Procedures, and in other EALs (.g., area, process, and/or effluent rad monitors, etc.)

Indications needed to monitor safety functions necessary for protection of the public include control room indications, computer generated indications and dedicated annunciation capability.

"Compensatory indications" in this context includes computer based information such as Plant Process Computer and SPDS.

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

NMP1 Basis Reference(s):

1. N1-OP-42 Process Computer/SPDS
2. NEI 99-01 IC SS6

Category: S – System Malfunction

Subcategory: 5 – Instrumentation

Initiating Condition: UNPLANNED loss of safety system annunciation or indication in the Control Room with either (1) a significant transient in progress, or (2) compensatory indicators are unavailable

EAL:

SA5.1	**Alert**
UNPLANNED loss of > approximately 75% of annunciation or indication on Control Room panels L, K, H, F and G for ≥ 15 min. (Note 4) **AND EITHER**: A significant transient is in progress, Table S-2 **OR** Compensatory indications are unavailable (Plant Process Computer, SPDS)	

Note 4: The ED should **not** wait until the applicable time has elapsed, but should declare the event as soon as it is determined that the condition has exceeded, or will likely exceed, the applicable time

Table S-2 Significant Transients
• Turbine runback > 25% thermal reactor power • Electric load rejection > 25% full electrical load • Reactor scram • ECCS injection • Thermal power oscillations > 10%

Mode Applicability:

1 - Power Operation, 2 - Hot Shutdown

Basis:

Plant-Specific

Plant Process Computer and SPDS are considered compensatory indication (ref. 1).

Significant transients are listed in Table S-2.

Generic

This EAL is intended to recognize the difficulty associated with monitoring changing plant conditions without the use of a major portion of the annunciation or indication equipment during a significant transient.

"Planned" loss of annunciators or indicators includes scheduled maintenance and testing activities.

Quantification is arbitrary, however, it is estimated that if approximately 75% of the safety system annunciators or indicators are lost, there is an increased risk that a degraded plant condition could go undetected. It is not intended that plant personnel perform a detailed count of the instrumentation lost but use the value as a judgment threshold for determining the severity of the plant conditions. It is also not intended that the Shift Manager be tasked with making a judgment decision as to whether additional personnel are required to provide increased monitoring of system operation.

It is further recognized that most plant designs provide redundant safety system indication powered from separate uninterruptible power supplies. While failure of a large portion of annunciators is more likely than a failure of a large portion of indications, the concern is included in this EAL due to difficulty associated with assessment of plant conditions. The loss of specific, or several, safety system indicators should remain a function of that specific system or component operability status. This will be addressed by the specific Technical Specification. The initiation of a Technical Specification imposed plant shutdown related to the instrument loss will be reported via 10 CFR 50.72. If the shutdown is not in compliance with the Technical Specification action, the UE is based on EAL SU4.1.

Annunciators or indicators for this EAL include those identified in the Abnormal Operating Procedures, in the Emergency Operating Procedures, and in other EALs (e.g., area, process, and/or effluent rad monitors, etc.).

"Compensatory indications" in this context includes computer based information such as Plant Process Computer and SPDS.

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

This Alert will be escalated to a Site Area Emergency if the operating crew cannot monitor the transient in progress due to a concurrent loss of compensatory indications with a significant transient in progress during the loss of annunciation or indication.

NMP1 Basis Reference(s):

1. N1-OP-42 Process Computer/SPDS
2. NEI 99-01 IC SA4

Category: S – System Malfunction

Subcategory: 5 – Instrumentation

Initiating Condition: UNPLANNED loss of safety system annunciation or indication in the Control Room for ≥ 15 min.

EAL:

SU5.1 **Unusual Event**
UNPLANNED loss of > approximately 75% of annunciation or indication on Control Room panels L, K, H, F and G for ≥ 15 min. (Note 4)

Note 4: The ED should **not** wait until the applicable time has elapsed, but should declare the event as soon as it is determined that the condition has exceeded, or will likely exceed, the applicable time

Mode Applicability:

1 - Power Operation, 2 - Hot Shutdown

Basis:

Plant-Specific

None

Generic

This EAL is intended to recognize the difficulty associated with monitoring changing plant conditions without the use of a major portion of the annunciation or indication equipment.

Recognition of the availability of computer based indication equipment is considered.

"Planned" loss of annunciators or indicators includes scheduled maintenance and testing activities.

Quantification is arbitrary, however, it is estimated that if approximately 75% of the safety system annunciators or indicators are lost, there is an increased risk that a degraded plant condition could go undetected. It is not intended that plant personnel perform a detailed count of the instrumentation lost but use the value as a judgment threshold for determining the severity of the plant conditions.

It is further recognized that plant design provides redundant safety system indication powered from separate uninterruptible power supplies. While failure of a large portion of annunciators is more likely than a failure of a large portion of indications, the concern is included in this EAL due to difficulty associated with assessment of plant conditions. The loss of specific, or several, safety system indicators should remain a function of that specific system or component operability status. This will be addressed by the specific Technical Specification. The initiation of a Technical Specification imposed plant shutdown related to the instrument loss will be reported via 10CFR 50.72. If the shutdown is not in compliance with the Technical Specification action, the UE is based on EAL SU4.1.

Annunciators or indicators for this EAL include those identified in the Abnormal Operating Procedures, in the Emergency Operating Procedures, and in other EALs (e.g., area, process, and/or effluent rad monitors, etc.).

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

This UE will be escalated to an Alert based on a concurrent loss of compensatory indications or if a significant transient is in progress during the loss of annunciation or indication.

NMP1 Basis Reference(s):

1. N1-OP-42 Process Computer/SPDS
2. NEI 99-01 IC SU3

Category: S – System Malfunction

Subcategory: 6 – Communications

Initiating Condition: Loss of **all** onsite or offsite communications capabilities

EAL:

SU6.1 Unusual Event

Loss of **all** Table S-3 onsite (internal) communication methods affecting the ability to perform routine operations

OR

Loss of **all** Table S-3 offsite (external) communication methods affecting the ability to perform offsite notifications

Table S-3 Communications Systems		
System	**Onsite** (internal)	**Offsite** (external)
PBX (normal dial telephones)	X	X
Gaitronics	X	
Hand-Held Portable Radio (station radio)	X	
Control Room installed satellite phones (non portable)		X
ENS		X
RECS		X
~~UHF radios~~		~~X~~

Mode Applicability:

1 - Power Operation, 2 - Hot Shutdown

Basis:

Plant-Specific

Onsite/offsite communications systems are listed in Table S-3 (ref. 1, 2).

This EAL is the hot condition equivalent of the cold condition EAL CU6.1.

Generic

The purpose of this EAL is to recognize a loss of communications capability that either defeats the plant operations staff ability to perform routine tasks necessary for plant operations or the ability to communicate issues with off-site authorities.

The loss of off-site communications ability is expected to be significantly more comprehensive than the condition addressed by 10 CFR 50.72.

The availability of one method of ordinary off-site communications is sufficient to inform federal, state, and local authorities of plant problems. This EAL is intended to be used only when extraordinary means (e.g., relaying of information from non-routine radio transmissions, individuals being sent to off-site locations, etc.) are being used to make communications possible.

NMP1 Basis Reference(s):

1. Nine Mile Point Nuclear Station Site Emergency Plan, Section 7.2
2. USAR 2.4.5 Lighting and Communication
3. NEI 99-01 IC SU6

Category: S – System Malfunction

Subcategory: 7 – Fuel Clad Degradation

Initiating Condition: Fuel clad degradation

EAL:

SU7.1	**Unusual Event**
Reactor coolant activity > 4 µCi/gm I-131 Equivalent	

Mode Applicability:

1 - Power Operation, 2 - Hot Shutdown

Basis:

Plant-Specific

This EAL addresses reactor coolant samples exceeding Technical Specification 3.2.4 (ref. 1). A reactor coolant sample analysis with specific activity in excess of the Technical Specification limit of 4 µCi/gm I-131 Equivalent is indicative of a degradation of the fuel clad, and is a precursor of more serious problems. This activity level for which operation is allowed to continue for up to 48 hours to accommodate short duration Iodine spikes following changes in thermal power.

Generic

This EAL is included because it is a precursor of more serious conditions and, as result, is considered to be a potential degradation of the level of safety of the plant.

Escalation of this EAL to the Alert level is via the EALs in Category F.

This threshold addresses coolant samples exceeding coolant technical specifications for transient iodine spiking limits.

NMP1 Basis Reference(s):

1. Technical Specification 3.2.4 Reactor Coolant System - RCS Specific Activity
2. NEI 99-01 IC SU4

Category: S – System Malfunction

Subcategory: 7 – Fuel Clad Degradation

Initiating Condition: Fuel clad degradation

EAL:

SU7.2	**Unusual Event**
Offgas radiation monitor RN-12A or RN-12B ≥ hi-hi alarm for ≥ 15 min.	

Mode Applicability:

1 - Power Operation, 2 - Hot Shutdown

Basis:

Plant-Specific

Elevated offgas radiation activity represents a potential degradation in the level of safety of the plant and a potential precursor of more serious problems. The Technical Specification allowable limit is an offgas level not to exceed 500,000 μCi/sec (recombiner discharge gross noble gases beta and/or gamma) (ref. 1).

The hi-hi alarm setpoint has been conservatively selected because it is operationally significant and is readily recognizable by Control Room operating staff (ref. 2). 15 minutes is allotted for operator action to reduce the offgas radiation levels and exclude transient conditions (ref. 3). The high offgas radiation alarm is set using methodology outlined in the ODCM (ref. 1).

The offgas system automatically isolates (BV 77-03) when both RN-12A and RN-12B alarm.

Generic

This EAL is included because it is a precursor of more serious conditions and, as result, is considered to be a potential degradation of the level of safety of the plant.

Escalation of this EAL to the Alert level is via the EALs in Category F.

This threshold addresses radiation monitor readings that provide indication of a degradation of fuel clad integrity.

NMP1 Basis Reference(s):

1. ODCM 3.6.14 & 15

2. N1-ARP-H1, Annunciator H1-2-7
3. N1-SOP-25.2 Fuel Failure or High Activity in RX Coolant or Off Gas
4. NEI 99-01 IC SU4

Category: S – System Malfunction

Subcategory: 8 – RCS Leakage

Initiating Condition: RCS leakage

EAL:

SU8.1	**Unusual Event**
Unidentified drywell leakage > 10 gpm	
OR	
Identified reactor coolant drywell leakage > 25 gpm	

Mode Applicability:

1 - Power Operation, 2 - Hot Shutdown

Basis:

Plant-Specific

Elevated RCS leakage may be detected by the following annunciators (ref. 1, 2):

- H2-1-1 DRYWELL FLOOR DRAIN LEVEL-HIGH
- H2-4-7 DRYWELL WATER LEAK DETECTION SYS

Once elevated drywell leakage is detected operators will monitor and record Drywell Floor and Equipment Drain Tank leakage to determine drywell unidentified and identified leakage values (ref. 3, 4).

Generic

This EAL is included as a UE because it may be a precursor of more serious conditions and, as result, is considered to be a potential degradation of the level of safety of the plant. The 10 gpm value for the unidentified or pressure boundary leakage was selected as it is observable with normal control room indications. Lesser values must generally be determined through time-consuming surveillance tests (e.g., mass balances).

Relief valve normal operation should be excluded from this EAL. However, a relief valve that operates and fails to close per design should be considered applicable to this EAL if the relief valve cannot be isolated.

The EAL for identified leakage is set at a higher value due to the lesser significance of identified leakage in comparison to unidentified or pressure boundary leakage. In either case, escalation of this EAL to the Alert level is via EALs in Category F.

NMP1 Basis Reference(s):

1. N1-ARP-H2 Annunciator H2-1-1 DRYWELL FLOOR DRAIN LEVEL-HIGH
2. N1-ARP-H2 Annunciator H2-4-7 DRYWELL WATER LEAK DETECTION SYS
3. S-ODP-OPS-0110 Containment Leakage Evaluation
4 N1-OP-08, Primary Containment Area Cooling System
5. NEI 99-01 IC SU5

Category F – Fission Product Barrier Degradation

EAL Group: Hot Conditions (RCS temperature > 212°F); EALs in this category are applicable only in one or more hot operating modes.

EALs in this category represent threats to the defense in depth design concept that precludes the release of highly radioactive fission products to the environment. This concept relies on multiple physical barriers any one of which, if maintained INTACT, precludes the release of significant amounts of radioactive fission products to the environment. The primary fission product barriers are:

A. Fuel Clad (FC): Zirconium tubes which house the ceramic uranium oxide pellets along with the end plugs which are welded into each end of the fuel rods comprise the FC barrier.

B. Reactor Coolant System (RCS): The reactor vessel shell, vessel head, CRD housings, vessel nozzles and penetrations, and all primary systems directly connected to the RPV up to the outermost Primary Containment isolation valve comprise the RCS barrier.

C. Containment (PC): The drywell, the suppression chamber/pool, their respective interconnecting paths, and other connections up to and including the outermost containment isolation valves comprise the Primary Containment barrier.

The EALs in this category require evaluation of the loss and potential loss thresholds listed in the fission product barrier matrix of Table F-1 (Attachment 2). "Loss" and "Potential Loss" signify the relative damage and threat of damage to the barrier. "Loss" means the barrier no longer assures containment of radioactive materials. "Potential Loss" means integrity of the barrier is threatened and could be lost if conditions continue to degrade. The number of barriers that are lost or potentially lost and the following criteria determine the appropriate emergency classification level:

Unusual Event:
Any loss or any potential loss of Containment

Alert:
Any loss or any potential loss of either Fuel Clad or RCS

Site Area Emergency:
Loss or potential loss of any two barriers

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

The logic used for Category F EALs reflects the following considerations:

- The Fuel Clad Barrier and the RCS Barrier are weighted more heavily than the Containment Barrier. UE EALs associated with RCS and Fuel Clad Barriers are addressed under Category S.
- At the Site Area Emergency level, there must be some ability to dynamically assess how far present conditions are from the threshold for a General Emergency. For example, if Fuel Clad and RCS Barrier "Loss" thresholds existed, that, in addition to off-site dose assessments, would require continual assessments of radioactive inventory and containment integrity. Alternatively, if both Fuel Clad and RCS Barrier "Potential Loss" thresholds existed, the ED would have more assurance that there was no immediate need to escalate to a General Emergency.
- The ability to escalate to higher emergency classification levels as an event deteriorates must be maintained. For example, RCS leakage steadily increasing would represent an increasing risk to public health and safety.
- The Containment Barrier should not be declared lost or potentially lost based on exceeding Technical Specification action statement criteria, unless there is an event in progress requiring mitigation by the Containment barrier.

Category: Fission Product Barrier Degradation

Subcategory: N/A

Initiating Condition: Loss of **ANY** two barriers and loss or potential loss of the third barrier

EAL:

FG1.1	**General Emergency**
Loss of **ANY** two fission product barriers **AND** Loss or potential loss of third fission product barrier (Table F-1)	

Mode Applicability:

1 - Power Operation, 2 - Hot Shutdown

Basis:

Plant-Specific

Fuel Clad, RCS and Containment comprise the fission product barriers. Table F-1 (Attachment 2) lists the fission product barrier thresholds, bases and references.

At the General Emergency classification level each barrier is weighted equally. A General Emergency is therefore appropriate for any combination of the following conditions:

- Loss of Fuel Clad, RCS and Containment barriers
- Loss of Fuel Clad and RCS barriers with potential loss of Containment barrier
- Loss of RCS and Containment barriers with potential loss of Fuel Clad barrier
- Loss of Fuel Clad and Containment barriers with potential loss of RCS barrier

Generic

None

NMP1 Basis Reference(s):

1. NEI 99-01 IC FG1

Category: Fission Product Barrier Degradation

Subcategory: N/A

Initiating Condition: Loss or potential loss of **ANY** two barriers

EAL:

FS1.1	**Site Area Emergency**
Loss or potential loss of **ANY** two fission product barriers (Table F-1)	

Mode Applicability:

1 - Power Operation, 2 - Hot Shutdown

Basis:

Plant-Specific

Fuel Clad, RCS and Containment comprise the fission product barriers. Table F-1 (Attachment 2) lists the fission product barrier thresholds, bases and references.

At the Site Area Emergency classification level, each barrier is weighted equally. A Site Area Emergency is therefore appropriate for any combination of the following conditions:

- One barrier loss and a second barrier loss (i.e., loss - loss)
- One barrier loss and a second barrier potential loss (i.e., loss - potential loss)
- One barrier potential loss and a second barrier potential loss (i.e., potential loss - potential loss)

At the Site Area Emergency classification level, the ability to dynamically assess the proximity of present conditions with respect to the threshold for a General Emergency is important. For example, the existence of Fuel Clad and RCS Barrier loss thresholds in addition to offsite dose assessments would require continual assessments of radioactive inventory and Containment integrity in anticipation of reaching a General Emergency classification. Alternatively, if both Fuel Clad and RCS potential loss thresholds existed, the Emergency Director would have greater assurance that escalation to a General Emergency is less IMMINENT.

Generic

None

NMP1 Basis Reference(s):

1. NEI 99-01 IC FS1

Category: Fission Product Barrier Degradation

Subcategory: N/A

Initiating Condition: **ANY** loss or **ANY** potential loss of **EITHER** Fuel Clad **OR** RCS

EAL:

FA1.1	**Alert**
ANY loss or **ANY** potential loss of **EITHER** Fuel Clad barrier **OR** RCS barrier (Table F-1)	

Mode Applicability:

1 - Power Operation, 2 - Hot Shutdown

Basis:

Plant-Specific

Fuel Clad, RCS and Containment comprise the fission product barriers. Table F-1 (Attachment 2) lists the fission product barrier thresholds, bases and references.

At the Alert classification level, Fuel Clad and RCS barriers are weighted more heavily than the Containment barrier. Unlike the Containment barrier, loss or potential loss of either the Fuel Clad or RCS barrier may result in the relocation of radioactive materials or degradation of core cooling capability. Note that the loss or potential loss of Containment barrier in combination with loss or potential loss of either Fuel Clad or RCS barrier results in declaration of a Site Area Emergency under EAL FS1.

Generic

None

NMP1 Basis Reference(s):

1. NEI 99-01 IC FA1

Category: Fission Product Barrier Degradation

Subcategory: N/A

Initiating Condition: **ANY** loss or **ANY** potential loss of Containment

EAL:

FU1.1	**Unusual Event**
ANY loss or **ANY** potential loss of Containment barrier (Table F-1)	

Mode Applicability:

1 - Power Operation, 2 - Hot Shutdown

Basis:

Plant-Specific

Fuel Clad, RCS and Containment comprise the fission product barriers. Table F-1 (Attachment 2) lists the fission product barrier thresholds, bases and references.

Fuel Clad and RCS barriers are weighted more heavily than the Containment barrier. Unlike the Fuel Clad and RCS barriers, the loss of either of which results in an Alert (EAL FA1.1), loss of the Containment barrier in and of itself does not result in the relocation of radioactive materials or the potential for degradation of core cooling capability. However, loss or potential loss of the Containment barrier in combination with the loss or potential loss of either the Fuel Clad or RCS barrier results in declaration of a Site Area Emergency under EAL FS1.1.

Generic

None

NMP1 Basis Reference(s):

1. NEI 99-01 IC FU1

Introduction

Table F-1 lists the threshold conditions that define the Loss and Potential Loss of the three fission product barriers (Fuel Clad, Reactor Coolant System, and Containment). The table is structured so that each of the three barriers occupies adjacent columns. Each fission product barrier column is further divided into two columns; one for Loss thresholds and one for Potential Loss thresholds.

The first column of the table (to the left of the Fuel Clad Loss column) lists the categories (types) of fission product barrier thresholds. The fission product barrier categories are:

A. RPV water level

B. Primary Containment Pressure / Temperature

C. Isolation

D. Rad

E. Judgment

Each category occupies a row in Table F-1 thus forming a matrix defined by the category rows and the Loss/Potential Loss columns. The intersection of each category row with each Loss/Potential Loss column forms a cell in which one or more fission product barrier thresholds appear. If NEI 99-01 does not define a threshold for a barrier Loss/Potential Loss, the word "None" is entered in the cell.

Thresholds are assigned sequential numbers within each Loss and Potential Loss column beginning with number one. In this manner, a threshold can be identified by its category title and number. For example, the first Fuel Clad barrier Loss in Category A would be assigned "FC Loss A.1," the third Containment barrier Potential Loss would be assigned "PC P-Loss B.3," etc.

If a cell in Table F-1 contains more than one numbered threshold, each of the numbered thresholds, if exceeded, signifies a Loss or Potential Loss of the barrier. It is not necessary to exceed all of the thresholds in a category before declaring a barrier Loss/Potential Loss.

Subdivision of Table F-1 by category facilitates association of plant conditions to the applicable fission product barrier Loss and Potential Loss thresholds. This structure promotes a systematic approach to assessing the classification status of the fission product barriers.

When equipped with knowledge of plant conditions related to the fission product barriers, the EAL-user first scans down the category column of Table F-1, locates the likely category and then reads across the row of fission product barrier Loss and Potential Loss thresholds in that category to determine if any threshold has been exceeded. If a threshold has not been exceeded in that category row, the EAL-user proceeds to the next likely category and continues review of the row of thresholds in the new category

If the EAL-user determines that any threshold has been exceeded, by definition, the barrier is lost or potentially lost – even if multiple thresholds in the same barrier column are exceeded; only that one barrier is lost or potentially lost. The EAL-user must examine each of the three fission product barriers to determine if other barrier thresholds in the category are lost or potentially lost. For example, if Primary Containment radiation is sufficiently high, a Loss of the Fuel Clad and RCS barriers and a Potential Loss of the Containment barrier can occur. Barrier Losses and Potential Losses are then applied to the algorithms given in EALs FG1.1, FS1.1, FA1.1 and FU1.1 to determine the appropriate emergency classification.

In the remainder of this Attachment, the Fuel Clad barrier threshold bases appear first, followed by the RCS barrier and finally the Containment barrier threshold bases. In each barrier, the bases are given according category Loss followed by category Potential Loss beginning with Category A, then B...E.

Table F-1 Fission Product Barrier Matrix

	Fuel Clad Barrier		Reactor Coolant System Barrier		Containment Barrier	
Category	**Loss**	**Potential Loss**	**Loss**	**Potential Loss**	**Loss**	**Potential Loss**
A **RPV Water Level**	1. Primary Containment Flooding is required	1. RPV water level **cannot** be restored and maintained above -84 in. following depressurization of the RPV or RPV water level **cannot** be determined	1. RPV water level **cannot** be restored and maintained above -84 in. or RPV water level **cannot** be determined	None	None	1. Primary Containment Flooding is required
B **Primary Containm ent Pressure / Temp.**	None	None	2. Primary Containment pressure > 3.5 psig due to RCS leakage	None	1. Primary Containment pressure rise followed by a rapid UNPLANNED drop in Primary Containment pressure 2. Primary Containment pressure response **not** consistent with LOCA conditions	2. Torus pressure > 35 psig and rising 3. Explosive mixture exists inside Primary Containment (≥ 6% H_2 and ≥ 5% O_2) 4. Torus water temperature and RPV pressure **cannot** be maintained below the Heat Capacity Temperature Limit (N1-EOP-4 Figure M)
C **Isolation**	None	None	3. Release pathway exists outside Primary Containment resulting from isolation failure in **ANY** of the following (excluding normal process system flowpaths from an UNISOLABLE system): • Main steam line • EC steam line • RWCU • Feedwater 4. RPV blowdown is required	1. UNISOLABLE primary system leakage outside Primary Containment as indicated by exceeding **EITHER**: Area temperature above **ANY** N1-EOP-5 Detail T alarm setpoint **OR** Area radiation above **ANY** N1-EOP-5 Detail R alarm setpoint	3. Failure of **all** Primary Containment isolation valves in **ANY** one line to close following auto or manual initiation **AND** Direct downstream pathway outside Primary Containment and to the environment exists 4. Intentional Primary Containment venting per EOPs 5. UNISOLABLE primary system leakage outside Primary Containment as indicated by exceeding **EITHER**: Maximum safe general area temperature of 135ºF **OR** Maximum safe area radiation of 8 R/hr	None
D **Rad**	2. Drywell radiation ≥ 3,000 R/hr 3. Reactor coolant activity > 300 μCi/gm I-131 Equivalent	None	5. Drywell radiation ≥ 80 R/hr	None	None	5. Drywell radiation ≥ 4.0E4 R/hr
E **Judgment**	4. **ANY** condition in the opinion of the Emergency Director that *indicates loss of the Fuel Clad* barrier	2. **ANY** condition in the opinion of the Emergency Director that *indicates potential loss of the* Fuel Clad barrier	6. **ANY** condition in the opinion of the Emergency Director that *indicates loss of the Reactor* Coolant System barrier	2 **ANY** condition in the opinion of the Emergency Director that indicates *potential loss of the Reactor* Coolant System barrier	6. **ANY** condition in the opinion of the Emergency Director that *indicates loss of the Containment* barrier	6. **ANY** condition in the opinion of the Emergency Director that indicates *potential loss of the Containment* barrier

Barrier: Fuel Clad

Category: A. RPV Water Level

Degradation Threat: Loss

Threshold:

1. Primary Containment Flooding is required

Basis:

Plant-Specific

Requirements for Primary Containment Flooding are established in EOP-2 Step L-l8; EOP-3 Steps L-8, L-l0 and L-l3; and EOP-7 Overrides 3 and 18. These EOPs provide instructions to ensure adequate core cooling by maintaining RPV water level above prescribed limits or operating sufficient RPV injection sources when level cannot be determined. SAP entry is required when (ref. 1, 2. 3, 4):

- RPV water level cannot be restored and maintained above -109 in. with insufficient Core Spray flow: The Minimum Steam Cooling RPV Water Level (MSCRWL) is the lowest RPV water level at which the covered portion of the reactor core will generate sufficient steam to preclude any clad temperature in the uncovered portion of the core from exceeding 1500°F. Core spray flow is insufficient if you cannot restore and maintain both Core Spray loop flows at or above 180 x 10^4 lbm/hr. Consistent with the EOP definition of "cannot be restored and maintained," the determination that the parameter cannot be restored and maintained above the limit may be made at, before, or after the parameter actually decreases to this point.
- RPV water level cannot be determined and it is determined that core damage is occurring: When RPV water level cannot be determined, EOPs require RPV flooding strategies. RPV water level indication provides the primary means of knowing if adequate core cooling is being maintained. When all means of determining RPV water level are unavailable, reliance on alternate means of assuring adequate core cooling must be attempted. The instructions in EOP-7 specify these means, which include blowdown of the RPV and injection into the

RPV at a rate needed to flood to the elevation of the main steam lines or hold RPV pressure above the Minimum Steam Cooling Pressure (in ATWS events)

This threshold is also a Potential Loss of the Containment barrier (PC P-Loss A.1). Since SAP entry occurs after core uncovery has occurred, a Loss of the RCS barrier exists (RCS Loss A.1). Primary Containment Flooding (SAP entry), therefore, represents a Loss of two barriers and a Potential Loss of a third, which requires a General Emergency classification.

Generic

This site specific value corresponds to the level used in EOPs to indicate challenge of core cooling. This is the minimum value to assure core cooling without further degradation of the clad.

NMP1 Basis Reference(s):

1. NER-1M-095, NMP1 Emergency Operating Procedures (EOP) Basis Document
2. N1-EOP-2 RPV Control
3. N1-EOP-3 Failure to Scram
4. N1-EOP-7 RPV Flooding
5. NEI 99-01 FC Loss 2

Barrier: Fuel Clad

Category: B. Primary Containment Pressure / Temperature

Degradation Threat: Loss

Threshold:

None

Barrier: Fuel Clad

Category: C. Isolation

Degradation Threat: Loss

Threshold:

None

Barrier: Fuel Clad

Category: D. Rad

Degradation Threat: Loss

Threshold:

2. Drywell radiation ≥ 3,000 R/hr

Basis:

Plant-Specific

It is important to recognize that the radiation monitor may be sensitive to shine from the RPV or RCS piping (caused by lower than normal RPV water level for example). The Drywell High Range Radiation Monitors are the following (ref. 1):

- RAM 201.7-36 Located: Az 340° , El 263' 6"
- RAM 201.7-37 Located: Az 310° , El 301' 0"

The Drywell High Range Radiation Monitors have a range of 1E0 to 1E8 R/hr on recorder RR 201.7-36C pens 1 and 2 (ref. 1).

The threshold value (3,090 R/hr rounded to 3,000 R/hr) was calculated assuming the instantaneous release and dispersal of the reactor coolant noble gas and iodine inventory associated with a concentration of 300 µCi/gm I-131 Equivalent (or approximately 5% clad failure) into the drywell atmosphere (ref. 2).

Generic

The 3,000 R/hr reading is a value which indicates the release of reactor coolant, with elevated activity indicative of fuel damage, into the drywell.

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 threshold D.5. Thus, this threshold indicates a loss of both Fuel Clad barrier and RCS barrier that appropriately escalates the emergency classification level to a Site Area Emergency.

There is no Potential Loss threshold associated with this item.

NMP1 Basis Reference(s):

1. N1-RSP-10C The Use and Routine Calibration of the General Atomic High Range Gamma Radiation Monitoring System
2. Calculation 1H21C003, Rev. 0
3. NEI 99-01 FC Loss 4

Barrier: Fuel Clad

Category: D. Rad

Degradation Threat: Loss

Threshold:

3. Reactor coolant activity > 300 µCi/gm I-131 Equivalent

Basis:

Plant-Specific

None

Generic

The site specific value corresponds to 300 µCi/gm I-131 Equivalent. Assessment by the EAL Task Force indicates that 300 µCi/gm I-131 Equivalent coolant activity is well above that expected for iodine spikes and corresponds to less than 5% fuel clad damage. This amount of radioactivity indicates significant clad damage and thus the Fuel Clad Barrier is considered lost.

There is no Potential Loss threshold associated with this item.

NMP1 Basis Reference(s):

1. NEI 99-01 FC Loss 1

Barrier: Fuel Clad

Category: E. Judgment

Degradation Threat: Loss

Threshold:

4. **ANY** condition in the opinion of the Emergency Director that indicates loss of the Fuel Clad barrier

Basis:

Plant-Specific

The Emergency Director judgment threshold addresses any other factors relevant to determining if the Fuel Clad barrier is lost. Such a determination should include IMMINENT barrier degradation, barrier monitoring capability and dominant accident sequences.

- IMMINENT barrier degradation exists if the degradation will likely occur within two hours based on a projection of current safety system performance. The term "IMMINENT" refers to recognition of the inability to reach safety acceptance criteria before completion of all checks.
- Barrier monitoring capability is decreased if there is a loss or lack of reliable indicators. This assessment should include instrumentation operability concerns, readings from portable instrumentation and consideration of offsite monitoring results.
- Dominant accident sequences lead to degradation of all fission product barriers and likely entry to the EOPs. The Emergency Director should be mindful of the Loss of AC power (Station Blackout) and ATWS EALs to assure timely emergency classification declarations.

Generic

These This threshold addresses any other factors that are to be used by the Emergency Director in determining whether the Fuel Clad barrier is lost. In addition, the inability to monitor the barrier should also be incorporated in this threshold as a factor in Emergency Director judgment that the barrier may be considered lost.

NMP1 Basis Reference(s):

1. NEI 99-01 FC Loss 6

Barrier: Fuel Clad

Category: A. RPV Water Level

Degradation Threat: Potential Loss

Threshold:

1. RPV water level **cannot** be restored and maintained above -84 in. following depressurization of the RPV or RPV water level **cannot** be determined

Basis:

Plant-Specific

An RPV water level instrument reading of -84 in. (rounded from 84 7/16") indicates RPV water level is at the top of active fuel. When RPV water level is at or above the top of active fuel, the core is completely submerged. Core submergence is the most desirable means of core cooling. When RPV water level is below the top of active fuel following depressurization of the RPV (automatically, manually or by failure of the RCS barrier), the uncovered portion of the core must be cooled by less reliable means (i.e., spray cooling). If core uncovery is threatened, the EOPs specify alternate, more extreme, RPV water level control measures in order to restore and maintain adequate core cooling (ref. 1).

Consistent with the EOP definition of "cannot be restored and maintained," the determination that RPV water level cannot be restored and maintained above the top of active fuel may be made at, before, or after RPV water level actually decreases to this point. (ref. 1)

When RPV water level cannot be determined, EOPs require RPV flooding strategies. RPV water level indication provides the primary means of knowing if adequate core cooling is being maintained. When all means of determining RPV water level are unavailable, the fuel clad barrier is threatened and reliance on alternate means of assuring adequate core cooling must be attempted. The instructions in EOP-7 specify these means, which include emergency depressurization of the RPV and injection into the RPV at a rate needed to flood to the elevation of the main steam lines or hold RPV pressure above the Minimum Steam Cooling Pressure (in ATWS events). (ref. 2) If RPV water level cannot be determined with respect to the top of active fuel, a potential loss of the fuel clad barrier exists.

Note that EOP-3 may require intentional uncovery of the core and control of RPV water level between -84 in. and -109 in., the Minimum Steam Cooling RPV Water Level (MSCRWL) (ref. 3). Under these conditions, a high-power ATWS event exists and requires at least a Site Area Emergency classification in accordance with the ATWS/Criticality EALs.

Generic

The site specific RPV water level threshold is the same as the RCS barrier Loss threshold A.1 and corresponds to the RPV water level at the top of the active fuel. Thus, this threshold indicates a Potential Loss of the Fuel Clad barrier and a Loss of RCS barrier that appropriately escalates the emergency classification level to a Site Area Emergency. This threshold is considered to be exceeded when, as specified in the site specific EOPs, that RPV water cannot be restored and maintained above the specified level following depressurization of the RPV (either manually, automatically or by failure of the RCS barrier).

NMP1 Basis Reference(s):

1. NER-1M-095, NMP1 Emergency Operating Procedures (EOP) Basis Document
2. N1-EOP-7 RPV Flooding
3. N1-EOP-3 Failure to Scram
4. NEI 99-01 FC Potential Loss 2

Barrier: Fuel Clad

Category: B. Primary Containment Pressure / Temperature

Degradation Threat: Potential Loss

Threshold:

None

Barrier: Fuel Clad

Category: C. Isolation

Degradation Threat: Potential Loss

Threshold:

None

Barrier: Fuel Clad

Category: D. Rad

Degradation Threat: Potential Loss

Threshold:

None

Barrier: Fuel Clad

Category: E. Judgment

Degradation Threat: Potential Loss

Threshold:

2. **ANY** condition in the opinion of the Emergency Director that indicates potential loss of the Fuel Clad barrier

Basis:

Plant-Specific

The Emergency Director judgment threshold addresses any other factors relevant to determining if the Fuel Clad barrier is potentially lost. Such a determination should include IMMINENT barrier degradation, barrier monitoring capability and dominant accident sequences.

- IMMINENT barrier degradation exists if the degradation will likely occur within two hours based on a projection of current safety system performance. The term "IMMINENT" refers to recognition of the inability to reach safety acceptance criteria before completion of all checks.
- Barrier monitoring capability is decreased if there is a loss or lack of reliable indicators. This assessment should include instrumentation operability concerns, readings from portable instrumentation and consideration of offsite monitoring results.
- Dominant accident sequences lead to degradation of all fission product barriers and likely entry to the EOPs. The Emergency Director should be mindful of the Loss of AC power (Station Blackout) and ATWS EALs to assure timely emergency classification declarations.

Generic

This threshold addresses any other factors that are to be used by the Emergency Director in determining whether the Fuel Clad barrier is potentially lost. In addition, the inability to monitor the barrier should also be incorporated in this threshold as a factor in Emergency Director judgment that the barrier may be considered potentially lost.

NMP1 Basis Reference(s):

1. NEI 99-01 FC Potential Loss 6

Barrier: Reactor Coolant System

Category: A. RPV Water Level

Degradation Threat: Loss

Threshold:

1. RPV water level **cannot** be restored and maintained above -84 in. or RPV water level **cannot** be determined

Basis:

Plant-Specific

An RPV water level instrument reading of -84 in. (rounded from 84 7/16") indicates RPV water level is at the top of active fuel (ref. 1). The top of the active fuel is significantly lower than the normal operating RPV water level control band. To reach this level, RPV inventory loss would have previously required isolation of the RCS and Containment (PC) barriers, and initiation of all ECCS. If RPV water level cannot be maintained above the top of active fuel, ECCS and other sources of RPV injection have been ineffective or incapable of reversing the decreasing level trend. The cause of the loss of RPV inventory is therefore assumed to be a Loss of Coolant Accident (LOCA). By definition, a LOCA event is a Loss of the RCS barrier.

Consistent with the EOP definition of "cannot be restored and maintained," the determination that RPV water level cannot be restored and maintained above the top of active fuel may be made at, before, or after RPV water level actually decreases to this point. (ref. 1)

When RPV water level cannot be determined, EOPs require RPV flooding strategies. The RPV flooding instructions in EOP-7 first specify blowdown of the RPV (ref. 2), which is defined to be a Loss of the RCS barrier (RCS Loss C.4).

Note that EOP-3 may require intentional uncovery of the core and control of RPV water level between -84 in. and -109 in., the Minimum Steam Cooling RPV Water Level (MSCRWL) (ref. 3). Under these conditions, a high-power ATWS event exists and requires at least a Site Area Emergency classification in accordance with the ATWS/Criticality EALs.

Generic

The Loss threshold RPV water level of -84 in. corresponds to the level that is used in EOPs to indicate challenge of core cooling.

The threshold value is the same as Fuel Clad Barrier Potential Loss threshold A.1 and corresponds to a challenge to core cooling. Thus, this threshold indicates a Loss of RCS barrier and Potential Loss of Fuel Clad barrier that appropriately escalates the emergency classification level to a Site Area Emergency.

Unlike the Fuel Clad barrier RPV water level Potential Loss threshold (top of the active fuel), the additional requirement that the RPV be depressurized is not associated with the RCS barrier Potential Loss. The significant loss of inventory that must occur to determine that RPV water level cannot be restored and maintained above the threshold is, by itself, a very strong indication that the RCS barrier is no longer capable of retaining sufficient inventory to keep the core submerged, and thus represents a Loss of the RCS Barrier.

There is no Potential Loss threshold associated with this item.

NMP1 Basis Reference(s):

1. NER-1M-095, NMP1 Emergency Operating Procedures (EOP) Basis Document
2. N1-EOP-7 RPV Flooding
3. N1-EOP-3 Failure to Scram
4. NEI 99-01 RCS Loss 2

Barrier: Reactor Coolant System

Category: B. Primary Containment Pressure / Temperature

Degradation Threat: Loss

Threshold:

2. Primary Containment pressure > 3.5 psig due to RCS leakage

Basis:

Plant-Specific

The drywell high pressure scram setpoint is an entry condition to the EOP flowcharts: EOP-2, RPV Control, and EOP-4, Primary Containment Control (ref. 1, 2). Normal Primary Containment (PC) pressure control functions such as operation of drywell cooling and venting through RBEVS are specified in EOP-4 in advance of less desirable but more effective functions such as operation of drywell or suppression chamber sprays.

In the NMP1 design basis, Primary Containment pressures above the drywell high pressure scram setpoint are assumed to be the result of a high-energy release into the containment for which normal pressure control systems are inadequate or incapable of reversing the increasing pressure trend. Pressures of this magnitude, however, can be caused by non-LOCA events such as a loss of drywell cooling or inability to control Primary Containment vent/purge (ref. 3).

The threshold phrase "...due to RCS leakage" focuses the barrier failure on the RCS instead of the non-LOCA malfunctions that may adversely affect Primary Containment pressure. Primary Containment pressure greater than 3.5 psig with corollary indications (e.g., elevated drywell temperature, indications of loss of RCS inventory) should, therefore, be considered a Loss of the RCS barrier. Loss of drywell cooling that results in pressure greater than 3.5 psig should not be considered an RCS barrier loss.

Generic

The Primary Containment pressure of 3.5 psig is based on the drywell high pressure set point which indicates a LOCA by automatically initiating the ECCS or equivalent makeup system.

There is no Potential Loss threshold associated with this item.

NMP1 Basis Reference(s):

1. N1-EOP-2 RPV RPV Control
2. N1-EOP-4 Primary Containment Control
3. NER-1M-095, NMP1 Emergency Operating Procedures (EOP) Basis Document
4. NEI 99-01 RCS Loss 1

Barrier: Reactor Coolant System

Category: C. Isolation

Degradation Threat: Loss

Threshold:

3. Release pathway exists outside Primary Containment resulting from isolation failure in **ANY** of the following systems (excluding normal process system flowpaths from an UNISOLABLE system):
    - Main steam line
    - EC steam line
    - RWCU
    - Feedwater

Basis:

Plant-Specific

The conditions of this threshold include required containment isolation failures allowing a flow path to the environment. A release pathway outside Primary Containment exists when flow is not prevented by downstream isolations. Emergency declaration under this threshold would not be required in the case of a failure of both isolation valves to close but no downstream flowpath exists. Similarly, if the emergency response requires the normal process flow of a system outside Primary Containment (e.g., EOP requirement to bypass MSIV low RPV water level interlocks and maintain the main condenser as a heat sink using main turbine bypass valves), the threshold is not met. The combination of these threshold conditions represent the loss of both the RCS and Containment (see PC Loss C.3) barriers and justifies declaration of a Site Area Emergency (i.e., Loss or Potential Loss of any two barriers). (ref. 1-3)

Even though RWCU and Feedwater systems do not contain steam, they are included in the list because an UNISOLABLE break could result in the high-pressure discharge of fluid that is flashed to steam from relatively large volume systems directly connected to the RCS.

Generic

An UNISOLABLE MSL break is a breach of the RCS barrier. Thus, this threshold is included for consistency with the Alert emergency classification level.

Other large high-energy line breaks such as EC steam line, Feedwater or RWCU that are UNISOLABLE also represent a significant loss of the RCS barrier and should be considered as MSL breaks for purposes of classification.

NMP1 Basis Reference(s):

1. USAR Section VIII.A Protective Systems
2. USAR Section V.E Emergency Cooling System
3. USAR Section X.B Reactor Cleanup System
4. NEI 99-01 RCS Loss 3A

Barrier: Reactor Coolant System

Category: C. Isolation

Degradation Threat: Loss

Threshold:

4. RPV blowdown is required

Basis:

Plant-Specific

RPV blowdown (Emergency RPV Depressurization) is specified in the EOP flowcharts (EOP-8 RPV Blowdown) when symbols containing the phrase "BLOW DOWN" are reached. The requirements for emergency RPV depressurization appear in the following EOPs (ref. 1-7):

- EOP-2 RPV Control
- EOP-3 Failure to Scram
- EOP-4 Primary Containment Control
- EOP-4.2 Hydrogen Control
- EOP-5 Secondary Containment Control
- EOP-6 Radioactivity Release Control
- EOP-9 Steam Cooling

RPV blowdown (Emergency RPV depressurization) is also performed upon entry to EOP-7 (ref. 8).

Generic

Plant symptoms requiring Emergency RPV Depressurization (RPV blowdown) per the EOP flowcharts are indicative of a loss of the RCS barrier. If Emergency RPV depressurization is required, the plant operators are directed to open electromatic relief valves (ERVs) and keep them open. Even though the RCS is being vented into the suppression pool, a loss of the RCS should be considered to exist due to the diminished effectiveness of the RCS pressure barrier to a release of fission products beyond its boundary.

NMP1 Basis Reference(s):

1. N1-EOP-2 RPV Control
2. N1-EOP-3 Failure to Scram
3. N1-EOP-4 Primary Containment Control
4. N1-EOP-4.2 Hydrogen Control
5. N1-EOP-5 Secondary Containment Control
6. N1-EOP-6 Radioactivity Release Control
7. N1-EOP-9 Steam Cooling
8. N1-EOP-7 RPV Flooding
9. NEI 99-01 RCS Loss 3B

Barrier: Reactor Coolant System

Category: D. Rad

Degradation Threat: Loss

Threshold:

5. Drywell radiation ≥ 80 R/hr

Basis:

Plant-Specific

It is important to recognize that the radiation monitor may be sensitive to shine from the RPV or RCS piping (caused by lower than normal RPV water level for example). The Drywell High Range Radiation Monitors are the following (ref. 1):

- RAM 201.7-36 Located: Az 340° , El 263' 6"
- RAM 201.7-37 Located: Az 310° , El 301' 0"

The Drywell High Range Radiation Monitors have a range of 1E0 to 1E8 R/hr on recorder RR 201.7-36C pens 1 and 2 (ref. 1).

The threshold value was calculated assuming the instantaneous release and dispersal of the reactor coolant noble gas and iodine inventory associated with normal operating concentrations (i.e., Technical Specification coolant activity limit of 4 μCi/gm I-131 Equivalent) into the drywell atmosphere (ref. 2). The reading is less than that specified for the Fuel Clad Loss because no damage to the fuel clad is assumed. Only leakage from the RCS is assumed in this RCS Loss. The referenced calculation resulted in a threshold value of 88.5 R/hr. A value of 80 R/hr is selected because it is observable on existing instrumentation.

Generic

The 80 R/hr reading is a value which indicates the release of reactor coolant to the Primary Containment.

This reading will be less than that specified for Fuel Clad barrier Loss threshold D.2. Thus, this threshold would be indicative of a RCS leak only. If the radiation monitor reading increased to that value specified by Fuel Clad Barrier threshold, fuel damage would also be indicated.

There is no Potential Loss threshold associated with this item.

NMP1 Basis Reference(s):

1. N1-RSP-10C The Use and Routine Calibration of the General Atomic High Range Gamma Radiation Monitoring System
2. Calculation 1H21C003, Rev. 0 CCN 008846 Rev. 0
3. NEI 99-01 RCS Loss 4

Barrier: Reactor Coolant System

Category: E. Judgment

Degradation Threat: Loss

Threshold:

6. **ANY** condition in the opinion of the Emergency Director that indicates loss of the RCS barrier

Basis:

Plant-Specific

The Emergency Director judgment threshold addresses any other factors relevant to determining if the RCS barrier is lost. Such a determination should include IMMINENT barrier degradation, barrier monitoring capability and dominant accident sequences.

- IMMINENT barrier degradation exists if the degradation will likely occur within two hours based on a projection of current safety system performance. The term "IMMINENT" refers to the recognition of the inability to reach safety acceptance criteria before completion of all checks.
- Barrier monitoring capability is decreased if there is a loss or lack of reliable indicators. This assessment should include instrumentation operability concerns, readings from portable instrumentation and consideration of offsite monitoring results.
- Dominant accident sequences lead to degradation of all fission product barriers and likely entry to the EOPs. The Emergency Director should be mindful of the Loss of AC power (Station Blackout) and ATWS EALs to assure timely emergency classification declarations.

Generic

This threshold addresses any other factors that are to be used by the Emergency Director in determining whether the RCS barrier is lost. In addition, the inability to monitor the barrier should also be incorporated in this threshold as a factor in Emergency Director judgment that the barrier may be considered lost.

NMP1 Basis Reference(s):

1. NEI 99-01 RCS Loss 6

Barrier: Reactor Coolant System

Category: A. RPV Water Level

Degradation Threat: Potential Loss

Threshold:

None

Barrier: Reactor Coolant System

Category: B. Primary Containment Pressure / Temperature

Degradation Threat: Potential Loss

Threshold:

None

Barrier: Reactor Coolant System

Category: C. Isolation

Degradation Threat: Potential Loss

Threshold:

1. UNISOLABLE primary system leakage outside Primary Containment as indicated by exceeding **EITHER**:

   ANY N1-EOP-5 Detail T area temperature alarm setpoint

   OR

   ANY N1-EOP-5 Detail R *area radiation alarm* setpoint

Basis:

Plant-Specific

The presence of elevated general area temperatures or radiation levels in the Reactor Building (RB) may be indicative of UNISOLABLE primary system leakage outside the Primary Containment. When parameters reach the threshold level, equipment failure or misoperation may be occurring. Elevated parameters may also adversely affect the ability to gain access to or operate equipment within the affected area. (ref. 1, 2)

In general, multiple indications should be used to determine if a primary system is discharging outside Primary Containment. For example, a high area radiation condition does not necessarily indicate that a primary system is discharging into the secondary containment since this may be caused by radiation shine from nearby steam lines or the movement of radioactive materials. Conversely, a high area radiation condition in conjunction with other indications (e.g. room flooding, high area temperatures, reports of steam in the secondary containment, an unexpected rise in feedwater flowrate, or unexpected main turbine control valve closure) may indicate that a primary system is discharging into the secondary containment.

Generic

EOP-5 temperature alarm setpoints or area radiation alarm setpoints in the areas of the main steam line tunnel, main turbine generator, RCIC, etc., indicate a direct path from the RCS to areas outside Primary Containment.

The indicators reaching the threshold barriers and confirmed to be caused by RCS leakage warrant an Alert classification. An UNISOLABLE leak which is indicated by a high alarm setpoint escalates to a Site Area Emergency when combined with Containment Barrier Loss threshold C.5 (after a containment isolation) and a General Emergency when the Fuel Clad Barrier criteria is also exceeded.

NMP1 Basis Reference(s):

1. N1-EOP-5 Secondary Containment Control
2. NER-1M-095, NMP1 Emergency Operating Procedures (EOP) Basis Document
3. NEI 99-01 RCS Potential Loss 3B

Barrier: Reactor Coolant System

Category: D. Rad

Degradation Threat: Potential Loss

Threshold:

None

Barrier: Reactor Coolant System

Category: E. Judgment

Degradation Threat: Potential Loss

Threshold:

2. **ANY** condition in the opinion of the Emergency Director that indicates potential loss of the RCS barrier

Basis:

Plant-Specific

The Emergency Director judgment threshold addresses any other factors relevant to determining if the RCS barrier is potentially lost. Such a determination should include IMMINENT barrier degradation, barrier monitoring capability and dominant accident sequences.

- IMMINENT barrier degradation exists if the degradation will likely occur within two hours based on a projection of current safety system performance. The term "IMMINENT" refers to the inability to reach final safety acceptance criteria before completing all checks.
- Barrier monitoring capability is decreased if there is a loss or lack of reliable indicators. This assessment should include instrumentation operability concerns, readings from portable instrumentation and consideration of offsite monitoring results.
- Dominant accident sequences lead to degradation of all fission product barriers and likely entry to the EOPs. The Emergency Director should be mindful of the Loss of AC power (Station Blackout) and ATWS EALs to assure timely emergency classification declarations.

Generic

This threshold addresses any other factors that are to be used by the Emergency Director in determining whether the RCS barrier is potentially lost. In addition, the inability to monitor the barrier should also be incorporated in this threshold as a factor in Emergency Director judgment that the barrier may be considered potentially lost.

NMP1 Basis Reference(s):

1. NEI 99-01 RCS Potential Loss 6

Barrier: Containment

Category: A. RPV Water Level

Degradation Threat: Loss

Threshold:

None

Barrier: Containment

Category: B. Primary Containment Pressure / Temperature

Degradation Threat: Loss

Threshold:

1. Primary Containment pressure rise followed by a rapid UNPLANNED drop in Primary Containment pressure

Basis:

Plant-Specific

None

Generic

Rapid UNPLANNED loss of pressure (i.e., not attributable to drywell spray or condensation effects) following an initial pressure increase from a high energy line break indicates a loss of containment integrity. Primary Containment pressure should increase as a result of mass and energy release into containment from a LOCA. Thus, Primary Containment pressure not increasing under these conditions indicates a loss of containment integrity.

This indicator relies on operator recognition of an unexpected response for the condition and therefore does not have a specific value associated with it. The unexpected response is important because it is the indicator for a containment bypass condition.

NMP1 Basis Reference(s):

1. NEI 99-01 CMT Loss 1A

Barrier: Containment

Category: B. Primary Containment Pressure / Temperature

Degradation Threat: Loss

Threshold:

2. Primary Containment pressure response **not** consistent with LOCA conditions

Basis:

Plant-Specific

USAR Sections VI.A.2.2 and VI.B.1.2 provide a summary of Primary Containment pressure response for the design basis loss of coolant accident and the conditions resulting in the release of RCS inventory to the containment (ref. 1, 2). The maximum calculated drywell pressure is 34 psig (unless the large rupture was preceded by a small break that prepurged the drywell of nitrogen in which case the peak pressure could reach 50 psig). These pressures are well below the design allowable drywell pressure of 62 psig. (ref. 1, 2)

Due to conservatisms in LOCA analyses, actual pressure response is expected to be less than the analyzed response. LOCA conditions are manifested on Control Room instrumentation by drywell pressure rising with torus pressure following and eventually equalizing (around 22 psig for the DBA LOCA) (ref. 2).

Generic

Rapid UNPLANNED loss of pressure (i.e., not attributable to drywell spray or condensation effects) following an initial pressure increase from a high energy line break indicates a loss of containment integrity. Primary Containment pressure should increase as a result of mass and energy release into containment from a LOCA. Thus, Primary Containment pressure not increasing under these conditions indicates a loss of containment integrity.

This indicator relies on operator recognition of an unexpected response for the condition and therefore does not have a specific value associated with it. The unexpected response is important because it is the indicator for a containment bypass condition.

NMP1 Basis Reference(s):

1. USAR Section VI.A.2.2 Loss-of-Coolant-Accident
2. USAR Section VI.B.1.2 Design Basis Accident (DBA)

3. NEI 99-01 CMT Loss 1B

Barrier: Containment

Category: C. Isolation

Degradation Threat: Loss

Threshold:

3. Failure of **all** Primary Containment isolation valves in **ANY** one line to close following auto or manual initiation **AND** Direct downstream pathway outside Primary Containment and to the environment exists

Basis:

Plant-Specific

This threshold addresses failure of open isolation devices which should close upon receipt of a manual or automatic Primary Containment isolation signal resulting in a significant radiological release pathway directly to the environment. The concern is the UNISOLABLE open pathway to the environment. A failure of the ability to isolate any one line indicates a breach of Primary Containment integrity.

As stated above, the adjective "Direct" modifies "pathway" to discriminate against release paths through interfacing liquid systems. Leakage into a closed system is to be considered only if the closed system is breached and thereby creates a significant pathway to the environment. Examples include UNISOLABLE Main steam line or Feedwater line breaks, UNISOLABLE RWCU system breaks, and UNISOLABLE Primary Containment atmosphere vent paths. If the main condenser is available with an UNISOLABLE main steam line, there may be releases through the steam jet air ejectors and gland seal exhausters. These pathways are monitored, however, and do not meet the intent of a nonisolable release path to the environment. These minor releases are assessed using the Category R EALs.

The existence of an in–line charcoal filter (RBEVS) does not make a release path indirect since the filter is not effective at removing fission noble gases. Typical filters have an efficiency of 95-99% removal of iodine. Given the magnitude of the core inventory of iodine, significant releases could still occur. In addition, since the fission product release would be driven by boiling in the reactor vessel, the high humidity in the release stream can be expected to render the filters ineffective in a short period.

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

EOP-4, Primary Containment Control, and EOP-4.2, Hydrogen Control, may specify Primary Containment venting and intentional bypassing of the containment isolation valve logic even if offsite radioactivity release rate limits are exceeded (ref. 1, 2). Under these conditions with a VALID containment isolation signal, the Containment barrier should be considered lost.

Generic

These thresholds address incomplete containment isolation that allows direct release to the environment.

The use of the modifier "direct" in defining the release path discriminates against release paths through interfacing liquid systems. The existence of an in–line charcoal filter does not make a release path indirect since the filter is not effective at removing fission product noble gases. Typical filters have an efficiency of 95-99% removal of iodine. Given the magnitude of the core inventory of iodine, significant releases could still occur. In addition, since the fission product release would be driven by boiling in the reactor vessel, the high humidity in the release stream can be expected to render the filters ineffective in a short period.

NMP1 Basis Reference(s):

1. N1-EOP-4 Primary Containment Control
2. N1-EOP-4.2 Hydrogen Control
3. NEI 99-01 CMT Loss 3A

Barrier: Containment

Category: C. Isolation

Degradation Threat: Loss

Threshold:

4. Intentional Primary Containment venting per EOPs

Basis:

Plant-Specific

EOP-4, Primary Containment Control, and EOP-4.2, Hydrogen Control, may specify Primary Containment venting and intentional bypassing of the containment isolation valve logic, even if offsite radioactivity release rate limits are exceeded (ref. 1, 2). The threshold is met when the operator begins venting the Primary Containment in accordance with EOP-4.1 Primary Containment Venting , not when actions are taken to bypass interlocks prior to opening the vent valves (ref. 3). Purge and vent actions specified in EOP-1 Attachment 10 to control Primary Containment pressure below the drywell high pressure scram setpoint by venting through RBEVS do not meet this threshold because such action is only permitted if offsite radioactivity release rates will remain below the ODCM limits (ref. 1, 2).

Generic

These thresholds address incomplete containment isolation that allows direct release to the environment.

Site specific EOPs may direct containment isolation valve logic(s) to be intentionally bypassed, regardless of radioactivity release rates. Under these conditions with a VALID containment isolation signal, the containment should also be considered lost if containment venting is actually performed.

Intentional venting of Primary Containment for Primary Containment pressure or combustible gas control per EOPs to the secondary containment and/or the environment is considered a loss of containment. Containment venting for pressure when not in an accident situation should not be considered.

NMP1 Basis Reference(s):

1. N1-EOP-4 Primary Containment Control
2. N1-EOP-4.2 Hydrogen Control

3. N1-EOP-4.1 Primary Containment Venting
4. NEI 99-01 CMT Loss 3B

Barrier: Containment

Category: C. Isolation

Degradation Threat: Loss

Threshold:

5. UNISOLABLE primary system leakage outside Primary Containment as indicated by exceeding **EITHER**:

    Maximum safe general area temperature of 135°F

    OR

    Maximum safe area radiation of 8 R/hr

Basis:

Plant-Specific

The presence of elevated general area temperatures or radiation levels in the Reactor Building (RB) may be indicative of UNISOLABLE primary system leakage outside the Primary Containment. The EOP maximum safe values define this Containment barrier threshold because they are indicative of problems in the secondary containment that are spreading and pose a threat to achieving a safe plant shutdown. This threshold addresses problematic discharges outside Primary Containment that may not originate from a high-energy line break. The locations into which the primary system discharge is of concern correspond to the areas addressed in EOP-5 Detail S (ref. 1).

A "Maximum Safe Value" is the highest value at which equipment necessary for the safe shutdown of the plant will operate and personnel can perform any actions necessary for the safe shutdown of the plant (ref. 2).

The maximum safe value for temperature is 135°F.

The maximum safe value for radiation is 8 R/hr.

In general, multiple indications should be used to determine if a primary system is discharging outside Primary Containment. For example, a high area radiation condition does not necessarily indicate that a primary system is discharging into the secondary containment since this may be caused by radiation shine from nearby steam lines or the movement of radioactive materials. Conversely, a high area radiation condition in conjunction with other indications (e.g. room flooding, high area temperatures, reports of steam in the secondary containment, an unexpected rise in feedwater flowrate, or unexpected main turbine control valve closure) may indicate that a primary system is discharging into the secondary containment.

Generic

This threshold addresses incomplete containment isolation that allows direct release to the environment.

In addition, The presence of area radiation or temperature Maximum Safe Values indicating UNISOLABLE primary system leakage outside the Primary Containment are addressed after a containment isolation. The indicators should be confirmed to be caused by RCS leakage.

There is no Potential Loss threshold associated with this item.

NMP1 Reference(s):

1. N1-EOP-5 Secondary Containment Control
2. NER-1M-095, NMP1 Emergency Operating Procedures (EOP) Basis Document
3. NEI 99-01 CMT Loss 3C

Barrier: Containment

Category: D. Rad

Degradation Threat: Loss

Threshold:

None

Barrier: Containment

Category: E. Judgment

Degradation Threat: Loss

Threshold:

6. **ANY** condition in the opinion of the Emergency Director that indicates loss of the Containment barrier

Basis:

Plant-Specific

The Emergency Director judgment threshold addresses any other factors relevant to determining if the Containment barrier is lost. Such a determination should include IMMINENT barrier degradation, barrier monitoring capability and dominant accident sequences.

- IMMINENT barrier degradation exists if the degradation will likely occur within two hours based on a projection of current safety system performance. The term "IMMINENT" refers to recognition of the inability to reach safety acceptance criteria before completion of all checks.
- Barrier monitoring capability is decreased if there is a loss or lack of reliable indicators. This assessment should include instrumentation operability concerns, readings from portable instrumentation and consideration of offsite monitoring results.
- Dominant accident sequences lead to degradation of all fission product barriers and likely entry to the EOPs. The Emergency Director should be mindful of the Loss of AC power (Station Blackout) and ATWS EALs to assure timely emergency classification declarations.

Generic

This threshold addresses any other factors that are to be used by the Emergency Director in determining whether the Containment barrier is lost. In addition, the inability to monitor the barrier should also be incorporated in this threshold as a factor in Emergency Director judgment that the barrier may be considered lost.

The Containment barrier should not be declared lost based on exceeding Technical Specification action statement criteria, unless there is an event in progress requiring mitigation by the Containment barrier. When no event is in progress (Loss or Potential Loss of either Fuel Clad and/or RCS) the Containment barrier status is addressed by Technical Specifications.

NMP1 Basis Reference(s):

1. NEI 99-01 CMT Loss 6

Barrier: Containment

Category: A. RPV Water Level

Degradation Threat: Potential Loss

Threshold:

1. Primary Containment Flooding is required

Basis:

Plant-Specific

Requirements for Primary Containment Flooding are established in EOP-2 Step L-l8; EOP-3 Steps L-8, L-10 and L-13; and EOP-7 Override 3 and 18. These EOPs provide instructions to ensure adequate core cooling by maintaining RPV water level above prescribed limits or operating sufficient RPV injection sources when level cannot be determined. SAP entry is required when (ref. 1, 2. 3, 4):

- RPV water level cannot be restored and maintained above -109 in. with insufficient Core Spray flow: The Minimum Steam Cooling RPV Water Level (MSCRWL) is the lowest RPV water level at which the covered portion of the reactor core will generate sufficient steam to preclude any clad temperature in the uncovered portion of the core from exceeding 1500°F. Core spray flow is insufficient if you cannot restore and maintain both Core Spray loop flows at or above 180 x 10^4 lbm/hr. Consistent with the EOP definition of "cannot be restored and maintained," the determination that the parameter cannot be restored and maintained above the limit may be made at, before, or after the parameter actually decreases to this point.
- RPV water level cannot be determined and it is determined that core damage is occurring: When RPV water level cannot be determined, EOPs require RPV flooding strategies. RPV water level indication provides the primary means of knowing if adequate core cooling is being maintained. When all means of determining RPV water level are unavailable, reliance on alternate means of assuring adequate core cooling must be attempted. The instructions in EOP-7 specify these means, which include blowdown of the RPV and injection into the

RPV at a rate needed to flood to the elevation of the main steam lines or hold RPV pressure above the Minimum Steam Cooling Pressure (in ATWS events)

This threshold is also a Loss of the Fuel Clad barrier (FC Loss A.1). Since Primary Containment Flooding occurs after core uncovery has occurred a Loss of the RCS barrier exists (RCS Loss A.1). Primary Containment Flooding (SAP entry), therefore, represents a Loss of two barriers and a Potential Loss of a third, which requires a General Emergency classification.

Generic

There is no Loss threshold associated with this item.

The potential loss requirement for drywell flooding indicates adequate core cooling cannot be established and maintained and that core melt is possible. Entry into Primary Containment Flooding procedures (SAPs) is a logical escalation in response to the inability to maintain adequate core cooling.

The condition in this potential loss threshold represents a potential core melt sequence which, if not corrected, could lead to vessel failure and increased potential for containment failure. In conjunction with Reactor Vessel water level "Loss" thresholds in the Fuel Clad and RCS barrier columns, this threshold will result in the declaration of a General Emergency -- loss of two barriers and the potential loss of a third.

NMP1 Basis Reference(s):

1. NER-1M-095, NMP1 Emergency Operating Procedures (EOP) Basis Document
2. N1-EOP-2 RPV Control
3. N1-EOP-3 Failure to Scram
4. N1-EOP-7 RPV Flooding
5. NEI 99-01 CMT Potential Loss 2

Barrier: Containment

Category: B. Primary Containment Pressure / Temperature

Degradation Threat: Potential Loss

Threshold:

2. Torus pressure > 35 psig and rising

Basis:

Plant-Specific

The internal design pressure of the primary containment is identified by two pressures, a drywell pressure of 62 psig and a torus pressure of 35 psig. The more limiting of the two pressures defines this potential loss threshold. If this threshold is exceeded, a challenge to the Primary Containment structure has occurred because assumptions used in the accident analysis are no longer VALID and an unanalyzed condition exists (ref. 1). This constitutes a Potential Loss of the Containment barrier even if a containment breach has not occurred.

Generic

The torus pressure of 35 psig is based on the torus internal design pressure.

NMP1 Basis Reference(s):

1. USAR Section VI.B.1.2 Design Basis Accident (DBA)
2. NEI 99-01 CMT Potential Loss 1A

Barrier: Containment

Category: B. Primary Containment Pressure / Temperature

Degradation Threat: Potential Loss

Threshold:

3. Explosive mixture exists inside Primary Containment (≥ 6% H_2 and ≥ 5% O_2)

Basis:

Plant-Specific

Explosive (deflagration) mixtures in the Primary Containment are assumed to be elevated concentrations of hydrogen and oxygen. BWR industry evaluation of hydrogen generation for development of EOPs/SAPs indicates that any hydrogen concentration above minimum detectable is not to be expected within the short term. Post-LOCA hydrogen generation primarily caused by radiolysis is a slowly evolving, long-term condition. Hydrogen concentrations that rapidly develop are most likely caused by metal-water reaction. A metal-water reaction is indicative of an accident more severe than accidents considered in the plant design basis and would be indicative, therefore, of a potential threat to Primary Containment integrity. Hydrogen concentration of approximately 6% is considered the global deflagration concentration limit (ref. 1).

Except for brief periods during plant startup and shutdown, oxygen concentration in the Primary Containment is maintained at insignificant levels by nitrogen inertion. The specified values for this Potential Loss threshold are the minimum global deflagration concentration limits (6% hydrogen and 5% oxygen, ref. 1) and readily recognizable because 6% hydrogen is well above the EOP-4.2 entry condition (ref. 2). The minimum global deflagration hydrogen/oxygen concentrations (6%/5%, respectively) require intentional Primary Containment venting, which is defined to be a Loss of Containment (PC Loss C.4).

If the hydrogen or oxygen monitor is unavailable, sampling and analysis may determine gas concentrations. The validity of sample results must be judged based upon plant conditions, since drawing and analyzing samples may take some time. If sample results cannot be relied upon and hydrogen concentrations cannot be determined by any other means, the concentrations must be considered "unknown." The monitors should not be considered "unavailable" until an attempt has been made to place them in service. (ref. 1)

Generic

BWRs specifically define the limits associated with explosive mixtures in terms of deflagration concentrations of hydrogen and oxygen.

NMP1 Basis Reference(s):

1. NER-1M-095, NMP1 Emergency Operating Procedures (EOP) Basis Document
2. N1-EOP-4.2 Hydrogen Control
3. NEI 99-01 CMT Potential Loss 1B

Barrier: Containment

Category: B. Primary Containment Pressure / Temperature

Degradation Threat: Potential Loss

Threshold:

4. Torus water temperature and RPV pressure **cannot** be maintained below the Heat Capacity Temperature Limit (N1-EOP-4 Figure M)

Basis:

Plant-Specific

The Heat Capacity Temperature Limit (HCTL) is given in N1-EOP-4 Figure M. This threshold is met when N1-EOP-4 Step TT-5 is reached and RPV blowdown is required (ref. 1).

Generic

The Heat Capacity Temperature Limit (HCTL) is the highest torus water temperature from which Emergency RPV Depressurization will not raise:

- Torus temperature above the design value (205°F),

  OR

- Torus pressure above Primary Containment Pressure Limit, before the rate of energy transfer from the RPV to the containment is greater than the capacity of the containment vent.

The HCTL is a function of RPV pressure and suppression pool water level. It is utilized to preclude failure of the containment and equipment in the containment necessary for the safe shutdown of the plant and therefore, the inability to maintain plant parameters below the limit constitutes a potential loss of Containment.

NMP1 Basis Reference(s):

1. N1-EOP-4 Primary Containment Control
2. NEI 99-01 CMT Potential Loss 1C

Barrier: Containment

Category: D. Rad

Degradation Threat: Potential Loss

Threshold:

5. Drywell radiation ≥ 4.0E4 R/hr

Basis:

Plant-Specific

It is important to recognize that the radiation monitor may be sensitive to shine from the RPV or RCS piping (caused by lower than normal RPV water level for example). The Drywell High Range Radiation Monitors are the following (ref. 1):

- RAM 201.7-36 Located: Az 340° , El 263' 6"
- RAM 201.7-37 Located: Az 310° , El 301' 0"

The Drywell High Range Radiation Monitors have a range of 1E0 to 1E8 R/hr on recorder RR 201.7-36C pens 1 and 2 (ref. 1).

The threshold value (4.1E4 R/hr rounded to 4.0E4 R/hr) was calculated assuming the instantaneous release and dispersal of the reactor coolant noble gas and iodine inventory associated with 20% fuel clad damage into the drywell atmosphere (ref. 2).

Generic

The 4.0 E4 R/hr reading is a value that indicates significant fuel damage well in excess of that required for loss of RCS and Fuel Clad.

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.

There is no Loss threshold associated with this item.

NMP1 Basis Reference(s):

1. N1-RSP-10C The Use and Routine Calibration of the General Atomic High Range Gamma Radiation Monitoring System
2. Calculation 1H21C003, Rev. 0 CCN 008830, Rev. 0
3. NEI 99-01 CMT Potential Loss 4

Barrier: Containment

Category: E. Judgment

Degradation Threat: Potential Loss

Threshold:

6. **ANY** condition in the opinion of the Emergency Director that indicates potential loss of the Containment barrier

Basis:

Plant-Specific

The Emergency Director judgment threshold addresses any other factors relevant to determining if the Containment barrier is potentially lost. Such a determination should include IMMINENT barrier degradation, barrier monitoring capability and dominant accident sequences.

- IMMINENT barrier degradation exists if the degradation will likely occur within two hours based on a projection of current safety system performance. The term "IMMINENT" refers to recognition of the inability to reach safety acceptance criteria before completion of all checks.
- Barrier monitoring capability is decreased if there is a loss or lack of reliable indicators. This assessment should include instrumentation operability concerns, readings from portable instrumentation and consideration of offsite monitoring results.
- Dominant accident sequences lead to degradation of all fission product barriers and likely entry to the EOPs. The Emergency Director should be mindful of the Loss of AC power (Station Blackout) and ATWS EALs to assure timely emergency classification declarations.

Generic

This threshold addresses any other factors that are to be used by the Emergency Director in determining whether the Containment barrier is potentially lost. In addition, the inability to monitor the barrier should also be incorporated in this threshold as a factor in Emergency Director judgment that the barrier may be considered potentially lost.

The Containment barrier should not be declared potentially lost based on exceeding Technical Specification action statement criteria, unless there is an event in progress requiring mitigation by the Containment barrier. When no event is in progress (Loss or Potential Loss of either Fuel Clad and/or RCS) the Containment barrier status is addressed by Technical Specifications.

NMP1 Basis Reference(s):

1. NEI 99-01 CMT Potential Loss 6

NINE MILE POINT NUCLEAR STATION

EMERGENCY PLAN MAINTENANCE PROCEDURE

EPMP-EPP-0102

REVISION 00 (Draft RAI 1-9-12)

UNIT 2 EMERGENCY CLASSIFICATION TECHNICAL BASES

TECHNICAL SPECIFICATION REQUIRED

Approved by: ______________________________ ____________
J. Kaminski Director Emergency Planning Date

THIS IS A COMPLETE REVISION

Effective Date:________________

PERIODIC REVIEW DUE DATE:___________

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ABBEVIATIONS / ACRONYMS

AC .... Alternating Current
APRM .... Average Power Range Meter
ATWS .... Anticipated Transient Without Scram
BWR .... Boiling Water Reactor
CDE .... Committed Dose Equivalent
CFR .... Code of Federal Regulations
DC .... Direct Current
EAL .... Emergency Action Level
ECCS .... Emergency Core Cooling System
ED .... Emergency Director
el .... elevation
EOF .... Emergency Operations Facility
EOP .... Emergency Operating Procedure
EPA .... Environmental Protection Agency
EPMP .... Emergency Plan Maintenance Procedure
EPRI .... Electric Power Research Institute
FAA .... Federal Aviation Administration
FBI .... Federal Bureau of Investigation
FEMA .... Federal Emergency Management Agency
GE .... General Emergency
GTS .... Standby Gas Treatment System
HCTL .... Heat Capacity Temperature Limit
HOO .... Headquarters (NRC) Operations Officer
HPCS .... High Pressure Core Spray
IC .... Initiating Condition
ISFSI .... INDEPENDENT SPENT FUEL STORAGE INSTALLATION
JAFNPP .... James A. FitzPatrick Nuclear Power Plant
LCO .... Limiting Condition of Operation
LOCA .... Loss of Coolant Accident
MSIV .... Main Steam Isolation Valve
MSL .... Main Steam Line
mR .... milliRoentgen
MSCP .... Minimum Steam Cooling Pressure
MSCRWL .... Minimum Steam Cooling RPV Water Level
MSIV .... Main Steam Isolation Valve
MSL .... Main Steam Line
NEI .... Nuclear Energy Institute
NESP .... National Environmental Studies Project
NRC .... Nuclear Regulatory Commission

ACRONYMS & ABBREVIATIONS (continued)

NORAD	North American Aerospace Defense Command
NUMARC	Nuclear Management and Resources Council
OBE	Operating Basis Earthquake
ODCM	Off-site Dose Calculation Manual
ORO	Off-site Response Organization
PAG	Protective Action Guideline
PC	Primary Containment
PGCC	Power Generator Control Complex
PSIG	Pounds per Square Inch Gauge
R	Roentgen
RB	Reactor Building
RCIC	Reactor Core Isolation Cooling
RCS	Reactor Coolant System
RE	Radiation Element
Rem	Roentgen Equivalent Man
RHR	Residual Heat Removal
RMS	Radiation Monitoring System
RPS	Reactor Protection System
RPV	Reactor Pressure Vessel
RWCU	Reactor Water Cleanup
SAE	Site Area Emergency
SC	Secondary Containment
SPDS	Safety Parameter Display System
TEDE	Total Effective Dose Equivalent
TSC	Technical Support Center
UE	Unusual Event
USAR	Updated Safety Analysis Report

1.0 PURPOSE

This document provides an explanation and rationale for each Emergency Action Level (EAL) included in the EAL Upgrade Project for Nine Mile Point Nuclear Station Unit 2 (NMP2). It should be used to facilitate review of the NMP2 EALs and provide historical documentation for future reference. Decision-makers responsible for implementation of EPIP-EPP-02, "Classification of Emergency Conditions at Unit 2," and the Emergency Action Level Matrices, may use this document as a technical reference in support of EAL interpretation. This information may assist the Emergency Director in making classifications, particularly those involving judgment or multiple events. The basis information may also be useful in training, for explaining event classifications to offsite officials, and facilitates regulatory review and approval of the classification scheme.

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

2.0 DISCUSSION

2.1 Background

EALs are the plant-specific indications, conditions or instrument readings that are utilized to classify emergency conditions defined in the Nine Mile Point Site Emergency Plan.

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

NEI 99-01 (NUMARC/NESP-007) Revision 4 was subsequently issued for industry implementation. Enhancements over earlier revisions included:

- Consolidating the system malfunction initiating conditions and example emergency action levels which address conditions that may be postulated to occur during plant shutdown conditions.
- Initiating conditions and example emergency action levels that fully address conditions that may be postulated to occur at permanently Defueled Stations and INDEPENDENT SPENT FUEL STORAGE INSTALLATIONs (ISFSIs).

- Simplifying the fission product barrier EAL threshold for a Site Area Emergency.

Subsequently, Revision 5 of NEI 99-01 has been issued which incorporates resolutions to numerous implementation issues including the NRC EAL FAQs. Using NEI 99-01 Revision 5 Final, February 2008 (ADAMS Accession Number ML080450149), NMP2 conducted an EAL implementation upgrade project that produced the EALs discussed herein.

2.2 Fission Product Barriers

Many of the EALs derived from the NEI methodology are fission product barrier based. That is, the conditions that define the EALs are based upon loss or potential loss of one or more of the three fission product barriers. "Loss" and "Potential Loss" signify the relative damage and threat of damage to the barrier. "Loss" means the barrier no longer assures containment of radioactive materials; "potential loss" implies an increased probability of barrier loss and decreased certainty of maintaining the barrier.

The primary fission product barriers are:

A. Fuel Clad (FC): Zirconium tubes which house the ceramic uranium oxide pellets along with the end plugs which are welded into each end of the fuel rods comprise the FC barrier.

B. Reactor Coolant System (RCS): The reactor vessel shell, vessel head, CRD housings, vessel nozzles and penetrations, and all primary systems directly connected to the RPV up to the outermost Primary Containment isolation valve comprise the RCS barrier.

C. Containment (PC): The drywell, the suppression chamber/pool, their respective interconnecting paths, and other connections up to and including the outermost containment isolation valves comprise the Primary Containment barrier.

2.3 Emergency Classification Based on Fission Product Barrier Degradation

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

Unusual Event:

Any loss or any potential loss of Containment

Alert:

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

Site Area Emergency:

Loss or potential loss of any two barriers

General Emergency:

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

2.4 EAL Relationship to EOPs

Where possible, the EALs have been made consistent with and utilize the conditions defined in the NMP2 Emergency Operating Procedures (EOPs). While the symptoms that drive operator actions specified in the EOPs are not indicative of all possible conditions which warrant emergency classification, they define the symptoms, independent of initiating events, for which reactor plant safety and/or fission product barrier integrity are threatened. When these symptoms are clearly representative of one of the NEI Initiating Conditions, they have been utilized as an EAL. This permits rapid classification of emergency situations based on plant conditions without the need for additional evaluation or event diagnosis. Although some of the EALs presented here are based on conditions defined in the EOPs, classification of emergencies using these EALs is not dependent upon EOP entry or execution. The EALs can be utilized independently or in conjunction with the EOPs.

2.5 Symptom-Based vs. Event-Based Approach

To the extent possible, the EALs are symptom-based. That is, the action level threshold is defined by values of key plant operating parameters that identify emergency or potential emergency conditions. This approach is appropriate because it allows the full scope of variations in the types of events to be classified as emergencies. However, a purely symptom-based approach is not sufficient to address all events for which emergency classification is appropriate. Particular events to which no predetermined symptoms can be ascribed have also been utilized as EALs since they may be indicative of potentially more serious conditions not yet fully realized.

2.6 EAL Organization

The NMP2 EAL scheme includes the following features:

- Division of the EAL set into three broad groups:

- EALs applicable under <u>all</u> plant operating modes – This group would be reviewed by the EAL-user any time emergency classification is considered.
- EALs applicable only under <u>hot</u> operating modes – This group would only be reviewed by the EAL-user when the plant is in Hot Shutdown, Startup or Power Operation mode.
- EALs applicable only under <u>cold</u> operating modes – This group would only be reviewed by the EAL-user when the plant is in Cold Shutdown, Refuel or Defueled mode.

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

- Within each of the above three groups, assignment of EALs to categories/subcategories – Category and subcategory titles are selected to represent conditions that are operationally significant to the EAL-user. Subcategories are used as necessary to further divide the EALs of a category into logical sets of possible emergency classification thresholds. The NMP2 EAL categories/subcategories and their relationship to NEI 99-01 Rev. 5 Recognition Categories are listed below.

EAL Groups, Categories and Subcategories

EAL Group/Category	EAL Subcategory
Any Operating Mode:	
R – Abnormal Radiation Levels / Radiological Effluents	1 – Offsite Rad Conditions 2 – Onsite Rad Conditions & Spent Fuel Events 3 – CR/CAS Rad
H – Hazards and Other Conditions Affecting Plant Safety	1 – Natural or Destructive Phenomena 2 – FIRE or EXPLOSION 3 – Hazardous Gas 4 – Security 5 – Control Room Evacuation 6 – Judgment
E – ISFSI	None
Cold Conditions:	
C – Cold Shutdown / Refueling System Malfunction	1 – Loss of AC Power 2 – Loss of DC Power 3 – RPV Level 4 – RCS Temperature 5 – Inadvertent Criticality 6 – Communications
Hot Conditions:	
S – System Malfunction	1 – Loss of AC Power 2 – Loss of DC Power 3 – Criticality & RPS Failure 4 – Inability to Reach or Maintain Shutdown Conditions 5 – Instrumentation 6 – Communications 7 – Fuel Clad Degradation 8 – RCS Leakage
F – Fission Product Barrier Degradation	None

The primary tool for determining the emergency classification level is the EAL Classification Matrix. The user of the EAL Classification Matrix may (but is not required to) consult the EAL Technical Bases Document in order to obtain additional information concerning the EALs under classification consideration. The user should consult Sections 2.7 and 2.8, and Attachments 1 and 2 of this document for such information.

2.7 Technical Bases Information

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

<u>Category Letter & Title</u>

<u>Subcategory Number & Title</u>

<u>Initiating Condition (IC)</u>

Site-specific description of the generic IC given in NEI 99-01 Rev. 5.

<u>EAL Identifier (enclosed in rectangle)</u>

Each EAL is assigned a unique identifier to support accurate communication of the emergency classification to onsite and offsite personnel. Four characters define each EAL identifier:

1. First character (letter): Corresponds to the EAL category as described above (R, H, E, C, S or F)
2. Second character (letter): The emergency classification (G, S, A or U)

   G = General Emergency

   S = Site Area Emergency

   A = Alert

   U = Unusual Event
3. Third character (number): Subcategory number within the given category. Subcategories are sequentially numbered beginning with the number one (1). If a category does not have a subcategory, this character is assigned the number one (1).

4. Fourth character (number): The numerical sequence of the EAL within the EAL subcategory. If the subcategory has only one EAL, it is given the number one (1).

Classification (enclosed in rectangle):

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

EAL (enclosed in rectangle)

Wording enclosed in the rectangle appears as it is displayed in the EAL Classification Matrix. Selected terms are highlighted for emphasis:

- Bold, uppercase print is assigned to: "**ANY**," EAL identifiers, and logic terms such as **AND**, **OR**, **EITHER**, etc. (When used as conjunctions, the words "and" and "or" are not highlighted.)
- Bold, mixed case print is assigned to: "**all**," "**only**," "**both**," table titles and column headings, numbers following the word "**ANY**," and negative terms (e.g., "**not**," "**cannot**," etc.)
- Uppercase print is assigned to acronyms, abbreviations, and terms defined in Section 4.0.

Mode Applicability

One or more of the following plant operating conditions comprise the mode to which each EAL is applicable: 1 - Power Operation, 2 - Startup, 3 - Hot Shutdown, 4 - Cold Shutdown, 5 - Refuel, D - Defueled, or All. (See Section 2.8 for operating mode definitions.)

Basis:

A Generic basis section provides a description of the rationale for the EAL as provided in NEI 99-01 Rev. 5. This is followed by a Plant-Specific basis section that provides NMP2-relevant information concerning the EAL.

NMP2 Basis Reference(s):

Site-specific source documentation from which the EAL is derived

2.8 Operating Mode Applicability (Technical Specifications Table 1.1-1)

1 Power Operation

Reactor mode switch is in RUN

2 Startup

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

3 Hot Shutdown

The mode switch is in SHUTDOWN, average reactor coolant temperature is > 200°F, and all reactor vessel head closure bolts are fully tensioned

4 Cold Shutdown

The mode switch is in SHUTDOWN, average reactor coolant temperature is ≤ 200°F, and all reactor vessel head closure bolts are fully tensioned

5 Refuel

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

D Defueled

All reactor fuel is removed from the RPV (full core off load during refueling or extended outage)

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

2.9 Validation of Indications, Reports and Conditions

All emergency classifications shall be based upon VALID indications, reports or conditions. An indication, report, or condition, is considered to be VALID when it is verified by (1) an instrument channel check, or (2) indications on related or redundant indicators, or (3) by direct observation by plant personnel, such that doubt related to the indicator's operability, the condition's existence, or the report's accuracy is removed. Implicit in this definition is the need for timely assessment.

2.10 Planned vs. UNPLANNED Events

Planned evolutions involve preplanning to address the limitations imposed by the condition, the performance of required surveillance testing, and the implementation of specific controls prior to knowingly entering the condition in accordance with the specific requirements of the site's Technical Specifications. Activities which cause the site to operate beyond that allowed by the site's Technical Specifications, planned or UNPLANNED, may result in an EAL threshold being met or exceeded. Planned evolutions to test, manipulate, repair, perform maintenance or modifications to systems and equipment that result in an EAL value being met or exceeded are not subject to classification and activation requirements as long as the evolution proceeds as planned and is within the operational limitations imposed by the specific operating license. However, these conditions may be subject to the reporting requirements of 10 CFR 50.72.

2.11 Classifying Transient Events

For some events, the condition may be corrected before a declaration has been made. The key consideration in this situation is to determine whether or not further plant damage occurred while the corrective actions were being taken. In some situations, this can be readily determined, in other situations, further analyses may be necessary (e.g., coolant radiochemistry following an ATWS event, plant structural examination following an *earthquake, etc.). Classify the event as indicated and terminate the emergency once* assessment shows that there were no consequences from the event and other termination criteria are met.

Existing guidance for classifying transient events addresses the period of time of event recognition and classification (15 minutes). However, in cases when EAL declaration criteria may be met momentarily during the normal expected response of the plant, declaration requirements should not be considered to be met when the conditions are a part of the designed plant response, or result from appropriate Operator actions.

There may be cases in which a plant condition that exceeded an EAL was not recognized at the time of occurrence but is identified well after the condition has occurred (e.g., as a result of routine log or record review), and the condition no longer exists. In these cases, an emergency should not be declared. Reporting requirements of 10 CFR 50.72 are applicable and the guidance of NUREG-1022, Event Reporting Guidelines 10 CFR 50.72 and 50.73, should be applied.

2.12 Multiple Simultaneous Events and IMMINENT EAL Thresholds

When multiple simultaneous events occur, the emergency classification level is based on the highest EAL reached. For example, two Alerts remain in the Alert category. Or, an Alert and a Site Area Emergency is a Site Area Emergency. Further guidance is provided in RIS 2007-02, Clarification of NRC Guidance for Emergency Notifications During Quickly Changing Events.

Since NMP2 is at a multi-unit site, emergency classification level upgrading must also consider the effects of a loss of a common system on more than one unit (e.g., potential for radioactive release from more than one core).

Although the majority of the EALs provide very specific thresholds, the Emergency Director (ED) must remain alert to events or conditions that lead to the conclusion that exceeding the EAL threshold is IMMINENT. If, in the judgment of the ED, an IMMINENT situation is at hand, the classification should be made as if the threshold has been exceeded. While this is particularly prudent at the higher emergency classes (the early classification may permit more effective implementation of protective measures), it is nonetheless applicable to all emergency classes.

2.13 Emergency Classification Level Downgrading

Another important aspect of usable EAL guidance is the consideration of what to do when the risk posed by an emergency is clearly decreasing. A combination approach involving recovery from General Emergencies and some Site Area Emergencies and termination from Unusual Events, Alerts, and certain Site Area Emergencies causing no long term plant damage appears to be the best choice. Downgrading to lower emergency classification levels adds notifications but may have merit under certain circumstances.

3.0 REFERENCES

3.1 Developmental

3.1.1 NEI 99-01 Rev. 5 Final, Methodology for Development of Emergency Action Levels, February 2008, ADAMS Accession Number ML080450149

3.1.2 NRC Regulatory Issue Summary (RIS) 2003-18, Supplement 2, Use of Nuclear Energy Institute (NEI) 99-01, Methodology for Development of Emergency Action Levels Revision 4, Dated January 2003 (December 12, 2005)

3.1.3 RIS 2007-02 Clarification of NRC Guidance for Emergency Notifications During Quickly Changing Events

3.1.4 Nine Mile Point Site Emergency Plan

3.2 Implementing

3.2.1 EPIP-EPP-02 Classification of Emergency Conditions at Unit 2

3.2.2 EAL Comparison Matrix

3.3 Commitments

None

4.0 DEFINITIONS (ref. 3.1.1 except as noted)

AFFECTING SAFE SHUTDOWN

Event in progress has adversely affected functions that are necessary to bring the plant to and maintain it in the applicable hot or cold shutdown condition. Plant condition applicability is determined by Technical Specification LCOs in effect.

Example 1: Event causes damage that results in entry into an LCO that requires the plant to be placed in hot shutdown. Hot shutdown is achievable, but cold shutdown is not. This event is not "AFFECTING SAFE SHUTDOWN."

Example 2: Event causes damage that results in entry into an LCO that requires the plant to be placed in cold shutdown. Hot shutdown is achievable, but cold shutdown is not. This event is "AFFECTING SAFE SHUTDOWN."

AIRLINER/LARGE AIRCRAFT

Any size or type of aircraft with the potential for causing significant damage to the plant (refer to the Security Plan for a more detailed definition).

BOMB

Refers to an explosive device suspected of having sufficient force to damage plant systems or structures.

CIVIL DISTURBANCE

A group of people violently protesting station operations or activities at the site.

CONFINEMENT BOUNDARY

The barrier(s) between areas containing radioactive substances and the environment.

CONTAINMENT CLOSURE

The procedurally defined actions taken to secure containment (primary or secondary) and its associated structures, systems, and components as a functional barrier to fission product release under existing plant conditions.

EXPLOSION

A rapid, violent, unconfined combustion, or catastrophic failure of pressurized/energized equipment that imparts energy of sufficient force to potentially damage permanent structures, systems, or components.

EXTORTION

An attempt to cause an action at the station by threat of force.

FIRE

Combustion characterized by heat and light. Sources of smoke such as slipping drive belts or overheated electrical equipment do not constitute FIREs. Observation of flame is preferred but is not required if large quantities of smoke and heat are observed.

HOSTAGE

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

HOSTILE ACTION

An act toward NMP2 or its personnel that includes the use of violent force to destroy equipment, take ~~FIREs~~HOSTAGEs, and/or intimidate the licensee to achieve an end. This includes attack by air, land, or water using guns, explosives, PROJECTILEs, vehicles, or other devices used to deliver destructive force. Other acts that satisfy the overall intent may be included.

HOSTILE ACTION should not be construed to include acts of civil disobedience or felonious acts that are not part of a concerted attack on NMP2. Non-terrorism-based EALs should be used to address such activities, (e.g., violent acts between individuals in the owner controlled area).

HOSTILE FORCE

One or more individuals who are engaged in a determined assault, overtly or by stealth and deception, equipped with suitable weapons capable of killing, maiming, or causing destruction.

IMMINENT

Mitigation actions have been ineffective, additional actions are not expected to be successful, and trended information indicates that the event or condition will occur. Where IMMINENT timeframes are specified, they shall apply.

INTACT

The RCS should be considered INTACT when the RCS pressure boundary is in its normal condition for the cold shutdown mode of operation (e.g., no freeze seals or nozzle dams).

INTRUSION

The act of entering without authorization. Discovery of a BOMB in a specified area is indication of INTRUSION into that area by a HOSTILE FORCE.

INDEPENDENT SPENT FUEL STORAGE INSTALLATION (ISFSI)

A complex that is designed and constructed for the interim storage of spent nuclear fuel and other radioactive materials associated with spent fuel storage.

NORMAL LEVELS

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

NORMAL PLANT OPERATIONS

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

PROJECTILE

An object directed toward NMP2 that could cause concern for its continued operability, reliability, or personnel safety.

PROTECTED AREA

The area which normally encompasses all controlled areas within the security PROTECTED AREA fence. NMP1 and NMP2 share a common PROTECTED AREA border. NMP1 and NMP2 PROTECTED AREA boundaries are illustrated in USAR Figure 1.2-1.

SABOTAGE

Deliberate damage, mis-alignment, or mis-operation of plant equipment with the intent to render the equipment inoperable. Equipment found tampered with or damaged due to malicious mischief may not meet the definition of SABOTAGE until this determination is made by security supervision.

SAFETY-RELATED STRUCTUREs, SYSTEMs and COMPONENTs (as defined in 10CFR50.2)

Those structures, systems and components that are relied upon to remain functional during and following design basis events to assure:

(1) The integrity of the reactor coolant pressure boundary;

(2) The capability to shut down the reactor and *maintain* it in a safe shutdown condition;

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

SECURITY CONDITION

Any security event as listed in the approved security contingency plan that constitutes a threat/compromise to site security, threat/risk to site personnel, or a potential degradation to the level of safety of the plant. A SECURITY CONDITION does not involve a HOSTILE ACTION.

SITE BOUNDARY

Per ODCM Figure D 1.0-1, the line around the Nine Mile Point Nuclear Station beyond which the land is not owned, leased or otherwise controlled by the owners and operators of Nine Mile Point Nuclear Station and James A. Fitzpatrick Nuclear Power Plant.

STRIKE ACTION

Work stoppage within the PROTECTED AREA by a body of workers to enforce compliance with demands made on NMP2. The STRIKE ACTION must threaten to interrupt NORMAL PLANT OPERATIONS.

UNISOLABLE

A breach or leak that cannot be promptly isolated.

UNPLANNED

A parameter change or an event, the reasons for which may be known or unknown, that is not the result of an intended evolution or expected plant response to a transient.

VALID

An indication, report, or condition, is considered to be VALID when it is verified by (1) an instrument channel check, or (2) indications on related or redundant indicators, or (3) by direct observation by plant personnel, such that doubt related to the indicator's operability, the condition's existence, or the report's accuracy is removed. Implicit in this definition is the need for timely assessment.

VISIBLE DAMAGE

Damage to equipment or structure that is readily observable without measurements, testing, or analysis. Damage is sufficient to cause concern regarding the continued operability or reliability of affected SAFETY-RELATED STRUCTURE, SYSTEM or COMPONENT. Example damage includes: deformation due to heat or impact, denting, penetration, rupture, cracking, paint blistering. Surface blemishes (e.g., paint chipping, scratches) should not be included.

VITAL AREA

Any areas, normally within the NMP2 PROTECTED AREA, that contains equipment, systems, components, or material, the failure, destruction, or release of which could directly or indirectly endanger the public health and safety by exposure to radiation.

5.0 NMP2-TO-NEI 99-01 EAL CROSSREFERENCE

This cross-reference is provided to facilitate association and location of a NMP2 EAL within the NEI 99-01 IC/EAL identification scheme. Further information regarding the development of the NMP2 EALs based on the NEI guidance can be found in the EAL Comparison Matrix.

NMP2	NEI 99-01	
EAL	IC	Example EAL
RG1.1	AG1	1
RG1.2	AG1	2
RG1.3	AG1	4
RS1.1	AS1	1
RS1.2	AS1	2
RS1.3	AS1	4
RA1.1	AA1	1
RA1.2	AA1	2
RA1.3	AA1	3
RU1.1	AU1	1
RU1.2	AU1	2
RU1.3	AU1	3
RA2.1	AA2	2
RA2.2	AA2	1
RU2.1	AU2	1
RU2.2	AU2	2
RA3.1	AA3	1
HA1.1	HA1	1
HA1.2	HA1	2

NMP2	NEI 99-01	
EAL	IC	Example EAL
HA1.3	HA1	3
HA1.4	HA1	4
HA1.5	HA1	6
HA1.6	HA1	5
HU1.1	HU1	1
HU1.2	HU1	2
HU1.3	HU1	3
HU1.4	HU1	4
HU1.5	HU1	5
HA2.1	HA2	1
HU2.1	HU2	1
HU2.2	HU2	2
HA3.1	HA3	1
HU3.1	HU3	1
HU3.2	HU3	2
HG4.1	HG1	1
HG4.2	HG1	2
HS4.1	HS4	1
HA4.1	HA4	1, 2
HU4.1	HU4	1, 2, 3
HS5.1	HS2	1
HA5.1	HA5	1
HG6.1	HG2	1
HS6.1	HS3	1
HA6.1	HA6	1

NMP2	NEI 99-01	
EAL	IC	Example EAL
HU6.1	HU5	1
EU1.1	E-HU1	1
CA1.1	CA3	1
CU1.1	CU3	1
CU2.1	CU7	1
CG3.1	CG1	1
CG3.2	CG1	2
CS3.1	CS1	1
CS3.2	CS1	2
CS3.3	CS1	3
CA3.1	CA1	1, 2
CU3.1	CU1	1
CU3.2	CU2	1
CU3.3	CU2	2
CA4.1	CA4	1, 2
CU4.1	CU4	1
CU4.2	CU4	2
CU5.1	CU8	1
CU6.1	CU6	1, 2
SG1.1	SG1	1
SS1.1	SS1	1
SA1.1	SA5	1
SU1.1	SU1	1
SS2.1	SS3	1
SG3.1	SG2	1

NMP2	NEI 99-01	
EAL	IC	Example EAL
SS3.1	SS2	1
SA3.1	SA2	1
SU3.1	SU8	1
SU4.1	SU2	1
SS5.1	SS6	1
SA5.1	SA4	1
SU5.1	SU3	1
SU6.1	SU6	1, 2
SU7.1	SU4	2
SU7.2	SU4	1
SU8.1	SU5	1, 2
FG1.1	FG1	1
FS1.1	FS1	1
FA1.1	FA1	1
FU1.1	FU1	1

6.0 ATTACHMENTS

6.1 Attachment 1, Emergency Action Level Technical Bases

6.2 Attachment 2, Fission Product Barrier Loss / Potential Loss Matrix and Basis

Category R – Abnormal Radiation Levels / Radiological Effluents

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

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

At lower levels, abnormal radioactivity releases may be indicative of a failure of containment systems or precursors to more significant releases. At higher release rates, offsite radiological conditions may result which require offsite protective actions. Elevated area radiation levels in plant may also be indicative of the failure of containment systems or preclude access to plant vital equipment necessary to ensure plant safety.

Events of this category pertain to the following subcategories:

1. Offsite Rad Conditions

Direct indication of effluent radiation monitoring systems provides a rapid assessment mechanism to determine releases in excess of classifiable limits. Projected offsite doses, actual offsite field measurements or measured release rates via sampling indicate doses or dose rates above classifiable limits.

2. Onsite Rad Conditions & Spent Fuel Events

Sustained general area radiation levels in excess of those indicating loss of control of radioactive materials or those levels which may preclude access to vital plant areas also warrant emergency classification.

3. CR/CAS Rad

Sustained general area radiation levels which may preclude access to areas requiring continuous occupancy also warrant emergency classification.

Category: R – Abnormal Radiation Levels / Radiological Effluents

Subcategory: 1 – Offsite Rad Conditions

Initiating Condition: Offsite dose resulting from an actual or IMMINENT release of gaseous radioactivity > 1,000 mRem TEDE or 5,000 mRem thyroid CDE for the actual or projected duration of the release using actual meteorology

EAL:

RG1.1 General Emergency

ANY monitor reading > Table R-1 "GE" column for ≥ 15 min. (Note 1)

- Do **not** delay declaration awaiting dose assessment results
- If dose assessment results are available, declaration should be based on dose assessment instead of radiation monitor values (see EAL RG1.2)

Note 1: The ED should **not** wait until the applicable time has elapsed, but should declare the event as soon as it is determined that the condition will likely exceed the applicable time

Table R-1 Effluent Monitor Classification Thresholds

Monitor	GE	SAE	Alert	UE
Gaseous				
Radwaste/RB Vent Effluent	5.5E+7 μCi/s	5.5E+6 μCi/s	200 x Alarm	2 x Alarm
Main Stack Effluent	1.0E+10 μCi/s	1.0E+9 μCi/s	200 x Alarm	2 x Alarm
Liquid				
Service Water Effluent	N/A	N/A	200 x DRMS High(red)	2 x DRMS High(red)
Liquid RadWaste Effluent	N/A	N/A	N/A	2 x DRMS High(red)
Cooling Tower Blowdown	N/A	N/A	200 x DRMS High(red)	2 x DRMS High(red)

Mode Applicability:

All

Basis:

Plant-Specific

The DRAGON computer code has been used to determine the threshold values in Table R-1 for the GE classification level. The methodology develops an isotopic concentration in the secondary containment that, when released through the Radwaste/RB Vent or the Main Stack, achieves 1,000 mRem TEDE or 5,000 mRem thyroid CDE at the SITE BOUNDARY. The nuclide inventory in the secondary containment was artificially created by postulating a source term in secondary containment based on main steam design isotopic distribution and adjusting the release rate from secondary containment until either the whole body or child thyroid dose limit at the SITE BOUNDARY is reached. This isotopic distribution is not intended to specify a particular accident as the initiating event.

Values have been calculated for the GEMs noble gas channel only since this is the reading that is readily available to the operator. Realistic, accident atmospheric dispersion (X/Q) factors have been applied. (ref. 1)

The SITE BOUNDARY is the line beyond which the land is not owned, leased, nor otherwise controlled by Constellation (ref. 2).

Liquid effluent radiation monitors are not addressed in Table R-1 at the Site Area Emergency and General Emergency levels because the dose assessment code used to calculate these Table R-1 readings only considers a release through the Radwaste/RB Vent or the Main Stack.

A radiation monitor reading is VALID when a release path is established. If the release path to the environment has been isolated, the radiation monitor reading is not VALID for classification.

<u>Generic</u>

This EAL addresses radioactivity releases that result in doses at or beyond the SITE BOUNDARY that exceed the EPA Protective Action Guides (PAGs). Public protective actions will be necessary. Releases of this magnitude are associated with the failure of plant systems needed for the protection of the public and likely involve fuel damage.

The monitor list in Table R-1 includes effluent monitors on all potential release pathways.

Since dose assessment is based on actual meteorology, whereas the monitor reading EAL is not, the results from these assessments may indicate that the classification is not warranted, or may indicate that a higher classification is warranted. For this reason, emergency implementing procedures should call for the timely performance of dose assessments using actual meteorology and release information. If the results of these dose assessments are available when the classification is made (e.g., initiated at a lower classification level), the dose assessment results override the monitor reading EAL.

NMP2 Basis Reference(s):

1. Calculation PR-C-24-X
2. NMP2 Offsite Dose Calculation Manual Figure D.1.0-1
3. NEI 99-01 IC AG1

Category: R – Abnormal Radiation Levels / Radiological Effluents

Subcategory: 1 – Offsite Rad Conditions

Initiating Condition: Offsite dose resulting from an actual or IMMINENT release of gaseous radioactivity > 1,000 mRem TEDE or 5,000 mRem thyroid CDE for the actual or projected duration of the release using actual meteorology

EAL:

RG1.2	**General Emergency**
Dose assessment using actual meteorology indicates doses > 1,000 mRem TEDE or 5,000 mRem thyroid CDE at or beyond the SITE BOUNDARY	

Mode Applicability:

All

Basis:

Plant-Specific

The 1,000 mRem TEDE dose is set at 100% of the EPA PAG, while the 5,000 mRem thyroid CDE was established in consideration of the 1:5 ratio of the EPA PAG for TEDE and thyroid CDE.

Dose assessment is performed in accordance with EPIP-EPP-08 "Offsite Dose Assessment and PAR" (ref. 1)

The SITE BOUNDARY is the line beyond which the land is not owned, leased, nor otherwise controlled by Constellation (ref. 2).

Generic

This EAL addresses radioactivity releases that result in doses at or beyond the SITE BOUNDARY that exceed the EPA Protective Action Guides (PAGs). Public protective actions will be necessary. Releases of this magnitude are associated with the failure of plant systems needed for the protection of the public and likely involve fuel damage.

Since dose assessment is based on actual meteorology, whereas the monitor reading EAL is not, the results from these assessments may indicate that the classification is not warranted, or may indicate that a higher classification is warranted. For this reason, emergency implementing procedures should call for the timely performance of dose assessments using actual meteorology and release information. If the results of these dose assessments are available when the classification is made (e.g., initiated at a lower classification level), the dose assessment results override the monitor reading EAL.

NMP2 Basis Reference(s):

Attachment 1 - Emergency Action Level Technical Bases

1. EPIP-EPP-08 Offsite Dose Assessment and PAR
2. NMP2 Offsite Dose Calculation Manual Figure D.1.0-1
3. NEI 99-01 IC AG1

Category: R – Abnormal Radiation Levels / Radiological Effluents

Subcategory: 1 – Offsite Rad Conditions

Initiating Condition: Offsite dose resulting from an actual or IMMINENT release of gaseous radioactivity > 1,000 mRem TEDE or 5,000 mRem thyroid CDE for the actual or projected duration of the release using actual meteorology

EAL:

RG1.3 **General Emergency**
Field survey results indicate closed window dose rates > 1,000 mRem/hr expected to continue for ≥ 60 min. at or beyond the SITE BOUNDARY (Note 1)
OR
Analyses of field survey samples indicate thyroid CDE > 5,000 mRem for 1 hr of inhalation at or beyond the SITE BOUNDARY (Note 1)

Note 1: The ED should **not** wait until the applicable time has elapsed, but should declare the event as soon as it is determined that the condition will likely exceed the applicable time

Mode Applicability:

All

Basis:

Plant-Specific

Real time field surveys and sample analysis is performed by offsite field monitoring teams per EPIP-EPP-07, "Downwind Radiological Monitoring" (ref. 1) and assessed for radiological dose consequences per EPIP-EPP-08 "Offsite Dose Assessment and PAR" (ref. 2).

The SITE BOUNDARY is the line beyond which the land is not owned, leased, nor otherwise controlled by Constellation (ref. 3).

Generic

This EAL addresses radioactivity releases that result in doses at or beyond the SITE BOUNDARY that exceed the EPA Protective Action Guides (PAGs). Public protective actions will be necessary. Releases of this magnitude are associated with the failure of plant systems needed for the protection of the public and likely involve fuel damage.

Since dose assessment is based on actual meteorology, whereas the monitor reading EAL is not, the results from these assessments may indicate that the classification is not warranted. For this reason, emergency implementing procedures should call for the timely performance of dose assessments using actual meteorology and release information. If the results of these dose assessments are available when the classification is made (e.g., initiated at a lower classification level), the dose assessment results override the monitor reading EAL.

NMP2 Basis Reference(s):

1. EPIP-EPP-07 Downwind Radiological Monitoring
2. EPIP-EPP-08 Offsite Dose Assessment and PAR
3. NMP2 Offsite Dose Calculation Manual Figure D.1.0-1
4. NEI 99-01 IC AG1

Category: R – Abnormal Radiation Levels / Radiological Effluents

Subcategory: 1 – Offsite Rad Conditions

Initiating Condition: Offsite dose resulting from an actual or IMMINENT release of gaseous radioactivity exceeds 100 mRem TEDE or 500 mRem thyroid CDE for the actual or projected duration of the release using actual meteorology

EAL:

RS1.1 Site Area Emergency

ANY monitor reading > Table R-1 "SAE" column for ≥ 15 min. (Note 1)

- Do **not** delay declaration awaiting dose assessment results
- If dose assessment results are available, declaration should be based on dose assessment instead of radiation monitor values (see EAL RS1.2)

Note 1: The ED should **not** wait until the applicable time has elapsed, but should declare the event as soon as it is determined that the condition will likely exceed the applicable time

Table R-1 Effluent Monitor Classification Thresholds

Monitor	GE	SAE	Alert	UE
<u>Gaseous</u>				
Radwaste/RB Vent Effluent	5.5E+7 µCi/s	5.5E+6 µCi/s	200 x Alarm	2 x Alarm
Main Stack Effluent	1.0E+10 µCi/s	1.0E+9 µCi/s	200 x Alarm	2 x Alarm
<u>Liquid</u>				
Service Water Effluent	N/A	N/A	200 x DRMS High(red)	2 x DRMS High(red)
Liquid RadWaste Effluent	N/A	N/A	N/A	2 x DRMS High(red)
Cooling Tower Blowdown	N/A	N/A	200 x DRMS High(red)	2 x DRMS High(red)

Mode Applicability:

All

Basis:

<u>Plant-Specific</u>

The DRAGON computer code has been used to determine the threshold values in Table R-1 for the SAE classification level. The methodology develops an isotopic concentration in the secondary containment that, when released through the Radwaste/RB Vent or the Main Stack, achieves 100 mRem TEDE or 500 mRem thyroid CDE at the SITE BOUNDARY. The nuclide inventory in the secondary containment was artificially created by postulating a source term in secondary containment based on main steam design isotopic distribution and adjusting the release rate from secondary containment until either the whole body or child thyroid dose limit at the SITE BOUNDARY is reached. This isotopic distribution is not intended to specify a particular accident as the initiating event.

Values have been calculated for the GEMs noble gas channel only since this is the reading that is readily available to the operator. Realistic, accident atmospheric dispersion (X/Q) factors have been applied. (ref. 1)

The SITE BOUNDARY is the line beyond which the land is not owned, leased, nor otherwise controlled by Constellation (ref. 2).

Liquid effluent radiation monitors are not addressed in Table R-1 at the Site Area Emergency and General Emergency levels because the dose assessment code used to calculate these Table R-1 readings only considers a release through the Radwaste/RB Vent or the Main Stack.

A radiation monitor reading is VALID when a release path is established. If the release path to the environment has been isolated, the radiation monitor reading is not VALID for classification.

Generic

This EAL addresses radioactivity releases that result in doses at or beyond the SITE BOUNDARY that exceed 10% of the EPA Protective Action Guides (PAGs). Releases of this magnitude are associated with the failure of plant systems needed for the protection of the public.

The site specific monitor list in Table R-1 includes effluent monitors on all potential release pathways.

Since dose assessment is based on actual meteorology, whereas the monitor reading EAL is not, the results from these assessments may indicate that the classification is not warranted, or may indicate that a higher classification is warranted. For this reason, emergency implementing procedures should call for the timely performance of dose assessments using actual meteorology and release information. If the results of these dose assessments are available when the classification is made (e.g., initiated at a lower classification level), the dose assessment results override the monitor reading EAL.

NMP2 Basis Reference(s):

1. Calculation PR-C-24-X
2. NMP2 Offsite Dose Calculation Manual Figure D.1.0-1
3. NEI 99-01 IC AS1

Category: R – Abnormal Radiation Levels / Radiological Effluents

Subcategory: 1 – Offsite Rad Conditions

Initiating Condition: Offsite dose resulting from an actual or IMMINENT release of gaseous radioactivity exceeds 100 mRem TEDE or 500 mRem thyroid CDE for the actual or projected duration of the release using actual meteorology

EAL:

RS1.2	**Site Area Emergency**
Dose assessment using actual meteorology indicates doses > 100 mRem TEDE or 500 mRem thyroid CDE at or beyond the SITE BOUNDARY	

Mode Applicability:

All

Basis:

Plant-Specific

The 100 mRem TEDE dose is set at 10% of the EPA PAG, while the 500 mRem thyroid CDE was established in consideration of the 1:5 ratio of the EPA PAG for TEDE and thyroid CDE.

Dose assessment is performed in accordance with EPIP-EPP-08 "Offsite Dose Assessment and PAR" (ref. 1)

The SITE BOUNDARY is the line beyond which the land is not owned, leased, nor otherwise controlled by Constellation (ref. 2).

Generic

This EAL addresses radioactivity releases that result in doses at or beyond the SITE BOUNDARY that exceed 10% of the EPA Protective Action Guides (PAGs). Releases of this magnitude are associated with the failure of plant systems needed for the protection of the public.

Since dose assessment is based on actual meteorology, whereas the monitor reading EAL is not, the results from these assessments may indicate that the classification is not warranted, or may indicate that a higher classification is warranted. For this reason, emergency implementing procedures should call for the timely performance of dose assessments using actual meteorology and release information. If the results of these dose assessments are available when the classification is made (e.g., initiated at a lower classification level), the dose assessment results override the monitor reading EAL.

NMP2 Basis Reference(s):

1. EPIP-EPP-08 Offsite Dose Assessment and PAR

2. NMP2 Offsite Dose Calculation Manual Figure D.1.0-1
3. NEI 99-01 IC AS1

Category: R – Abnormal Radiation Levels / Radiological Effluents

Subcategory: 1 – Offsite Rad Conditions

Initiating Condition: Offsite dose resulting from an actual or IMMINENT release of gaseous radioactivity exceeds 100 mRem TEDE or 500 mRem thyroid CDE for the actual or projected duration of the release using actual meteorology

EAL:

RS1.3	**Site Area Emergency**
Field survey results indicate closed window dose rates > 100 mRem/hr expected to continue for ≥ 60 min. at or beyond the SITE BOUNDARY (Note 1)	
OR	
Analyses of field survey samples indicate thyroid CDE > 500 mRem for 1 hr of inhalation at or beyond the SITE BOUNDARY (Note 1)	

Note 1: The ED should **not** wait until the applicable time has elapsed, but should declare the event as soon as it is determined that the condition will likely exceed the applicable time

Mode Applicability:

All

Basis:

Plant-Specific

Real time field surveys and sample analysis is performed by offsite field monitoring teams per EPIP-EPP-07, "Downwind Radiological Monitoring" (ref. 1) and assessed for radiological dose consequences per EPIP-EPP-08 "Offsite Dose Assessment and PAR" (ref. 2).

The SITE BOUNDARY is the line beyond which the land is not owned, leased, nor otherwise controlled by Constellation (ref. 3).

Generic

This EAL addresses radioactivity releases that result in doses at or beyond the SITE BOUNDARY that exceed 10% of the EPA Protective Action Guides (PAGs). Releases of this magnitude are associated with the failure of plant systems needed for the protection of the public.

Since dose assessment is based on actual meteorology, whereas the monitor reading EAL is not, the results from these assessments may indicate that the classification is not warranted, or may indicate that a higher classification is warranted. For this reason, emergency implementing procedures should call for the timely performance of dose assessments using actual meteorology and release information. If the results of these dose assessments are available when the classification is made (e.g., initiated at a lower classification level), the dose assessment results override the monitor reading EAL.

NMP2 Basis Reference(s):

1. EPIP-EPP-07 Downwind Radiological Monitoring
2. EPIP-EPP-08 Offsite Dose Assessment and PAR
3. NMP2 Offsite Dose Calculation Manual Figure D.1.0-1
4. NEI 99-01 IC AS1

Category: R – Abnormal Radiation Levels / Radiological Effluents

Subcategory: 1 – Offsite Rad Conditions

Initiating Condition: **ANY** release of gaseous or liquid radioactivity to the environment > 200 times the ODCM for 15 minutes or longer

EAL:

RA1.1 **Alert**
ANY gaseous monitor reading > Table R-1 "Alert" column for ≥ 15 min. (Note 2)

Note 2: The ED should **not** wait until the applicable time has elapsed, but should declare the event as soon as it is determined that the release duration has exceeded, or will likely exceed, the applicable time. In the absence of data to the contrary, assume that the release duration has exceeded the applicable time if an ongoing release is detected and the release start time is unknown.

Table R-1 Effluent Monitor Classification Thresholds

Monitor	GE	SAE	Alert	UE
Gaseous				
Radwaste/RB Vent Effluent	5.5E+7 µCi/s	5.5E+6 µCi/s	200 x Alarm	2 x Alarm
Main Stack Effluent	1.0E+10 µCi/s	1.0E+9 µCi/s	200 x Alarm	2 x Alarm
Liquid				
Service Water Effluent	N/A	N/A	200 x DRMS High(red)	2 x DRMS High(red)
Liquid RadWaste Effluent	N/A	N/A	N/A	2 x DRMS High(red)
Cooling Tower Blowdown	N/A	N/A	200 x DRMS High(red)	2 x DRMS High(red)

Mode Applicability: All

Basis:

Plant-Specific

The value shown for each monitor in Table R-1 is two hundred times the high (red) alarm setpoint for the Digital Radiation Monitoring System (DRMS). The DRMS high (red) alarm setpoints for the listed monitors are conservatively set to ensure ODCM radioactivity release limits are not exceeded (ref. 1). Instrumentation that may be used to assess this EAL is listed below (ref. 2):

- Radwaste/Reactor Building Vent Effluent Monitoring System

  monitor: 2RMS-PNL180C

  recorder: 2RMS-RR170/180

annunciator: 851248

- Main Stack Effluent Monitoring System

  monitor: 2RMS-PNL170C

  recorder: 2RMS-RR170/180

  annunciator: 851256

A radiation monitor reading is VALID when a release path is established. If the release path to the environment has been isolated, the radiation monitor reading is not VALID for classification.

Generic

The Emergency Director should not wait until the applicable time has elapsed, but should declare the event as soon as it is determined that the condition will likely exceed the applicable time.

This EAL addresses an actual or substantial potential decrease in the level of safety of the plant as indicated by a radiological release that exceeds regulatory commitments for an extended period of time.

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

The 200 x DRMS high (red) multiples are specified only to distinguish between non-emergency conditions. While these multiples obviously correspond to an off-site dose or dose rate, the emphasis in classifying these events is the degradation in the level of safety of the Releases should not be prorated or averaged. For example, a release exceeding 600x ODCM for 5 minutes does not meet the threshold.

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

This EAL is intended for sites that have established effluent monitoring on non-routine release pathways for which a discharge permit would not normally be prepared.

NMP2 Basis Reference(s):

1. NMP2 Off-Site Dose Calculation Manual Sections D.3.1.1, D.3.2.1, D.3.3.1, D.3.3.2
2. N2-OP-79 Radiation Monitoring System
3. NEI 99-01 IC AA1

Category: R – Abnormal Radiation Levels / Radiological Effluents

Subcategory: 1 – Offsite Rad Conditions

Initiating Condition: **ANY** release of gaseous or liquid radioactivity to the environment > 200 times the ODCM for 15 minutes or longer

EAL:

RA1.2 Alert

ANY liquid monitor reading > Table R-1 "Alert" column for ≥ 15 min. (Note 2)

Note 2: The ED should **not** wait until the applicable time has elapsed, but should declare the event as soon as it is determined that the release duration has exceeded, or will likely exceed, the applicable time. In the absence of data to the contrary, assume that the release duration has exceeded the applicable time if an ongoing release is detected and the release start time is unknown.

Table R-1 Effluent Monitor Classification Thresholds

Monitor	GE	SAE	Alert	UE
Gaseous				
Radwaste/RB Vent Effluent	5.5E+7 μCi/s	5.5E+6 μCi/s	200 x Alarm	2 x Alarm
Main Stack Effluent	1.0E+10 μCi/s	1.0E+9 μCi/s	200 x Alarm	2 x Alarm
Liquid				
Service Water Effluent	N/A	N/A	200 x DRMS High(red)	2 x DRMS High(red)
Liquid RadWaste Effluent	N/A	N/A	N/A	2 x DRMS High(red)
Cooling Tower Blowdown	N/A	N/A	200 x DRMS High(red)	2 x DRMS High(red)

Mode Applicability:

All

Basis:

Plant-Specific

The value shown for each monitor in Table R-1 is two hundred times the high (red) alarm setpoint for the Digital Radiation Monitoring System (DRMS). The DRMS high (red) alarm setpoints for the listed monitors are conservatively set to ensure ODCM radioactivity release limits are not exceeded (ref. 1). Instrumentation that may be used to assess this EAL is listed below (ref. 2):

- Service Water Effluent Loop A/B Radiation

  monitor: 2SWP*RE146A/B

recorder: 2SWP*RR146A/B

annunciator: 851258

- Cooling Tower Blowdown Line

  monitor: CWS-RE 157

  annunciator: 851258

The designation "N/A" in Table R-1 indicates that the listed instrument range is insufficient to indicate the specified value and therefore no value is used.

A radiation monitor reading is VALID when a release path is established. If the release path to the environment has been isolated, the radiation monitor reading is not VALID for classification.

Generic

The Emergency Director should not wait until the applicable time has elapsed, but should declare the event as soon as it is determined that the condition will likely exceed the applicable time.

This EAL addresses an actual or substantial potential decrease in the level of safety of the plant as indicated by a radiological release that exceeds regulatory commitments for an extended period of time.

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

The 200 x DRMS high (red) multiples are specified only to distinguish between non-emergency conditions. While these multiples obviously correspond to an off-site dose or dose rate, the emphasis in classifying these events is the degradation in the level of safety of the plant, not the magnitude of the associated dose or dose rate.

Releases should not be prorated or averaged. For example, a release exceeding 600x ODCM for 5 minutes does not meet the threshold.

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

This EAL addresses radioactivity releases, that for whatever reason, cause effluent radiation monitor readings to exceed the threshold identified in the EAL established by the radioactivity discharge permit. This value may be associated with a planned batch release, or a continuous release path.

NMP2 Basis Reference(s):

1. NMP2 Off-Site Dose Calculation Manual Sections D.3.1.1, D.3.2.1, D.3.3.1, D.3.3.2
2. N2-OP-79 Radiation Monitoring System
3. NEI 99-01 IC AA1

Category: R – Abnormal Radiation Levels / Radiological Effluents

Subcategory: 1 – Offsite Rad Conditions

Initiating Condition: **ANY** release of gaseous or liquid radioactivity to the environment > 200 times the ODCM for 15 minutes or longer

EAL:

RA1.3	**Alert**
Confirmed sample analyses for gaseous or liquid releases indicate concentrations or release rates > 200 x ODCM limits for ≥ 15 min. (Note 2)	

Note 2: The ED should **not** wait until the applicable time has elapsed, but should declare the event as soon as it is determined that the release duration has exceeded, or will likely exceed, the applicable time. In the absence of data to the contrary, assume that the release duration has exceeded the applicable time if an ongoing release is detected and the release start time is unknown.

Mode Applicability:

All

Basis:

Plant-Specific

Confirmed sample analyses in excess of two hundred times the site Offsite Dose Calculation Manual (ODCM) limits that continue for 15 minutes or longer represent an uncontrolled situation and hence, a potential degradation in the level of safety. This event escalates from the Unusual Event by raising the magnitude of the release by a factor of 100 over the Unusual Event level (i.e., 200 times ODCM). Prorating the 500 mRem/yr basis of the 10 CFR 20 non-occupational MPC limits for both time (8766 hr/yr) and the 200 multiplier, the associated Exclusion Area Boundary dose rate would be approximately 10 mRem/hr. If sample analysis indicates the threshold is met and nothing is done within 15 minutes to effect a release reduction, the ED can conclude that the EAL threshold is met without second sample results.

Generic

The Emergency Director should not wait until the applicable time has elapsed, but should declare the event as soon as it is determined that the condition will likely exceed the applicable time.

This EAL addresses an actual or substantial potential decrease in the level of safety of the plant as indicated by a radiological release that exceeds regulatory commitments for an extended period of time.

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

The 200 x ODCM limit are specified only to distinguish between non-emergency conditions. While these multiples obviously correspond to an off-site dose or dose rate, the emphasis in classifying these events is the degradation in the level of safety of the plant, not the magnitude of the associated dose or dose rate.

Releases should not be prorated or averaged. For example, a release exceeding 600x ODCM for 5 *minutes does not meet the threshold.*

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

This EAL addresses uncontrolled releases that are detected by sample analyses, particularly on unmonitored pathways, e.g., spills of radioactive liquids into storm drains, heat exchanger leakage.

NMP2 Basis Reference(s):

1. NMP2 Off-Site Dose Calculation Manual
2. NEI 99-01 IC AA1

Category: R – Abnormal Radiation Levels / Radiological Effluents

Subcategory: 1 – Offsite Rad Conditions

Initiating Condition: **ANY** release of gaseous or liquid radioactivity to the environment > 2 times the ODCM for 60 minutes or longer

EAL:

RU1.1 Unusual Event

ANY gaseous monitor reading > Table R-1 "UE" column for ≥ 60 min. (Note 2)

Note 2: The ED should **not** wait until the applicable time has elapsed, but should declare the event as soon as it is determined that the release duration has exceeded, or will likely exceed, the applicable time. In the absence of data to the contrary, assume that the release duration has exceeded the applicable time if an ongoing release is detected and the release start time is unknown.

Table R-1 Effluent Monitor Classification Thresholds

Monitor	GE	SAE	Alert	UE
Gaseous				
Radwaste/RB Vent Effluent	5.5E+7 µCi/s	5.5E+6 µCi/s	200 x Alarm	2 x Alarm
Main Stack Effluent	1.0E+10 µCi/s	1.0E+9 µCi/s	200 x Alarm	2 x Alarm
Liquid				
Service Water Effluent	N/A	N/A	200 x DRMS High(red)	2 x DRMS High(red)
Liquid RadWaste Effluent	N/A	N/A	N/A	2 x DRMS High(red)
Cooling Tower Blowdown	N/A	N/A	200 x DRMS High(red)	2 x DRMS High(red)

Mode Applicability:

All

Basis:

Plant-Specific

The value shown for each monitor in Table R-1 is two times the high (red) alarm setpoint for the Digital Radiation Monitoring System (DRMS). The DRMS high (red) alarm setpoints for the listed monitors are conservatively set to ensure ODCM radioactivity release limits are not exceeded (ref. 1). Instrumentation that may be used to assess this EAL is listed below (ref. 2):

- Radwaste/Reactor Building Vent Effluent Monitoring System

  monitor: 2RMS-PNL180C

recorder: 2RMS-RR170/180

annunciator: 851248

- Main Stack Effluent Monitoring System

  monitor: 2RMS-PNL170C

  recorder: 2RMS-RR170/180

  annunciator: 851256

A radiation monitor reading is VALID when a release path is established. If the release path to the environment has been isolated, the radiation monitor reading is not VALID for classification.

Generic

The Emergency Director should not wait until the applicable time has elapsed, but should declare the event as soon as it is determined that the condition will likely exceed the applicable time.

This EAL addresses a potential decrease in the level of safety of the plant as indicated by a radiological release that exceeds regulatory commitments for an extended period of time.

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

The 2 x DRMS (red) multiples are specified only to distinguish between non-emergency conditions. While these multiples obviously correspond to an off-site dose or dose rate, the emphasis in classifying these events is the degradation in the level of safety of the plant, not the magnitude of the associated dose or dose rate.

Releases should not be prorated or averaged. For example, a release exceeding 4x ODCM for 30 minutes does not meet the threshold.

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

This EAL addresses radioactivity releases, that for whatever reason, cause effluent radiation monitor readings to exceed the threshold identified in the IC.

This EAL is intended for sites that have established effluent monitoring on non-routine release pathways for which a discharge permit would not normally be prepared.

NMP2 Basis Reference(s):

1. NMP2 Off-Site Dose Calculation Manual Sections D.3.1.1, D.3.2.1, D.3.3.1, D.3.3.2
2. N2-OP-79 Radiation Monitoring System
3. NEI 99-01 IC AU1

Category: R – Abnormal Radiation Levels / Radiological Effluents

Subcategory: 1 – Offsite Rad Conditions

Initiating Condition: **ANY** release of gaseous or liquid radioactivity to the environment > 2 times the ODCM for 60 minutes or longer

EAL:

RU1.2 Unusual Event

ANY liquid monitor reading > Table R-1 "UE" column for ≥ 60 min. (Note 2)

Note 2: The ED should **not** wait until the applicable time has elapsed, but should declare the event as soon as it is determined that the release duration has exceeded, or will likely exceed, the applicable time. In the absence of data to the contrary, assume that the release duration has exceeded the applicable time if an ongoing release is detected and the release start time is unknown.

Table R-1 Effluent Monitor Classification Thresholds

Monitor	GE	SAE	Alert	UE
Gaseous				
Radwaste/RB Vent Effluent	5.5E+7 μCi/s	5.5E+6 μCi/s	200 x Alarm	2 x Alarm
Main Stack Effluent	1.0E+10 μCi/s	1.0E+9 μCi/s	200 x Alarm	2 x Alarm
Liquid				
Service Water Effluent	N/A	N/A	200 x DRMS High(red)	2 x DRMS High(red)
Liquid RadWaste Effluent	N/A	N/A	N/A	2 x DRMS High(red)
Cooling Tower Blowdown	N/A	N/A	200 x DRMS High(red)	2 x DRMS High(red)

Mode Applicability:

All

Basis:

Plant-Specific

The value shown for each monitor in Table R-1 is two times the high (red) alarm setpoint for the Digital Radiation Monitoring System (DRMS). The DRMS high (red) alarm setpoints for the listed monitors are conservatively set to ensure ODCM radioactivity release limits are not exceeded (ref. 1). Instrumentation that may be used to assess this EAL is listed below (ref. 2):

- Service Water Effluent Loop A/B Radiation

  monitor: 2SWP*RE146A/B

  recorder: 2SWP*RR146A/B

annunciator: 851258

- Liquid Effluent Line

  monitor: LWS-RE206

  annunciator: 851258

- Cooling Tower Blowdown Line

  monitor: CWS-RE 157

  annunciator: 851258

A radiation monitor reading is VALID when a release path is established. If the release path to the environment has been isolated, the radiation monitor reading is not VALID for classification.

Generic

The Emergency Director should not wait until the applicable time has elapsed, but should declare the event as soon as it is determined that the condition will likely exceed the applicable time.

This IC addresses a potential decrease in the level of safety of the plant as indicated by a radiological release that exceeds regulatory commitments for an extended period of time.

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

The 2 x ODCM limit multiples are specified only to distinguish between non-emergency conditions. While these multiples obviously correspond to an off-site dose or dose rate, the emphasis in classifying these events is the degradation in the level of safety of the plant, not the magnitude of the associated dose or dose rate.

Releases should not be prorated or averaged. For example, a release exceeding 4x ODCM for 30 minutes does not meet the threshold.

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

This EAL addresses radioactivity releases, that for whatever reason, cause effluent radiation monitor readings to exceed the threshold identified in the EAL established by the radioactivity discharge permit. This value may be associated with a planned batch release, or a continuous release path.

NMP2 Basis Reference(s):

1. NMP2 Off-Site Dose Calculation Manual Sections D.3.1.1, D.3.2.1, D.3.3.1, D.3.3.2
2. N2-OP-79 Radiation Monitoring System
3. NEI 99-01 IC AU1

Category: R – Abnormal Radiation Levels / Radiological Effluents

Subcategory: 1 – Offsite Rad Conditions

Initiating Condition: **ANY** release of gaseous or liquid radioactivity to the environment > 2 times the ODCM for 60 minutes or longer

EAL:

RU1.3	**Unusual Event**
Confirmed sample analyses for gaseous or liquid releases indicate concentrations or release rates > 2 x ODCM limits for ≥ 60 min. (Note 2)	

Note 2: The ED should **not** wait until the applicable time has elapsed, but should declare the event as soon as it is determined that the release duration has exceeded, or will likely exceed, the applicable time. In the absence of data to the contrary, assume that the release duration has exceeded the applicable time if an ongoing release is detected and the release start time is unknown.

Mode Applicability:

All

Basis:

Plant-Specific

Releases in excess of two times the site Offsite Dose Calculation Manual (ODCM) (ref. 1) instantaneous limits that continue for 60 minutes or longer represent an uncontrolled situation and hence, a potential degradation in the level of safety. The final integrated dose (which is very low in the Unusual Event emergency class) is not the primary concern here; it is the degradation in plant control implied by the fact that the release was not isolated within 60 minutes. Therefore, it is not intended that the release be averaged over 60 minutes. For example, a release of 4 times the ODCM limit for 30 minutes does not exceed this initiating condition. Further, the ED should not wait until 60 minutes has elapsed, but should declare the event as soon as it is determined that the release duration has or will likely exceed 60 minutes.

Generic

The Emergency Director should not wait until the applicable time has elapsed, but should declare the event as soon as it is determined that the condition will likely exceed the applicable time.

This EAL addresses a potential decrease in the level of safety of the plant as indicated by a radiological release that exceeds regulatory commitments for an extended period of time.

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

The 2 x ODCM limit multiples are specified only to distinguish between non-emergency conditions. While these multiples obviously correspond to an off-site dose or dose rate, the emphasis in classifying these events is the degradation in the level of safety of the plant, not the magnitude of the associated dose or dose rate.

Releases should not be prorated or averaged. For example, a release exceeding 4x ODCM for 30 minutes does not meet the threshold.

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

This EAL addresses uncontrolled releases that are detected by sample analyses, particularly on unmonitored pathways, e.g., spills of radioactive liquids into storm drains, heat exchanger leakage in water systems, etc.

NMP2 Basis Reference(s):

1. NMP2 Off-Site Dose Calculation Manual
2. NEI 99-01 IC AU1

Category: R – Abnormal Radiation Levels / Radiological Effluents

Subcategory: 2 – Onsite Rad Conditions & Spent Fuel Events

Initiating Condition: Damage to irradiated fuel or loss of water level that has resulted or will result in the uncovering of irradiated fuel outside the Reactor Vessel

EAL:

RA2.1 Alert

Alarm on **ANY** of the following radiation monitors due to damage to irradiated fuel or loss of water level:

- 2RMS-RE111
- 2RMS-RE112
- 2RMS-RE113
- 2RMS-RE114
- 2RMS-RE140
- 2HVR*RE14A
- 2HVR*RE14B

Mode Applicability:

All

Basis:

Plant-Specific

This EAL is defined by the specific areas where irradiated fuel is located such as the reactor cavity, RPV or Spent Fuel Pool.

The bases for the area radiation high alarms and the Above Refuel Floor HVAC Exhaust (2HVR*RE14A/B) high alarms are a spent fuel handling accident and are, therefore, appropriate for this EAL.

Elevated readings on the ventilation monitors may also be indication of a radioactivity release from the fuel, confirming that damage has occurred. However, elevated background at the monitor due to water level lowering may mask elevated ventilation exhaust airborne activity and needs to be considered.

However, while radiation monitors may detect a rise in dose rate due to a drop in the water level, it might not be a reliable indication of whether or not the fuel is covered. For example, the monitor could in fact be properly responding to a known event involving transfer or relocation of a source stored in or near the Spent Fuel Pool or responding to a planned evolution such as removal of the RPV head. Interpretation of these EAL thresholds requires some understanding of the actual radiological conditions present in the vicinity of the monitors.

Generic

This EAL addresses increases in radiation dose rates within plant buildings, and may be a precursor to a radioactivity release to the environment. These events represent a loss of control over radioactive material and represent an actual or substantial potential degradation in the level of safety of the plant.

This EAL addresses radiation monitor indications of fuel uncovery and/or fuel damage.

Increased ventilation monitor readings may be indication of a radioactivity release from the fuel, confirming that damage has occurred. Increased background at the ventilation monitor due to water level decrease may mask increased ventilation exhaust airborne activity and needs to be considered.

While a radiation monitor could detect an increase in dose rate due to a drop in the water level, it might not be a reliable indication of whether or not the fuel is covered.

Escalation of this emergency classification level, if appropriate, would be based on RS1.1, RS1.2, RS1.3, RG1.1, RG1.2 or RG1.3.

NMP2 Basis Reference(s):

1. N2-SOP-39 Refuel Floor Events
2. N2-ARP-01 Annunciator Response Procedures for annunciator 851254
3. NEI 99-01 IC AA2

Category: R – Abnormal Radiation Levels / Radiological Effluents

Subcategory: 2 – Onsite Rad Conditions & Spent Fuel Events

Initiating Condition: Damage to irradiated fuel or loss of water level that has resulted or will result in the uncovering of irradiated fuel outside the Reactor Vessel

EAL:

RA2.2	**Alert**
A water level drop in a reactor refueling pathway that will result in irradiated fuel becoming uncovered	

Mode Applicability:

All

Basis:

Plant-Specific

The reactor cavity and Spent Fuel Pool comprise the reactor refueling pathway (ref. 1).

The movement of irradiated fuel assemblies requires a minimum water level of 22 ft 3 in. above the RPV flange and the top of spent fuel in the SFP. During refueling activities, this maintains sufficient water level in the reactor cavity and SFP to retain iodine fission product activity in the water in the event of a fuel handling accident (ref. 2, 3).

Allowing level to decrease could result in spent fuel being uncovered, reducing spent fuel decay heat removal and creating an extremely hazardous radiation environment.

There is no indication that water level in the spent fuel pool has dropped to the level of the fuel other than by visual observation by personnel on the refueling floor. N2-SOP-39, Refuel Floor Events, provides appropriate instructions to report a visual observation of irradiated fuel uncovery (ref. 4).

Generic

This event represents a loss of control over radioactive material and represents an actual or substantial potential degradation in the level of safety of the plant.

Escalation of this emergency classification level, if appropriate, would be based on RS1.1, RS1.2, RS1.3, RG1.1, RG1.2 or RG1.3.

NMP2 Basis Reference(s):

1. USAR Section 9.1.2
2. Improved Technical Specifications Nine Mile Point Nuclear Station, Unit No. 2, 3.7.6
3. Improved Technical Specifications Nine Mile Point Nuclear Station, Unit No. 2, 3.9.6
~~3.~~4. N2-SOP-39, Refuel Floor Events
~~4.~~5. NEI 99-01 IC AA2

Category: R – Abnormal Radiation Levels / Radiological Effluents

Subcategory: 2 – Onsite Rad Conditions & Spent Fuel Events

Initiating Condition: UNPLANNED rise in plant radiation levels

EAL:

RU2.1 Unusual Event

UNPLANNED water level drop in a reactor refueling pathway as indicated by inability to restore and maintain SFP level > low water level alarm (Note 3)

AND

Area radiation monitor reading rise on **ANY** of the following:

- 2RMS-RE111
- 2RMS-RE112
- 2RMS-RE113
- 2RMS-RE114
- 2RMS-RE140

Note 3: If loss of water level in the refueling pathway occurs while in Mode 4, 5 or D, consider classification under EALs CU3.1, CU3.2 or CU3.3

Mode Applicability:

All

Basis:

Plant-Specific

The reactor cavity and Spent Fuel Pool (SFP) comprise the reactor refueling pathway (ref. 1).

The SFP is normally filled to a level of 352 ft 10 in. Level switches 2SFC*LS55A and B are set at 2 inches below the normal water level (or 352 ft 8 in.) and activate annunciators 873317 and 875117 in the Control Room. (ref. 2, 3)

The phrase "... inability to restore and maintain level >..." allows the operator to visually observe the low water level condition, if possible, and to attempt water level restoration actions as long as water level remains above the top of irradiated fuel. Water level restoration operations are performed in accordance with N2-OP-38 (ref. 4).

Technical Specifications requires that:

- SFP water level be maintained 22 ft 3 in. above irradiated fuel seated in the storage racks during movement of irradiated fuel assemblies in the SFP (ref. 5).

- RPV water level be maintained 22 ft 3 in. above the top of the RPV flange during movement of irradiated fuel assemblies in the RPV (ref. 6).

The listed Area radiation monitors are located in the proximity of where spent fuel may be located and have been selected to be indicative of a decrease in radiation shielding due to decreasing refueling pathway water level (ref. 1). While a radiation monitor could detect a rise in dose due to a drop in the water level, it might not be a reliable indication, in and of itself, of whether or not the fuel is uncovered. For example, the reading on an area radiation monitor located on the refuel bridge may rise due to planned evolutions such as RPV head lift or a fuel assembly being raised on fuel grapple. Elevated radiation monitor indications will need to be combined with another indicator (or personnel report) of water loss.

This event escalates to an Alert if irradiated fuel outside the RPV is uncovered.

Generic

This EAL addresses increased radiation levels as a result of water level decreases above irradiated fuel or events that have resulted, or may result, in UNPLANNED increases in radiation dose rates within plant buildings. These radiation increases represent a loss of control over radioactive material and represent a potential degradation in the level of safety of the plant.

The refueling pathway is a combination of cavities, tubes, canals and pools. While a radiation monitor could detect an increase in dose rate due to a drop in the water level, it might not be a reliable indication of whether or not the fuel is covered.

For refueling events where the water level drops below the RPV flange classification would be via EAL CU3.1, CU3.2 or CU3.3. This event escalates to an Alert per EAL RA2.1 if irradiated fuel outside the reactor vessel is uncovered. For events involving irradiated fuel in the reactor vessel, escalation would be via the Fission Product Barrier Table for events in operating modes 1-4.

NMP2 Basis Reference(s):

1. USAR Section 9.1.2
2. N2-ARP-01 Annunciator Response Procedures for annunciator 873317
3. N2-ARP-01 Annunciator Response Procedures for annunciator 875117
4. N2-OP-38 Spent Fuel Pool Cooling and Cleanup System
5. Improved Technical Specifications Nine Mile Point Nuclear Station, Unit No. 2, 3.7.6
6. Improved Technical Specifications Nine Mile Point Nuclear Station, Unit No. 2, 3.9.6
7. N2-SOP-39 Refuel Floor Events
8. NEI 99-01 IC AU2

Category: R – Radioactivity Release / Area Radiation
Subcategory: 2 – Onsite Rad Conditions & Spent Fuel Events
Initiating Condition: UNPLANNED rise in plant radiation levels
EAL:

RU2.2	**Unusual Event**
UNPLANNED area radiation readings rise by a factor of 1,000 over NORMAL LEVELS	

Mode Applicability:

All

Basis:

Plant-Specific

Assessment of this EAL may be made with survey readings using portable instruments as well as installed radiation monitors.

Generic

This EAL addresses increased radiation levels as a result of water level decreases above irradiated fuel or events that have resulted, or may result, in UNPLANNED increases in radiation dose rates within plant buildings. These radiation increases represent a loss of control over radioactive material and represent a potential degradation in the level of safety of the plant.

This EAL addresses increases in plant radiation levels that represent a loss of control of radioactive material resulting in a potential degradation in the level of safety of the plant.

This EAL excludes radiation level increases that result from planned activities such as use of radiographic sources and movement of radioactive waste materials. A specific list of ARMs is not required as it would restrict the applicability of the threshold. The intent is to identify loss of control of radioactive material in any monitored area.

NMP2 Basis Reference(s):

1. NEI 99-01 IC AU2

Category: R – Abnormal Radiation Levels / Radiological Effluents

Subcategory: 3 – CR/CAS Rad

Initiating Condition: Rise in radiation levels within the facility that impedes operation of systems required to maintain plant safety functions

EAL:

RA3.1 Alert

Dose rates > 15 mRem/hr in **EITHER** of the following areas requiring continuous occupancy to maintain plant safety functions:

Control Room

OR

CAS

Mode Applicability:

All

Basis:

Plant-Specific

The Control Room and Central Alarm Station (CAS) must be continuously occupied in all plant operating modes at NMP2. CAS is included in this EAL because of its importance to permitting access to areas required to assure safe plant operation.

Area Radiation Monitor (ARM) 2RMS-RE129 monitors radiation levels in the Control Room at 306' elevation. This is one of three Control Building ARMs that actuate Control Room annunciator 851246, CONTROL BLDG AREA RADN MON ACTVATED, giving personnel sufficient warning of changing levels (ref. 1). There is no area radiation monitoring system at NMP2 for the CAS. Abnormal radiation levels may be initially detected by routine radiological surveys.

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

Emergency may be indicated by other EAL categories.

This EAL could result in declaration of an Alert at NMP2 due to a radioactivity release or radiation shine resulting from a major accident at the NMP1 or JAFNPP. Such a declaration would be appropriate if the increase impairs safe plant operation.

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

Generic

This EAL addresses increased radiation levels that: impact continued operation in areas requiring continuous occupancy to maintain safe operation or to perform a safe shutdown.

The cause and/or magnitude of the increase in radiation levels is not a concern of this EAL. The Emergency Director must consider the source or cause of the increased radiation levels and determine if any other EAL may be involved.

Areas requiring continuous occupancy include the Control Room and any other control stations that are staffed continuously, such as the security alarm station CAS.

NMP2 Basis Reference(s):

1. N2-ARP-01 Annunciator Response Procedures for annunciator 851246
2. NEI 99-01 IC AA3

Category H – Hazards and Other Conditions Affecting Plant Safety

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

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

The events of this category pertain to the following subcategories:

1. Natural or Destructive Phenomena

Natural events include hurricanes, earthquakes or tornados that have potential to cause plant structure or equipment damage of sufficient magnitude to threaten personnel or plant safety. Non-naturally occurring events that can cause damage to plant facilities and include aircraft crashes, missile impacts, etc.

2. FIRE or EXPLOSION

FIREs can pose significant hazards to personnel and reactor safety. Appropriate for classification are FIREs within the site PROTECTED AREA or which may affect operability of equipment needed for safe shutdown

3. Hazardous Gas

Non-naturally occurring events that can cause damage to plant facilities and include toxic, asphyxiant, corrosive or flammable gas leaks.

4. Security

Unauthorized entry attempts into the PROTECTED AREA, BOMB threats, SABOTAGE attempts, and actual security compromises threatening loss of physical control of the plant.

5. Control Room Evacuation

Events that are indicative of loss of Control Room habitability. If the Control Room must be evacuated, additional support for monitoring and controlling plant functions is necessary through the emergency response facilities.

6. Judgment

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

Category: H – Hazards and Other Conditions Affecting Plant Safety

Subcategory: Natural or Destructive Phenomena

Initiating Condition: Natural or destructive phenomena affecting VITAL AREAs

EAL:

HA1.1 Alert

Seismic event > OBE (0.075g)
as indicated by **EITHER**:

Computer Point ERSNC02, OBE Detected

OR

ANY amber LED light lit at the Seismic Monitor Panel, Response Spectrum Annunciator

AND

Earthquake confirmed by **ANY** of the following:

- Earthquake felt in plant
- JAFNPP seismic instrumentation
- Control Room indication of degraded performance of systems required for the safe shutdown of the plant

Mode Applicability:

All

Basis:

Plant-Specific

This EAL is based on the USAR design basis operating earthquake of 0.075g (ref. 1, 2). Seismic events of this magnitude can cause damage to plant safety functions.

The method of detection relies on actuation of the NMP2 seismic monitor OBE alarm confirmed by one or more indications such as shift operators on duty in the Control Room determining that the ground motion was felt~~,~~ or degraded system performance ~~or corroborated by the NEIC~~.

NMP2 seismic instrumentation actuates at 0.01g upon sensing any seismic activity (ref. 2).

NMP1 and NMP2 share a common PROTECTED AREA border. Consideration should be given to the opposite unit when classifying under this EAL.

Generic

These EALs escalate from HU1.1 in that the occurrence of the event has resulted in VISIBLE DAMAGE to plant structures or areas containing equipment necessary for a safe shutdown, or has caused damage to the safety systems in those structures evidenced by control room indications of degraded system response or performance. The occurrence of VISIBLE DAMAGE and/or degraded system response is intended to discriminate against lesser events. The initial report should not be interpreted as mandating a lengthy damage assessment prior to classification. No attempt is made in this EAL to assess the actual magnitude of the damage. The significance here is not that a particular system or structure was damaged, but rather, that the event was of sufficient magnitude to cause this degradation.

Escalation of this emergency classification level, if appropriate, would be based on System Malfunction EALs.

Seismic events of this magnitude can result in a VITAL AREA being subjected to forces beyond design limits, and thus damage may be assumed to have occurred to plant safety systems.

NMP2 Basis Reference(s):

1. USAR Section 3.7A.1.1
2. N2-SOP-90 Natural Events
3. USAR Section 2.1.1.1
4. NEI 99-01 IC HA1

Category: H – Hazards and Other Conditions Affecting Plant Safety

Subcategory: 1 – Natural or Destructive Phenomena

Initiating Condition: Natural or destructive phenomena affecting VITAL AREAs

EAL:

HA1.2 **Alert**
Tornado striking **OR** Sustained high winds > 90 mph resulting in **EITHER**: VISIBLE DAMAGE to **ANY** SAFETY-RELATED STRUCTURE, SYSTEM or COMPONENT within **ANY** Table H-1 area **OR** Control Room indication of degraded performance of **ANY** SAFETY-RELATED STRUCTURE, SYSTEM or COMPONENT within **ANY** Table H-1 area

Table H-1 Safe Shutdown Areas
• Reactor Building (including Primary Containment) • Control Room • Diesel Generator Engine and Board Rooms • Standby Switchgear and Battery Rooms • HPCS Switchgear and Battery Rooms • Remote Shutdown Rooms • Control Building HVAC Rooms • Service Water Pump Rooms • Electrical Protection Assembly Room • PGCC Relay Room

Mode Applicability:

All

Basis:

Plant-Specific

All Category 1 structures are designed for a wind velocity of 90 mph (ref. 1). This EAL is based on the structural design basis of 90 mph or impact by tornado. Wind loads of this magnitude can cause damage to safety functions.

Weather conditions are monitored at three locations:

- The 200 foot high Primary OR Main Meteorological Tower located 0.6 miles west-southwest of NMP2
- The 90 foot Backup Tower located east of JAFNPP
- The 30 foot Inland Tower located at the Oswego County Airport near Fulton

Meteorological parameters such as wind speed are sent to the Control Rooms and Technical Support Centers (TSC) at NMP1, NMP2, JAFNPP and the Emergency Operations Facility (EOF). Data from sensors mounted on these towers are sent to both digital and analog systems for display, processing and storage. Wind speed and wind direction, as well as wind speed deviation and differential temperatures are monitored in NMP2 Control Room and recorded on strip chart recorders. (ref. 2)

Wind speed can be measured up to 100 mph.

Weather information may be obtained from (ref. 4):

- National Weather Service: 716-565-9001 or 800-462-7751
- Accu-Weather: 815-235-8650 or 814-237-5803

The PROTECTED AREA Boundary is depicted in USAR Figure 1.2-1, Plot Plan (ref. 3).

This threshold addresses events that may have resulted in a Safe Shutdown Area being subjected to forces beyond design limits and thus damage may be assumed to have occurred to plant safety systems. Safe Shutdown Areas are areas that house equipment the operation of which may be needed to ensure the reactor safely reaches and is maintained in cold shutdown. Safe Shutdown Areas include structures that contain the equipment of concern. The Alert classification is appropriate if relevant plant parameters indicate that the performance of safety systems in the affected Safe Shutdown Areas has been degraded. No attempt should be made to fully inventory the actual magnitude of the damage or quantify the degradation of safety system performance prior to declaration of an Alert under this threshold.

Table H-1 Safe Shutdown Areas include all structures containing Category I equipment and systems needed for safe shutdown (ref. 5).

NMP1 and NMP2 share a common PROTECTED AREA border. Consideration should be given to the opposite unit when classifying under this EAL.

Generic

This EAL escalates from HU1.2 in that the occurrence of the event has resulted in VISIBLE DAMAGE to plant structures or areas containing equipment necessary for a safe shutdown, or has caused damage to the safety systems in those structures evidenced by control room indications of degraded system response or performance. The occurrence of VISIBLE DAMAGE and/or degraded system response is intended to discriminate against lesser events. The initial report should not be interpreted as mandating a lengthy damage assessment prior to classification. No attempt is made in this EAL to assess the actual magnitude of the damage. The significance here is not that a particular system or structure was damaged, but rather, that the event was of sufficient magnitude to cause this degradation.

Escalation of this emergency classification level, if appropriate, would be based on System Malfunction EALs.

This EAL is based on a tornado striking (touching down) or high winds that have caused VISIBLE DAMAGE to structures containing functions or systems required for safe shutdown of the plant.

NMP2 Basis Reference(s):

1. USAR Section 3.3.1.1
2. N2-OP-102 Meteorological Monitoring
3. USAR Figure 1.2-1
4. N2-SOP-64 High Winds
5. USAR 9B and Figure 9B.6-1
6. NEI 99-01 IC HA1

Category: H – Hazards and Other Conditions Affecting Plant Safety

Subcategory: 1 – Natural or Destructive Phenomena

Initiating Condition: Natural or destructive phenomena affecting VITAL AREAs

EAL:

HA1.3 Alert

Internal flooding
resulting in **EITHER**:

An electrical shock hazard that precludes access to operate or monitor **ANY** SAFETY-RELATED STRUCTURE, SYSTEM or COMPONENT within **ANY** Table H-1 area

OR

Control Room indication of degraded performance of **ANY** SAFETY-RELATED STRUCTURE, SYSTEM or COMPONENT within **ANY** Table H-1 area

Table H-1 Safe Shutdown Areas
• Reactor Building (including Primary Containment) • Control Room • Diesel Generator Engine and Board Rooms • Standby Switchgear and Battery Rooms • HPCS Switchgear and Battery Rooms • Remote Shutdown Rooms • Control Building HVAC Rooms • Service Water Pump Rooms • Electrical Protection Assembly Room • PGCC Relay Room

Mode Applicability:

All

Basis:

Plant-Specific

This threshold addresses the affect of flooding caused by internal events such as component failures, Circulating, Component Cooling or Service Water line ruptures, equipment misalignment, FIRE suppression system actuation, and outage activity mishaps.

Table H-1 Safe Shutdown Areas include all structures containing Category I equipment and systems needed for safe shutdown (ref. 1).

Uncontrolled internal flooding that has degraded safety-related equipment or created a safety hazard precluding access necessary for the safe operation or monitoring of safety equipment warrants declaration of an Alert.

Generic

Escalation of this emergency classification level, if appropriate, would be based on System Malfunction EALs.

This EAL addresses the effect of internal flooding caused by events such as component failures, equipment misalignment, or outage activity mishaps. It is based on the degraded performance of systems, or has created industrial safety hazards (e.g., electrical shock) that preclude necessary access to operate or monitor safety equipment. The inability to access, operate or monitor safety equipment represents an actual or substantial potential degradation of the level of safety of the plant.

Flooding as used in this EAL describes a condition where water is entering the room faster than installed equipment is capable of removal, resulting in a rise of water level within the room. Classification of this EAL should not be delayed while corrective actions are being taken to isolate the water source.

NMP2 Basis Reference(s):

1. USAR 9B and Figure 9B.6-1
2. NEI 99-01 IC HA1

Category: H – Hazards and Other Conditions Affecting Plant Safety

Subcategory: 1 – Natural or Destructive Phenomena

Initiating Condition: Natural or destructive phenomena affecting VITAL AREAs

EAL:

HA1.4 Alert

Turbine failure-generated PROJECTILEs
resulting in **EITHER**:

VISIBLE DAMAGE to or penetration of **ANY** SAFETY-RELATED STRUCTURE, SYSTEM or COMPONENT within **ANY** Table H-1 area

OR

Control Room indication of degraded performance of **ANY** SAFETY-RELATED STRUCTURE, SYSTEM or COMPONENT within **ANY** Table H-1 area

Table H-1 Safe Shutdown Areas
• Reactor Building (including Primary Containment)
• Control Room
• Diesel Generator Engine and Board Rooms
• Standby Switchgear and Battery Rooms
• HPCS Switchgear and Battery Rooms
• Remote Shutdown Rooms
• Control Building HVAC Rooms
• Service Water Pump Rooms
• Electrical Protection Assembly Room
• PGCC Relay Room

Mode Applicability:

All

Basis:

Plant-Specific

The turbine generator stores large amounts of rotational kinetic energy in its rotor. In the unlikely event of a major mechanical failure, this energy may be transformed into both rotational and translational energy of rotor fragments. These fragments may impact the surrounding stationary parts. If the energy-absorbing capability of these stationary turbine generator parts is insufficient, external PROJECTILEs will be released. These ejected

PROJECTILEs may impact various plant structures, including those housing safety related equipment.

Table H-1 Safe Shutdown Areas include all structures containing Category I equipment and systems needed for safe shutdown (ref. 1).

Generic

This EAL escalates from HU1.4 in that the occurrence of the event has resulted in VISIBLE DAMAGE to plant structures or areas containing equipment necessary for a safe shutdown, or has caused damage to the safety systems in those structures evidenced by control room indications of degraded system response or performance. The occurrence of VISIBLE DAMAGE and/or degraded system response is intended to discriminate against lesser events. The initial report should not be interpreted as mandating a lengthy damage assessment prior to classification. No attempt is made in this EAL to assess the actual magnitude of the damage. The significance here is not that a particular system or structure was damaged, but rather, that the event was of sufficient magnitude to cause this degradation.

Escalation of this emergency classification level, if appropriate, would be based on System Malfunction EALs.

This EAL addresses the threat to safety related equipment imposed by PROJECTILEs generated by main turbine rotating component failures. Therefore, this EAL is consistent with the definition of an Alert in that the potential exists for actual or substantial potential degradation of the level of safety of the plant.

NMP2 Basis Reference(s):

1. USAR 9B and Figure 9B.6-1
2. NEI 99-01 IC HA1

Category: H – Hazards and Other Conditions Affecting Plant Safety
Subcategory: 1 – Natural or Destructive Phenomena
Initiating Condition: Natural or destructive phenomena affecting VITAL AREAs
EAL:

HA1.5	**Alert**
Lake water level > 254 ft **OR** Intake water level < 233 ft	

Mode Applicability:

All

Basis:

Plant-Specific

This threshold covers high and low water level conditions that may have resulted in a plant VITAL AREA being subjected to levels beyond design limits, and thus damage may be assumed to have occurred to plant safety systems.

The high lake level is based upon the maximum probable flood level (ref. 1).

The low forebay water level corresponds to the minimum intake bay water level which provides adequate submergence to the service water pumps (ref. 2, 3).

Generic

This EAL addresses other site specific phenomena that result in VISIBLE DAMAGE to VITAL AREAs or results in indication of damage to SAFETY STRUCTURES, SYSTEMS, or COMPONENTS containing functions and systems required for safe shutdown of the plant that can also be precursors of more serious events.

NMP2 Basis Reference(s):

1. USAR Section 2.4.5.2
2. USAR Section 2.4.1.1
3. USAR Section 9.2.5.3.1
4. N2-OSP-LOG-W001, Weekly Checks
5. NEI 99-01 IC HA1

Category: H – Hazards and Other Conditions Affecting Plant Safety

Subcategory: 1 – Natural or Destructive Phenomena

Initiating Condition: Natural or destructive phenomena affecting VITAL AREAs

EAL:

HA1.6 Alert

Vehicle crash
resulting in **EITHER**:

VISIBLE DAMAGE to **ANY** SAFETY-RELATED STRUCTURE, SYSTEM or COMPONENT within **ANY** Table H-1 area

OR

Control Room indication of degraded performance of **ANY** SAFETY-RELATED STRUCTURE, SYSTEM or COMPONENT within **ANY** Table H-1 area

Table H-1 Safe Shutdown Areas
• Reactor Building (including Primary Containment) • Control Room • Diesel Generator Engine and Board Rooms • Standby Switchgear and Battery Rooms • HPCS Switchgear and Battery Rooms • Remote Shutdown Rooms • Control Building HVAC Rooms • Service Water Pump Rooms • Electrical Protection Assembly Room • PGCC Relay Room

Mode Applicability:

All

Basis:

Plant-Specific

This EAL is intended to address crashes of vehicle types large enough to cause significant damage to plant structures containing functions and systems required for safe shutdown of the plant. Vehicle types include automobiles, aircraft, trucks, cranes, forklifts, waterborne craft, etc.

Table H-1 Safe Shutdown Areas include all structures containing Category I equipment and systems needed for safe shutdown (ref. 1).

Generic

The occurrence of VISIBLE DAMAGE and/or degraded system response is intended to discriminate against lesser events. The initial report should not be interpreted as mandating a lengthy damage assessment prior to classification. No attempt is made in this EAL to assess the actual magnitude of the damage. The significance here is not that a particular system or structure was damaged, but rather, that the event was of sufficient magnitude to cause this degradation.

Escalation of this emergency classification level, if appropriate, would be based on System Malfunction EALs.

This EAL addresses vehicle crashes within the PROTECTED AREA that results in VISIBLE DAMAGE to VITAL AREAs or indication of damage to SAFETY STRUCTURES, SYSTEMS, or COMPONENTS containing functions and systems required for safe shutdown of the plant.

NMP2 Basis Reference(s):

1. USAR 9B and Figure 9B.6-1
2. NEI 99-01 IC HA1

Category: H – Hazards and Other Conditions Affecting Plant Safety

Subcategory: 1 – Natural or Destructive Phenomena

Initiating Condition: Natural or destructive phenomena affecting the PROTECTED AREA

EAL:

HU1.1 Unusual Event

Seismic event identified by **ANY two** of the following:

- Annunciator 842121 SEISMIC ACCELERATION EXCEEDED indicates seismic event detected
- Confirmation of earthquake received on NMP-1 or JAFNPP seismic instrumentation
- Earthquake felt in plant

Mode Applicability:

All

Basis:

Plant-Specific

The NMP2 seismic instrumentation actuates at 0.01 g causing (ref. 1-4):

- Power to remote acceleration sensor units
- Activation of MRS1 recorders
- EVENT alarm light on PWRS1 to light
- EVENT INDICATOR on PWRS1 to turn from black to white
- Annunciator 842121 on panel 2CEC-PNL842 to be received

Annunciator 842121 provides the most direct indication in the Control Room that a seismic event has occurred. The EVENT alarm light and EVENT INDICATOR are located on 2CES-PNL889 in the relay room (ref. 4). Other methods are indication received from NMP-1 or JAFNPP instrumentation.

Evaluation of the magnitude of the event will require evaluation of data recorded by the Seismic Monitoring Recorders.

NMP1 and NMP2 share a common PROTECTED AREA border. Consideration should be given to the opposite unit when classifying under this EAL.

Generic

This EAL is categorized on the basis of the occurrence of an event of sufficient magnitude to be of concern to plant operators.

Damage may be caused to some portions of the site, but should not affect ability of safety functions to operate.

As defined in the EPRI-sponsored Guidelines for Nuclear Plant Response to an Earthquake, dated October 1989, a "felt earthquake" is: An earthquake of sufficient intensity such that: (a) the vibratory ground motion is felt at the nuclear plant site and recognized as an earthquake based on a consensus of control room operators on duty at the time, and (b) for plants with operable seismic instrumentation, the seismic switches of the plant are activated.

NMP2 Basis Reference(s):

1. USAR Section 3.7
2. Technical Requirements Manual Section 3.3.7.2
3. N2-OP-90 Seismic Monitor
4. N2-SOP-90 Natural Events
5. USAR Section 2.1.1.1
6. NEI 99-01 IC HU1

Category: H – Hazards and Other Conditions Affecting Plant Safety

Subcategory: 1 – Natural or Destructive Phenomena

Initiating Condition: Natural or destructive phenomena affecting the PROTECTED AREA

EAL:

HU1.2	**Unusual Event**
Tornado striking within PROTECTED AREA boundary	
OR	
Sustained high winds > 90 mph	

Mode Applicability:

All

Basis:

Plant-Specific

All Category 1 safe shutdown structures are designed for a wind velocity of 90 mph, 30 feet above ground using a gust factor of 1.1 (ref. 1).

Weather conditions are monitored at three locations:

- The 200 foot high Primary OR Main Meteorological Tower located 0.6 miles west-southwest of NMP2
- The 90 foot Backup Tower located east of JAFNPP
- The 30 foot Inland Tower located at the Oswego County Airport near Fulton

Meteorological parameters such as wind speed are sent to the Control Rooms and Technical Support Centers (TSC) at NMP1, NMP2, JAFNPP and the Emergency Operations Facility (EOF). Data from sensors mounted on these towers are sent to both digital and analog systems for display, processing and storage. Wind speed and wind direction, as well as wind speed deviation and differential temperatures are monitored in NMP2 Control Room and recorded on strip chart recorders. (ref. 2)

Wind speed can be measured up to 100 mph.

Weather information may be obtained from (ref. 3):

- National Weather Service: 716-565-9001 or 800-462-7751
- Accu-Weather: 815-235-8650 or 814-237-5803

NMP1 and NMP2 share a common PROTECTED AREA border. Consideration should be given to the opposite unit when classifying under this EAL.

Generic

This EAL is categorized on the basis of the occurrence of an event of sufficient magnitude to be of concern to plant operators.

This EAL is based on a tornado striking (touching down) or high winds within the PROTECTED AREA.

Escalation of this emergency classification level, if appropriate, would be based on VISIBLE DAMAGE, or by other in plant conditions, via EAL HA1.2.

NMP2 Basis Reference(s):

1. USAR Section 3.3.1.1
2. N2-OP-102 Meteorological Monitoring
3. N2-SOP-90 Natural Events
4. NEI 99-01 IC HU1

Category: H – Hazards and Other Conditions Affecting Plant Safety

Subcategory: 1 – Natural or Destructive Phenomena

Initiating Condition: Natural or destructive phenomena affecting the PROTECTED AREA

EAL:

HU1.3	**Unusual Event**
Internal flooding that has the potential to affect **ANY** SAFETY-RELATED STRUCTURE, SYSTEM or COMPONENT required by Technical Specifications for the current operating mode in **ANY** Table H-1 area	

Table H-1 Safe Shutdown Areas
• Reactor Building (including Primary Containment) • Control Room • Diesel Generator Engine and Board Rooms • Standby Switchgear and Battery Rooms • HPCS Switchgear and Battery Rooms • Remote Shutdown Rooms • Control Building HVAC Rooms • Service Water Pump Rooms • Electrical Protection Assembly Room • PGCC Relay Room

Mode Applicability:

All

Basis:

Plant-Specific

This threshold addresses the affect of flooding caused by internal events such as component failures, Circulating, Component Cooling or Service Water line ruptures, equipment misalignment, FIRE suppression system actuation, and outage activity mishaps.

Table H-1 Safe Shutdown Areas include all structures containing Category I equipment and systems needed for safe shutdown (ref. 1).

Flooding as used in this EAL describes a condition where water is entering the room faster than installed equipment is capable of removal, resulting in a rise of water level within the room. Classification of this EAL should not be delayed while corrective actions are being taken to isolate the water source.

Generic

This EAL is categorized on the basis of the occurrence of an event of sufficient magnitude to be of concern to plant operators.

This EAL addresses the effect of internal flooding caused by events such as component failures, equipment misalignment, or outage activity mishaps.

Escalation of this emergency classification level, if appropriate, would be based VISIBLE DAMAGE via EAL HA1.3, or by other plant conditions.

NMP2 Basis Reference(s):

1. USAR 9B and Figure 9B.6-1
2. NEI 99-01 IC HU1

Category: H – Hazards and Other Conditions Affecting Plant Safety

Subcategory: 1 – Natural or Destructive Phenomena

Initiating Condition: Natural or destructive phenomena affecting the PROTECTED AREA

EAL:

HU1.4 Unusual Event

Turbine failure resulting in **ANY** of the following:

- Casing penetration
- Damage to turbine seals
- Damage to generator seals

Mode Applicability:

All

Basis:

Plant-Specific

The turbine generator stores large amounts of rotational kinetic energy in its rotor. In the unlikely event of a major mechanical failure, this energy may be transformed into both rotational and translational energy of rotor fragments. These fragments may impact the surrounding stationary parts. If the energy-absorbing capability of these stationary turbine generator parts is insufficient, external PROJECTILEs will be released. These ejected PROJECTILEs may impact various plant structures, including those housing safety related equipment.

In the event of PROJECTILE ejection, the probability of a strike on a plant region is a function of the energy and direction of an ejected PROJECTILE and of the orientation of the turbine with respect to the plant region.

Failure of turbine or generator seals may be indicated by a loss of seal oil pressure or loss of condenser vacuum (ref. 2, 3).

Generic

These EALs are categorized on the basis of the occurrence of an event of sufficient magnitude to be of concern to plant operators.

This EAL addresses main turbine rotating component failures of sufficient magnitude to cause observable damage to the turbine casing or to the seals of the turbine generator. Generator seal damage observed after generator purge does not meet the intent of this EAL because it did not impact normal operation of the plant.

Of major concern is the potential for leakage of combustible fluids (lubricating oils) and gases (hydrogen cooling) to the plant environs. Actual FIRES and flammable gas build up are appropriately classified via EAL HU2.1 and EAL HU3.1.

This EAL is consistent with the definition of a UE while maintaining the anticipatory nature desired and recognizing the risk to non-safety related equipment.

Escalation of this emergency classification level, if appropriate, would be to EAL HA1.4 based on damage done by PROJECTILES generated by the failure or in conjunction with a steam generator tube rupture. These latter events would be classified by the Category R EALs or Category F EALs.

NMP2 Basis Reference(s):

1. N2-OP-21 Main Turbine System
2. N2-SOP-21 Turbine Trip
3. N2-ARP-01 Annunciator Response Procedures for annunciator 851102
4. N2-ARP-01 Annunciator Response Procedures for annunciator 851140
5. N2-SOP-09 Loss of Condenser Vacuum
6. NEI 99-01 IC HU1

Category: H – Hazards and Other Conditions Affecting Plant Safety

Subcategory: 1 – Natural or Destructive Phenomena

Initiating Condition: Natural or destructive phenomena affecting the PROTECTED AREA

EAL:

HU1.5 Unusual Event
Lake water level > 248.2 ft
OR
Intake water level < 237 ft

Mode Applicability:

All

Basis:

Plant-Specific

This threshold addresses high and low lake water level conditions that could be a precursor of more serious events.

The high lake level is based upon the maximum attainable uncontrolled lake water level as specified in the USAR. Dams on the St. Lawrence River, under the authority of the International St. Lawrence River Board of Control, are now used to regulate the lake level. The low limit is set for el 74.37 m (244 ft) on April 1 and is maintained at or above that elevation during the entire navigation season (April 1 to November 30). The upper limit of the lake level is el 75.59 m (248.2 ft) (ref. 1).

The low level is based on intake water level and corresponds to the design minimum lake level. The probable minimum low water level of Lake Ontario at the site has been determined to be 72.0 m (236.3 ft) resulting from a setdown caused by a Probable Maximum Wind Storm concurrent with the lowest probable lake level. (ref. 2)

Generic

This EAL is categorized on the basis of the occurrence of an event of sufficient magnitude to be of concern to plant operators.

This EAL addresses other site specific phenomena that can also be precursors of more serious events.

NMP2 Basis Reference(s):

1. USAR Section 2.4.1.2
2. USAR Section 2.4.11.2
3. N2-OSP-LOG-W001, Weekly Checks
4. NEI 99-01 IC HU1

Category: H – Hazards and Other Conditions Affecting Plant Safety

Subcategory: 2 – FIRE or EXPLOSION

Initiating Condition: FIRE or EXPLOSION affecting the operability of plant safety systems required to establish or maintain safe shutdown

EAL:

HA2.1 Alert

FIRE or EXPLOSION
resulting in **EITHER**:

VISIBLE DAMAGE to **ANY** SAFETY-RELATED STRUCTURE, SYSTEM or COMPONENT within **ANY** Table H-1 area

OR

Control Room indication of degraded performance of **ANY** SAFETY-RELATED STRUCTURE, SYSTEM or COMPONENT within **ANY** Table H-1 area

Table H-1 Safe Shutdown Areas
• Reactor Building (including Primary Containment) • Control Room • Diesel Generator Engine and Board Rooms • Standby Switchgear and Battery Rooms • HPCS Switchgear and Battery Rooms • Remote Shutdown Rooms • Control Building HVAC Rooms • Service Water Pump Rooms • Electrical Protection Assembly Room • PGCC Relay Room

Mode Applicability:

All

Basis:

Plant-Specific

Table H-1 Safe Shutdown Areas include all structures containing Category I equipment and systems needed for safe shutdown (ref. 1).

Generic

VISIBLE DAMAGE is used to identify the magnitude of the FIRE or EXPLOSION and to discriminate against minor FIREs and EXPLOSIONs.

The reference to structures containing safety systems or components is included to discriminate against FIREs or EXPLOSIONs in areas having a low probability of affecting safe operation. The significance here is not that a safety system was degraded but the fact that the FIRE or EXPLOSION was large enough to cause damage to these systems.

The use of VISIBLE DAMAGE should not be interpreted as mandating a lengthy damage assessment prior to classification. The declaration of an Alert and the activation of the Technical Support Center will provide the Emergency Director with the resources needed to perform detailed damage assessments.

The Emergency Director also needs to consider any security aspects of the EXPLOSION.

Escalation of this emergency classification level, if appropriate, will be based on EALs in Category S, Category F or Category R.

NMP2 Basis Reference(s):

1. USAR 9B and USAR Figure 9B.6-1
2. NEI 99-01 IC HA2

Category: H – Hazards and Other Conditions Affecting Plant Safety

Subcategory: 2 – FIRE or EXPLOSION

Initiating Condition: FIRE within the PROTECTED AREA **not** extinguished within 15 min. of detection or EXPLOSION within the PROTECTED AREA

EAL:

HU2.1 **Unusual Event**
FIRE **not** extinguished within 15 min. of Control Room notification or verification of a Control Room FIRE alarm in **ANY** Table H-1 area or Turbine Building (Note 4)

Note 4: The ED should **not** wait until the applicable time has elapsed, but should declare the event as soon as it is determined that the condition has exceeded, or will likely exceed, the applicable time.

Table H-1 Safe Shutdown Areas
• Reactor Building (including Primary Containment) • Control Room • Diesel Generator Engine and Board Rooms • Standby Switchgear and Battery Rooms • HPCS Switchgear and Battery Rooms • Remote Shutdown Rooms • Control Building HVAC Rooms • Service Water Pump Rooms • Electrical Protection Assembly Room • PGCC Relay Room

Mode Applicability:

All

Basis:

Plant-Specific

Table H-1 Safe Shutdown Areas include all structures containing Category I equipment and systems needed for safe shutdown (ref. 1). The Turbine Building is included because it is immediately adjacent to one or more Table H-1 areas and a FIRE within the Turbine Building may potentially impact safe shutdown equipment should the FIRE not be controlled.

Generic

This EAL addresses the magnitude and extent of FIREs that may be potentially significant precursors of damage to safety systems. It addresses the FIRE, and not the degradation in performance of affected systems that may result.

As used here, detection is visual observation and either report by plant personnel or sensor alarm indication.

The 15 minute time period begins with a credible notification that a FIRE is occurring, or indication of a FIRE detection system alarm/actuation. Verification of a FIRE detection system alarm/actuation includes actions that can be taken within the control room or other nearby site specific location to ensure that it is not spurious. An alarm is assumed to be an indication of a FIRE unless it is disproved within the 15 minute period by personnel dispatched to the scene. In other words, a personnel report from the scene may be used to disprove a sensor alarm if received within 15 minutes of the alarm, but shall not be required to verify the alarm.~~The purpose of this threshold is to address the magnitude and extent of FIREs that may be potentially significant precursors to damage to safety systems. As used here, notification is visual observation and either report by plant personnel or sensor alarm indication. The 15-minute period to extinguish the FIRE begins with a credible notification that a FIRE is occurring or indication of a VALID FIRE detection system alarm. Determination of a VALID FIRE detection system alarm includes actions that can be taken within the Control Room or at nearby FIRE Panels to determine that the alarm is not spurious. These actions include the use of direct or indirect indications such as redundant alarms or instrumentation readings associated with the area to ensure the alarm is not spurious and is an indication of a FIRE. An alarm verified in this manner is assumed to be an indication of a FIRE unless personnel dispatched to the scene disprove the alarm within the 15-minute period. The report, however, shall not be required to verify the alarm. If the alarm cannot be verified by redundant Control Room or nearby FIRE Panel indications, notification from the field that a FIRE exists would be required to start both the 15-minute classification and FIRE extinguishment clocks.~~

The intent of this 15 minute duration is to size the FIRE and to discriminate against small FIREs that are readily extinguished (e.g., smoldering waste paper basket).

NMP2 Basis Reference(s):

1. USAR 9B and Figure 9B.6-1
2. NEI 99-01 IC HU2

Category: H – Hazards and Other Conditions Affecting Plant Safety

Subcategory: 2 – FIRE or EXPLOSION

Initiating Condition: FIRE within the PROTECTED AREA **not** extinguished within 15 min. of detection or EXPLOSION within the PROTECTED AREA

EAL:

HU2.2	**Unusual Event**
EXPLOSION of sufficient force to damage permanent structures or equipment within the PROTECTED AREA	

Mode Applicability:

All

Basis:

Plant-Specific

While some EXPLOSIONs may also result in FIREs that exceed EAL HU2.1, no FIRE is necessary to declare an emergency in the event of an EXPLOSION. If a FIRE also occurs as a result or with an EXPLOSION, declare the Unusual Event based on the EXPLOSION and monitor the progress of the FIRE for potential escalation due to FIRE damage.

NMP1 and NMP2 share a common PROTECTED AREA border. NMP1 and NMP2 PROTECTED AREA boundaries are illustrated in USAR Figure 1.2-1 (ref. 1).

Generic

This EAL addresses the magnitude and extent of EXPLOSIONs that may be potentially significant precursors of damage to safety systems. It addresses the EXPLOSION, and not the degradation in performance of affected systems that may result.

This EAL addresses only those EXPLOSIONs of sufficient force to damage permanent structures or equipment within the PROTECTED AREA.

No attempt is made to assess the actual magnitude of the damage. The occurrence of the EXPLOSION is sufficient for declaration.

The Emergency director also needs to consider any security aspects of the EXPLOSION, if applicable.

Escalation of this emergency classification level, if appropriate, would be based on EAL HA2.1.

NMP2 Basis Reference(s):

1. USAR Figure 1.2-1
2. NEI 99-01 IC HU2

Category: H – Hazards and Other Conditions Affecting Plant Safety

Subcategory: 3 – Hazardous Gas

Initiating Condition: Access to a VITAL AREA is prohibited due to toxic, corrosive, asphyxiant or flammable gases which jeopardize operation of operable equipment required to maintain safe operations or safely shutdown the reactor

EAL:

HA3.1 Alert

Access to **ANY** Table H-1 area is prohibited due to toxic, corrosive, asphyxiant or flammable gases which jeopardize operation of systems required to maintain safe operations or safely shutdown the reactor ~~**ANY** SAFETY-RELATED STRUCTURE, SYSTEM or COMPONENT~~ (Note 5)

Note 5: If the equipment in the stated area was already inoperable, or out of service, before the event occurred, then EAL HA3.1 should **not** be declared as it will have **no** adverse impact on the ability of the plant to safely operate or safely shutdown beyond that already allowed by Technical Specifications at the time of the event.

Table H-1 Safe Shutdown Areas
• Reactor Building (including Primary Containment) • Control Room • Diesel Generator Engine and Board Rooms • Standby Switchgear and Battery Rooms • HPCS Switchgear and Battery Rooms • Remote Shutdown Rooms • Control Building HVAC Rooms • Service Water Pump Rooms • Electrical Protection Assembly Room • PGCC Relay Room

Mode Applicability:

All

Basis:

Plant-Specific

Table H-1 Safe Shutdown Areas include all structures containing Category I equipment and systems needed for safe shutdown (ref. 1).

For areas that contain no safety-related structure, system or component that would potentially be required to be operated or for which the structure, system or component was already out of service or inoperable before the event, this EAL would not be applicable.

For purposes of this EAL, any gas (CO_2 included) is considered toxic when oxygen concentrations in the affected areas have been or could be expected to be reduced to <19.5% or toxicity of the gas will be injurious to persons inhaling it. For discharges of Halon, NMP's systems are designed for discharge concentration from 5% up to 6.5%. In accordance with NFPA 12 A, Halon 1301 Fire Extinguishing Systems, exposures to levels of up to 7% produce little if any noticeable effect (ref. 2).

Generic

Gases in a Safe Shutdown AREA can affect the ability to safely operate or safely shutdown the reactor.

The fact that SCBA may be worn does not eliminate the need to declare the event.

Declaration should not be delayed for confirmation from atmospheric testing if the atmosphere poses an immediate threat to life and health or an immediate threat of severe exposure to gases. This could be based upon documented analysis, indication of personal ill effects from exposure, or operating experience with the hazards.

If the equipment in the stated area was already inoperable, or out of service, before the event occurred, then this EAL should not be declared as it will have no adverse impact on the ability of the plant to safely operate or safely shutdown beyond that already allowed by Technical Specifications at the time of the event.

An asphyxiant is a gas capable of reducing the level of oxygen in the body to dangerous levels. Most commonly, asphyxiants work by merely displacing air in an enclosed environment. This reduces the concentration of oxygen below the normal level of around 19%, which can lead to breathing difficulties, unconsciousness or even death.

An uncontrolled release of flammable gasses within a facility structure has the potential to affect safe operation of the plant by limiting either operator or equipment operations due to the potential for ignition and resulting equipment damage/personnel injury. Flammable gasses, such as hydrogen and acetylene, are routinely used to maintain plant systems (hydrogen) or to repair equipment/components (acetylene - used in welding). This EAL assumes concentrations of flammable gasses which can ignite/support combustion.

Escalation of this emergency classification level, if appropriate, will be based on EALs in Category S, Category F or Category R.

NMP2 Basis Reference(s):

1. USAR 9B and Figure 9B.6-1
2. NFPA 12 A Halon 1301 Fire Extinguishing Systems
3. NEI 99-01 IC HA3

Category: H – Hazards and Other Conditions Affecting Plant Safety

Subcategory: 3 – Hazardous Gas

Initiating Condition: Release of toxic, corrosive, asphyxiant or flammable gases deemed detrimental to NORMAL PLANT OPERATIONS

EAL:

HU3.1	**Unusual Event**
Toxic, corrosive, asphyxiant or flammable gases in amounts that have or could adversely affect NORMAL PLANT OPERATIONS	

Mode Applicability:

All

Basis:

Plant-Specific

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

For purposes of this EAL, any gas (CO_2 included) is considered toxic when oxygen concentrations in the affected areas have been or could be expected to be reduced to <19.5% or toxicity of the gas will be injurious to persons inhaling it. For discharges of Halon, NMP's systems are designed for discharge concentration from 5% up to 6.5%. In accordance with NFPA 12 A, Halon 1301 Fire Extinguishing Systems, exposures to levels of up to 7% produce little if any noticeable effect (ref. 1).

Generic

This EAL is based on the release of toxic, corrosive, asphyxiant or flammable gases of sufficient quantity to affect NORMAL PLANT OPERATIONS.

The fact that SCBA may be worn does not eliminate the need to declare the event.

This EAL is not intended to require significant assessment or quantification. It assumes an uncontrolled process that has the potential to affect plant operations. This would preclude small or incidental releases, or releases that do not impact structures needed for plant operation.

An asphyxiant is a gas capable of reducing the level of oxygen in the body to dangerous levels.

Most commonly, asphyxiants work by merely displacing air in an enclosed environment. This reduces the concentration of oxygen below the normal level of around 19%, which can lead to breathing difficulties, unconsciousness or even death.

Escalation of this emergency classification level, if appropriate, would be based on EAL HA3.1.

NMP2 Basis Reference(s):

1. NFPA 12 A Halon 1301 Fire Extinguishing Systems
2. NEI 99-01 IC HU3

Category: H – Hazards and Other Conditions Affecting Plant Safety

Subcategory: 3 – Hazardous Gas

Initiating Condition: *Release of toxic, corrosive, asphyxiant or flammable gases* deemed detrimental to NORMAL PLANT OPERATIONS

EAL:

HU3.2	**Unusual Event**
Recommendation by local, county or state officials to evacuate or shelter site personnel based on an offsite event	

Mode Applicability:

All

Basis:

Plant-Specific

A recommendation by offsite officials that a potential evacuation of site personnel may be required based on an offsite event assumes that the plant lies within an evacuation area established by offsite officials due to a release of toxic, corrosive, asphyxiant or flammable gas. In this case, it can be assumed that an actual or potential release of such hazardous gas is anticipated to enter the PROTECTED AREA in amounts that could affect the health of plant personnel or NORMAL PLANT OPERATIONS..

Generic

Escalation of this emergency classification level, if appropriate, would be based on EAL HA3.1.

NMP2 Basis Reference(s):

1. NEI 99-01 IC HU3

Category: H – Hazards and Other Conditions Affecting Plant Safety

Subcategory: 4 – Security

Initiating Condition: HOSTILE ACTION resulting in loss of physical control of the facility

EAL:

HG4.1	**General Emergency**
A HOSTILE ACTION has occurred such that plant personnel are unable to operate equipment required to maintain safety functions	

Mode Applicability:

All

Basis:

Plant-Specific

Safety functions include:

- Reactivity control – ability to shut down the reactor and keep it shutdown
- RPV level control – ability to cool the core
- Decay heat removal – ability to maintain a heat sink

Generic

This EAL encompasses conditions under which a HOSTILE ACTION has resulted in a loss of physical control of VITAL AREAs (containing vital equipment or controls of vital equipment) required to maintain safety functions and control of that equipment cannot be transferred to and operated from another location.

If control of the plant equipment necessary to maintain safety functions can be transferred to another location, then the threshold is not met.

NMP2 Basis Reference(s):

1. NEI 99-01 IC HG1

Category: H – Hazards and Other Conditions Affecting Plant Safety

Subcategory: 4 – Security

Initiating Condition: HOSTILE ACTION resulting in loss of physical control of the facility

EAL:

HG4.2 General Emergency

A HOSTILE ACTION has caused failure of Spent Fuel Cooling systems

AND

IMMINENT fuel damage is likely

Mode Applicability:

All

Basis:

Plant-Specific

None

Generic

This EAL addresses failure of spent fuel cooling systems as a result of HOSTILE ACTION if IMMINENT fuel damage is likely.

NMP2 Basis Reference(s):

1. NEI 99-01 IC HG1

Category: H – Hazards and Other Conditions Affecting Plant Safety

Subcategory: 4 – Security

Initiating Condition: HOSTILE ACTION within the PROTECTED AREA

EAL:

HS4.1	**Site Area Emergency**
A HOSTILE ACTION is occurring or has occurred within the PROTECTED AREA as reported by the Security Site Supervisor	

Mode Applicability:

All

Basis:

Generic

This condition represents an escalated threat to plant safety above that contained in the Alert in that a HOSTILE FORCE has progressed from the Owner Controlled Area to the PROTECTED AREA.

This EAL addresses the contingency for a very rapid progression of events, such as that experienced on September 11, 2001. It is not premised solely on the potential for a radiological release. Rather the issue includes the need for rapid assistance due to the possibility for significant and indeterminate damage from additional air, land or water attack elements.

The fact that the site is under serious attack with minimal time available for further preparation or additional assistance to arrive requires Offsite Response Organization (ORO) readiness and preparation for the implementation of protective measures.

This EAL addresses the potential for a very rapid progression of events due to a HOSTILE ACTION. It is not intended to address incidents that are accidental events or acts of civil disobedience, such as small aircraft impact, hunters, or physical disputes between employees within the PROTECTED AREA. Those events are adequately addressed by other EALs.

Escalation of this emergency classification level, if appropriate, would be based on actual plant status after impact or progression of attack.

NMP2 Basis Reference(s):

1. NMP Site Security Plan
2. NEI 99-01 IC HS4

Category: H – Hazards and Other Conditions Affecting Plant Safety

Subcategory: 4 – Security

Initiating Condition: HOSTILE ACTION within the Owner Controlled Area or airborne attack threat

EAL:

HA4.1 Alert
A HOSTILE ACTION is occurring or has occurred within the Owner Controlled Area as reported by the Security Site Supervisor
OR
A validated notification from NRC of an AIRLINER attack threat within 30 min. of the site

Mode Applicability:

All

Basis:

Plant-Specific

None

Generic

Note: Timely and accurate communication between the Security Site Supervisor and the Control Room is crucial for the implementation of effective Security EALs.

This EAL addresses the contingency for a very rapid progression of events, such as that experienced on September 11, 2001. They are not premised solely on the potential for a radiological release. Rather the issue includes the need for rapid assistance due to the possibility for significant and indeterminate damage from additional air, land or water attack elements.

The fact that the site is under serious attack or is an identified attack target with minimal time available for further preparation or additional assistance to arrive requires a heightened state of readiness and implementation of protective measures that can be effective (such as on-site evacuation, dispersal or sheltering).

First Condition

This condition addresses the potential for a very rapid progression of events due to a HOSTILE ACTION. It is not intended to address incidents that are accidental events or acts of civil disobedience, such as small aircraft impact, hunters, or physical disputes between employees within the Owner Controlled Area. Those events are adequately addressed by other EALs.

Note that this condition is applicable for any HOSTILE ACTION occurring, or that has occurred, in the Owner Controlled Area.

Second Condition

This condition addresses the immediacy of an expected threat arrival or impact on the site within a relatively short time.

The intent of this condition is to ensure that notifications for the AIRLINER attack threat are made in a timely manner and that Offsite Response Organizations (OROs) and plant personnel are at a state of heightened awareness regarding the credible threat. AIRLINER is meant to be a LARGE AIRCRAFT with the potential for causing significant damage to the plant.

This condition is met when a plant receives information regarding an AIRLINER attack threat from NRC and the AIRLINER is within 30 minutes of the plant. Only the plant to which the specific threat is made need declare the Alert.

The NRC Headquarters Operations Officer (HOO) will communicate to the licensee if the threat involves an AIRLINER (AIRLINER is meant to be a LARGE AIRCRAFT with the potential for causing significant damage to the plant). The status and size of the plane may be provided by NORAD through the NRC.

NMP2 Basis Reference(s):

1. NMP Site Security Plan
2. NEI 99-01 IC HA4

Category: H – Hazards and Other Conditions Affecting Plant Safety

Subcategory: 4 – Security

Initiating Condition: Confirmed SECURITY CONDITION or threat which indicates a potential degradation in the level of safety of the plant

EAL:

HU4.1	**Unusual Event**

A SECURITY CONDITION that does **not** involve a HOSTILE ACTION as reported by the Security Site Supervisor

OR

A credible site-specific security threat notification

OR

A validated notification from NRC providing information of an aircraft threat

Mode Applicability:

All

Basis:

Plant-Specific

If the Security Site Supervisor determines that a threat notification is credible, the Security Site Supervisor will notify the Operations Shift Manager that a "Credible Threat" condition exists for NMP2. Generally, NMP2 Security Procedures address standard practices for determining credibility. The three main criteria for determining credibility are: technical feasibility, operational feasibility, and resolve. For NMP2, a validated notification delivered by the FBI, the NRC or similar agency is treated as credible.

Generic

Note: Timely and accurate communication between Security the Site Supervisor and the Control Room is crucial for the implementation of effective Security EALs.

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 in some cases under 10 CFR 50.72. Security events assessed as HOSTILE ACTIONs are classifiable under EAL HA4.1, EAL HS4.1 and EAL HG4.1.

A higher initial classification could be made based upon the nature and timing of the security threat and potential consequences. The licensee shall consider upgrading the emergency response status and emergency classification level in accordance with the NMP Site Security and Plan.

First Condition

Reference is made to security shift supervision because these individuals are the designated personnel on-site qualified and trained to confirm that a security event is occurring or has occurred. Training on security event classification confirmation is closely controlled due to the strict secrecy controls placed on the NMP Site Security Plan.

This threshold is based on the NMP Site Security Plan. The NMP Site Security Plan is based on guidance provided by NEI 03-12.

Second Condition

This threshold is included to ensure that appropriate notifications for the security threat are made in a timely manner. This includes information of a credible threat. Only the plant to which the specific threat is made need declare the Unusual Event.

The determination of "credible" is made through use of information found in the NMP Site Security Plan .

Third Condition

The intent of this EAL is to ensure that notifications for the aircraft threat are made in a timely manner and that Offsite Response Organizations and plant personnel are at a state of heightened awareness regarding the credible threat. It is not the intent of this EAL to replace existing non-hostile related EALs involving aircraft.

This EAL is met when a plant receives information regarding an aircraft threat from NRC. Validation is performed by calling the NRC or by other approved methods of authentication. Only the plant to which the specific threat is made need declare the Unusual Event.

The NRC Headquarters Operations Officer (HOO) will communicate to the licensee if the threat involves an AIRLINER (AIRLINER is meant to be a LARGE AIRCRAFT with the potential for causing significant damage to the plant). The status and size of the plane may be provided by NORAD through the NRC.

Escalation to Alert emergency classification level via EAL HA4.1 would be appropriate if the threat involves an AIRLINER within 30 minutes of the plant.

NMP2 Basis Reference(s):

1. NMP Site Security Plan
2. NEI 99-01 IC HU4

Category: H – Hazards and Other Conditions Affecting Plant Safety

Subcategory: 5 – Control Room Evacuation

Initiating Condition: Control Room evacuation has been initiated and plant control **cannot** be established

EAL:

HS5.1 Site Area Emergency

Control Room evacuation has been initiated

AND

Control of the plant **cannot** be established within 15 min. ~~(Note 4)~~

~~Note 4: The ED should **not** wait until the applicable time has elapsed, but should declare the event as soon as it is determined that the condition has exceeded, or will likely exceed, the applicable time.~~

Mode Applicability:

All

Basis:

Plant-Specific

N2-SOP-78, Control Room Evacuation, provides specific instructions for evacuating the Control Room/Building and establishing plant control in alternate locations.

Generic

The intent of this EAL is to capture those events where control of the plant cannot be reestablished in a timely manner. In this case, expeditious transfer of control of safety systems has not occurred (although fission product barrier damage may not yet be indicated).

The intent of the EAL is to establish control of important plant equipment and knowledge of important plant parameters in a timely manner. Primary emphasis should be placed on those components and instruments that supply protection for and information about safety functions. Typically, these safety functions are reactivity control (ability to reach and maintain reactor shutdown), reactor water level (ability to cool the core), and decay heat removal (ability to maintain a heat sink).

The determination of whether or not control is established at the remote shutdown panel is based on Emergency Director (ED) judgment. The Emergency Director is expected to make a reasonable, informed judgment within the site specific time for transfer that the licensee has control of the plant from the remote shutdown panel.

Escalation of this emergency classification level, if appropriate, would be by EALs in Category F or Category R.

NMP2 Basis Reference(s):

1. N2-SOP-78 Control Room Evacuation

2. USAR Section 9B.8.2.2
3. NEI 99-01 IC HS2

Category: H – Hazards and Other Conditions Affecting Plant Safety

Subcategory: 5 – Control Room Evacuation

Initiating Condition: Control Room evacuation has been initiated

EAL:

HA5.1	**Alert**
Control Room evacuation has been initiated	

Mode Applicability:

All

Basis:

Plant-Specific

N2-SOP-78, Control Room Evacuation, provides specific instructions for evacuating the Control Room/Building and establishing plant control in alternate locations.

Generic

With the control room evacuated, additional support, monitoring and direction through the Technical Support Center and/or other emergency response facilities may be necessary.

Inability to establish plant control from outside the control room will escalate this event to a Site Area Emergency.

NMP2 Basis Reference(s):

1. N2-SOP-78 Control Room Evacuation
2. USAR Section 9B.8.2.2
3. NEI 99-01 IC HA5

Category: H – Hazards and Other Conditions Affecting Plant Safety

Subcategory: 6 – Judgment

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

EAL:

HG6.1	**General Emergency**
Other conditions exist which in the judgment of the Emergency Director indicate that events are in progress or have occurred which involve actual or IMMINENT substantial core degradation or melting with potential for loss of containment integrity or HOSTILE ACTION that results in an actual loss of physical control of the facility. Releases can be reasonably expected to exceed EPA Protective Action Guideline exposure levels (1,000 mRem TEDE or 5,000 mRem thyroid CDE) offsite for more than the immediate site area	

Mode Applicability:

All

Basis:

Plant-Specific

None

Generic

This EAL addresses unanticipated conditions not addressed explicitly elsewhere but that warrant declaration of an emergency because conditions exist which are believed by the Emergency Director to fall under the emergency classification level description for General Emergency.

NMP2 Basis Reference(s):

1. NEI 99-01 IC HG2

Category: H – Hazards and Other Conditions Affecting Plant Safety

Subcategory: 6 – Judgment

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

EAL:

HS6.1 **Site Area Emergency**
Other conditions exist which in the judgment of the Emergency Director indicate that events are in progress or have occurred which involve actual or likely major failures of plant functions needed for protection of the public or HOSTILE ACTION that results in intentional damage or malicious acts; (1) toward site personnel or equipment that could lead to the likely failure of or; (2) that prevent effective access to equipment needed for the protection of the public. **ANY** releases are **not** expected to result in exposure levels which exceed EPA Protective Action Guideline exposure levels (1,000 mRem TEDE or 5,000 mRem thyroid CDE) beyond the SITE BOUNDARY

Mode Applicability:

All

Basis:

Plant-Specific

None

Generic

This EAL addresses unanticipated conditions not addressed explicitly elsewhere but that warrant declaration of an emergency because conditions exist which are believed by the Emergency Director to fall under the emergency classification level description for Site Area Emergency.

NMP2 Basis Reference(s):

1. NEI 99-01 IC HS3

Category: H – Hazards and Other Conditions Affecting Plant Safety

Subcategory: 6 – Judgment

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

EAL:

HA6.1 **Alert**
Other conditions exist which in the judgment of the Emergency Director indicate that events are in progress or have occurred which involve an actual or potential substantial degradation of the level of safety of the plant or a security event that involves probable life threatening risk to site personnel or damage to site equipment because of HOSTILE ACTION. **ANY** releases are expected to be limited to small fractions of the EPA Protective Action Guideline exposure levels (1,000 mRem TEDE or 5,000 mRem thyroid CDE)

Mode Applicability:

All

Basis:

Plant-Specific

None

Generic

This EAL addresses unanticipated conditions not addressed explicitly elsewhere but that warrant declaration of an emergency because conditions exist which are believed by the Emergency Director to fall under the Alert emergency classification level.

NMP2 Basis Reference(s):

1. NEI 99-01 IC HA6

Category: H – Hazards and Other Conditions Affecting Plant Safety

Subcategory: 6 – Judgment

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

EAL:

HU6.1	**Unusual Event**
Other conditions exist which in the judgment of the Emergency Director indicate that events are in progress or have occurred which indicate a potential degradation of the level of safety of the plant or indicate a security threat to facility protection has been initiated. **No** releases of radioactive material requiring offsite response or monitoring are expected unless further degradation of safety systems occurs	

Mode Applicability:

All

Basis:

<u>Plant-Specific</u>

None

<u>Generic</u>

This EAL addresses unanticipated conditions not addressed explicitly elsewhere but that warrant declaration of an emergency because conditions exist which are believed by the Emergency Director to fall under the UE emergency classification level.

NMP2 Basis Reference(s):

1. NEI 99-01 IC HU5

Category E – INDEPENDENT SPENT FUEL STORAGE INSTALLATION (ISFSI)

EAL Group: Not Applicable (the EAL in this category is applicable independent of plant operating mode)

An INDEPENDENT SPENT FUEL STORAGE INSTALLATION (ISFSI) is a complex that is designed and constructed for the interim storage of spent nuclear fuel and other radioactive materials associated with spent fuel storage. A significant amount of the radioactive material contained within a cask/canister must escape its packaging and enter the biosphere for there to be a significant environmental effect resulting from an accident involving the dry storage of spent nuclear fuel. Formal offsite planning is not required because the postulated worst-case accident involving an ISFSI has insignificant consequences to the public health and safety.

A Notification of Unusual Event is declared on the basis of the occurrence of an event of sufficient magnitude that a loaded cask CONFINEMENT BOUNDARY is damaged or violated. This includes classification based on a loaded fuel storage cask/canister CONFINEMENT BOUNDARY loss leading to the degradation of the fuel during storage or posing an operational safety problem with respect to its removal from storage.

A hostile security event that leads to a potential loss in the level of safety of the ISFSI is a classifiable event under Security category EAL HA4.1.

Minor surface damage that does not affect storage cask/canister boundary is excluded from the scope of these EALs.

Category: E – ISFSI

Subcategory: Not Applicable

Initiating Condition: Damage to a loaded cask CONFINEMENT BOUNDARY

EAL:

EU1.1	**Unusual Event**

Damage to a loaded cask CONFINEMENT BOUNDARY as indicated by measured dose rates > then **ANY** of the following:

- 400 mRem/hr at 3 feet from the HSM surface
- 100 mRem/hr outside HSM door on centerline
- 20 mRem/hr end shield wall exterior

Mode Applicability:

All

Basis:

Plant-Specific

The NMP site ISFSI utilizes the NUHOMS Horizontal Modular Storage System.

This EAL addresses any condition which indicates a loss of a cask CONFINEMENT BOUNDARY and thus a potential degradation in the level of safety of the ISFSI. The cask CONFINEMENT BOUNDARY is the NUHOMS 61BT Dry Shielded Canister (DSC). The DSC is the pressure-retaining component of the storage system (ref. 1). Each loaded DSC is housed within a Horizontal Storage Module (HSM). Indication of a loss of CONFINEMENT BOUNDARY is any increase in external HSM radiation levels in excess of Technical Specification limits (ref. 2).

Generic

An UE in this EAL is categorized on the basis of the occurrence of an event of sufficient magnitude that a loaded cask CONFINEMENT BOUNDARY is damaged or violated. This includes classification based on a loaded fuel storage cask CONFINEMENT BOUNDARY loss leading to the degradation of the fuel during storage or posing an operational safety problem with respect to its removal from storage.

NMP2 Basis Reference(s):

1. CDP No. N1-07-092/N2-07-070 Nine Mile Point Nuclear Station - Conceptual Design, Independent Spent Fuel Storage Installation

2. Transnuclear, Inc. Standardized NUHOMS Horizontal Modular Storage System Certificate of Compliance No. 1004, Attachment A Technical Specifications Section 1.2.7 HSM Dose Rates with a Loaded 24P, 52B or 61BT DSC
3. NEI 99-01 IC E-HU1

Category C – Cold Shutdown / Refueling System Malfunction

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

Category C EALs are directly associated with cold shutdown or refueling system safety functions. Given the variability of plant configurations (e.g., systems out-of-service for maintenance, containment open, reduced AC power redundancy, time since shutdown) during these periods, the consequences of any given initiating event can vary greatly. For example, a loss of decay heat removal capability that occurs at the end of an extended outage has less significance than a similar loss occurring during the first week after shutdown. Compounding these events is the likelihood that instrumentation necessary for assessment may also be inoperable. The cold shutdown and refueling system malfunction EALs are based on performance capability to the extent possible with consideration given to RCS integrity, containment closure, and fuel clad integrity for the applicable operating modes (4 - Cold Shutdown, 5 - Refuel, D – Defueled).

The events of this category pertain to the following subcategories:

1. Loss of AC Power

Loss of emergency plant electrical power can compromise plant safety system operability including decay heat removal and emergency core cooling systems which may be necessary to ensure fission product barrier integrity. This category includes loss of onsite and offsite power sources for the 4.16 KV emergency buses.

2. Loss of DC Power

Loss of emergency plant electrical power can compromise plant safety system operability including decay heat removal and emergency core cooling systems which may be necessary to ensure fission product barrier integrity. This category includes loss of power to the 125 VDC buses.

3. RPV Level

RPV water level is a measure of inventory available to ensure adequate core cooling and, therefore, maintain fuel clad integrity. The RPV provides a volume for the coolant that covers the reactor core. The RPV and associated pressure piping (reactor coolant system) together provide a barrier to limit the release of radioactive material should the reactor fuel clad integrity fail.

4. RCS Temperature

Uncontrolled or inadvertent temperature or pressure increases are indicative of a potential loss of safety functions.

5. Inadvertent Criticality

Inadvertent criticalities pose potential personnel safety hazards as well being indicative of losses of reactivity control.

6. Communications

Certain events that degrade plant operator ability to effectively communicate with essential personnel within or external to the plant warrant emergency classification.

Category: C – Cold Shutdown / Refueling System Malfunction

Subcategory: 1 – Loss of AC Power

Initiating Condition: Loss of **all** offsite and **all** onsite AC power to 4.16 KV emergency buses for ≥ 15 min.

EAL:

CA1.1	**Alert**
Loss of **all** offsite and **all** onsite AC power, Table C-1, to 4.16 KV emergency buses 2ENS*SWG101 and *2ENS*SWG103 for ≥ 15 min. (Note 4)	

Note 4: The ED should **not** wait until the applicable time has elapsed, but should declare the event as soon as it is determined that the condition has exceeded, or will likely exceed, the applicable time.

Table C-1 AC Power Sources	
Onsite	• 2EGS*EG1 • 2EGS*EG3
Offsite	• Reserve Transformer A • Reserve Transformer B • Aux Boiler Transformer

Mode Applicability:

4 - Cold Shutdown, 5 - Refuel, D - Defueled

Basis:

Plant-Specific

2ENS*SWG101, 2ENS*SWG102, and 2ENS*SWG103 are the 4.16 KV emergency buses. Bus 2ENS*SWG101 is dedicated to Division I of the On-site Emergency AC Electrical Distribution System, bus 2ENS*SWG102 is dedicated to Division III (HPCS), and bus 2ENS*SWG103 is dedicated to Division II. Buses 2ENS*SWG101 and *SWG103 feed all Station redundant safety-related loads, except the HPCS system loads. The HPCS system loads are fed by bus 2ENS*SWG102 (ref. 1, 2).

- All three divisions are normally energized by the On-site Normal AC Electrical Distribution System via the off-site power sources through the reserve station service transformers 2RTX-XSR1A and 2RTX-XSR1B.

- 2ENS*SWG102 from transformer 2RTX-XSR1A
- 2ENS*SWG103 from transformer 2RTX-XSR1B.

- Buses 2ENS*SWG101 and 2ENS*SWG103 each have a backup source, the Auxiliary Boiler Transformer 2ABS-X1. Also, 2ENS*SWG101 and *SWG103 each have a feeder to a normal AC (stub) bus, NNS-SWG014 and NNS-SWG015 respectively.
- Bus 2ENS*SWG102 has a backup connection to the Reserve Station Service Transformer 2RTX-XSR1B, if required.
- Each of the three 4.16 KV emergency buses has a standby diesel generator (2EGS*EG1, 2EGS*EG3, 2EGS*EG2) to carry its loads in case of a LOOP or in case of a sustained degraded voltage condition on the offsite source (ref. 3, 4).

Consideration should be given to operable loads necessary to remove decay heat or provide RPV makeup capability when evaluating loss of all AC power to vital buses. Even though an essential bus may be energized, if necessary loads (i.e., loads that if lost would inhibit decay heat removal capability or RPV makeup capability) are not operable on the energized bus then the bus should not be considered operable.

The fifteen-minute interval was selected as a threshold to exclude transient power losses.

This EAL is the cold condition equivalent of the hot condition loss of all AC power EAL SS1.1.

Generic

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.

The event can be classified as an Alert when in cold shutdown, refuel, or defueled mode because of the significantly reduced decay heat and lower temperature and pressure, increasing the time to restore one of the emergency buses, relative to that specified for the Site Area Emergency EAL.

Escalating to Site Area Emergency, if appropriate, is by EALs in Category R.

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

NMP2 Basis Reference(s):

1. USAR Section 8.2
2. USAR Section 8.3

3. N2-SOP-03 Loss of AC Power
4. N2-SOP-01 Station Blackout
5. NEI 99-01 IC CA3

Category: C – Cold Shutdown / Refueling System Malfunction

Subcategory: 1 – Loss of AC Power

Initiating Condition: AC power capability to 4.16 KV emergency buses reduced to a single power source for ≥ 15 min. such that **ANY** additional single failure would result in a complete loss of **all** 4.16 KV emergency bus power

EAL:

CU1.1 Unusual Event

AC power capability to 4.16 KV emergency buses 2ENS*SWG101 and 2ENS*SWG103 reduced to a single power source, Table C-1, for ≥ 15 min. (Note 4)

AND

ANY additional single power source failure will result in a loss of **all** power to 4.16 KV emergency buses 2ENS*SWG101 and 2ENS*SWG103

Note 4: The ED should **not** wait until the applicable time has elapsed, but should declare the event as soon as it is determined that the condition has exceeded, or will likely exceed, the applicable time.

Table C-1 AC Power Sources	
Onsite	• 2EGS*EG1 • 2EGS*EG3
Offsite	• Reserve Transformer A • Reserve Transformer B • Aux Boiler Transformer

Mode Applicability:

4 - Cold Shutdown, 5 - Refuel, D - Defueled

Basis:

Plant-Specific

2ENS*SWG101, 2ENS*SWG102, and 2ENS*SWG103 are the 4.16 KV emergency buses. Bus 2ENS*SWG101 is dedicated to Division I of the On-site Emergency AC Electrical Distribution System, bus 2ENS*SWG102 is dedicated to Division III (HPCS), and bus 2ENS*SWG103 is dedicated to Division II. Buses 2ENS*SWG101 and 2ENS*SWG103 feed all Station redundant safety-related loads, except the HPCS system loads. The HPCS system loads are fed by bus 2ENS*SWG102 (ref. 1, 2).

- All three divisions are normally energized by the On-site Normal AC Electrical Distribution System via the off-site power sources through the reserve station service transformers 2RTX-XSR1A and 2RTX-XSR1B.
    - 2ENS*SWG102 from transformer 2RTX-XSR1A
    - 2ENS*SWG103 from transformer 2RTX-XSR1B.
- Buses 2ENS*SWG101 and 2ENS*SWG103 each have a backup source, the Auxiliary Boiler Transformer 2ABS-X1. Also, 2ENS*SWG101 and 2ENS*SWG103 each have a feeder to a normal AC (stub) bus, NNS-SWG014 and NNS-SWG015 respectively.
- Bus 2ENS*SWG102 has a backup connection to the Reserve Station Service Transformer 2RTX-XSR1B, if required.
- Each of the three 4.16 KV emergency buses has a standby diesel generator (2EGS*EG1, 2EGS*EG3, 2EGS*EG2) to carry its loads in case of a LOOP or in case of a sustained degraded voltage condition on the offsite source (ref. 3, 4).

The fifteen-minute interval was selected as a threshold to exclude transient power losses. If multiple sources fail to energize the unit safety-related buses within 15 minutes, an Unusual Event is declared under this EAL. The subsequent loss of the single remaining power source escalates the event to an Alert under EAL CA1.1.

<u>Generic</u>

The condition indicated by this EAL is the degradation of the off-site and on-site AC power systems such that any additional single failure would result in a complete loss of 4.16 KV emergency bus AC power to one or both units. This condition could occur due to a loss of off-site power with a concurrent failure of all but one emergency generator to supply power to its emergency bus. The subsequent loss of this single power source would escalate the event to an Alert in accordance with EAL CA1.1.

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

NMP2 Basis Reference(s):

1. USAR Section 8.2
2. USAR Section 8.3
3. N2-SOP-03 Loss of AC Power

4. N2-SOP-01 Station Blackout
5. NEI 99-01 IC CU3

Category: C – Cold Shutdown / Refueling System Malfunction

Subcategory: 2 – Loss of DC Power

Initiating Condition: Loss of **required** DC power for ≥ 15 min.

EAL:

CU2.1	**Unusual Event**
< 105 VDC on **required** 125 VDC emergency buses for ≥ 15 min. (Note 4)	

Note 4: The ED should **not** wait until the applicable time has elapsed, but should declare the event as soon as it is determined that the condition has exceeded, or will likely exceed, the applicable time.

Mode Applicability:

4 - Cold Shutdown, 5 - Refuel

Basis:

Plant-Specific

The emergency 125 VDC power system includes three electrically independent and separate switchgears (2BYS*SWG002A, 2BYS*SWG002B and 2CES*IPNL414). Division I ((2BYS*SWG002A) and Division II (2BYS*SWG002B) feed the redundant emergency DC loads associated with Divisions I and II of the emergency onsite AC system, respectively. Division III (2CES*PNP414) feeds the emergency DC loads associated with Division III (HPCS system).

Each emergency 125 VDC distribution system has a battery and a battery charger that are normally connected to the bus such that these two sources of power are operating in parallel. The charger is normally supplying system electrical loads with the battery on a float charge. Should both battery chargers for any particular battery be out of service at any point in the DC load cycle, the battery is capable of starting and operating its associated loads for 2 hr according to a precalculated load profile without the battery terminal voltage falling below minimum acceptable level, 105 VDC. (ref. 1, 2, 3)

In Cold Shutdown mode and Refuel mode, requirements on emergency 125 VDC power are relaxed. The term "required" in this EAL signifies the minimum Technical Specifications requirements for shutdown conditions (ref. 2):

- One Division I or Division II DC electrical power subsystem; and

- Division III DC electrical power subsystem when the HPCS system is required to be operable.

This EAL is the cold condition equivalent of the hot condition loss of DC power EAL SS2.1.

Generic

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.

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

NMP2 Basis Reference(s):

1. USAR Section 8.3.2.1.2
2. Improved Technical Specifications Nine Mile Point Nuclear Station, Unit No. 2, 3.8.5
3. N2-SOP-04 Loss of DC Power
4. NEI 99-01 IC CU7

Category: C – Cold Shutdown / Refueling System Malfunction

Subcategory: 3 – RPV Level

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

EAL:

CG3.1 General Emergency
RPV level < -14 in. for ≥ 30 min. (Note 4)
AND
ANY Containment Challenge Indication, Table C-3

Note 4: The ED should **not** wait until the applicable time has elapsed, but should declare the event as soon as it is determined that the condition has exceeded, or will likely exceed, the applicable time.

Table C-3 Containment Challenge Indications
• CONTAINMENT CLOSURE **not** established • Explosive mixture exists inside Primary Containment ($H_2 \geq 6\%$ and $O_2 \geq 5\%$) • UNPLANNED rise in Primary Containment pressure • RB area radiation > 8.00E+3 mR/hr

Mode Applicability:

4 - Cold Shutdown, 5 - Refuel

Basis:

<u>Plant-Specific</u>

When RPV level drops the top of active fuel (an indicated RPV level of -14 in.), core uncovery starts to occur (ref. 1, 2).

Four conditions are associated with a challenge to Primary Containment integrity:

- CONTAINMENT CLOSURE is the procedurally defined actions taken to secure containment (primary or secondary) and its associated structures, systems, and components as a functional barrier to fission product release under existing plant conditions. This definition is less restrictive than Technical Specification criteria governing Primary and Secondary Containment operability. If the Technical

Specification criteria are met, therefore, CONTAINMENT CLOSURE has been established. (ref. 3, 4, 5)

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

The specified values for this threshold are the minimum global deflagration concentration limits (6% hydrogen and 5% oxygen), and readily recognizable because 6% hydrogen is well above the EOP flowchart entry condition. The minimum global deflagration hydrogen/oxygen concentrations (6%/5%, respectively) require intentional Primary Containment venting, which is defined to be a loss of the Primary Containment barrier. (ref. 6, 7)

The USAR requires the $H_2/0_2$ analyzers to be able to provide and record combustible gas concentration in the Primary Containment within 90 minutes following a LOCA with safety system injection. The $H_2/0_2$ analyzers are normally in standby and require a 30 minute warrn-up/self-test period before they start providing data. (ref. 6)

If the hydrogen or oxygen monitor is unavailable, sampling and analysis may determine gas concentrations. The validity of sample results must be judged based upon plant conditions, since drawing and analyzing samples may take some time. If sample results cannot be relied upon and hydrogen concentrations cannot be determined by any other means, the concentrations must be considered "unknown." The monitors should not be considered "unavailable" until an attempt has been made to place them in service. (ref. 2)

- Any UNPLANNED rise in Primary Containment pressure in the Cold Shutdown or Refuel mode indicates CONTAINMENT CLOSURE cannot be assured and the Primary Containment cannot be relied upon as a barrier to fission product release.
- RB (Reactor Building) area radiation monitors should provide indication of increased release that may be indicative of a challenge to CONTAINMENT CLOSURE. The EOP Maximum Safe Operating level is 8.00E+3 mR/hr and is indicative of problems in the secondary containment that are spreading. The locations into which the primary system discharge is of concern correspond to the areas addressed in Detail S of N2-EOP-SC (ref. 7).

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

Generic

This EAL represents the inability to restore and maintain RPV water level to above the top of active fuel with containment challenged. Fuel damage is probable if RPV water level cannot be restored, as available decay heat will cause boiling, further reducing the RPV water level. With the Containment breached or challenged then the potential for unmonitored fission product release to the environment is high. This represents a direct path for radioactive inventory to be released to the environment. This is consistent with the definition of a GE. The GE is declared on the occurrence of the loss or IMMINENT loss of function of all three barriers.

A number of variables can have a significant impact on heat removal capability challenging the fuel clad barrier. Examples include: mid-loop, reduced level/flange level, head in place, cavity flooded, RCS venting strategy, decay heat removal system design, vortexing pre-disposition, steam generator U-tube draining

Analysis indicates that core damage may occur within an hour following continued core uncovery therefore, 30 minutes was conservatively chosen.

If CONTAINMENT CLOSURE is re-established prior to exceeding the 30 minute core uncovery time limit then escalation to General Emergency would not occur.

NMP2 Basis Reference(s):

1. N2-EOP-RPV RPV Control
2. NER-2M-039, NMP2 Emergency Operating Procedures (EOP) Basis Document
3. NIP-OUT-01 Shutdown Safety
4. Improved Technical Specifications Nine Mile Point Nuclear Station, Unit No. 2, 3.6.1.1
5. Improved Technical Specifications Nine Mile Point Nuclear Station, Unit No. 2, 3.6.4.1

6. N2-EOP-PCH Hydrogen Control
7. N2-EOP-SC Secondary Containment Control
8. NEI 99-01 IC CG1

Category: C – Cold Shutdown / Refueling System Malfunction

Subcategory: 3 – RPV Level

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

EAL:

CG3.2	**General Emergency**

RPV water level **cannot** be monitored with core uncovery indicated by **ANY** of the following for ≥ 30 min. (Note 4):

- **ANY** UNPLANNED RPV leakage indication, Table C-2
- Erratic Source Range Monitor indication

AND

ANY Containment Challenge Indication, Table C-3

Note 4: The ED should **not** wait until the applicable time has elapsed, but should declare the event as soon as it is determined that the condition has exceeded, or will likely exceed, the applicable time.

Table C-2 RPV Leakage Indications
• Drywell equipment drain sump level rise
• Drywell floor drain sump level rise
• Reactor building equipment sump level rise
• Reactor Building floor drain sump level rise
• Suppression Pool level rise
• UNPLANNED rise in RPV make-up rate
• Observation of UNISOLABLE RCS leakage

Table C-3 Containment Challenge Indications
• CONTAINMENT CLOSURE **not** established
• Explosive mixture exists inside Primary Containment (H_2 ≥ 6% and O_2 ≥ 5%)
• UNPLANNED rise in Primary Containment pressure
• RB area radiation > 8.00E+3 mR/hr

Mode Applicability:

4 - Cold Shutdown, 5 - Refuel

Basis:

Plant-Specific

If RPV water level monitoring capability is unavailable, all RPV water level indication would be unavailable and, the RPV inventory loss must be detected by Table C-2, RPV Leakage Indications. Level increases must be evaluated against other potential sources of leakage such as cooling water sources inside the drywell to ensure they are indicative of RPV leakage. Drywell equipment and floor drain sump level rise is the normal method of monitoring and calculating leakage from the RPV. A Reactor Building equipment or floor drain sump level rise may also be indicative of RPV inventory losses external to the Primary Containment from systems connected to the RPV. With RHR System operating in the Shutdown Cooling mode, an UNPLANNED rise in suppression pool level could be indicative of RHR valve misalignment or leakage. If the make-up rate to the RPV unexplainably rises above the pre-established rate, a loss of RPV inventory may be occurring even if the source of the leakage cannot be immediately identified. Visual observation of leakage from systems connected to the RCS in areas outside the Primary Containment that cannot be isolated could be indicative of a loss of RPV inventory. (ref. 1, 2, 3)

Four channels of log count rate meters are available in the Control Room to detect erratic source range monitor indications (ref. 4):

- SRM A & C on 2CEC*PNL606
- SRM B & D on 2CEC*PNL633

Post-TMI studies indicated that the installed nuclear instrumentation will operate erratically when the core is uncovered and that source range monitors can be used as a tool for making such determinations. Figure C-2 shows the response of the source range monitor during the first few hours of the TMI-2 accident. The instrument reported an increasing signal about 30 minutes into the accident. At this time, the reactor coolant pumps were running and the core was adequately cooled as indicated by the core outlet thermocouples. Hence, the increasing signal was the result of an increasing two-phase void fraction in the reactor core and vessel downcomer and the reduced shielding that the two-phase mixture provides to the source range monitor.

Four conditions are associated with a challenge to Primary Containment integrity:

- CONTAINMENT CLOSURE is the procedurally defined actions taken to secure containment (primary or secondary) and its associated structures, systems, and components as a functional barrier to fission product release under existing plant conditions. This definition is less restrictive than Technical Specification criteria governing Primary and Secondary Containment operability. If the Technical Specification criteria are met, therefore, CONTAINMENT CLOSURE has been established. (ref. 5, 9, 10)
- Explosive (deflagration) mixtures in the Primary Containment are assumed to be elevated concentrations of hydrogen and oxygen. BWR industry evaluation of hydrogen generation for development of EOPs/SAGs indicates that any hydrogen concentration above minimum detectable is not to be expected within the short term. Post-LOCA hydrogen generation primarily caused by radiolysis is a slowly evolving, long-term condition. Hydrogen concentrations that rapidly develop are most likely caused by metal-water reaction. A metal-water reaction is indicative of an accident more severe than accidents considered in the plant design basis and would be indicative, therefore, of a potential threat to Primary Containment integrity. Hydrogen concentration of approximately 6% is considered the global deflagration concentration limit.

  The specified values for this threshold are the minimum global deflagration concentration limits (6% hydrogen and 5% oxygen), and readily recognizable because 6% hydrogen is well above the EOP flowchart entry condition. The minimum global deflagration hydrogen/oxygen concentrations (6%/5%, respectively) require intentional Primary Containment venting, which is defined to be a loss of the Primary Containment barrier. (ref. 6, 7)

  The USAR requires the $H_2/0_2$ analyzers to be able to provide and record combustible gas concentration in the Primary Containment within 90 minutes following a LOCA with safety system injection. The $H_2/0_2$ analyzers are normally in standby and require a 30 minute warrn-up/self-test period before they start providing data. (ref. 6)

If the hydrogen or oxygen monitor is unavailable, sampling and analysis may determine gas concentrations. The validity of sample results must be judged based upon plant conditions, since drawing and analyzing samples may take some time. If sample results cannot be relied upon and hydrogen concentrations cannot be determined by any other means, the concentrations must be considered "unknown." The monitors should not be considered "unavailable" until an attempt has been made to place them in service. (ref. 7)

- Any UNPLANNED rise in Primary Containment pressure in the Cold Shutdown or Refuel mode indicates CONTAINMENT CLOSURE cannot be assured and the Primary Containment cannot be relied upon as a barrier to fission product release.
- RB (Reactor Building) area radiation monitors should provide indication of increased release that may be indicative of a challenge to CONTAINMENT CLOSURE. The EOP Maximum Safe Operating level is 8.00E+3 mR/hr and is indicative of problems in the secondary containment that are spreading. The locations into which the primary system discharge is of concern correspond to the areas addressed in Detail S of N2-EOP-SC (ref. 8).

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

Generic

This EAL represents the inability to restore and maintain RPV water level to above the top of active fuel with containment challenged. Fuel damage is probable if RPV water level cannot be restored, as available decay heat will cause boiling, further reducing the RPV water level. With the Containment breached or challenged then the potential for unmonitored fission product release to the environment is high. This represents a direct path for radioactive inventory to be released to the environment. This is consistent with the definition of a GE. The GE is declared on the occurrence of the loss or IMMINENT loss of function of all three barriers.

A number of variables can have a significant impact on heat removal capability challenging the fuel clad barrier. Examples include: initial RPV water level, shutdown heat removal system design.

Analysis indicates that core damage may occur within an hour following continued core uncovery therefore, 30 minutes was conservatively chosen.

If CONTAINMENT CLOSURE is re-established prior to exceeding the 30 minute core uncovery time limit then escalation to General Emergency would not occur.

Sump and tank level increases must be evaluated against other potential sources of leakage such as cooling water sources inside the containment to ensure they are indicative of RCS leakage.

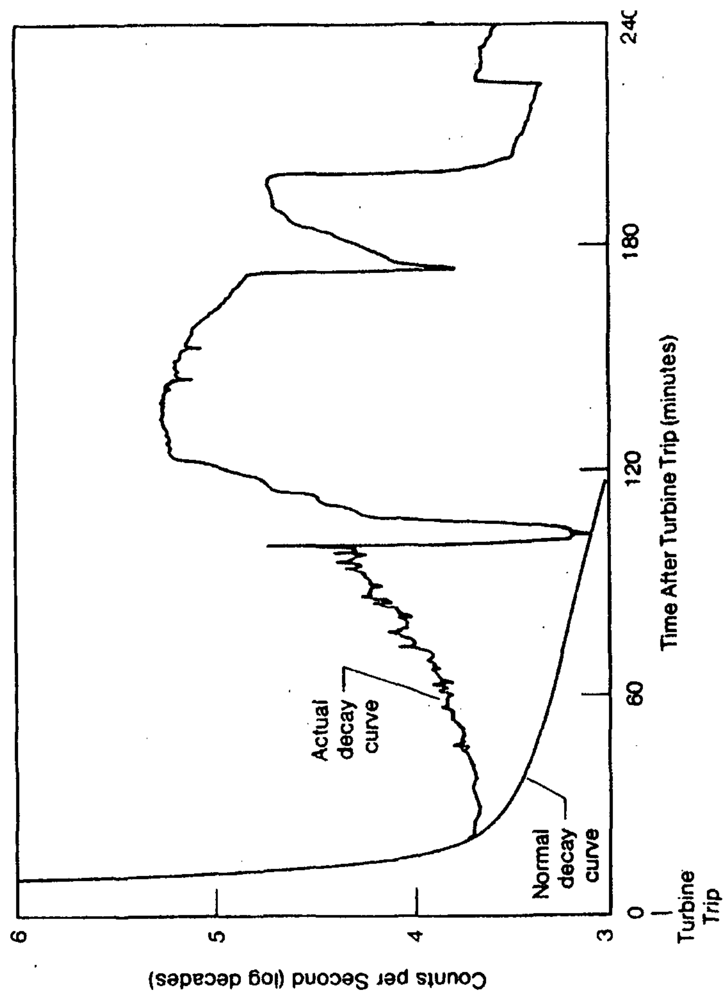
As water level in the RPV lowers, the dose rate above the core will increase. The dose rate due to this core shine should result in site specific monitor indication and possible alarm.

Post-TMI studies indicated that the installed nuclear instrumentation will operate erratically when the core is uncovered and that this should be used as a tool for making such determinations.

NMP2 Basis Reference(s):

1. USAR Section 5.2.5
2. USAR Section 7.6.1.3
3. N2-EOP-PC Primary Containment Control
4. N2-OP-92 Neutron Monitoring
5. Improved Technical Specifications Nine Mile Point Nuclear Station, Unit No. 2, 3.6.4.1
6. N2-EOP-PCH Hydrogen Control
7. NER-2M-039, NMP2 Emergency Operating Procedures (EOP) Basis Document
8. N2-EOP-SC Secondary Containment Control
9. NIP-OUT-01 Shutdown Safety
10. Improved Technical Specifications Nine Mile Point Nuclear Station, Unit No. 2, 3.6.1.1
11. NEI 99-01 IC CG1

Figure C-2: Response of the TMI-2 Source Range Measurement During the First Six Hours of the Accident



Category: C – Cold Shutdown / Refueling System Malfunction

Subcategory: 3 – RPV Level

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

EAL:

CS3.1	**Site Area Emergency**
With CONTAINMENT CLOSURE **not** established, RPV water level < 11.8 in.	

Mode Applicability:

4- Cold Shutdown, 5 - Refuel

Basis:

Plant-Specific

When RPV water level decreases to 11.8 in., water level is six inches below the low-low-low ECCS actuation setpoint (ref. 1).

The inability to restore and maintain level after reaching this setpoint infers a failure of the RCS barrier and Potential Loss of the Fuel Clad barrier.

CONTAINMENT CLOSURE is the procedurally defined actions taken to secure containment (primary or secondary) and its associated structures, systems, and components as a functional barrier to fission product release under existing plant conditions. This definition is less restrictive than Technical Specification criteria governing Primary and Secondary Containment operability. If the Technical Specification criteria are met, therefore, CONTAINMENT CLOSURE has been established. (ref. 2, 3, 4)

Generic

Under the conditions specified by this EAL, continued decrease in RPV level is indicative of a loss of inventory control. Inventory loss may be due to an RCS breach, pressure boundary leakage, or continued boiling in the RPV. Thus, declaration of a Site Area Emergency is warranted.

Escalation to a General Emergency is via EAL CG3.1, EAL CG3.2, RG1.1, RG1.2 or RG1.3.

NMP2 Basis Reference(s):

1. N2-OP-33 High Pressure Core Spray
2. NIP-OUT-01 Shutdown Safety
3. Improved Technical Specifications Nine Mile Point Nuclear Station, Unit No. 2, 3.6.1.1
4. Improved Technical Specifications Nine Mile Point Nuclear Station, Unit No. 2, 3.6.4.1
5. NEI 99-01 IC CS1

Category: C – Cold Shutdown / Refueling System Malfunction

Subcategory: 3 – RPV Level

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

EAL:

CS3.2	**Site Area Emergency**
With CONTAINMENT CLOSURE established, RPV water level < -14 in.	

Mode Applicability:

4 - Cold Shutdown, 5 - Refuel

Basis:

Plant-Specific

When RPV level drops the top of active fuel (an indicated RPV level of -14 in.), core uncovery starts to occur (ref. 1, 2).

CONTAINMENT CLOSURE is the procedurally defined actions taken to secure containment (primary or secondary) and its associated structures, systems, and components as a functional barrier to fission product release under existing plant conditions. This definition is less restrictive than Technical Specification criteria governing Primary and Secondary Containment operability. If the Technical Specification criteria are met, therefore, CONTAINMENT CLOSURE has been established. (ref. 3, 4, 5)

Generic

Under the conditions specified by this EAL, continued decrease in RPV level is indicative of a loss of inventory control. Inventory loss may be due to an RCS breach, pressure boundary leakage, or continued boiling in the RPV. Thus, declaration of a Site Area Emergency is warranted.

Escalation to a General Emergency is via EAL CG3.1, EAL CG3.2, RG1.1, RG1.2 or RG1.3.

NMP2 Basis Reference(s):

1. N2-EOP-RPV RPV Control
2. NER-2M-039, NMP2 Emergency Operating Procedures (EOP) Basis Document
3. NIP-OUT-01 Shutdown Safety
4. Improved Technical Specifications Nine Mile Point Nuclear Station, Unit No. 2, 3.6.1.1
5. Improved Technical Specifications Nine Mile Point Nuclear Station, Unit No. 2, 3.6.4.1
6. NEI 99-01 IC CS1

Category: C – Cold Shutdown / Refueling System Malfunction

Subcategory: 3 – RPV Level

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

EAL:

CS3.3	**Site Area Emergency**
RPV water level **cannot** be monitored for ≥ 30 min. (Note 4) with a loss of RPV inventory as indicated by **ANY** of the following: • **ANY** UNPLANNED RPV leakage indication, Table C-2 • Erratic Source Range Monitor indication	

Note 4: The ED should **not** wait until the applicable time has elapsed, but should declare the event as soon as it is determined that the condition has exceeded, or will likely exceed, the applicable time.

Table C-2 RPV Leakage Indications
• Drywell equipment drain sump level rise
• Drywell floor drain sump level rise
• Reactor building equipment sump level rise
• Reactor Building floor drain sump level rise
• Suppression Pool level rise
• UNPLANNED rise in RPV make-up rate
• Observation of UNISOLABLE RCS leakage

Mode Applicability:

4 - Cold Shutdown, 5 - Refuel

Basis:

Plant-Specific

If RPV water level monitoring capability is unavailable, all RPV water level indication would be unavailable and, the RPV inventory loss must be detected by Table C-2, RPV Leakage Indications. Level increases must be evaluated against other potential sources of leakage such as cooling water sources inside the drywell to ensure they are indicative of RPV leakage. Drywell equipment and floor drain sump level rise is the normal method of monitoring and calculating leakage from the RPV. A Reactor Building equipment or floor drain sump level rise may also be indicative of RPV inventory losses external to the

Primary Containment from systems connected to the RPV. With RHR System operating in the Shutdown Cooling mode, an UNPLANNED rise in suppression pool level could be indicative of RHR valve misalignment or leakage. If the make-up rate to the RPV unexplainably rises above the pre-established rate, a loss of RPV inventory may be occurring even if the source of the leakage cannot be immediately identified. Visual observation of leakage from systems connected to the RCS in areas outside the Primary Containment that cannot be isolated could be indicative of a loss of RPV inventory. (ref. 1, 2, 3)

Four channels of log count rate meters are available in the Control Room to detect erratic source range monitor indications (ref. 4):

- SRM A & C on 2CEC*PNL606
- SRM B & D on 2CEC*PNL633

Post-TMI studies indicated that the installed nuclear instrumentation will operate erratically when the core is uncovered and that source range monitors can be used as a tool for making such determinations. Figure C-2 shows the response of the source range monitor during the first few hours of the TMI-2 accident. The instrument reported an increasing signal about 30 minutes into the accident. At this time, the reactor coolant pumps were running and the core was adequately cooled as indicated by the core outlet thermocouples. Hence, the increasing signal was the result of an increasing two-phase void fraction in the reactor core and vessel downcomer and the reduced shielding that the two-phase mixture provides to the source range monitor.

Generic

Under the conditions specified by this EAL, continued decrease in RPV level is indicative of a loss of inventory control. Inventory loss may be due to an RCS breach, pressure boundary leakage, or continued boiling in the RPV. Thus, declaration of a Site Area Emergency is warranted.

Escalation to a General Emergency is via EAL CG3.1, EAL CG3.2, RG1.1, RG1.2 or RG1.3.

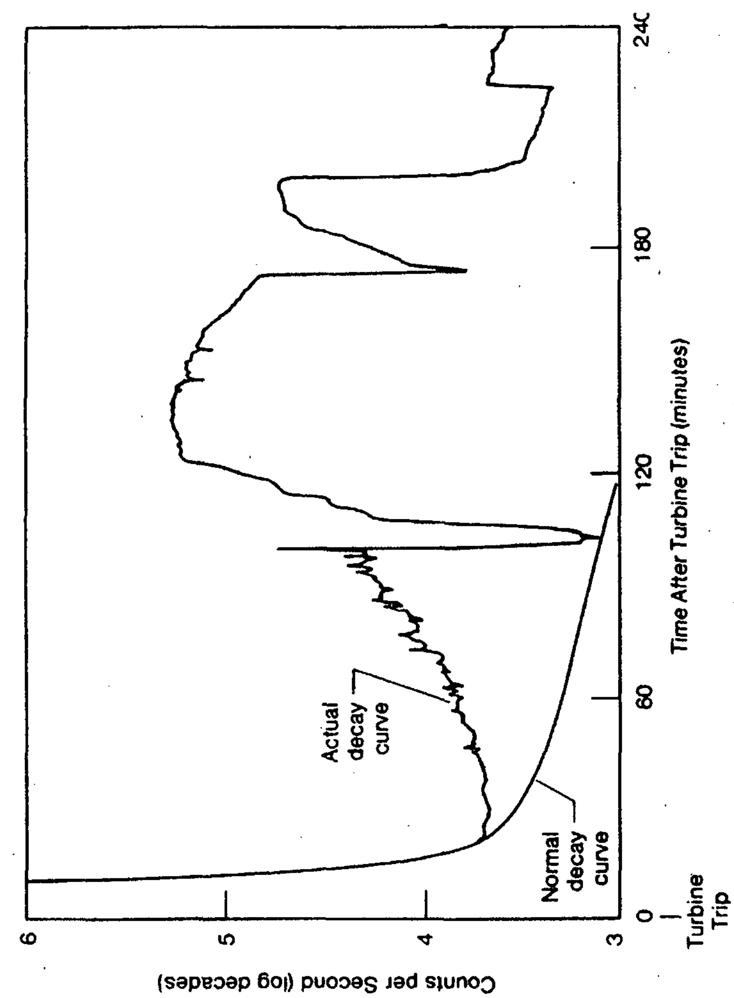
The 30-minute duration allows sufficient time for actions to be performed to recover inventory control equipment.

As water level in the RPV lowers, the dose rate above the core will increase. The dose rate due to this core shine should result in site specific monitor indication and possible alarm.

NMP2 Basis Reference(s):

1. USAR Section 5.2.5
2. USAR Section 7.6.1.3
3. N2-EOP-PC Primary Containment Control
4. N2-OP-92 Neutron Monitoring
5. NEI 99-01 IC CS1

Figure C-2: Response of the TMI-2 Source Range Measurement During the First Six Hours of the Accident



Category: C – Cold Shutdown / Refueling System Malfunction

Subcategory: 3 – RPV Level

Initiating Condition: Loss of RPV inventory

EAL:

CA3.1 **Alert**	
RPV water level < 17.8 in.	
OR	
RPV water level **cannot** be monitored for ≥ 15 min. with **ANY** UNPLANNED RPV leakage indication, Table C-2 (Note 4)	

Note 4: The ED should **not** wait until the applicable time has elapsed, but should declare the event as soon as it is determined that the condition has exceeded, or will likely exceed, the applicable time.

Table C-2 RPV Leakage Indications
• Drywell equipment drain sump level rise
• Drywell floor drain sump level rise
• Reactor building equipment sump level rise
• Reactor Building floor drain sump level rise
• Suppression Pool level rise
• UNPLANNED rise in RPV make-up rate
• Observation of UNISOLABLE RCS leakage

Mode Applicability:

4 - Cold Shutdown, 5 - Refuel

Basis:

Plant-Specific

The threshold RPV water level of 17.8 in. is the low-low-low ECCS actuation setpoint (ref. 1).

Figure C-1 illustrates the RPV water level instrument ranges (ref. 2, 3).

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

In the second condition of this EAL, all RPV water level indication would be unavailable and, the RPV inventory loss must be detected by Table C-2, RPV Leakage Indications. Level increases must be evaluated against other potential sources of leakage such as cooling water sources inside the drywell to ensure they are indicative of RPV leakage. Drywell equipment and floor drain sump level rise is the normal method of monitoring and calculating leakage from the RPV. A Reactor Building equipment or floor drain sump level rise may also be indicative of RPV inventory losses external to the Primary Containment from systems connected to the RPV. With RHR System operating in the Shutdown Cooling mode, an UNPLANNED rise in suppression pool level could be indicative of RHR valve misalignment or leakage. If the make-up rate to the RPV unexplainably rises above the pre-established rate, a loss of RPV inventory may be occurring even if the source of the leakage cannot be immediately identified. Visual observation of leakage from systems connected to the RCS in areas outside the Primary Containment that cannot be isolated could be indicative of a loss of RPV inventory. (ref. 4, 5, 6)

Depending on the configuration of the reactor cavity and Spent Fuel Pool (gates installed or removed) and the status of refueling operations (all spent fuel seated in storage racks/RPV or a bundle raised on the fuel grapple), a loss of inventory may reduce water shielding above irradiated components or spent fuel. EALs in Subcategory R.2 should be assessed for emergency classification due to the radiological consequences of such events.

Generic

This EAL serves as a precursor to a loss of ability to adequately cool the fuel. The magnitude of this loss of water indicates that makeup systems have not been effective and may not be capable of preventing further RPV water level decrease and potential core uncovery. This condition will result in a minimum emergency classification level of an Alert.

The inability to restore and maintain level after reaching this setpoint would be indicative of a failure of the RCS barrier.

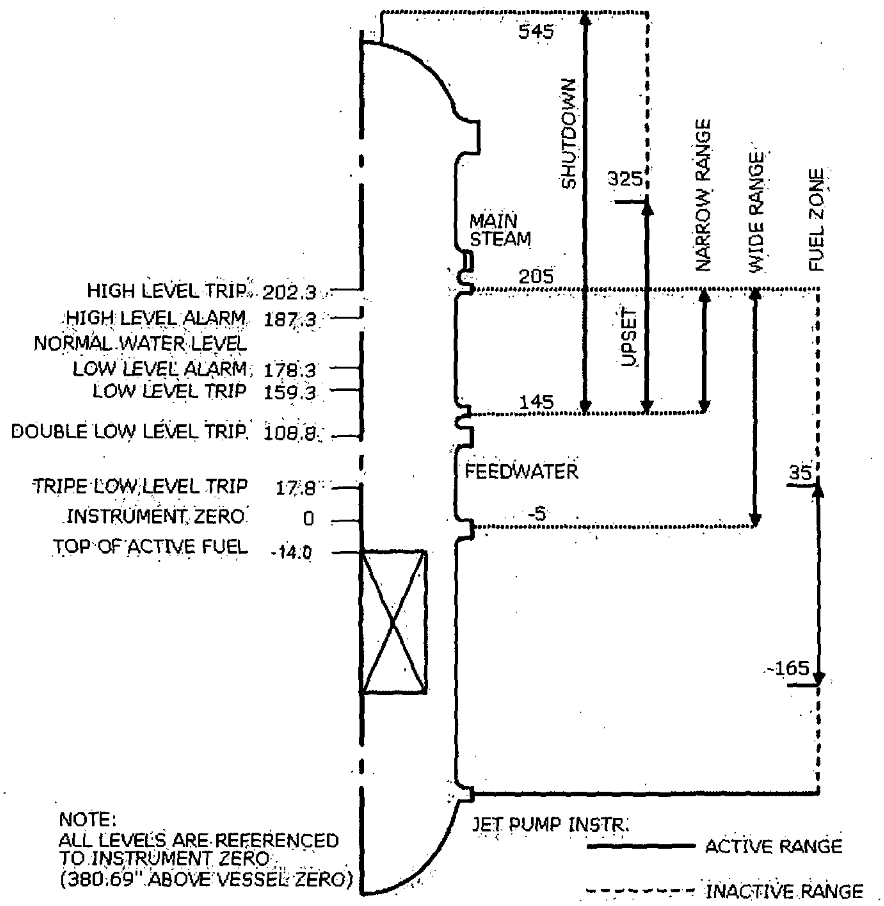
If RPV water level continues to lower then escalation to Site Area Emergency will be via EAL CS3.1, EAL CS3.2 or EAL CS3.3.

NMP2 Basis Reference(s):

1. N2-OP-33 High Pressure Core Spray
2. N2-EOP-RPV RPV Control
3. N2-OP-34 Nuclear Boiler, Automatic Depressurization, and Safety Relief Valves

4. USAR Section 5.2.5
5. USAR Section 7.6.1.3
6. N2-EOP-PC Primary Containment Control
7. NEI 99-01 IC CA1

Figure C-1 RPV Water Level Instrumentation Ranges (ref. 2, 3)



Category: C – Cold Shutdown / Refueling System Malfunction

Subcategory: 3 – RPV Water Level

Initiating Condition: RCS leakage

EAL:

CU3.1	**Unusual Event**
RCS leakage results in the inability to maintain or restore RPV water level > 159.3 in. for ≥ 15 min. (Note 4)	

Note 4: The ED should **not** wait until the applicable time has elapsed, but should declare the event as soon as it is determined that the condition has exceeded, or will likely exceed, the applicable time

Mode Applicability:

4 - Cold Shutdown

Basis:

Plant-Specific

Figure C-1 illustrates the RPV water level instrument ranges (ref. 1, 2).

159.3 in. is the RPV low water level scram setpoint (ref. 1).

RPV water level is monitored from -165 in. to +545 in. to ensure adequate coverage for expected and postulated conditions of RPV water level. RPV water level measurement is derived by the differential pressure that exists between a reference leg and variable leg. All level instruments are referenced to an "instrument zero", which is 380.69 inches above "vessel zero". The instrument zero is the top of the reactor vessel upper grid (top guide). RPV water level monitoring is subdivided into five ranges identified as:

- Narrow provides indication and control signals for normal plant operation and protection system actuation.
- Wide provides indication and control signals for transient conditions below the normal operating band and emergency equipment actuation.
- Upset provides indication for transient conditions above normal operating band.
- Shutdown provides indication for vessel flood up and activities.
- Fuel Zone provides indication for long term accident conditions where reactor level cannot be restored.

The shutdown range level indication is utilized during cold reactor startup and vessel flood up for refueling. The shutdown range instrument uses a single level transmitter (2ISC*LT105) to provide an input to a level indicator on 2CES*PNL851 (Computer Point A486). (ref. 3)

This Cold Shutdown EAL represents the hot condition EAL SU8.1, in which RCS leakage is associated with Technical Specification limits. In Cold Shutdown, these limits are not applicable; hence, the use of RPV level as the parameter of concern in this EAL (ref.).

Generic

This EAL is considered to be a potential degradation of the level of safety of the plant. The inability to maintain or restore level is indicative of loss of RCS inventory.

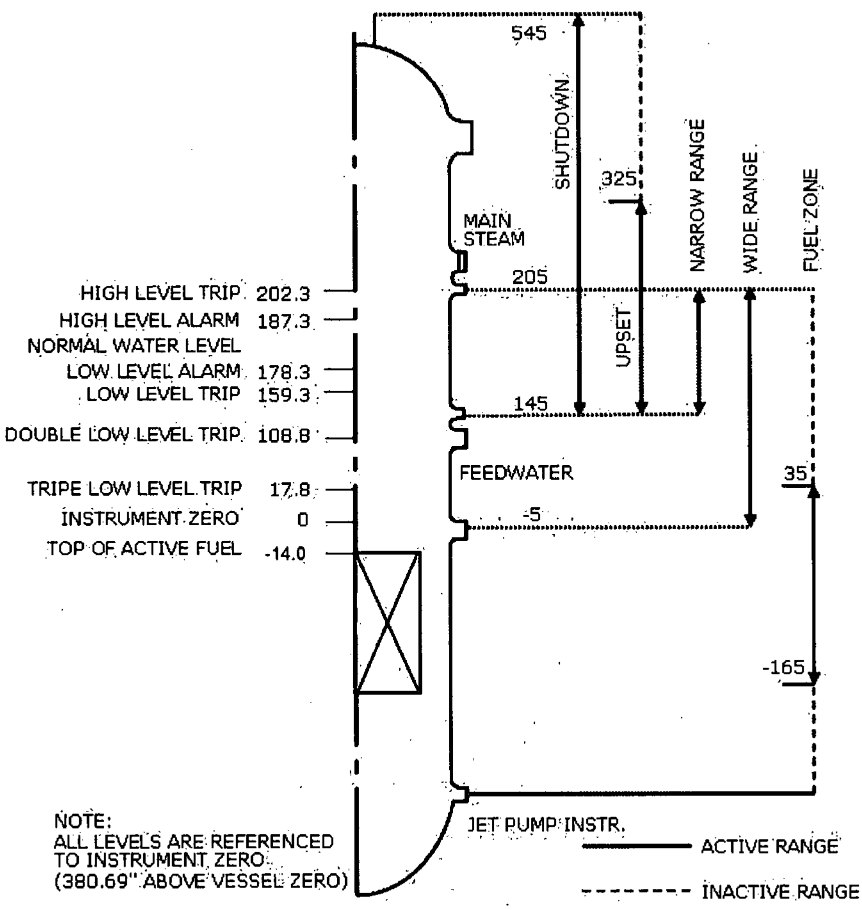
Relief valve normal operation should be excluded from this EAL. However, a relief valve that operates and fails to close per design should be considered applicable to this EAL if the relief valve cannot be isolated.

Prolonged loss of RCS inventory may result in escalation to the Alert emergency classification level via either EAL CA2.1 or EAL CA3.1.

NMP2 Basis Reference(s):

1. N2-EOP-RPV RPV Control
2. N2-OP-34 Nuclear Boiler, Automatic Depressurization, and Safety Relief Valves
3. NIP-OUT-01 Shutdown Safety
4. NEI 99-01 IC CU1

Figure C-1 RPV Water Level Instrumentation Ranges (ref. 1, 2)



Category: C – Cold Shutdown / Refueling System Malfunction

Subcategory: 3 – RPV Water Level

Initiating Condition: RCS Leakage

EAL:

CU3.2 Unusual Event

UNPLANNED RPV water level drop below **EITHER** of the following for ≥ 15 min. (Note 4):

- 364 in. (RPV flange)
- RPV water level band (when the RPV water level band is established below the RPV flange)

Note 4: The ED should **not** wait until the applicable time has elapsed, but should declare the event as soon as it is determined that the condition has exceeded, or will likely exceed, the applicable time

Mode Applicability:

5 - Refuel

Basis:

Plant-Specific

The RPV flange level is at 364 in. or 330 ft 10 in. el (ref. 1).

Figure C-1 illustrates the RPV water level instrument ranges (ref. 2, 3).

RPV water level is monitored from -165 in. to +545 in. to ensure adequate coverage for expected and postulated conditions of RPV water level. RPV water level measurement is derived by the differential pressure that exists between a reference leg and variable leg. All level instruments are referenced to an "instrument zero", which is 380.69 inches above "vessel zero". The instrument zero is the top of the reactor vessel upper grid (top guide). RPV water level monitoring is subdivided into five ranges identified as:

- Narrow provides indication and control signals for normal plant operation and protection system actuation.
- Wide provides indication and control signals for transient conditions below the normal operating band and emergency equipment actuation.
- Upset provides indication for transient conditions above normal operating band.
- Shutdown provides indication for vessel flood up and activities.
- Fuel Zone provides indication for long term accident conditions where reactor level

cannot be restored.

The shutdown range level indication is utilized during cold reactor startup and vessel flood up for refueling. The shutdown range instrument uses a single level transmitter (2ISC*LT105) to provide an input to a level indicator on 2CES*PNL851 (Computer Point A486). (ref. 4)

This Cold Shutdown EAL represents the hot condition EAL SU8.1, in which RCS leakage is associated with Technical Specification limits. In Cold Shutdown, these limits are not applicable; hence, the use of RPV water level as the parameter of concern in this EAL (ref. 5).

Generic

This EAL is a precursor of more serious conditions and considered to be a potential degradation of the level of safety of the plant.

Refueling evolutions that decrease RPV water level below the RPV flange are carefully planned and procedurally controlled. An UNPLANNED event that results in water level decreasing below the RPV flange, or below the planned RPV water level for the given evolution (if the planned RPV water level is already below the RPV flange), warrants declaration of a UE due to the reduced RCS inventory that is available to keep the core covered.

The allowance of 15 minutes was chosen because it is reasonable to assume that level can be restored within this time frame using one or more of the redundant means of refill that should be available. If level cannot be restored in this time frame then it may indicate a more serious condition exists.

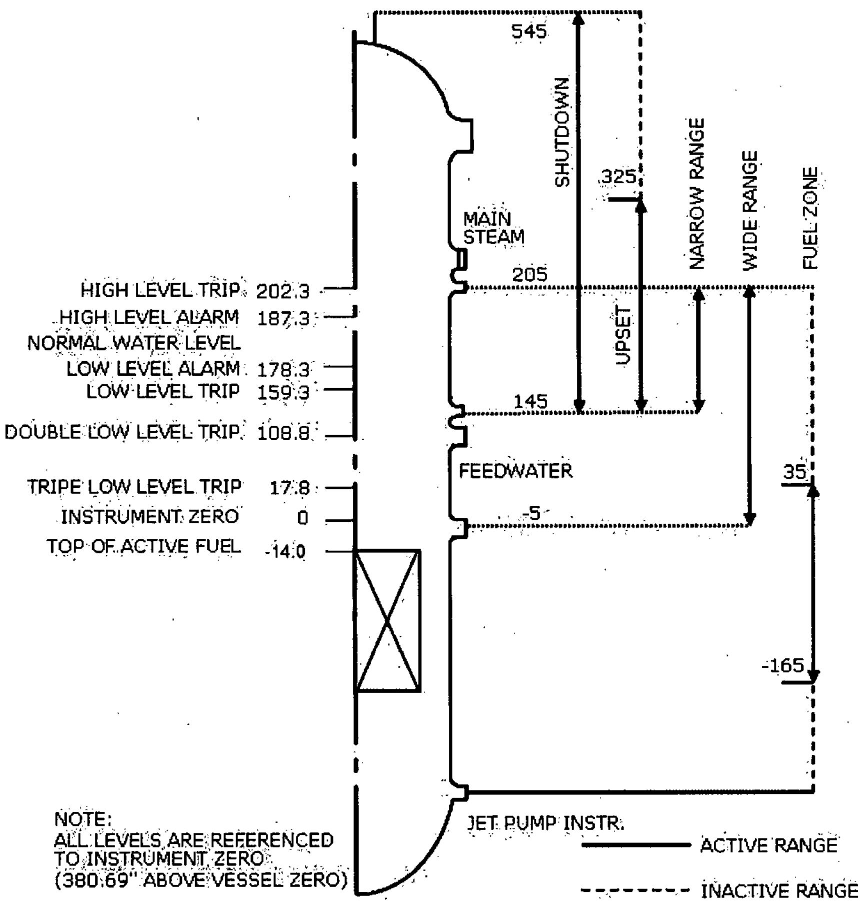
Continued loss of RCS Inventory will result in escalation to the Alert emergency classification level via either EAL CA2.1 or EAL CA3.1.

This EAL involves a decrease in RCS level below the top of the RPV flange that continues for 15 minutes due to an UNPLANNED event. This EAL is not applicable to decreases in flooded reactor cavity level, which is addressed by EAL RU2.1, until such time as the level decreases to the level of the vessel flange.

NMP2 Basis Reference(s):

1. N2-SOP-31R Refueling Operations Alternate Shutdown Cooling
2. N2-EOP-RPV RPV Control
3. N2-OP-34 Nuclear Boiler, Automatic Depressurization, and Safety Relief Valves
4. NIP-OUT-01 Shutdown Safety
5. Improved Technical Specifications Nine Mile Point Nuclear Station, Unit No. 2, 3.4.7
6. NEI 99-01 IC CU2

Figure C-1 RPV Water Level Instrumentation Ranges (ref. 2, 3)



Category: C – Cold Shutdown / Refueling System Malfunction

Subcategory: 3 – RPV Water Level

Initiating Condition: RCS Leakage

EAL:

CU3.3	**Unusual Event**
RPV water level **cannot** be monitored with a loss of RPV inventory as indicated by **ANY** UNPLANNED RPV leakage indication, Table C-2	

Table C-2 RPV Leakage Indications
• Drywell equipment drain sump level rise
• Drywell floor drain sump level rise
• Reactor building equipment sump level rise
• Reactor Building floor drain sump level rise
• Suppression Pool level rise
• UNPLANNED rise in RPV make-up rate
• Observation of UNISOLABLE RCS leakage

Mode Applicability:

5 - Refuel

Basis:

Plant-Specific

In this EAL, all RPV water level indication would be unavailable and, the RPV inventory loss must be detected by Table C-2, RPV Leakage Indications. Level increases must be evaluated against other potential sources of leakage such as cooling water sources inside the drywell to ensure they are indicative of RPV leakage. Drywell equipment and floor drain sump level rise is the normal method of monitoring and calculating leakage from the RPV. A Reactor Building equipment or floor drain sump level rise may also be indicative of RPV inventory losses external to the Primary Containment from systems connected to the RPV. With RHR System operating in the Shutdown Cooling mode, an UNPLANNED rise in suppression pool level could be indicative of RHR valve misalignment or leakage. If the make-up rate to the RPV unexplainably rises above the pre-established rate, a loss of RPV inventory may be occurring even if the source of the leakage cannot be immediately identified. Visual observation of leakage from systems connected to the RCS in areas outside the Primary Containment that cannot be isolated could be indicative of a loss of RPV inventory. (ref. 1, 2, 3)

Depending on the configuration of the reactor cavity and Spent Fuel Pool (gates installed or removed) and the status of refueling operations (all spent fuel seated in storage racks/RPV or a bundle raised on the fuel grapple), a loss of inventory may reduce water shielding above irradiated components or spent fuel. EALs in Subcategory R.2 should be assessed for emergency classification due to the radiological consequences of such events.

Generic

This EAL is a precursor of more serious conditions and considered to be a potential degradation of the level of safety of the plant.

Refueling evolutions that decrease RPV water level below the RPV flange are carefully planned and procedurally controlled. An UNPLANNED event that results in water level decreasing below the RPV flange, or below the planned RPV water level for the given evolution (if the planned RPV water level is already below the RPV flange), warrants declaration of a UE due to the reduced RPV inventory that is available to keep the core covered.

Continued loss of RCS Inventory will result in escalation to the Alert emergency classification level via either EAL CA3.1 or EAL CA4.1.

This EAL addresses conditions in the refueling mode when normal means of core temperature indication and RCS level indication may not be available. Redundant means of RPV water level indication will normally be installed (including the ability to monitor level visually) to assure that the ability to monitor level will not be interrupted. However, if all level indication were to be lost during a loss of RPV inventory event, the operators would need to determine that RPV inventory loss was occurring by observing sump and tank level changes. Sump and tank level increases must be evaluated against other potential sources of leakage such as cooling water sources inside the containment to ensure they are indicative of RCS leakage.

NMP2 Basis Reference(s):

1. USAR Section 5.2.5
2. USAR Section 7.6.1.3
3. N2-EOP-PC Primary Containment Control
4. NEI 99-01 IC CU2

Category: C – Cold Shutdown / Refueling System Malfunction

Subcategory: 4 – RCS Temperature

Initiating Condition: Inability to maintain plant in cold shutdown

EAL:

CA4.1 Alert

An UNPLANNED event results in **EITHER**:

RCS temperature > 200°F for > Table C-4 duration

OR

RPV pressure increase > 10 psi due to an UNPLANNED loss of decay heat removal capability

Table C-4 RCS Reheat Duration Thresholds		
RCS Status	CONTAINMENT CLOSURE Status	Duration
INTACT	N/A	60 min.*
Not INTACT	Established	20 min.*
	Not established	0 min.

* If an RCS heat removal system is in operation within this time frame and RCS temperature is being reduced, the EAL is **not** applicable.

Mode Applicability:

4 - Cold Shutdown, 5 – Refuel

Basis:

Plant-Specific

Several instruments are capable of providing indication of RCS temperature with respect to the Technical Specification cold shutdown temperature limit (200°F). These include (ref. 2):

- Recirc operating – Temperature Recorder B35-R650 at P602:
  - Loop A: Channel 1, RCS LOOP A SUCTION
  - Loop B: Channel 6, RCS LOOP B SUCTION
- Shutdown cooling operating – Temperature Recorder E12-R601 at P601

- Loop A: Point 1, RHR INLET TO HX A
- Loop B: Point 2, RHR INLET TO HX B

If Rx Recirc or Shutdown Cooling pumps are not in operation and reactor coolant temperature is greater than or equal to 212°F, RCS temperature can be obtained by converting the RPV pressure to temperature using the saturated steam tables.

If RCS temperature exceeds 200°F, an operating mode change occurs. Although the event may have originated in cold conditions, the emergency classification shall be based on the operating mode that existed at the time the event occurred (prior to any protective system or operator action initiated in response to the condition). For events that occur in Cold Shutdown or Refuel, escalation is via EALs that have Cold Shutdown or Refuel for mode applicability, even if Hot Shutdown (or a higher mode) is entered during any subsequent heat-up. In particular, the fission product barrier EALs are applicable only to events that initiate in Hot Shutdown or higher.

The RCS should be considered INTACT when the RCS pressure boundary is in its normal condition for the cold shutdown mode of operation (e.g., no freeze seals or nozzle dams).

CONTAINMENT CLOSURE is the procedurally defined actions taken to secure containment (primary or secondary) and its associated structures, systems, and components as a functional barrier to fission product release under existing plant conditions. This definition is less restrictive than Technical Specification criteria governing Primary and Secondary Containment operability. If the Technical Specification criteria are met, therefore, CONTAINMENT CLOSURE has been established. (ref. 3, 4, 5)

The pressure rise of greater than 10 psig infers an RCS temperature in excess of the Technical Specification cold shutdown limit (200°F) for which this EAL would otherwise permit up to sixty minutes to restore RCS cooling before declaration of an Alert (RCS INTACT). This EAL therefore covers situations in which it is determined that, due to high decay heat loads, the time provided to reestablish temperature control should be less than sixty minutes (as indicated by significant RCS re-pressurization).

Wide range pressure indication (0-1200 psig) is capable of measuring pressure changes of 10 psig (ref. 6).

If RCS temperature exceeds 200°F, an operating mode change occurs. Although the event may have originated in cold conditions, the emergency classification shall be based on the operating mode that existed at the time the event occurred (prior to any protective system or operator action initiated in response to the condition). For events that occur in Cold Shutdown or Refuel, escalation is via EALs that have Cold Shutdown or Refuel for mode applicability, even if Hot Shutdown (or a higher mode) is entered during any subsequent heat-up. In particular, the fission product barrier EALs are applicable only to events that initiate in Hot Shutdown or higher.

Escalation to a Site Area Emergency would be under EAL CS3.1 should boiling result in significant RPV water level loss leading to core uncovery.

Generic

The RCS Reheat Duration Thresholds table addresses complete loss of functions required for core cooling for greater than 60 minutes during refuel and cold shutdown modes when RCS integrity is established. The 60 minute time frame should allow sufficient time to restore cooling without there being a substantial degradation in plant safety.

The RCS Reheat Duration Thresholds table also addresses the complete loss of functions required for core cooling for greater than 20 minutes during Refuel and cold shutdown modes when CONTAINMENT CLOSURE is established but RCS integrity is not established. The allowed 20 minute time frame was included to allow operator action to restore the heat removal function, if possible.

Finally, complete loss of functions required for core cooling during Refuel and cold shutdown modes when neither CONTAINMENT CLOSURE nor RCS integrity are established is addressed. No delay time is allowed because the evaporated reactor coolant that may be released into the Containment during this heatup condition could also be directly released to the environment.

The note (*) indicates that this EAL is not applicable if actions are successful in restoring an RCS heat removal system to operation and RCS temperature is being reduced within the specified time frame.

The 10 psig pressure increase addresses situations where, due to high decay heat loads, the time provided to restore temperature control, should be less than 60 minutes. The RPV pressure setpoint was chosen because it is the lowest pressure that the site can read on installed Control Board instrumentation that is equal to or greater than 10 psig.

Escalation to Site Area Emergency would be via EAL CS3.1 should boiling result in significant RPV level loss leading to core uncovery.

A loss of Technical Specification components alone is not intended to constitute an Alert. The same is true of a momentary UNPLANNED excursion above the Technical Specification cold shutdown temperature limit when the heat removal function is available.

The Emergency Director must remain alert to events or conditions that lead to the conclusion that exceeding the EAL is IMMINENT. If, in the judgment of the Emergency Director, an IMMINENT situation is at hand, the classification should be made as if the threshold has been exceeded.

NMP2 Basis Reference(s):

1. Technical Specifications Table 1.1-1
2. N2-OSP-RCS-@001 RCS Pressure/Temperature Verification
3. NIP-OUT-01 Shutdown Safety
4. Improved Technical Specifications Nine Mile Point Nuclear Station, Unit No. 2, 3.6.1.1
5. Improved Technical Specifications Nine Mile Point Nuclear Station, Unit No. 2, 3.6.4.1
6. N2-OP-34 Nuclear Boiler, Automatic Depressurization and Safety Relief Valves, Attachment 1
7. NEI 99-01 IC CA4

Category: C – Cold Shutdown / Refueling System Malfunction

Subcategory: 4 – RCS Temperature

Initiating Condition: UNPLANNED loss of decay heat removal capability

EAL:

CU4.1	**Unusual Event**
UNPLANNED event results in RCS temperature > 200°F	

Mode Applicability:

4 - Cold Shutdown, 5 - Refuel

Basis:

Plant-Specific

Several instruments are capable of providing indication of RCS temperature with respect to the Technical Specification cold shutdown temperature limit (200°F). These include (ref. 2):

- Recirc operating – Temperature Recorder B35-R650 at P602:
  - Loop A: Channel 1, RCS LOOP A SUCTION
  - Loop B: Channel 6, RCS LOOP B SUCTION
- Shutdown cooling operating – Temperature Recorder E12-R601 at P601
  - Loop A: Point 1, RHR INLET TO HX A
  - Loop B: Point 2, RHR INLET TO HX B

If Rx Recirc or Shutdown Cooling pumps are not in operation and reactor coolant temperature is greater than or equal to 212°F, RCS temperature can be obtained by converting the RPV pressure to temperature using the saturated steam tables.

If RCS temperature exceeds 200°F, an operating mode change occurs. Although the event may have originated in cold conditions, the emergency classification shall be based on the operating mode that existed at the time the event occurred (prior to any protective system or operator action initiated in response to the condition). For events that occur in Cold Shutdown or Refuel, escalation is via EALs that have Cold Shutdown or Refuel for mode applicability, even if Hot Shutdown (or a higher mode) is entered during any subsequent heat-up. In particular, the fission product barrier EALs are applicable only to events that initiate in Hot Shutdown or higher.

Generic

This EAL is a precursor of more serious conditions and, as a result, is considered to be a potential degradation of the level of safety of the plant. In cold shutdown the ability to remove decay heat relies primarily on forced cooling flow. Operation of the systems that provide this forced cooling may be jeopardized due to the unlikely loss of electrical power or RCS inventory. Since the RCS usually remains INTACT in the cold shutdown mode a large inventory of water is available to keep the core covered.

During refueling the level in the RPV will normally be maintained above the RPV flange. Refueling evolutions that decrease water level below the RPV flange are carefully planned and procedurally controlled. Loss of forced decay heat removal at reduced inventory may result in more rapid increases in RCS/RPV temperatures depending on the time since shutdown.

Normal means of core temperature indication and RPV water level indication may not be available in the Refuel mode. Redundant means of RPV water level indication are therefore procedurally installed to assure that the ability to monitor level will not be interrupted. Escalation to Alert would be via EAL CA3.1 based on an inventory loss or EAL CA4.1 based on exceeding its temperature duration or pressure criteria.

NMP2 Basis Reference(s):

1. Technical Specifications Table 1.1-1
2. N2-OSP-RCS-@001 RCS Pressure/Temperature Verification
3. NEI 99-01 IC CU4

Category: C – Cold Shutdown / Refueling System Malfunction
Subcategory: 4 – RCS Temperature
Initiating Condition: UNPLANNED loss of decay heat removal capability
EAL:

CU4.2	**Unusual Event**
Loss of **all** RCS temperature and RPV water level indication for ≥ 15 min. (Note 4)	

Note 4: The ED should **not** wait until the applicable time has elapsed, but should declare the event as soon as it is determined that the condition has exceeded, or will likely exceed, the applicable time

Mode Applicability:

4 - Cold Shutdown, 5 - Refuel

Basis:

Plant-Specific

Several instruments are capable of providing indication of RCS temperature with respect to the Technical Specification cold shutdown temperature limit (200°F). These include (ref. 2):

- Recirc operating – Temperature Recorder B35-R650 at P602:
  - Loop A: Channel 1, RCS LOOP A SUCTION
  - Loop B: Channel 6, RCS LOOP B SUCTION
- Shutdown cooling operating – Temperature Recorder E12-R601 at P601
  - Loop A: Point 1, RHR INLET TO HX A
  - Loop B: Point 2, RHR INLET TO HX B

If Rx Recirc or Shutdown Cooling pumps are not in operation and reactor coolant temperature is greater than or equal to 212°F, RCS temperature can be obtained by converting the RPV pressure to temperature using the saturated steam tables.

RPV water level is monitored from -165 in. to +545 in. to ensure adequate coverage for expected and postulated conditions of RPV water level. RPV water level measurement is derived by the differential pressure that exists between a reference leg and variable leg. All level instruments are referenced to an "instrument zero", which is 380.69 inches above "vessel zero". The instrument zero is the top of the reactor vessel upper grid (top guide).

RPV water level monitoring is subdivided into five ranges identified as:

- Narrow provides indication and control signals for normal plant operation and protection system actuation.
- Wide provides indication and control signals for transient conditions below the normal operating band and emergency equipment actuation.
- Upset provides indication for transient conditions above normal operating band.
- Shutdown provides indication for vessel flood up and activities.
- Fuel Zone provides indication for long term accident conditions where reactor level cannot be restored.

The shutdown range level indication is utilized during cold reactor startup and vessel flood up for refueling. The shutdown range instrument uses a single level transmitter (2ISC*LT105) to provide an input to a level indicator on 2CES*PNL851 (Computer Point A486). (ref. 3)

Although the event may have originated in cold conditions, the emergency classification shall be based on the operating mode that existed at the time the event occurred (prior to any protective system or operator action initiated in response to the condition). For events that occur in Cold Shutdown or Refuel, escalation is via EALs that have Cold Shutdown or Refuel for mode applicability, even if Hot Shutdown (or a higher mode) is entered during any subsequent heat-up. In particular, the fission product barrier EALs are applicable only to events that initiate in Hot Shutdown or higher.

Generic

This EAL is a precursor of more serious conditions and, as a result, is considered to be a potential degradation of the level of safety of the plant. In cold shutdown the ability to remove decay heat relies primarily on forced cooling flow. Operation of the systems that provide this forced cooling may be jeopardized due to the unlikely loss of electrical power or RPV inventory. Since the RCS usually remains INTACT in the cold shutdown mode a large inventory of water is available to keep the core covered.

During refueling the level in the RPV will normally be maintained above the RPV flange. Refueling evolutions that decrease water level below the RPV flange are carefully planned and procedurally controlled. Loss of forced decay heat removal at reduced inventory may result in more rapid increases in RPV temperatures depending on the time since shutdown.

Normal means of core temperature indication and RPV water level indication may not be available in the Refuel mode. Redundant means of RPV water level indication are therefore procedurally installed to assure that the ability to monitor level will not be interrupted. However, if all level and temperature indication were to be lost in either the cold shutdown of refueling modes, this EAL would result in declaration of a UE if both temperature and level indication cannot be restored within 15 minutes from the loss of both means of indication. Escalation to Alert would be via EAL CA3.1 based on an inventory loss or EAL CA4.1 based on exceeding its temperature criteria.

NMP2 Basis Reference(s):

1. Technical Specifications Table 1.1-1
2. N2-OSP-RCS-@001 RCS Pressure/Temperature Verification
3. NIP-OUT-01 Shutdown Safety
4. NEI 99-01 IC CU4

Category: C – Cold Shutdown / Refueling System Malfunction

Subcategory: 5 – Inadvertent Criticality

Initiating Condition: Inadvertent criticality

EAL:

CU5.1	**Unusual Event**
An UNPLANNED sustained positive period observed on nuclear instrumentation	

Mode Applicability:

4 - Cold Shutdown, 5 - Refuel

Basis:

Plant-Specific

The term "sustained" is used to allow exclusion of expected short-term positive periods from planned fuel bundle or control rod movements during core alteration. These short-term positive periods are the result of the rise in neutron population due to subcritical multiplication.

Generic

This EAL addresses criticality events that occur in Cold Shutdown or Refueling modes such as fuel mis-loading events and inadvertent dilution events. This EAL indicates a potential degradation of the level of safety of the plant, warranting a UE classification.

Escalation would be by Emergency Director judgment.

NMP2 Basis Reference(s):

1. NEI 99-01 IC CU8

Category: C – Cold Shutdown / Refueling System Malfunction

Subcategory: 6 – Communications

Initiating Condition: Loss of **all** onsite or offsite communications capabilities

EAL:

CU6.1 Unusual Event

Loss of **all** Table C-5 onsite (internal) communication methods affecting the ability to perform routine operations

OR

Loss of **all** Table C-5 offsite (external) communication methods affecting the ability to perform offsite notifications

Table C-5 Communications Systems		
System	**Onsite** (internal)	**Offsite** (external)
PBX (normal dial telephones)	X	X
Gaitronics	X	
Station radio (portable)	X	
Control Room installed satellite phones (non portable)		X
ENS		X
RECS		X
~~UHF radios~~		~~X~~

Mode Applicability:

4 - Cold Shutdown, 5 - Refuel, D - Defueled

Basis:

Plant-Specific

Onsite/offsite communications systems are listed in Table C-2 (ref. 1, 2, 3).

This EAL is the cold condition equivalent of the hot condition EAL SU6.1.

Generic

The purpose of this EAL is to recognize a loss of communications capability that either defeats the plant operations staff ability to perform routine tasks necessary for plant operations or the ability to communicate issues with off-site authorities. The loss of off-site communications ability is expected to be significantly more comprehensive than the condition addressed by 10 CFR 50.72.

The availability of one method of ordinary off-site communications is sufficient to inform federal, state, and local authorities of plant issues. This EAL is intended to be used only when extraordinary means (e.g., relaying of information from radio transmissions, individuals being sent to off-site locations, etc.) are being utilized to make communications possible.

NMP2 Basis Reference(s):

1. USAR Section 9.5.2
2. Nine Mile Point Nuclear Station Site Emergency Plan, Section 7.2
3. N2-OP-76 Plant Communications
4. NEI 99-01 IC CU6

Category S – System Malfunction

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

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

The events of this category pertain to the following subcategories:

1. Loss of AC Power

Loss of emergency plant electrical power can compromise plant safety system operability including decay heat removal and emergency core cooling systems which may be necessary to ensure fission product barrier integrity. This category includes loss of onsite and offsite power sources for the 4.16KV emergency buses.

2. Loss of DC Power

Loss of emergency plant electrical power can compromise plant safety system operability including decay heat removal and emergency core cooling systems which may be necessary to ensure fission product barrier integrity. This category includes loss of power to the 125 VDC buses.

3. Criticality & RPS Failure

Inadvertent criticalities pose potential personnel safety hazards as well being indicative of losses of reactivity control.

Events related to failure of the Reactor Protection System (RPS) to initiate and complete reactor scrams. In the plant licensing basis, postulated failures of the RPS to complete a reactor scram comprise a specific set of analyzed events referred to as Anticipated Transient Without Scram (ATWS) events. For EAL classification however, ATWS is intended to mean any scram failure event that does not achieve reactor shutdown. If RPS actuation fails to assure reactor shutdown, positive control of reactivity is at risk and could cause a threat to fuel clad, RCS and containment integrity.

4. Inability to Reach or Maintain Shutdown Conditions

System malfunctions may lead to failure of the plant to be brought to the required plant operating condition required by technical specifications if a limiting condition for operation (LCO) is not met.

5. Instrumentation

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

6. Communications

Certain events that degrade plant operator ability to effectively communicate with essential personnel within or external to the plant warrant emergency classification.

7. Fuel Clad Degradation

During normal operation, reactor coolant fission product activity is very low. Small concentrations of fission products in the coolant are primarily from the fission of tramp uranium in the fuel clad or minor perforations in the clad itself. Any significant increase from these base-line levels (~5% clad failures) is indicative of fuel failures and is covered under Category F, Fission Product Barrier Degradation. However, lesser amounts of clad damage may result in coolant activity exceeding Technical Specification limits. These fission products will be circulated with the reactor coolant and can be detected by coolant sampling and/or the Letdown radiation monitor.

8. RCS Leakage

The RPV provides a volume for the coolant that covers the reactor core. The RPV and associated pressure piping (reactor coolant system) together provide a barrier to limit the release of radioactive material should the reactor fuel clad integrity fail.

Excessive RCS leakage greater than Technical Specification limits are utilized to indicate potential pipe cracks that may propagate to an extent threatening fuel clad, RCS and containment integrity.

Category: S –System Malfunction

Subcategory: 1 – Loss of Power

Initiating Condition: Prolonged loss of **all** offsite and **all** onsite AC power to 4.16 KV emergency buses

EAL:

SG1.1 General Emergency

Loss of **all** offsite and **all** onsite AC power, Table S-1, to 4.16 KV emergency buses 2ENS*SWG101 and 2ENS *SWG103

AND EITHER:

Restoration of 4.16 KV emergency bus 2ENS*SWG101 or 2ENS *SWG103 within 4 hours is **not** likely

OR

RPV water level **cannot** be restored and maintained above -14 in. or RPV water level **cannot** be determined

Table S-1 AC Power Sources	
Onsite	• 2EGS*EG1 • 2EGS*EG3 • 2EGS*EG2 (with 2ENS*SWG102 crosstied to 2ENS*SWG101 or 2ENS*SWG103)
Offsite	• Reserve Transformer A • Reserve Transformer B • Aux Boiler Transformer

Mode Applicability:

1 - Power Operation, 2 - Startup, 3 - Hot Shutdown

Basis:

Plant-Specific

2ENS*SWG101, *SWG102, and *SWG103 are the 4.16 KV emergency buses. Bus 2ENS*SWG101 is dedicated to Division I of the On-site Emergency AC Electrical Distribution System, bus 2ENS*SWG102 is dedicated to Division III (HPCS), and bus 2ENS*SWG103 is dedicated to Division II. Buses 2ENS*SWG101 and *SWG103 feed all

Station redundant safety-related loads, except the HPCS system loads. The HPCS system loads are fed by bus 2ENS*SWG102 (ref. 1, 2).

- All three divisions are normally energized by the On-site Normal AC Electrical Distribution System via the off-site power sources through the reserve station service transformers 2RTX-XSR1A and 2RTX-XSR1B.
  - 2ENS*SWG102 from transformer 2RTX-XSR1A
  - 2ENS*SWG103 from transformer 2RTX-XSR1B.
- Buses 2ENS*SWG101 and *SWG103 each have a backup source, the Auxiliary Boiler Transformer 2ABS-X1. Also, 2ENS*SWG101 and *SWG103 each have a feeder to a normal AC (stub) bus, NNS-SWG014 and NNS-SWG015 respectively.
- Bus 2ENS*SWG102 has a backup connection to the Reserve Station Service Transformer 2RTX-XSR1B, if required.
- Each of the three 4.16 KV emergency buses has a standby diesel generator (2EGS*EG1, 2EGS*EG3, 2EGS*EG2) to carry its loads in case of a LOOP or in case of a sustained degraded voltage condition on the offsite source (ref. 3, 4). 2EGS*EG2 (Division III) is capable of powering either the Division I or Division II 4.16 KV emergency bus through manual breaker alignments. The availability of 2EGS*EG2 as an onsite AC power source in Table S-1 only applies if 2EGS*EG2 is aligned to energize 2ENS*SWG101 or 2ENS*SWG103.

Consideration should be given to operable loads necessary to remove decay heat or provide RPV makeup capability when evaluating loss of all AC power to vital buses. Even though an essential bus may be energized, if necessary loads (i.e., loads that if lost would inhibit decay heat removal capability or RPV makeup capability) are not operable on the energized bus then the bus should not be considered operable.

Four hours is the station blackout coping period (ref. 4, 5).

An RPV water level instrument reading of -14 in. indicates RPV water level is at the top of active fuel. When RPV water level is at or above the top of active fuel, the core is completely submerged. Core submergence is the most desirable means of core cooling. When RPV water level is below the top of active fuel, the uncovered portion of the core must be cooled by less reliable means (i.e., steam cooling or spray cooling). If core uncovery is threatened, the EOPs specify alternate, more extreme, RPV water level control measures in order to restore and maintain adequate core cooling (ref. 6). Since core uncovery begins if RPV water level drops to -14 in., the level is indicative of a challenge to core cooling and the Fuel Clad barrier.

Consistent with the EOP definition of "cannot be restored and maintained," the determination that RPV water level cannot be restored and maintained above the top of active fuel may be made at, before, or after RPV water level actually decreases to this point. (ref. 6)

When RPV water level cannot be determined, EOPs require RPV flooding strategies. RPV water level indication provides the primary means of knowing if adequate core cooling is being maintained. When all means of determining RPV water level are unavailable, the fuel clad barrier is threatened and reliance on alternate means of assuring adequate core cooling must be attempted. The instructions in EOP-C4 specify these means, which include emergency depressurization of the RPV and injection into the RPV at a rate needed to flood to the elevation of the main steam lines or hold RPV pressure above the Minimum Steam Cooling Pressure (in ATWS events). (ref. 7) If RPV water level cannot be determined with respect to the top of active fuel, a potential loss of the Fuel Clad barrier exists.

Note that EOP-C5 may require intentional uncovery of the core and control of RPV water level between -14 in. and -39 in., the Minimum Steam Cooling RPV Water Level (MSCRWL) (ref. 8). Under these conditions, a high-power ATWS event exists and requires at least a Site Area Emergency classification in accordance with the ATWS/Criticality EALs.

Generic

Loss of all AC power to emergency busses 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 to emergency buses will lead to loss of fuel clad, RCS, and containment, thus warranting declaration of a General Emergency.

This EAL is specified to assure that in the unlikely event of a prolonged loss of all AC power to 4.16 KV emergency buses, timely recognition of the seriousness of the event occurs and that declaration of a General Emergency occurs as early as is appropriate, based on a reasonable assessment of the event trajectory.

The likelihood of restoring at least one emergency bus should be based on a realistic appraisal of the situation since a delay in an upgrade decision based on only a chance of mitigating the event could result in a loss of valuable time in preparing and implementing public protective actions.

In addition, under these conditions, fission product barrier monitoring capability may be degraded.

NMP2 Basis Reference(s):

1. USAR Section 8.2
2. USAR Section 8.3
3. N2-SOP-03 Loss of AC Power
4. N2-SOP-01 Station Blackout
5. USAR Section 8.3.1.5.2
6. NER-2M-039, NMP2 Emergency Operating Procedures (EOP) Basis Document
7. N2-EOP-C4 RPV Flooding
8. N2-EOP-C5 Failure to Scram
9. NEI 99-01 IC SG1

Category: S – System Malfunction

Subcategory: 1 – Loss of AC Power

Initiating Condition: Loss of **all** offsite and **all** onsite AC power to 4.16 KV emergency buses for ≥ 15 min.

EAL:

SS1.1	**Site Area Emergency**
Loss of **all** offsite and **all** onsite AC power, Table S-1, to 4.16 KV emergency buses 2ENS*SWG101 and 2ENS*SWG103 for ≥ 15 min. (Note 4)	

Note 4: The ED should **not** wait until the applicable time has elapsed, but should declare the event as soon as it is determined that the condition has exceeded, or will likely exceed, the applicable time.

Table S-1 AC Power Sources	
Onsite	• 2EGS*EG1 • 2EGS*EG3 • 2EGS*EG2 (with 2ENS*SWG102 crosstied to 2ENS*SWG101 or 2ENS*SWG103)
Offsite	• Reserve Transformer A • Reserve Transformer B • Aux Boiler Transformer

Mode Applicability:

1 - Power Operation, 2 - Startup, 3 - Hot Shutdown

Basis:

Plant-Specific

2ENS*SWG101, *SWG102, and *SWG103 are the 4.16 KV emergency buses. Bus 2ENS*SWG101 is dedicated to Division I of the On-site Emergency AC Electrical Distribution System, bus 2ENS*SWG102 is dedicated to Division III (HPCS), and bus 2ENS*SWG103 is dedicated to Division II. Buses 2ENS*SWG101 and *SWG103 feed all Station redundant safety-related loads, except the HPCS system loads. The HPCS system loads are fed by bus 2ENS*SWG102 (ref. 1, 2).

- All three divisions are normally energized by the On-site Normal AC Electrical

Distribution System via the off-site power sources through the reserve station service transformers 2RTX-XSR1A and 2RTX-XSR1B.

- 2ENS*SWG102 from transformer 2RTX-XSR1A
- 2ENS*SWG103 from transformer 2RTX-XSR1B.

- Buses 2ENS*SWG101 and *SWG103 each have a backup source, the Auxiliary Boiler Transformer 2ABS-X1. Also, 2ENS*SWG101 and *SWG103 each have a feeder to a normal AC (stub) bus, NNS-SWG014 and NNS-SWG015 respectively.
- Bus 2ENS*SWG102 has a backup connection to the Reserve Station Service Transformer 2RTX-XSR1B, if required.
- Each of the three 4.16 KV emergency buses has a standby diesel generator (2EGS*EG1, 2EGS*EG3, 2EGS*EG2) to carry its loads in case of a LOOP or in case of a sustained degraded voltage condition on the offsite source (ref. 3, 4). 2EGS*EG2 (Division III) is capable of powering either the Division I or Division II 4.16 KV emergency bus through manual breaker alignments. It is unlikely that these actions could be performed within the fifteen-minute interval of this EAL. The availability of 2EGS*EG2 as an onsite AC power source in Table S-1 only applies if 2EGS*EG2 is aligned to energize 2ENS*SWG101 or 2ENS*SWG103.

Consideration should be given to operable loads necessary to remove decay heat or provide RPV makeup capability when evaluating loss of all AC power to vital buses. Even though an essential bus may be energized, if necessary loads (i.e., loads that if lost would inhibit decay heat removal capability or RPV makeup capability) are not operable on the energized bus then the bus should not be considered operable.

The fifteen-minute interval was selected as a threshold to exclude transient power losses.

<u>Generic</u>

Loss of all AC power to emergency busses 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 to 4.16 KV emergency buses will lead to loss of Fuel Clad, RCS, and Containment, thus this event can escalate to a General Emergency.
Fifteen minutes was selected as a threshold to exclude transient or momentary losses of off-site power.

Escalation to General Emergency is via EALs in Category F or EAL SG1.1.

NMP2 Basis Reference(s):

1. USAR Section 8.2
2. USAR Section 8.3
3. N2-SOP-03 Loss of AC Power
4. N2-SOP-01 Station Blackout
5. NEI 99-01 IC SS1

Category: S – System Malfunction

Subcategory: 1 – Loss of AC Power

Initiating Condition: AC power capability to 4.16 KV emergency buses reduced to a single power source for ≥15 min. such that **ANY** additional single failure would result in a complete loss of **all** 4.16 KV emergency bus power

EAL:

SA1.1 **Alert**
AC power capability to 4.16 KV emergency buses 2ENS*SWG101 and 2ENS*SWG103 reduced to a single power source, Table S-1, for ≥ 15 min. (Note 4)
AND
ANY additional single power source failure will result in a loss of **all** power to 4.16 KV emergency buses 2ENS*SWG101 and 2ENS*SWG103

Note 4: The ED should **not** wait until the applicable time has elapsed, but should declare the event as soon as it is determined that the condition has exceeded, or will likely exceed, the applicable time.

Table S-1 AC Power Sources	
Onsite	• 2EGS*EG1 • 2EGS*EG3 • 2EGS*EG2 (with 2ENS*SWG102 crosstied to 2ENS*SWG101 or 2ENS*SWG103)
Offsite	• Reserve Transformer A • Reserve Transformer B • Aux Boiler Transformer

Mode Applicability:

1 - Power Operation, 2 - Startup, 3 - Hot Shutdown

Basis:

Plant-Specific

2ENS*SWG101, *SWG102, and *SWG103 are the 4.16 KV emergency buses. Bus 2ENS*SWG101 is dedicated to Division I of the On-site Emergency AC Electrical Distribution System, bus 2ENS*SWG102 is dedicated to Division III (HPCS), and bus

2ENS*SWG102 is dedicated to Division II. Buses 2ENS*SWG101 and *SWG103 feed all Station redundant safety-related loads, except the HPCS system loads. The HPCS system loads are fed by bus 2ENS*SWG102 (ref. 1, 2).

- All three divisions are normally energized by the On-site Normal AC Electrical Distribution System via the off-site power sources through the reserve station service transformers 2RTX-XSR1A and 2RTX-XSR1B.
  - 2ENS*SWG102 from transformer 2RTX-XSR1A
  - 2ENS*SWG103 from transformer 2RTX-XSR1B.
- Buses 2ENS*SWG101 and *SWG103 each have a backup source, the Auxiliary Boiler Transformer 2ABS-X1. Also, 2ENS*SWG101 and *SWG103 each have a feeder to a normal AC (stub) bus, NNS-SWG014 and NNS-SWG015 respectively.
- Bus 2ENS*SWG102 has a backup connection to the Reserve Station Service Transformer 2RTX-XSR1B, if required.
- Each of the three 4.16 KV emergency buses has a standby diesel generator (2EGS*EG1, 2EGS*EG3, 2EGS*EG2) to carry its loads in case of a LOOP or in case of a sustained degraded voltage condition on the offsite source (ref. 3, 4). 2EGS*EG2 (Division III) is capable of powering either the Division I or Division II 4.16 KV emergency bus through manual breaker alignments. It is unlikely that these actions could be performed within the fifteen-minute interval of this EAL. The availability of 2EGS*EG2 as an onsite AC power source in Table S-1 only applies if 2EGS*EG2 is aligned to energize 2ENS*SWG101 or 2ENS*SWG103.

The fifteen-minute interval was selected as a threshold to exclude transient power losses. If the capability for multiple sources to energize the unit vital buses within 15 minutes is not restored, an Alert is declared under this EAL. The subsequent loss of the single remaining power source escalates the event to a Site Area Emergency under EAL SS1.1.

<u>Generic</u>

The condition indicated by this EAL is the degradation of the off-site and on-site AC power systems such that any additional single failure would result in a complete loss of 4.16 KV emergency bus AC power to one or both units. This condition could occur due to a loss of off-site power with a concurrent failure of all but one emergency generator to supply power to its emergency buses.

Another related condition could be the loss of all off-site power and loss of on-site emergency generators with only one train of 4.16 KV emergency buses being backfed from the unit main generator, or the loss of on-site emergency generators with only one train of 4.16 KV emergency buses being backfed from off-site power. The subsequent loss of this single power source would escalate the event to a Site Area Emergency in accordance with EAL SS1.1.
Fifteen minutes was selected as a threshold to exclude transient or momentary losses of power.

NMP2 Basis Reference(s):

1. USAR Section 8.2
2. USAR Section 8.3
3. N2-SOP-03 Loss of AC Power
4. N2-SOP-01 Station Blackout
5. NEI 99-01 IC SA5

Category: S – System Malfunction

Subcategory: 1 – Loss of AC Power

Initiating Condition: Loss of **all** offsite AC power to 4.16KV vital buses for ≥ 15 min.

EAL:

SU1.1 Unusual Event
Loss of **all** offsite AC power, Table S-1, to 4.16 KV emergency buses 2ENS*SWG101 and 2ENS*SWG103

Table S-1 AC Power Sources	
Onsite	• 2EGS*EG1 • 2EGS*EG3 • 2EGS*EG2 (with 2ENS*SWG102 crosstied to 2ENS*SWG101 or 2ENS*SWG103)
Offsite	• Reserve Transformer A • Reserve Transformer B • Aux Boiler Transformer

Mode Applicability:

1 - Power Operation, 2 - Startup, 3 - Hot Shutdown

Basis:

Plant-Specific

2ENS*SWG101, *SWG102, and 2ENS*SWG103 are the 4.16 KV emergency buses. Bus 2ENS*SWG101 is dedicated to Division I of the On-site Emergency AC Electrical Distribution System, bus 2ENS*SWG102 is dedicated to Division III (HPCS), and bus 2ENS*SWG103 is dedicated to Division II. Buses 2ENS*SWG101 and *SWG103 feed all Station redundant safety-related loads, except the HPCS system loads. The HPCS system loads are fed by bus 2ENS*SWG102 (ref. 1, 2).

- All three divisions are normally energized by the On-site Normal AC Electrical Distribution System via the off-site power sources through the reserve station service transformers 2RTX-XSR1A and 2RTX-XSR1B.

- 2ENS*SWG102 from transformer 2RTX-XSR1A
- 2ENS*SWG103 from transformer 2RTX-XSR1B.

- Buses 2ENS*SWG101 and *SWG103 each have a backup source, the Auxiliary Boiler Transformer 2ABS-X1. Also, 2ENS*SWG101 and *SWG103 each have a feeder to a normal AC (stub) bus, NNS-SWG014 and NNS-SWG015 respectively.
- Bus 2ENS*SWG102 has a backup connection to the Reserve Station Service Transformer 2RTX-XSR1B, if required.
- Each of the three 4.16 KV emergency buses has a standby diesel generator (2EGS*EG1, 2EGS*EG3, 2EGS*EG2) to carry its loads in case of a LOOP or in case of a sustained degraded voltage condition on the offsite source (ref. 3, 4). 2EGS*EG2 (Division III) is capable of powering either the Division I or Division II 4.16 KV emergency bus through manual breaker alignments. It is unlikely that these actions could be performed within the fifteen-minute interval of this EAL. The availability of 2EGS*EG2 as an onsite AC power source in Table S-1 only applies if 2EGS*EG2 is aligned to energize 2ENS*SWG101 or 2ENS*SWG103.

~~The fifteen-minute interval was selected as a threshold to exclude transient power losses.~~

The NMP2 electrical distribution configuration precludes restoration of offsite power sources within 15 minutes in all instances, once lost. Therefore no time component is allocated for this EAL threshold.

Generic

Prolonged loss of off-site 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 to emergency busses.

~~The NMP2 electrical distribution configuration precludes restoration of offsite power sources within 15 minutes in all instances, once lost. Therefore no time component is allocated for this EAL threshold.~~

NMP2 Basis Reference(s):

1. USAR Section 8.2
2. USAR Section 8.3
3. N2-SOP-03 Loss of AC Power
4. N2-SOP-01 Station Blackout

5. NEI 99-01 IC SU1

Category: S – System Malfunction

Subcategory: 2 – Loss of DC Power

Initiating Condition: Loss of **all** emergency DC power for ≥ 15 min.

EAL:

SS2.1	**Site Area Emergency**
< 105 VDC on **both** 2BYS*SWG002A and 2BYS*SWG002B for ≥ 15 min. (Note 4)	

Note 4: The ED should **not** wait until the applicable time has elapsed, but should declare the event as soon as it is determined that the condition has exceeded, or will likely exceed, the applicable time.

Mode Applicability:

1 - Power Operation, 2 - Startup, 3 - Hot Shutdown

Basis:

Plant-Specific

The emergency 125 VDC power system includes three electrically independent and separate switchgears (2BYS*SWG002A, 2BYS*SWG002B and 2CES*IPNL414). Division I ((2BYS*SWG002A) and Division II (2BYS*SWG002B) feed the redundant emergency DC loads associated with Divisions I and II of the emergency onsite AC system, respectively. Division III (2CES*PNP414) feeds the emergency DC loads associated with Division III (HPCS system). 2CES*IPNL414 is not included in this EAL because it only supplies power to HPCS loads.

Each emergency 125 VDC distribution system has a battery and a battery charger that are normally connected to the bus such that these two sources of power are operating in parallel. The charger is normally supplying system electrical loads with the battery on a float charge. Should both battery chargers for any particular battery be out of service at any point in the DC load cycle, the battery is capable of starting and operating its associated loads for 2 hr according to a precalculated load profile without the battery terminal voltage falling below minimum acceptable level, 105 VDC. (ref. 1, 2, 3)

This EAL is the hot condition equivalent of the cold condition loss of DC power EAL CU2.1.

Generic

Loss of all DC power compromises ability to monitor and control plant safety functions. Prolonged loss of all DC power will cause core uncovering and loss of containment integrity when there is significant decay heat and sensible heat in the reactor system.

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

Escalation to a General Emergency would occur by EALs in Category R and Category F.

NMP2 Basis Reference(s):

1. USAR Section 8.3.2.1.2
2. Improved Technical Specifications Nine Mile Point Nuclear Station, Unit No. 2, 3.8.4
3. N2-SOP-04 Loss of DC Power
4. NEI 99-01 IC SS3

Category: S – System Malfunction

Subcategory: 3 – Criticality & RPS Failure

Initiating Condition: Automatic scram and **all** manual actions fail to shut down the reactor and indication of an extreme challenge to the ability to cool the core exists

EAL:

SG3.1 General Emergency

An automatic scram fails to shut down the reactor as indicated by reactor power > 4%

AND

All manual actions fail to shut down the reactor as indicated by reactor power > 4%

AND EITHER of the following exist or have occurred:

RPV water level **cannot** be restored and maintained above -39 in. or RPV water level **cannot** be determined

OR

Suppression pool temperature and RPV pressure **cannot** be maintained below the Heat Capacity Temperature Limit (N2-EOP-PC Figure M)

Mode Applicability:

1 - Power Operation, 2 - Startup

Basis:

Plant-Specific

This EAL addresses the following:

- Any automatic reactor scram signal followed by a manual scram that fails to shut down the reactor to an extent the reactor is producing energy in excess of the heat load for which the safety systems were designed (EAL SS3.1), and
- Indications that either core cooling is extremely challenged or heat removal is extremely challenged.

Reactor shutdown achieved by use of the alternate control rod insertion methods of EOP-C5 is also credited as a successful manual scram provided reactor power can be reduced below the APRM downscale trip setpoint before indications of an extreme challenge to either core cooling or heat removal exist (ref. 1, 2).

The APRM downscale trip setpoint (4%) is a minimum reading on the power range scale that indicates power production (ref. 1, 2). It also approximates the decay heat which the shutdown systems were designed to remove and is indicative of a condition requiring immediate response to prevent subsequent core damage. At or below the APRM downscale trip setpoint, plant response will be similar to that observed during a normal shutdown. Nuclear instrumentation (APRM/IRM) indications or other reactor parameters (e.g., number of open SRVs, number of open main turbine bypass valves, main steam flow, RPV pressure and suppression pool temperature trend, etc.) can be used to determine if reactor power is greater than 4% power (ref. 2).

The combination of failure of both front line and backup protection systems to function in response to a plant transient, along with the continued production of heat, poses a direct threat to the Fuel Clad and RCS barriers.

By definition, an operating mode change occurs when the Mode Switch is moved from the startup/hot standby or run position to the shutdown position. The plant operating mode that existed at the time the event occurs (i.e., Power Operation or Startup), however, requires emergency classification of at least an Alert. The operating mode change associated with movement of the Mode Switch, by itself, does not justify failure to declare an emergency for ATWS events.

Indication that core cooling is extremely challenged is manifested by:

- RPV level cannot be restored and maintained above -39 in. (ref. 1, 2). The Minimum Steam Cooling RPV Water Level (MSCRWL) is the lowest RPV water level at which the covered portion of the reactor core will generate sufficient steam to preclude any clad temperature in the uncovered portion of the core from exceeding 1500°F. Consistent with the EOP definition of "cannot be restored and maintained," the determination that RPV level cannot be restored and maintained above the MSCRWL may be made at, before, or after RPV level actually decreases to this point.

  When RPV water level cannot be determined, EOPs require RPV flooding strategies. RPV water level indication provides the primary means of knowing if adequate core cooling is being maintained. When all means of determining RPV water level are unavailable, the fuel clad barrier is threatened and reliance on alternate means of assuring adequate core cooling must be attempted. The instructions in N2-EOP-C4 specify these means, which include emergency depressurization of the RPV and injection into the RPV at a rate needed to flood to the elevation of the main steam lines or hold RPV pressure above the Minimum Steam Cooling Pressure (in ATWS events) (ref. 3).

- The HCTL is the highest wetwell temperature from which emergency RPV depressurization will not raise:
  - Suppression chamber temperature above the design value (270°F), or
  - Suppression chamber pressure above Primary Containment Pressure Limit before the rate of energy transfer from the RPV to the containment is greater than the capacity of the containment vent.

  The HCTL is a function of RPV pressure and suppression pool water level. It is utilized to preclude failure of the containment and equipment in the containment necessary for the safe shutdown of the plant. Plant parameters in excess of the HCTL could be a precursor of primary containment failure. (ref. 2)

  The HCTL is given in N2-EOP-PC Figure M. This threshold is met when RPV BLOW DOWN is required in N2-EOP-PC, Step SPT-6 (ref. 4). This condition addresses loss of functions required for hot shutdown with the reactor at pressure and temperature.

<u>Generic</u>

Under these conditions, the reactor is producing more heat than the maximum decay heat load for which the safety systems are designed and efforts to bring the reactor subcritical are unsuccessful.

The reactor should be considered shutdown when it producing less heat than the maximum decay heat load for which the safety systems are designed (4% power). In the event either of these challenges exists at a time that the reactor has not been brought below the power associated with the safety system design a core melt sequence exists. In this situation, core degradation can occur rapidly. For this reason, the General Emergency declaration is intended to be anticipatory of the fission product barrier table declaration to permit maximum off-site intervention time.

NMP2 Basis Reference(s):

1. N2-EOP-C5 Failure to Scram
2. NER-2M-039, NMP2 Emergency Operating Procedures (EOP) Basis Document
3. N2-EOP-C4 RPV Flooding

4. N2-EOP-PC Primary Containment Control
5. NEI 99-01 IC SG2

Category: S – System Malfunction

Subcategory: 3 – Criticality & RPS Failure

Initiating Condition: Automatic scram fails to shut down the reactor and manual actions taken from the reactor control console are **not** successful in shutting down the reactor

EAL:

SS3.1 Site Area Emergency

An automatic scram failed to shut down the reactor as indicated by reactor power > 4%

AND

Manual actions taken at the reactor control console (mode switch in shutdown, manual scram push buttons and ARI) failed to shut down the reactor as indicated by reactor power > 4%

Mode Applicability:

1 - Power Operation, 2 - Startup

Basis:

Plant-Specific

This EAL addresses any automatic reactor scram signal followed by a manual scram that failed to shut down the reactor to an extent the reactor is producing energy in excess of the heat load for which the safety systems were designed.

For the purposes of emergency classification at the Site Area Emergency level, successful manual scram actions are those which can be quickly performed from the reactor control console (i.e., Mode Switch, manual scram pushbuttons and ARI actuation). Reactor shutdown achieved by use of the alternate control rod insertion methods of EOP-C5 does not constitute a successful manual scram (ref. 1, 2).

The APRM downscale trip setpoint (4%) is a minimum reading on the power range scale that indicates power production (ref. 1). It also approximates the decay heat which the shutdown systems were designed to remove and is indicative of a condition requiring immediate response to prevent subsequent core damage. At or below the APRM downscale trip setpoint, plant response will be similar to that observed during a normal shutdown. Nuclear instrumentation (APRM/IRM) indications or other reactor parameters (e.g., number of open SRVs, number of open main turbine bypass valves, main steam flow, RPV pressure and wetwell temperature trend, etc.) can be used to determine if reactor power is greater than 4% power.

By definition, an operating mode change occurs when the Mode Switch is moved from the startup/hot standby or run position to the shutdown position. The plant operating mode that existed at the time the event occurs (i.e., Power Operation or Startup), however, requires emergency classification of at least an Alert. The operating mode change associated with movement of the Mode Switch, by itself, does not justify failure to declare an emergency for ATWS events.

Escalation of this event to a General Emergency would be under EAL SG3.1 or Emergency Director judgment.

<u>Generic</u>

Under these conditions, the reactor is producing more heat than the maximum decay heat load for which the safety systems are designed and efforts to bring the reactor subcritical are unsuccessful. A Site Area Emergency is warranted because conditions exist that lead to IMMINENT loss or potential loss of both fuel clad and RCS.

The reactor should be considered shutdown when it producing less heat than the maximum decay heat load for which the safety systems are designed (4% power).

Manual scram actions taken at the reactor control console are any set of actions by the reactor operator(s) at which causes or should cause control rods to be rapidly inserted into the core and shuts down the reactor.

Manual scram actions are not considered successful if action away from the reactor control console is required to scram the reactor. This EAL is still applicable even if actions taken away from the reactor control console are successful in shutting the reactor down because the design limits of the fuel may have been exceeded or because of the gross failure of the Reactor Protection System to shutdown the plant.

Escalation of this event to a General Emergency would be due to a prolonged condition leading to an extreme challenge to either core-cooling or heat removal.

NMP2 Basis Reference(s):

1. N2-EOP-C5 Failure to Scram
2. NER-2M-039, NMP2 Emergency Operating Procedures (EOP) Basis Document
3. NEI 99-01 IC SS2

Category: S – System Malfunction

Subcategory: 3 – Criticality & RPS Failure

Initiating Condition: Automatic scram failed to shut down the reactor and the manual actions taken from the reactor control console are successful in shutting down the reactor

EAL:

SA3.1 Alert

An automatic scram failed to shut down the reactor

AND

Manual actions taken at the reactor control console (mode switch in shutdown, manual scram push buttons or ARI) successfully shut down the reactor as indicated by reactor power ≤ 4%

Mode Applicability:

1 - Power Operation, 2 - Startup

Basis:

Plant-Specific

The first condition of this EAL identifies the need to cease critical reactor operations by actuation of the automatic Reactor Protection System (RPS) scram function. A reactor scram is automatically initiated by the Reactor Protection System (RPS) when certain continuously monitored parameters exceed predetermined setpoints. A reactor scram may be the result of manual or automatic action in response to ~~any of the following~~various plant conditions (ref. 1):

~~Parameter~~	~~Setpoint~~	~~Bypassed~~
~~SRM Upscale Trip~~	~~≥ 2 x 105 cps~~	~~Shorting links installed or joystick to BYPASS~~
~~IRM Upscale Trip~~	~~≥ 120/125 percent of scale~~	~~Reactor mode switch in RUN or joystick to BYPASS~~
~~IRM Inop~~	~~Not in OPERATE Detector HV low Module unplugged~~	~~Reactor mode switch in RUN or joystick to BYPASS~~
~~APRM Upscale Neutron Flux (Setdown)~~	~~≥ 15%~~	~~Reactor mode switch in RUN or joystick to BYPASS~~

~~Parameter~~	~~Setpoint~~	~~Bypassed~~
~~APRM Upscale Neutron Flux~~	~~≥ 118%~~	~~Reactor mode switch not in RUN or joystick to BYPASS~~
~~Thermal~~	~~.58 (W - ΔW) + 59% 113.5% maximum~~	~~Joystick in BYPASS~~
~~APRM Inop~~	~~Critical Self-test fault detected APRM keylock switch in INOP Watchdog Timer timed out Loss of input Power Note: Low LPRM count INOP does not insert a scram signal.~~	~~Joystick in BYPASS~~
~~OPRM Upscale~~	~~PBA: N ≥ 16 and the amplitude of the oscillation is ≥ 1.5 ABA: Oscillation Peak ≥ 1.3 GRBA: Cell magnitude change > 1.3~~	~~< 30% Reactor power as indicated by APRM's OR when core drive flow is > 60% of rated~~
~~Reactor Pressure High~~	~~≥ 1052 psig~~	~~N/A~~
~~Reactor Water Level Low~~	~~≤ Low Level (159.3")~~	~~N/A~~
~~Turbine Stop Valve Closure~~	~~5% closed~~	~~< 30% power*~~
~~Turbine Control Valve Fast Closure~~	~~530 psig ETS oil pressure~~	~~< 30% power*~~
~~MSIV Closure~~	~~8% Closed~~	~~Reactor mode switch not in RUN~~
~~Scram Discharge Volume Level High~~	~~Level Switch - 48.5 inches Level Transmitter - 43.4 inches~~	~~Key-lock switch and Reactor mode switch in SHUTDOWN or REFUEL~~
~~Drywell Pressure High~~	~~≥ 1.68 psig~~	~~N/A~~
~~Manual Scram Pushbuttons~~	~~N/A~~	~~N/A~~
~~Mode Switch in SHUTDOWN~~	~~N/A~~	~~After 10 seconds~~

Following a successful reactor scram, rapid insertion of the control rods occurs. Nuclear power promptly drops to a fraction of the original power level and then decays to a level several decades less with a negative period. The reactor power drop continues until reactor power reaches the point at which the influence of source neutrons on reactor power starts to be observable. A predictable post-scram response from an automatic reactor scram signal should therefore consist of a prompt drop in reactor power as sensed by the nuclear instrumentation and a lowering of power into the source range. A successful scram has therefore occurred when there is sufficient rod insertion from the trip of RPS to bring the reactor power to or below the APRM downscale trip setpoint of 4%. For the purposes of this EAL, a successful automatic initiation of ARI that reduces reactor power to or below 4% is a not a successful automatic scram. If automatic actuation of ARI has occurred and caused reactor shutdown, the automatic RPS scram must have failed. ARI is a backup means of inserting control rods in the unlikely event that an automatic RPS scram signal exists but the reactor continues to generate significant power. (ref. 2, 3)

For the purposes of emergency classification at the Alert level, successful manual scram actions are those which can be quickly performed from the reactor control console (i.e., mode switch, manual scram pushbuttons, and manual ARI actuation). Reactor shutdown achieved by use of the alternate control rod insertion methods of EOP-C5 does not constitute a successful manual scram (ref. 2).

Following any automatic RPS scram signal EOPs prescribe insertion of redundant manual scram signals to back up the automatic RPS scram function and ensure reactor shutdown is achieved. Even if the first subsequent manual scram signal inserts all control rods to the full-in position immediately after the initial failure of the automatic scram, the lowest level of classification that must be declared is an Alert.

If the operator determines the reactor must be scrammed before one of the RPS setpoints is reached, procedures require that the Mode Switch first be placed in the shutdown position. Although manipulation of the Mode Switch is a manual action, the RPS logic trains are actuated as with an automatic RPS-initiated scram. If reactor power remains above the APRM downscale trip setpoint after the Mode Switch is placed in shutdown, RPS has failed and, as a minimum, an Alert emergency declaration is required. If subsequent actuation of the reactor scram pushbuttons and manual initiation of ARI do not reduce reactor power to or below the APRM downscale trip setpoint, a Site Area Emergency declaration is required under EAL SS3.1.

In the event that the operator identifies a reactor scram is IMMINENT and initiates a successful manual reactor scram before the automatic scram setpoint is reached, no declaration is required. The successful manual scram of the reactor before it reaches its automatic scram setpoint or reactor scram signals caused by instrumentation channel failures do not lead to a potential fission product barrier loss. If manual reactor scram actions fail to reduce reactor power to or below 4%, the event escalates to the Site Area Emergency under EAL SS3.1.

By procedure, operator actions include the initiation of an immediate manual scram following receipt of an automatic scram signal. If there are no clear indications that the automatic scram failed (such as a time delay following indications that a scram setpoint was exceeded), it may be difficult to determine if the reactor was shut down because of automatic scram or manual actions. If a subsequent review of the scram actuation indications reveals that the automatic scram did not cause the reactor to be shut down, consideration should be given to evaluating the fuel for potential damage and the reporting requirements of 50.72 should be considered for the transient event.

By definition, an operating mode change occurs when the Mode Switch is moved from the startup/hot standby or run position to the shutdown position. The plant operating mode that existed at the time the event occurs (i.e., Power Operation or Startup), however, requires emergency classification of at least an Alert. The operating mode change associated with movement of the Mode Switch, by itself, does not justify failure to declare an emergency for ATWS events.

Generic

The reactor should be considered shutdown when it producing less heat than the maximum decay heat load for which the safety systems are designed (4% power).

Manual scram actions taken at the reactor control console are any set of actions by the reactor operator(s) which causes or should cause control rods to be rapidly inserted into the core and shuts down the reactor.

This condition indicates failure of the automatic protection system to scram the reactor. This condition is more than a potential degradation of a safety system in that a front line automatic protection system did not function in response to a scram signal. Thus the plant safety has been compromised because of the failure of RPS to automatically shut down the plant. An Alert is indicated because conditions may exist that lead to potential loss of fuel clad barrier or RCS barrier and because of the failure of the Reactor Protection System to automatically shut down the plant.

If manual actions taken at the reactor control console fail to shut down the reactor, the event would escalate to a Site Area Emergency.

NMP2 Basis Reference(s):

1. Improved Technical Specifications Nine Mile Point Nuclear Station, Unit No. 2, Table 3.3.1.1-1
2. N2-EOP-C5 Failure to Scram
3. NER-2M-039, NMP2 Emergency Operating Procedures (EOP) Basis Document
4. NEI 99-01 IC SA2

Category: S – System Malfunction

Subcategory: 3 – Criticality & RPS Failure

Initiating Condition: Inadvertent criticality

EAL:

SU3.1	**Unusual Event**
An UNPLANNED sustained positive period observed on nuclear instrumentation	

Mode Applicability:

3 - Hot Shutdown

Basis:

Plant-Specific

The term "sustained" is used to allow exclusion of expected short-term positive periods from planned fuel bundle or control rod movements during core alteration. These short-term positive periods are the result of the rise in neutron population due to subcritical multiplication.

Generic

This EAL addresses inadvertent criticality events. While the primary concern of this EAL is criticality This EAL addresses inadvertent criticality events. This EAL indicates a potential degradation of the level of safety of the plant, warranting a UE classification. This EAL excludes inadvertent criticalities that occur during planned reactivity changes associated with reactor startups (e.g., criticality earlier than estimated).

Escalation would be by EALs in Category F, as appropriate to the operating mode at the time of the event.

NMP2 Basis Reference(s):

1. NEI 99-01 IC SU8

Category: S – System Malfunction

Subcategory: 4 – Inability to Reach or Maintain Shutdown Conditions

Initiating Condition: Inability to reach required shutdown within Technical Specification limits

EAL:

SU4.1	**Unusual Event**
Plant is **not** brought to required operating mode within Technical Specifications LCO required action completion time	

Mode Applicability:

1 - Power Operation, 2 - Startup, 3 - Hot Shutdown

Basis:

Plant-Specific

Limiting Conditions for Operation (LCOs) specify minimum requirements for ensuring safe operation of the unit. The actions associated with an LCO state conditions that typically describe the ways in which the requirements of the LCO can fail to be met. Specified with each stated condition are required action completion times. (ref. 1)

Generic

Limiting Conditions of Operation (LCOs) require the plant to be brought to a required operating 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 four 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 required action completion time in the Technical Specifications. An immediate UE is required when the plant is not brought to the required operating mode within the allowable required action completion time in the Technical Specifications. Declaration of a UE is based on the time at which the LCO-specified required action completion time period elapses under the site Technical Specifications and is not related to how long a condition may have existed.

NMP2 Basis Reference(s):

1. Improved Technical Specifications Nine Mile Point Nuclear Station, Unit No. 2, 1.3
2. NEI 99-01 IC SU2

Category: S – System Malfunction

Subcategory: 5 – Instrumentation

Initiating Condition: Inability to monitor a significant transient in progress

EAL:

SS5.1 Site Area Emergency

Loss of > approximately 75% of annunciation or indication on all of the following Control Room panels for ≥ 15 min. (Note 4):

- 2CEC*PNL601
- 2CEC*PNL602
- 2CEC*PNL603
- 2CEC*PNL851
- 2CEC*PNL852

AND

A significant transient is in progress, Table S-2

AND

Compensatory indications are unavailable (Plant Process Computer, SPDS)

Note 4: The ED should **not** wait until the applicable time has elapsed, but should declare the event as soon as it is determined that the condition has exceeded, or will likely exceed, the applicable time

Table S-2 Significant Transients
• Automatic turbine runback > 25% thermal reactor power
• Electric load rejection > 25% full electrical load
• Reactor scram
• ECCS injection
• Thermal power oscillations > 10%

Mode Applicability:

1 - Power Operation, 2 - Startup, 3 - Hot Shutdown

Basis:

Plant-Specific

Plant Process Computer and SPDS are considered compensatory indication.

Significant transients are listed in Table S-2.

Generic

This EAL is intended to recognize the threat to plant safety associated with the complete loss of capability of the control room staff to monitor plant response to a significant transient.

"Planned" and "UNPLANNED" *actions are not differentiated since the* loss *of* instrumentation *of* this magnitude is of such significance during a transient that the cause of the loss is not an ameliorating factor.

Quantification is arbitrary, however, it is estimated that if approximately 75% of the safety system annunciators or indicators are lost, there is an increased risk that a degraded plant condition could go undetected. It is not intended that plant personnel perform a detailed count of the instrumentation lost but use the value as a judgment threshold for determining the severity of the plant conditions. It is also not intended that the Shift Manager be tasked with making a judgment decision as to whether additional personnel are required to provide increased monitoring of system operation.

It is further recognized that most plant designs provide redundant safety system indication powered from separate uninterruptible power supplies. While failure of a large portion of annunciators is more likely than a failure of a large portion of indications, the concern is included in this EAL due to difficulty associated with assessment of plant conditions. The loss of specific, or several, safety system indicators should remain a function of that specific system or component operability status. This will be addressed by the specific Technical Specification. The initiation of a Technical Specification imposed plant shutdown related to the instrument loss will be reported via 10 CFR 50.72. If the shutdown is not in compliance with the Technical Specification action, the NOUE is based on EAL SU4.1

A Site Area Emergency is considered to exist if the control room staff cannot monitor safety functions needed for protection of the public while a significant transient is in progress.

Annunciators for this EAL are limited to include those identified in the Abnormal Operating Procedures, in the Emergency Operating Procedures, and in other EALs (.g., area, process, and/or effluent rad monitors, etc.)

Indications needed to monitor safety functions necessary for protection of the public include control room indications, computer generated indications and dedicated annunciation capability.

"Compensatory indications" in this context includes computer based information such as Plant Process Computer and SPDS.

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

NMP2 Basis Reference(s):

1. USAR Figure 1.2-15
2. N2-OP-91A Process Computer
3. N2-OP-91B Safety Parameter Display System (SPDS)
4. SOP-78A EOP Key Parameter Alternate Instrumentation
5. NEI 99-01 IC SS6

Category: S – System Malfunction

Subcategory: 5 – Instrumentation

Initiating Condition: UNPLANNED loss of safety system annunciation or indication in the Control Room with either (1) a significant transient in progress, or (2) compensatory indicators are unavailable

EAL:

SA5.1 Alert

UNPLANNED loss of > approximately 75% of annunciation or indication on all of the following Control Room panels for ≥ 15 min. (Note 4):

- 2CEC*PNL601
- 2CEC*PNL602
- 2CEC*PNL603
- 2CEC*PNL851
- 2CEC*PNL852

AND EITHER:

A significant transient is in progress, Table S-2

OR

Compensatory indications are unavailable (Plant Process Computer, SPDS)

Note 4: The ED should **not** wait until the applicable time has elapsed, but should declare the event as soon as it is determined that the condition has exceeded, or will likely exceed, the applicable time

Table S-2 Significant Transients
• Automatic turbine runback > 25% thermal reactor power
• Electric load rejection > 25% full electrical load
• Reactor scram
• ECCS injection
• Thermal power oscillations > 10%

Mode Applicability:

1 - Power Operation, 2 - Startup, 3 - Hot Shutdown

Basis:

Plant-Specific

Plant Process Computer and SPDS are considered compensatory indication.

Significant transients are listed in Table S-2.

Generic

This EAL is intended to recognize the difficulty associated with monitoring changing plant conditions without the use of a major portion of the annunciation or indication equipment during a significant transient.

"Planned" loss of annunciators or indicators includes scheduled maintenance and testing activities.

Quantification is arbitrary, however, it is estimated that if approximately 75% of the safety system annunciators or indicators are lost, there is an increased risk that a degraded plant condition could go undetected. It is not intended that plant personnel perform a detailed count of the instrumentation lost but use the value as a judgment threshold for determining the severity of the plant conditions. It is also not intended that the Shift Manager be tasked with making a judgment decision as to whether additional personnel are required to provide increased monitoring of system operation.

It is further recognized that most plant designs provide redundant safety system indication powered from separate uninterruptible power supplies. While failure of a large portion of annunciators is more likely than a failure of a large portion of indications, the concern is included in this EAL due to difficulty associated with assessment of plant conditions. The loss of specific, or several, safety system indicators should remain a function of that specific system or component operability status. This will be addressed by the specific Technical Specification. The initiation of a Technical Specification imposed plant shutdown related to the instrument loss will be reported via 10 CFR 50.72. If the shutdown is not in compliance with the Technical Specification action, the UE is based on EAL SU4.1.

Annunciators or indicators for this EAL include those identified in the Abnormal Operating Procedures, in the Emergency Operating Procedures, and in other EALs (e.g., area, process, and/or effluent rad monitors, etc.).

"Compensatory indications" in this context includes computer based information such as Plant Process Computer and SPDS.

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

This Alert will be escalated to a Site Area Emergency if the operating crew cannot monitor the transient in progress due to a concurrent loss of compensatory indications with a significant transient in progress during the loss of annunciation or indication.

NMP2 Basis Reference(s):

1. USAR Figure 1.2-15
2. N2-OP-91A Process Computer
3. N2-OP-91B Safety Parameter Display System (SPDS)
4. SOP-78A EOP Key Parameter Alternate Instrumentation
5. NEI 99-01 IC SA4

Category: S – System Malfunction

Subcategory: 5 – Instrumentation

Initiating Condition: UNPLANNED loss of safety system annunciation or indication in the Control Room for ≥ 15 min.

EAL:

SU5.1 Unusual Event

UNPLANNED loss of > approximately 75% of annunciation or indication on all of the following Control Room panels for ≥ 15 min. (Note 4):

- 2CEC*PNL601
- 2CEC*PNL602
- 2CEC*PNL603
- 2CEC*PNL851
- 2CEC*PNL852

Note 4: The ED should **not** wait until the applicable time has elapsed, but should declare the event as soon as it is determined that the condition has exceeded, or will likely exceed, the applicable time

Mode Applicability:

1 - Power Operation, 2 - Startup, 3 - Hot Shutdown

Basis:

Plant-Specific

None

Generic

This EAL is intended to recognize the difficulty associated with monitoring changing plant conditions without the use of a major portion of the annunciation or indication equipment.

Recognition of the availability of computer based indication equipment is considered.

"Planned" loss of annunciators or indicators includes scheduled maintenance and testing activities.

Quantification is arbitrary, however, it is estimated that if approximately 75% of the safety system annunciators or indicators are lost, there is an increased risk that a degraded plant condition could go undetected. It is not intended that plant personnel perform a detailed count of the instrumentation lost but use the value as a judgment threshold for determining the severity of the plant conditions.

It is further recognized that plant design provides redundant safety system indication powered from separate uninterruptible power supplies. While failure of a large portion of annunciators is more likely than a failure of a large portion of indications, the concern is included in this EAL due to difficulty associated with assessment of plant conditions. The loss of specific, or several, safety system indicators should remain a function of that specific system or component operability status. This will be addressed by the specific Technical Specification. The initiation of a Technical Specification imposed plant shutdown related to the instrument loss will be reported via 10 CFR 50.72. If the shutdown is not in compliance with the Technical Specification action, the UE is based on EAL SU4.1.

Annunciators or indicators for this EAL include those identified in the Abnormal Operating Procedures, in the Emergency Operating Procedures, and in other EALs (e.g., area, process, and/or effluent rad monitors, etc.).

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

This UE will be escalated to an Alert based on a concurrent loss of compensatory indications or if a significant transient is in progress during the loss of annunciation or indication.

NMP2 Basis Reference(s):

1. USAR Figure 1.2-15
2. N2-OP-91A Process Computer
3. N2-OP-91B Safety Parameter Display System (SPDS)
4. SOP-78A EOP Key Parameter Alternate Instrumentation
5. NEI 99-01 IC SU3

Category: S – System Malfunction

Subcategory: 6 – Communications

Initiating Condition: Loss of **all** onsite or offsite communications capabilities

EAL:

SU6.1 Unusual Event

Loss of **all** Table S-3 onsite (internal) communication methods affecting the ability to perform routine operations

OR

Loss of **all** Table S-3 offsite (external) communication methods affecting the ability to perform offsite notifications

Table S-3 Communications Systems		
System	**Onsite** (internal)	**Offsite** (external)
PBX (normal dial telephones)	X	X
Gaitronics	X	
Station radio (portable)	X	
Control Room installed satellite phones (non portable)		X
ENS		X
RECS		X
~~UHF radios~~		~~X~~

Mode Applicability:

1 - Power Operation, 2 - Startup, 3 - Hot Shutdown

Basis:

<u>Plant-Specific</u>

Onsite/offsite communications systems are listed in Table S-3 (ref. 1, 2, 3).

This EAL is the hot condition equivalent of the cold condition EAL CU6.1.

<u>Generic</u>

The purpose of this EAL is to recognize a loss of communications capability that either defeats the plant operations staff ability to perform routine tasks necessary for plant operations or the ability to communicate issues with off-site authorities.

The loss of off-site communications ability is expected to be significantly more comprehensive than the condition addressed by 10 CFR 50.72.

The availability of one method of ordinary off-site communications is sufficient to inform federal, state, and local authorities of plant problems. This EAL is intended to be used only when extraordinary means (e.g., relaying of information from non-routine radio transmissions, individuals being sent to off-site locations, etc.) are being used to make communications possible.

NMP2 Basis Reference(s):

1. USAR Section 9.5.2
2. Nine Mile Point Nuclear Station Site Emergency Plan, Section 7.2
3. N2-OP-76 Plant Communications
4. NEI 99-01 IC SU6

Category: S – System Malfunction

Subcategory: 7 – Fuel Clad Degradation

Initiating Condition: Fuel clad degradation

EAL:

SU7.1	**Unusual Event**
Reactor coolant activity > 4 μCi/gm I-131 Equivalent	

Mode Applicability:

1 - Power Operation, 2 - Startup, 3 - Hot Shutdown

Basis:

Plant-Specific

This EAL addresses reactor coolant samples exceeding Technical Specification 3.4.8 (ref. 1). A reactor coolant sample analysis with specific activity in excess of the Technical Specification limit of 4 μCi/gm I-131 Equivalent is indicative of a degradation of the fuel clad, and is a precursor of more serious problems. This activity level for which operation is allowed to continue for up to 48 hours to accommodate short duration Iodine spikes following changes in thermal power.

Generic

This EAL is included because it is a precursor of more serious conditions and, as result, is considered to be a potential degradation of the level of safety of the plant.

Escalation of this EAL to the Alert level is via the EALs in Category F.

This threshold addresses coolant samples exceeding coolant technical specifications for transient iodine spiking limits.

NMP2 Basis Reference(s):

1. Improved Technical Specifications Nine Mile Point Nuclear Station, Unit No. 2, 3.4.8
2. Improved Technical Specifications Nine Mile Point Nuclear Station, Unit No. 2, 3.4.8.A.1
3. NEI 99-01 IC SU4

Category: S – System Malfunction

Subcategory: 7 – Fuel Clad Degradation

Initiating Condition: Fuel clad degradation

EAL:

SU7.2	**Unusual Event**
Offgas radiation DRMS high (red) alarm for ≥ 15 min.	

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Mode Applicability:

1 - Power Operation, 2 - Startup, 3 - Hot Shutdown

Basis:

Plant-Specific

Elevated offgas radiation activity represents a potential degradation in the level of safety of the plant and a potential precursor of more serious problems. The Technical Specification allowable limit is an offgas level not to exceed 350,000 μCi/sec (ref. 1). The DRMS alarm setpoint has been conservatively selected because it is operationally significant and is readily recognizable by Control Room operating staff. 15 minutes is allotted for operator action to reduce the offgas radiation levels and exclude TRANSIENT conditions (ref. 2, 3, 4). The high offgas radiation alarm is set using methodology outlined in the ODCM (ref. 5).

Generic

This EAL is included because it is a precursor of more serious conditions and, as result, is considered to be a potential degradation of the level of safety of the plant.

Escalation of this EAL to the Alert level is via the EALs in Category F.

This threshold addresses radiation monitor readings that provide indication of a degradation of fuel clad integrity.

NMP2 Basis Reference(s):

1. Improved Technical Specifications Nine Mile Point Nuclear Station, Unit No.2, 3.7.4
2. N2-ARP-01 Annunciator Response Procedures for annunciator 851253
3. N2-ARP-01 Annunciator Response Procedures for annunciator 851326
4. N2-SOP-17 Fuel Failure or High Activity in Rx Coolant or Offgas
5. Offsite Dose Calculation Manual 3.3.2
6. NEI 99-01 IC SU4

Category: S – System Malfunction
Subcategory: 8 – RCS Leakage
Initiating Condition: RCS leakage

EAL:

SU8.1	**Unusual Event**
Unidentified or reactor coolant pressure boundary leakage > 10 gpm **OR** Identified reactor coolant leakage > 25 gpm	

Mode Applicability:

1 - Power Operation, 2 - Startup, 3 - Hot Shutdown

Basis:

Plant-Specific

Elevated RCS leakage may be detected by the following annunciators (ref. 1-4):

- 873115 DRWL FLR DRN LEAK RATE HIGH (setpoint 4 gpm)
- 873111 DRWL FLR DRN TANK 1 LEVELHI-HI
- 873105 DRWL EQPT DRN TANK 1 LEVEL HI-HI
- 873110 DRWL EQPT DRN DAILY LK RATE HIGH

The Plant Process Computer monitors unidentified and identified leakage over six minute intervals (Computer Point DERXA01) as well as a twenty-four hour average (Computer Point 2DER-FI101). Leak rates can also be verified by alternate measurements according to N2-OSP-LOG-S001, Attachments 6 and 7 (ref. 5, 6).

Generic

This EAL is included as a UE because it may be a precursor of more serious conditions and, as result, is considered to be a potential degradation of the level of safety of the plant. The 10 gpm value for the unidentified or pressure boundary leakage was selected as it is observable with normal control room indications. Lesser values must generally be determined through time-consuming surveillance tests (e.g., mass balances).

Relief valve normal operation should be excluded from this EAL. However, a relief valve that operates and fails to close per design should be considered applicable to this EAL if the relief valve cannot be isolated.

The EAL for identified leakage is set at a higher value due to the lesser significance of identified leakage in comparison to unidentified or pressure boundary leakage. In either case, escalation of this EAL to the Alert level is via EALs in Category F.

NMP2 Basis Reference(s):

1. N2-ARP-01 Annunciator Response Procedures for annunciator 873115
2. N2-ARP-01 Annunciator Response Procedures for annunciator 873111
3. N2-ARP-01 Annunciator Response Procedures for annunciator 873105
4. N2-ARP-01 Annunciator Response Procedures for annunciator 873110
5. N2-OSP-LOG-S001 Shift Checks - Mode 1
6. N2-OP-67 Drywell Equipment and Floor Drains System
7. NEI 99-01 IC SU5

Category F – Fission Product Barrier Degradation

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

EALs in this category represent threats to the defense in depth design concept that precludes the release of highly radioactive fission products to the environment. This concept relies on multiple physical barriers any one of which, if maintained INTACT, precludes the release of significant amounts of radioactive fission products to the environment. The primary fission product barriers are:

A. Fuel Clad (FC): Zirconium tubes which house the ceramic uranium oxide pellets along with the end plugs which are welded into each end of the fuel rods comprise the FC barrier.
B. Reactor Coolant System (RCS): The reactor vessel shell, vessel head, CRD housings, vessel nozzles and penetrations, and all primary systems directly connected to the RPV up to the outermost Primary Containment isolation valve comprise the RCS barrier.
C. Containment (PC): The drywell, the suppression chamber/pool, their respective interconnecting paths, and other connections up to and including the outermost containment isolation valves comprise the Primary Containment barrier.

The EALs in this category require evaluation of the loss and potential loss thresholds listed in the fission product barrier matrix of Table F-1 (Attachment 2). "Loss" and "Potential Loss" signify the relative damage and threat of damage to the barrier. "Loss" means the barrier no longer assures containment of radioactive materials. "Potential Loss" means integrity of the barrier is threatened and could be lost if conditions continue to degrade. The number of barriers that are lost or potentially lost and the following criteria determine the appropriate emergency classification level:

Unusual Event:
Any loss or any potential loss of Containment

Alert:
Any loss or any potential loss of either Fuel Clad or RCS

Site Area Emergency:
Loss or potential loss of any two barriers

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

The logic used for Category F EALs reflects the following considerations:

- The Fuel Clad Barrier and the RCS Barrier are weighted more heavily than the Containment Barrier. UE EALs associated with RCS and Fuel Clad Barriers are addressed under Category S.
- At the Site Area Emergency level, there must be some ability to dynamically assess how far present conditions are from the threshold for a General Emergency. For example, if Fuel Clad and RCS Barrier "Loss" thresholds existed, that, in addition to off-site dose assessments, would require continual assessments of radioactive inventory and containment integrity. Alternatively, if both Fuel Clad and RCS Barrier "Potential Loss" thresholds existed, the ED would have more assurance that there was no immediate need to escalate to a General Emergency.
- The ability to escalate to higher emergency classification levels as an event deteriorates must be maintained. For example, RCS leakage steadily increasing would represent an increasing risk to public health and safety.
- The Containment Barrier should not be declared lost or potentially lost based on exceeding Technical Specification action statement criteria, unless there is an event in progress requiring mitigation by the Containment barrier.

Category: Fission Product Barrier Degradation

Subcategory: N/A

Initiating Condition: Loss of **ANY** two barriers and loss or potential loss of the third barrier

EAL:

FG1.1	**General Emergency**
Loss of **ANY** two fission product barriers **AND** Loss or potential loss of third fission product barrier (Table F-1)	

Mode Applicability:

1 - Power Operation, 2 - Startup, 3 - Hot Shutdown

Basis:

Plant-Specific

Fuel Clad, RCS and Containment comprise the fission product barriers. Table F-1 (Attachment 2) lists the fission product barrier thresholds, bases and references.

At the General Emergency classification level each barrier is weighted equally. A General Emergency is therefore appropriate for any combination of the following conditions:

- Loss of Fuel Clad, RCS and Containment barriers
- Loss of Fuel Clad and RCS barriers with potential loss of Containment barrier
- Loss of RCS and Containment barriers with potential loss of Fuel Clad barrier
- Loss of Fuel Clad and Containment barriers with potential loss of RCS barrier

Generic

None

NMP2 Basis Reference(s):

1. NEI 99-01 IC FG1

Category: Fission Product Barrier Degradation

Subcategory: N/A

Initiating Condition: Loss or potential loss of **ANY** two barriers

EAL:

FS1.1	**Site Area Emergency**
Loss or potential loss of **ANY** two fission product barriers (Table F-1)	

Mode Applicability:

1 - Power Operation, 2 - Startup, 3 - Hot Shutdown

Basis:

Plant-Specific

Fuel Clad, RCS and Containment comprise the fission product barriers. Table F-1 (Attachment 2) lists the fission product barrier thresholds, bases and references.

At the Site Area Emergency classification level, each barrier is weighted equally. A Site Area Emergency is therefore appropriate for any combination of the following conditions:

- One barrier loss and a second barrier loss (i.e., loss - loss)
- One barrier loss and a second barrier potential loss (i.e., loss - potential loss)
- One barrier potential loss and a second barrier potential loss (i.e., potential loss - potential loss)

At the Site Area Emergency classification level, the ability to dynamically assess the proximity of present conditions with respect to the threshold for a General Emergency is important. For example, the existence of Fuel Clad and RCS Barrier loss thresholds in addition to offsite dose assessments would require continual assessments of radioactive inventory and Containment integrity in anticipation of reaching a General Emergency classification. Alternatively, if both Fuel Clad and RCS potential loss thresholds existed, the Emergency Director would have greater assurance that escalation to a General Emergency is less IMMINENT.

Generic

None

NMP2 Basis Reference(s):

1. NEI 99-01 IC FS1

Category: Fission Product Barrier Degradation

Subcategory: N/A

Initiating Condition: **ANY** loss or **ANY** potential loss of **EITHER** Fuel Clad **OR** RCS

EAL:

FA1.1	**Alert**
ANY loss or **ANY** potential loss of **EITHER** Fuel Clad barrier **OR** RCS barrier (Table F-1)	

Mode Applicability:

1 - Power Operation, 2 - Startup, 3 - Hot Shutdown

Basis:

Plant-Specific

Fuel Clad, RCS and Containment comprise the fission product barriers. Table F-1 (Attachment 2) lists the fission product barrier thresholds, bases and references.

At the Alert classification level, Fuel Clad and RCS barriers are weighted more heavily than the Containment barrier. Unlike the Containment barrier, loss or potential loss of either the Fuel Clad or RCS barrier may result in the relocation of radioactive materials or degradation of core cooling capability. Note that the loss or potential loss of Containment barrier in combination with loss or potential loss of either Fuel Clad or RCS barrier results in declaration of a Site Area Emergency under EAL FS1.

Generic

None

NMP2 Basis Reference(s):

1. NEI 99-01 IC FA1

Category: Fission Product Barrier Degradation

Subcategory: N/A

Initiating Condition: **ANY** loss or **ANY** potential loss of Containment

EAL:

FU1.1	**Unusual Event**
ANY loss or **ANY** potential loss of Containment barrier (Table F-1)	

Mode Applicability:

1 - Power Operation, 2 - Startup, 3 - Hot Shutdown

Basis:

Plant-Specific

Fuel Clad, RCS and Containment comprise the fission product barriers. Table F-1 (Attachment 2) lists the fission product barrier thresholds, bases and references.

Fuel Clad and RCS barriers are weighted more heavily than the Containment barrier. Unlike the Fuel Clad and RCS barriers, the loss of either of which results in an Alert (EAL FA1.1), loss of the Containment barrier in and of itself does not result in the relocation of radioactive materials or the potential for degradation of core cooling capability. However, loss or potential loss of the Containment barrier in combination with the loss or potential loss of either the Fuel Clad or RCS barrier results in declaration of a Site Area Emergency under EAL FS1.1.

Generic

None

NMP2 Basis Reference(s):

1. NEI 99-01 IC FU1

Introduction

Table F-1 lists the threshold conditions that define the Loss and Potential Loss of the three fission product barriers (Fuel Clad, Reactor Coolant System, and Containment). The table is structured so that each of the three barriers occupies adjacent columns. Each fission product barrier column is further divided into two columns; one for Loss thresholds and one for Potential Loss thresholds.

The first column of the table (to the left of the Fuel Clad Loss column) lists the categories (types) of fission product barrier thresholds. The fission product barrier categories are:

A. RPV Level

B. Primary Containment Pressure / Temperature

C. Isolation

D. Rad

E. Judgment

Each category occupies a row in Table F-1 thus forming a matrix defined by the categories. The intersection of each row with each Loss/Potential Loss column forms a cell in which one or more fission product barrier thresholds appear. If NEI 99-01 does not define a threshold for a barrier Loss/Potential Loss, the word "None" is entered in the cell.

Thresholds are assigned sequential numbers within each Loss and Potential Loss column beginning with number one. In this manner, a threshold can be identified by its category title and number. For example, the first Fuel Clad barrier Loss in Category A would be assigned "FC Loss A.1," the third Containment barrier Potential Loss would be assigned "PC P-Loss B.3," etc.

If a cell in Table F-1 contains more than one numbered threshold, each of the numbered thresholds, if exceeded, signifies a Loss or Potential Loss of the barrier. It is not necessary to exceed all of the thresholds in a category before declaring a barrier Loss/Potential Loss.

Subdivision of Table F-1 by category facilitates association of plant conditions to the applicable fission product barrier Loss and Potential Loss thresholds. This structure promotes a systematic approach to assessing the classification status of the fission product barriers.

When equipped with knowledge of plant conditions related to the fission product barriers, the EAL-user first scans down the category column of Table F-1, locates the likely category and then reads across the fission product barrier Loss and Potential Loss thresholds in that category to determine if a threshold has been exceeded. If a threshold has not been exceeded, the EAL-user proceeds to the next likely category and continues review of the thresholds in the new category

If the EAL-user determines that any threshold has been exceeded, by definition, the barrier is lost or potentially lost – even if multiple thresholds in the same barrier column are exceeded; only that one barrier is lost or potentially lost. The EAL-user must examine each of the three fission product barriers to determine if other barrier thresholds in the category are lost or potentially lost. For example, if Primary Containment radiation is sufficiently high, a Loss of the Fuel Clad and RCS barriers and a Potential Loss of the Containment barrier can occur. Barrier Losses and Potential Losses are then applied to the algorithms given in EALs FG1.1, FS1.1, FA1.1 and FU1.1 to determine the appropriate emergency classification.

In the remainder of this Attachment, the Fuel Clad barrier threshold bases appear first, followed by the RCS barrier and finally the Containment barrier threshold bases. In each barrier, the bases are given according to category Loss followed by category Potential Loss beginning with Category A, then B,..., E.

Attachment 2 - Fission Product Barrier Loss / Potential Loss Matrix and Basis

Table F-1 Fission Product Barrier Matrix

	Fuel Clad Barrier		Reactor Coolant System Barrier		Containment Barrier	
Category	Loss	Potential Loss	Loss	Potential Loss	Loss	Potential Loss
A RPV Level	1. Primary Containment Flooding is required	1. RPV water level **cannot** be restored and maintained above -14 in. following depressurization of the RPV or RPV water level **cannot** be determined	1. RPV water level **cannot** be restored and maintained above -14 in. or RPV water level **cannot** be determined	None	None	1. Primary Containment Flooding is required
B Primary Containment Pressure / Temp.	None	None	2. Primary Containment pressure > 1.68 psig due to RCS leakage	None	1. Primary Containment pressure rise followed by a rapid UNPLANNED drop in Primary Containment pressure 2. Primary Containment pressure response **not** consistent with LOCA conditions	2. Primary Containment pressure > 45 psig and rising 3. Explosive mixture exists inside Primary Containment ($\geq$ 6% H_2 and $\geq$ 5% O_2) 4. Suppression pool temperature and RPV pressure cannot be maintained below the Heat Capacity Temperature Limit (N2-EOP-PC Figure M)
C Isolation	None	None	3. Release pathway exists outside Primary Containment resulting from isolation failure in **ANY** of the following (excluding normal process system flowpaths from an UNISOLABLE system): • Main steam line • RCIC steam line • RWCU • Feedwater 4. RPV blowdown is required	1. UNISOLABLE primary system leakage outside Primary Containment as indicated by exceeding **EITHER**: RB area temperature above an isolation setpoint **OR** RB area radiation above an alarm setpoint	3. Failure of **all** Primary Containment isolation valves in **ANY** one line to close following auto or manual initiation **AND** Direct downstream pathway outside Primary Containment and to the environment exists 4. Intentional Primary Containment venting per EOPs 5. UNISOLABLE primary system leakage outside Primary Containment as indicated by exceeding **EITHER**: RB area maximum safe temperature value (N2-EOP-SC Detail S) **OR** RB area radiation > 8.00E+3 mR/hr	None
D Rad	2. Drywell area radiation $\geq$ 3100 R/hr (3.1 E6 mRem/hr) 3. Reactor coolant activity > 300 μCi/gm I-131 Equivalent	None	5. Drywell area radiation $\geq$ 41 R/hr (4.1 E4 mRem/hr)	None	None	5. Drywell area radiation $\geq$ 6.0 E4 R/hr (6.0 E7 mRem/hr)
E Judgment	4. **ANY** condition in the opinion of the Emergency Director that indicates loss of the Fuel Clad barrier	2. **ANY** condition in the opinion of the Emergency Director that indicates potential loss of the Fuel Clad barrier	6. **ANY** condition in the opinion of the Emergency Director that indicates loss of the Reactor Coolant System barrier	2. **ANY** condition in the opinion of the Emergency Director that indicates potential loss of the Reactor Coolant System barrier	6. **ANY** condition in the opinion of the Emergency Director that indicates loss of the Containment barrier	6. **ANY** condition in the opinion of the Emergency Director that indicates potential loss of the Containment barrier

Barrier: Fuel Clad
Category: A. RPV Water Level
Degradation Threat: Loss
Threshold:

1. Primary Containment Flooding is required

Basis:

Plant-Specific

Requirements for Primary Containment Flooding are established in EOP-RPV Step L-I6; EOP-C5 Steps L-8, L-I0 and L-I8; and EOP-C4 Override 1. These EOPs provide instructions to ensure adequate core cooling by maintaining RPV water level above prescribed limits or operating sufficient RPV injection sources when level cannot be determined. SAP entry is required when (ref. 1):

- RPV water level cannot be restored and maintained above -39 in. with insufficient Core Spray Cooling: The Minimum Steam Cooling RPV Water Level (MSCRWL) is the lowest RPV water level at which the covered portion of the reactor core will generate sufficient steam to preclude any clad temperature in the uncovered portion of the core from exceeding 1500°F. Core Spray Cooling is insufficient if RPV water level cannot be restored and maintained at or above -62 in. with at least 6350 gpm core spray loop flow. Consistent with the EOP definition of "cannot be restored and maintained," the determination that the parameter cannot be restored and maintained above the limit may be made at, before, or after the parameter actually decreases to this point.
- RPV water level cannot be determined and it is determined that core damage is occurring: When RPV water level cannot be determined, EOPs require RPV flooding strategies. RPV water level indication provides the primary means of knowing if adequate core cooling is being maintained. When all means of determining RPV water level are unavailable, reliance on alternate means of assuring adequate core cooling must be attempted. The instructions in EOP-C4 specify these means, which include emergency depressurization of the RPV and

injection into the RPV at a rate needed to flood to the elevation of the main steam lines or hold RPV pressure above the Minimum Steam Cooling Pressure (in ATWS events)

This threshold is also a Potential Loss of the Containment barrier (PC P-Loss A.1). Since SAP entry occurs after core uncovery has occurred, a Loss of the RCS barrier exists (RCS Loss A.1). Primary Containment Flooding (SAP entry), therefore, represents a Loss of two barriers and a Potential Loss of a third, which requires a General Emergency classification.

<u>Generic</u>

This site specific value corresponds to the level used in EOPs to indicate challenge of core cooling. This is the minimum value to assure core cooling without further degradation of the clad.

NMP2 Basis Reference(s):

1. NER-2M-039, NMP2 Emergency Operating Procedures (EOP) Basis Document
2. N2-EOP-C4 RPV Flooding
3. NEI 99-01 FC Loss 2

Barrier: Fuel Clad

Category: B. Primary Containment Pressure / Temperature

Degradation Threat: Loss

Threshold:

None

Barrier: Fuel Clad

Category: C. Isolation

Degradation Threat: Loss

Threshold:

None

Barrier: Fuel Clad

Category: D. Rad

Degradation Threat: Loss

Threshold:

2. Drywell area radiation ≥ 3100 R/hr (3.1 E6 mRem/hr)

Basis:

Plant-Specific

It is important to recognize that the radiation monitor may be sensitive to shine from the RPV or RCS piping (caused by lower than normal RPV water level for example). The Drywell High Range Radiation Monitors are the following (ref. 1):

- 2CEC*PNL880D: DRMS 2RMS*RE1B/D

  RMS*RUZ1A

  RMS*RUZ1B

- 2CEC*PNL880B: DRMS 2RMS*RE1A/C

  RMS*RUZ1C

  RMS*RUZ1D

Figure F-1 illustrates the location of the following four detectors inside the drywell (ref. 1):

- 2RMS*RE1A P.C. 268 170EAZ
- 2RMS*RE1C P.C. 267 024EAZ
- 2RMS*RE1B P.C. 268 245EAZ
- 2RMS*RE1D P.C. 268 353EAZ

The threshold value was calculated assuming the instantaneous release and dispersal of the reactor coolant noble gas and iodine inventory associated with a concentration of 300 μCi/gm I-131 Equivalent (or approximately 5% clad failure) into the drywell atmosphere (ref. 2).

Generic

The 3100 R/hr (3.1 E6 mRem/hr) reading is a value which indicates the release of reactor coolant, with elevated activity indicative of fuel damage, into the drywell.

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 threshold D.5. Thus, this threshold indicates a loss of both Fuel Clad barrier and RCS barrier that appropriately escalates the emergency classification level to a Site Area Emergency.

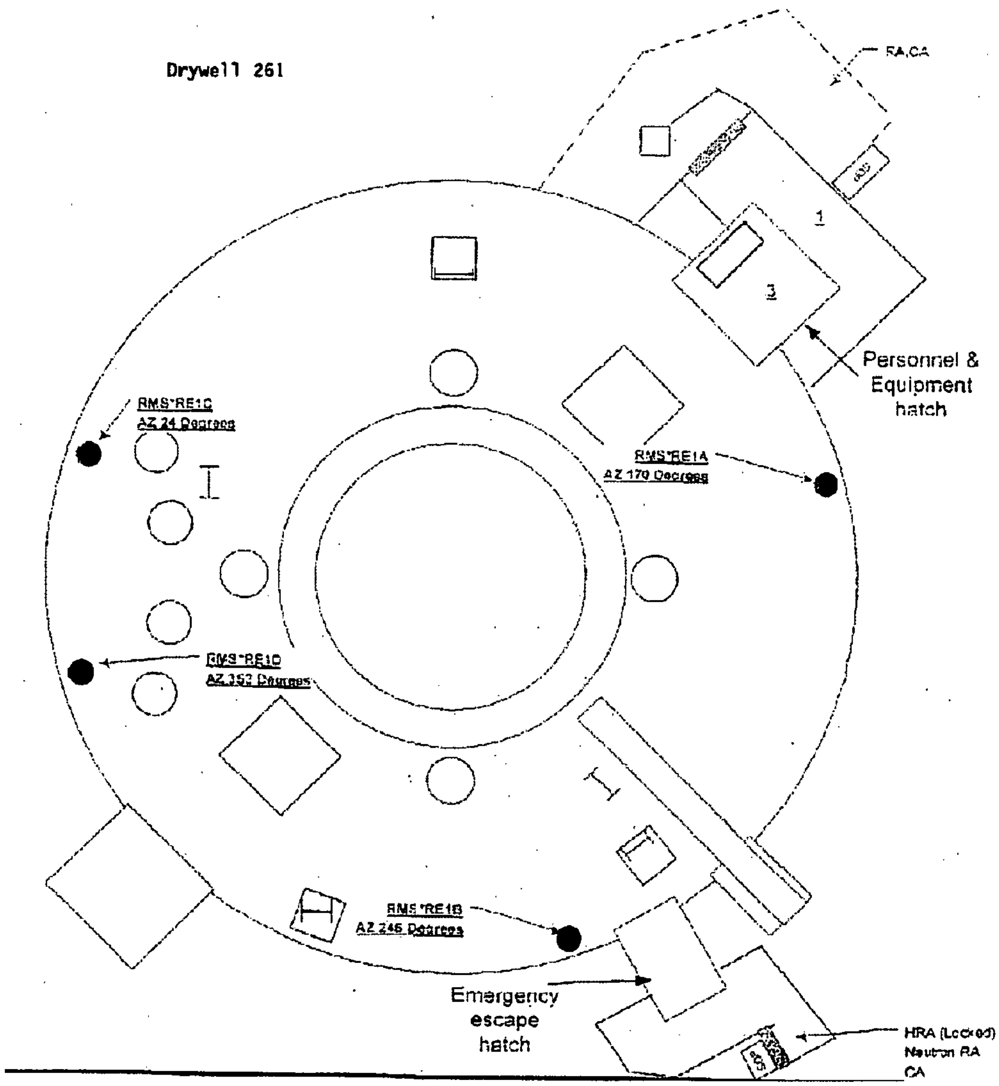
There is no Potential Loss threshold associated with this item.

NMP2 Basis Reference(s):

1. N2-RSP-RMS-R106 Channel Calibration Test of the Drywell High Range Area Radiation Monitors
2. Calculation PR-C-24-0
3. NEI 99-01 FC Loss 4

Figure F-1: Drywell High Range Radiation Monitor Detector Locations (ref. 1)

Drywell 261



Barrier: Fuel Clad

Category: D. Rad

Degradation Threat: Loss

Threshold:

3. Reactor coolant activity > 300 μCi/gm I-131 Equivalent

Basis:

Plant-Specific

None

Generic

The site specific value corresponds to 300 μCi/gm I-131 Equivalent. Assessment by the EAL Task Force indicates that 300 μCi/gm I-131 Equivalent coolant activity is well above that expected for iodine spikes and corresponds to less than 5% fuel clad damage. This amount of radioactivity indicates significant clad damage and thus the Fuel Clad Barrier is considered lost.

There is no Potential Loss threshold associated with this item.

NMP2 Basis Reference(s):

1. General Electric NEDO-22215, Procedures for the Determination of the Extent of Core Damage Under Accident Conditions
2. NEI 99-01 FC Loss 1

Barrier: Fuel Clad
Category: E. Judgment
Degradation Threat: Loss
Threshold:

4. **ANY** condition in the opinion of the Emergency Director that indicates loss of the Fuel Clad barrier

Basis:

Plant-Specific

The Emergency Director judgment threshold addresses any other factors relevant to determining if the Fuel Clad barrier is lost. Such a determination should include IMMINENT barrier degradation, barrier monitoring capability and dominant accident sequences.

- IMMINENT barrier degradation exists if the degradation will likely occur within two hours based on a projection of current safety system performance. The term "IMMINENT" refers to recognition of the inability to reach safety acceptance criteria before completion of all checks.
- Barrier monitoring capability is decreased if there is a loss or lack of reliable indicators. This assessment should include instrumentation operability concerns, readings from portable instrumentation and consideration of offsite monitoring results.
- Dominant accident sequences lead to degradation of all fission product barriers and likely entry to the EOPs. The Emergency Director should be mindful of the Loss of AC power (Station Blackout) and ATWS EALs to assure timely emergency classification declarations.

Generic

This threshold addresses any other factors that are to be used by the Emergency Director in determining whether the Fuel Clad barrier is lost. In addition, the inability to monitor the barrier should also be incorporated in this threshold as a factor in Emergency Director judgment that the barrier may be considered lost.

NMP2 Basis Reference(s):

1. NEI 99-01 FC Loss 6

Barrier: Fuel Clad

Category: A. RPV Level

Degradation Threat: Potential Loss

Threshold:

1. RPV water level **cannot** be restored and maintained above -14 in. following depressurization of the RPV or **cannot** be determined

Basis:

Plant-Specific

An RPV water level instrument reading of -14 in. indicates RPV water level is at the top of active fuel. When RPV water level is at or above the top of active fuel, the core is completely submerged. Core submergence is the most desirable means of core cooling. When RPV water level is below the top of active fuel following depressurization of the RPV (automatically, manually or by failure of the RCS barrier), the uncovered portion of the core must be cooled by less reliable means (i.e., spray cooling). If core uncovery is threatened, the EOPs specify alternate, more extreme, RPV water level control measures in order to restore and maintain adequate core cooling (ref. 1).

Consistent with the EOP definition of "cannot be restored and maintained," the determination that RPV water level cannot be restored and maintained above the top of active fuel may be made at, before, or after RPV water level actually decreases to this point. (ref. 1)

When RPV water level cannot be determined, EOPs require RPV flooding strategies. RPV water level indication provides the primary means of knowing if adequate core cooling is being maintained. When all means of determining RPV water level are unavailable, the fuel clad barrier is threatened and reliance on alternate means of assuring adequate core cooling must be attempted. The instructions in EOP-C4 specify these means, which include emergency depressurization of the RPV and injection into the RPV at a rate needed to flood to the elevation of the main steam lines or hold RPV pressure above the Minimum Steam Cooling Pressure (in ATWS events). (ref. 2) If RPV water level cannot be determined with respect to the top of active fuel, a potential loss of the fuel clad barrier exists.

Note that EOP-C5 may require intentional uncovery of the core and control of RPV water level between -14 in. and -39 in., the Minimum Steam Cooling RPV Water Level (MSCRWL) (ref. 3). Under these conditions, a high-power ATWS event exists and requires at least a Site Area Emergency classification in accordance with the ATWS/Criticality EALs.

Generic

The site specific RPV water level threshold is the same as the RCS barrier Loss threshold A.1 and corresponds to the RPV water level at the top of the active fuel. Thus, this threshold indicates a Potential Loss of the Fuel Clad barrier and a Loss of RCS barrier that appropriately escalates the emergency classification level to a Site Area Emergency. This threshold is considered to be exceeded when, as specified in the site specific EOPs, that RPV water cannot be restored and maintained above the specified level following depressurization of the RPV (either manually, automatically or by failure of the RCS barrier).

NMP2 Basis Reference(s):

1. NER-2M-039, NMP2 Emergency Operating Procedures (EOP) Basis Document
2. N2-EOP-C4 RPV Flooding
3. N2-EOP-C5 Failure to Scram
4. NEI 99-01 FC Potential Loss 2

Barrier: Fuel Clad

Category: B. Primary Containment Pressure / Temperature

Degradation Threat: Potential Loss

Threshold:

None

Barrier: Fuel Clad

Category: C. Isolation

Degradation Threat: Potential Loss

Threshold:

None

Barrier: Fuel Clad

Category: D. Rad

Degradation Threat: Potential Loss

Threshold:

None

Barrier: Fuel Clad

Category: E. Judgment

Degradation Threat: Potential Loss

Threshold:

2. **ANY** condition in the opinion of the Emergency Director that indicates potential loss of the Fuel Clad barrier

Basis:

Plant-Specific

The Emergency Director judgment threshold addresses any other factors relevant to determining if the Fuel Clad barrier is potentially lost. Such a determination should include IMMINENT barrier degradation, barrier monitoring capability and dominant accident sequences.

- IMMINENT barrier degradation exists if the degradation will likely occur within two hours based on a projection of current safety system performance. The term "IMMINENT" refers to recognition of the inability to reach safety acceptance criteria before completion of all checks.
- Barrier monitoring capability is decreased if there is a loss or lack of reliable indicators. This assessment should include instrumentation operability concerns, readings from portable instrumentation and consideration of offsite monitoring results.
- Dominant accident sequences lead to degradation of all fission product barriers and likely entry to the EOPs. The Emergency Director should be mindful of the Loss of AC power (Station Blackout) and ATWS EALs to assure timely emergency classification declarations.

Generic

This threshold addresses any other factors that are to be used by the Emergency Director in determining whether the Fuel Clad barrier is potentially lost. In addition, the inability to monitor the barrier should also be incorporated in this threshold as a factor in Emergency Director judgment that the barrier may be considered potentially lost.

NMP2 Basis Reference(s):

1. NEI 99-01 FC Potential Loss 6

Barrier: Reactor Coolant System

Category: A. RPV Level

Degradation Threat: Loss

Threshold:

1. RPV water level **cannot** be restored and maintained above -14 in. or **cannot** be determined

Basis:

Plant-Specific

An RPV water level instrument reading of -14 in. indicates RPV water level is at the top of active fuel (ref. 1). The top of the active fuel is significantly lower than the normal operating RPV water level control band. To reach this level, RPV inventory loss would have previously required isolation of the RCS and Containment (PC) barriers, and initiation of all ECCS. If RPV water level cannot be maintained above the top of active fuel, ECCS and other sources of RPV injection have been ineffective or incapable of reversing the decreasing level trend. The cause of the loss of RPV inventory is therefore assumed to be a Loss of Coolant Accident (LOCA). By definition, a LOCA event is a Loss of the RCS barrier.

Consistent with the EOP definition of "cannot be restored and maintained," the determination that RPV water level cannot be restored and maintained above the top of active fuel may be made at, before, or after RPV water level actually decreases to this point. (ref. 1)

When RPV level cannot be determined, EOPs require RPV flooding strategies. The RPV flooding instructions in EOP-C4 first specify emergency depressurization of the RPV (ref. 2), which is defined to be a Loss of the RCS barrier (RCS Loss C.4).

Note that EOP-C5 may require intentional uncovery of the core and control of RPV water level between -14 in. and -39 in., the Minimum Steam Cooling RPV Water Level (MSCRWL) (ref. 3). Under these conditions, a high-power ATWS event exists and requires at least a Site Area Emergency classification in accordance with the ATWS/Criticality

EALs.

Generic

The Loss threshold RPV water level of 161 in. corresponds to the level that is used in EOPs to indicate challenge of core cooling.

This threshold is the same as Fuel Clad Barrier Potential Loss threshold A.1 and corresponds to a challenge to core cooling. Thus, this threshold indicates a Loss of RCS barrier and Potential Loss of Fuel Clad barrier that appropriately escalates the emergency classification level to a Site Area Emergency.

Unlike the Fuel Clad barrier RPV water level Potential Loss threshold (top of the active fuel), the additional requirement that the RPV be depressurized is not associated with the RCS barrier Potential Loss. The significant loss of inventory that must occur to determine that RPV water level cannot be restored and maintained above the threshold is, by itself, a very strong indication that the RCS barrier is no longer capable of retaining sufficient inventory to keep the core submerged, and thus represents a Loss of the RCS Barrier.

There is no Potential Loss threshold associated with this item.

NMP2 Basis Reference(s):

1. NER-2M-039, NMP2 Emergency Operating Procedures (EOP) Basis Document
2. N2-EOP-C4 RPV Flooding
3. N2-EOP-C5 Failure to Scram
4. NEI 99-01 RCS Loss 2

Barrier: Reactor Coolant System

Category: B. Primary Containment Pressure / Temperature

Degradation Threat: Loss

Threshold:

2. Primary Containment pressure > 1.68 psig due to RCS leakage

Basis:

Plant-Specific

The drywell high pressure scram setpoint is an entry condition to the EOP flowcharts: EOP-RPV, RPV Control, and EOP-PC, Primary Containment Control (ref. 1, 2). Normal Primary Containment (PC) pressure control functions such as operation of drywell cooling and venting through GTS are specified in EOP-PC in advance of less desirable but more effective functions such as operation of drywell or suppression chamber sprays.

In the NMP2 design basis, Primary Containment pressures above the drywell high pressure scram setpoint are assumed to be the result of a high-energy release into the containment for which normal pressure control systems are inadequate or incapable of reversing the increasing pressure trend. Pressures of this magnitude, however, can be caused by non-LOCA events such as a loss of drywell cooling or inability to control Primary Containment vent/purge (ref. 3, 4).

The threshold phrase "...due to RCS leakage" focuses the barrier failure on the RCS instead of the non-LOCA malfunctions that may adversely affect Primary Containment pressure. Primary Containment pressure greater than 1.68 psig with corollary indications (e.g., elevated drywell temperature, indications of loss of RCS inventory) should, therefore, be considered a Loss of the RCS barrier. Loss of drywell cooling that results in pressure greater than 1.68 psig should not be considered an RCS barrier loss.

Generic

The Primary Containment pressure of 1.68 psig is based on the drywell high pressure set point which indicates a LOCA by automatically initiating the ECCS or equivalent makeup system.

There is no Potential Loss threshold associated with this item.

NMP2 Basis Reference(s):

1. N2-EOP RPV RPV Control
2. N2-EOP-PC Primary Containment Control
3. NER-2M-039, NMP2 Emergency Operating Procedures (EOP) Basis Document
4. USAR Section 6.2
5. NEI 99-01 RCS Loss 1

Barrier: Reactor Coolant System

Category: C. Isolation

Degradation Threat: Loss

Threshold:

3. Release pathway exists outside Primary Containment resulting from isolation failure in **ANY** of the following systems (excluding normal process system flowpaths from an UNISOLABLE system):
   - Main steam line
   - RCIC steam line
   - RWCU
   - Feedwater

Basis:

Plant-Specific

The conditions of this threshold include required containment isolation failures allowing a flow path to the environment. A release pathway outside Primary Containment exists when flow is not prevented by downstream isolations. Emergency declaration under this threshold would not be required in the case of a failure of both isolation valves to close but no downstream flowpath exists. Similarly, if the emergency response requires the normal process flow of a system outside Primary Containment (e.g., EOP requirement to bypass MSIV low RPV water level interlocks and maintain the main condenser as a heat sink using main turbine bypass valves), the threshold is not met. The combination of these threshold conditions represent the loss of both the RCS and Containment (see PC Loss C.3) barriers and justifies declaration of a Site Area Emergency (i.e., Loss or Potential Loss of any two barriers). (ref. 1-4)

Even though RWCU and Feedwater systems do not contain steam, they are included in the list because an UNISOLABLE break could result in the high-pressure discharge of fluid that is flashed to steam from relatively large volume systems directly connected to the RCS.

Generic

An UNISOLABLE MSL break is a breach of the RCS barrier. Thus, this threshold is included for consistency with the Alert emergency classification level.

Other large high-energy line breaks such as Feedwater, RWCU, or RCIC that are UNISOLABLE also represent a significant loss of the RCS barrier and should be considered as MSL breaks for purposes of classification.

NMP2 Basis Reference(s):

1. USAR Section 5.4.5
2. USAR Section 5.4.6
3. USAR Section 5.4.8
4. USAR Section 5.4.9
5. NEI 99-01 RCS Loss 3A

Barrier: Reactor Coolant System
Category: C. Isolation
Degradation Threat: Loss
Threshold:

4. RPV blowdown is required

Basis:

Plant-Specific

RPV blowdown (Emergency RPV Depressurization) is specified in the EOP flowcharts when symbols containing the phrase “BLOW DOWN” are reached. The requirements for emergency RPV depressurization appear in the following EOPs (ref. 1-7):

- EOP-RPV RPV Control
- EOP-PC Primary Containment Control
- EOP-SC Secondary Containment Control
- EOP-RR Radioactivity Release Control
- EOP-PCH Hydrogen Control
- EOP-C3 Steam Cooling
- EOP-C5 Failure to Scram

RPV blowdown (Emergency RPV Depressurization) is also performed upon entry to EOP-C4 (ref. 8).

Generic

Plant symptoms requiring Emergency RPV Depressurization (RPV blowdown) per the EOP flowcharts are indicative of a loss of the RCS barrier. If Emergency RPV depressurization is required, the plant operators are directed to open safety relief valves (SRVs) and keep them open. Even though the RCS is being vented into the suppression pool, a loss of the RCS should be considered to exist due to the diminished effectiveness of the RCS pressure barrier to a release of fission products beyond its boundary.

NMP2 Basis Reference(s):

1. N2-EOP-RPV RPV Control
2. N2-EOP-PC primary Containment Control
3. N2-EOP-SC Secondary Containment Control
4. N2-EOP-RR Radioactivity Release Control
5. N2-EOP-PCH Hydrogen Control
6. N2-EOP-C3 Steam Cooling
7. N2-EOP-C5 Failure to Scram
8. N2-EOP-C4 RPV Flooding
9. NEI 99-01 RCS Loss 3

Barrier: Reactor Coolant System

Category: D. Rad

Degradation Threat: Loss

Threshold:

5. Drywell area radiation ≥ 41 R/hr (4.1 E4 mRem/hr)

Basis:

Plant-Specific

It is important to recognize that the radiation monitor may be sensitive to shine from the RPV or RCS piping (caused by lower than normal RPV water level for example). The Drywell High Range Radiation Monitors are the following (ref. 1):

- 2CEC*PNL880D: DRMS 2RMS*RE1B/D

  RMS*RUZ1A

  RMS*RUZ1B

- 2CEC*PNL880B: DRMS 2RMS*RE1A/C

  RMS*RUZ1C

  RMS*RUZ1D

Figure F-1 illustrates the location of the following four detectors inside the drywell (ref. 1):

- 2RMS*RE1A P.C. 268 170EAZ
- 2RMS*RE1C P.C. 267 024EAZ
- 2RMS*RE1B P.C. 268 245EAZ
- 2RMS*RE1D P.C. 268 353EAZ

The threshold value was calculated assuming the instantaneous release and dispersal of the reactor coolant noble gas and iodine inventory associated with normal operating concentrations (i.e., within Technical Specifications) into the drywell atmosphere (ref. 2). The reading is less than that specified for the Fuel Clad Loss because no damage to the fuel clad is assumed in this RCS Loss. Only leakage from the RCS is assumed in this EAL.

Generic

The 41 R/hr reading is a value which indicates the release of reactor coolant to the Primary Containment.

This reading will be less than that specified for Fuel Clad barrier Loss threshold D.2. Thus, this threshold would be indicative of a RCS leak only. If the radiation monitor reading increased to that value specified by Fuel Clad Barrier threshold, fuel damage would also be indicated.

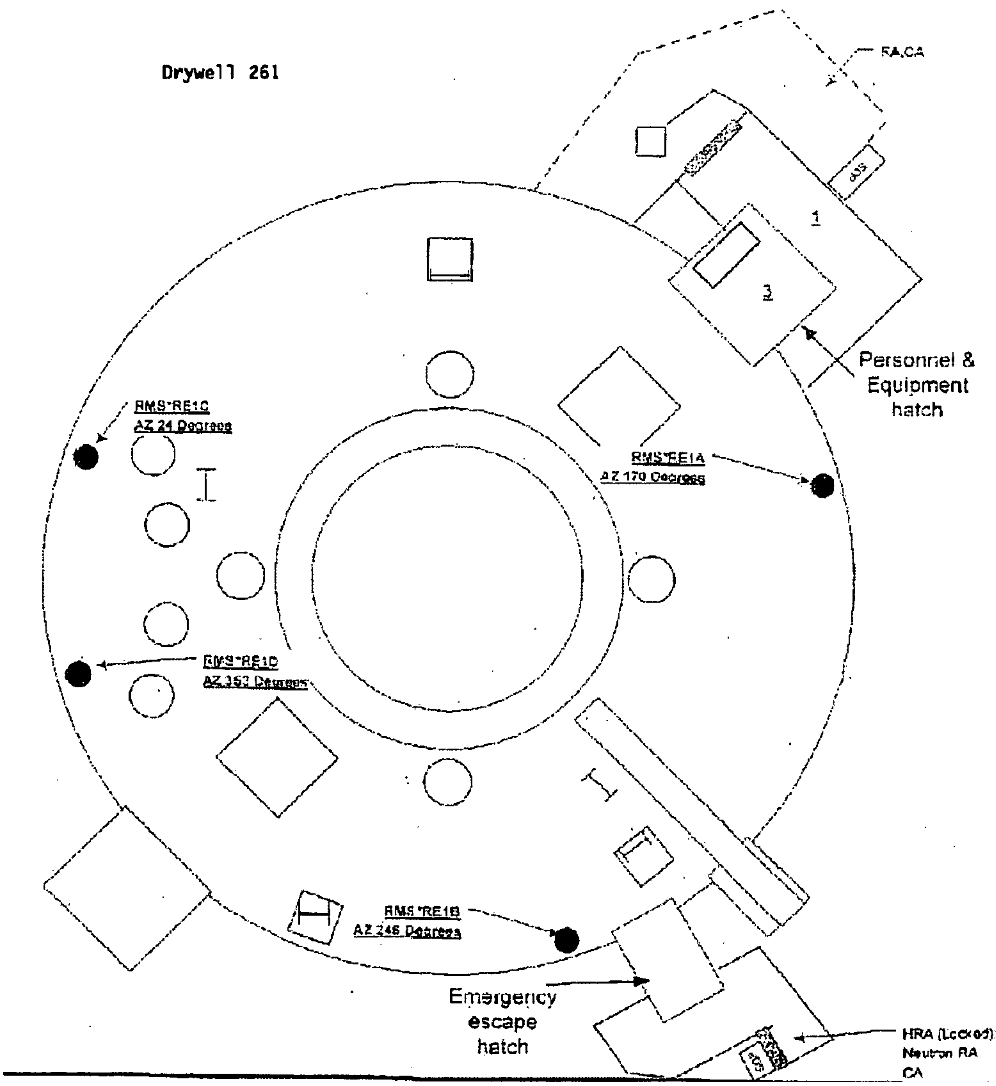
There is no Potential Loss threshold associated with this item.

NMP2 Basis Reference(s):

1. N2-RSP-RMS-R106 Channel Calibration Test of the Drywell High Range Area Radiation Monitors
2. Calculation PR-C-24-0
3. NEI 99-01 RCS Loss 4

Figure F-1: Drywell High Range Radiation Monitor Detector Locations (ref. 1)

Drywell 261



Barrier: Reactor Coolant System

Category: E. Judgment

Degradation Threat: Loss

Threshold:

6. **ANY** condition in the opinion of the Emergency Director that indicates loss of the RCS barrier

Basis:

Plant-Specific

The Emergency Director judgment threshold addresses any other factors relevant to determining if the RCS barrier is lost. Such a determination should include IMMINENT barrier degradation, barrier monitoring capability and dominant accident sequences.

- IMMINENT barrier degradation exists if the degradation will likely occur within two hours based on a projection of current safety system performance. The term "IMMINENT" refers to the recognition of the inability to reach safety acceptance criteria before completion of all checks.
- Barrier monitoring capability is decreased if there is a loss or lack of reliable indicators. This assessment should include instrumentation operability concerns, readings from portable instrumentation and consideration of offsite monitoring results.
- Dominant accident sequences lead to degradation of all fission product barriers and likely entry to the EOPs. The Emergency Director should be mindful of the Loss of AC power (Station Blackout) and ATWS EALs to assure timely emergency classification declarations.

Generic

This threshold addresses any other factors that are to be used by the Emergency Director in determining whether the RCS barrier is lost. In addition, the inability to monitor the barrier should also be incorporated in this threshold as a factor in Emergency Director judgment that the barrier may be considered lost.

NMP2 Basis Reference(s):

1. NEI 99-01 RCS Loss 6

Barrier: Reactor Coolant System

Category: A. RPV Level

Degradation Threat: Potential Loss

Threshold:

None

Barrier: Reactor Coolant System

Category: B. Primary Containment Pressure / Temperature

Degradation Threat: Potential Loss

Threshold:

None

Barrier: Reactor Coolant System

Category: C. Isolation

Degradation Threat: Potential Loss

Threshold:

1. UNISOLABLE primary system leakage outside Primary Containment as indicated by exceeding **EITHER**:

   RB area temperature above an isolation setpoint

   OR

   RB area radiation above an alarm setpoint

Basis:

Plant-Specific

The presence of elevated general area temperatures or radiation levels in the Reactor Building (RB) may be indicative of UNISOLABLE primary system leakage outside the Primary Containment. When parameters reach the threshold level, equipment failure or misoperation may be occurring. Elevated parameters may also adversely affect the ability to gain access to or operate equipment within the affected area. (ref. 1, 2)

In general, multiple indications should be used to determine if a primary system is discharging outside Primary Containment. For example, a high area radiation condition does not necessarily indicate that a primary system is discharging into the secondary containment since this may be caused by radiation shine from nearby steam lines or the movement of radioactive materials. Conversely, a high area radiation condition in conjunction with other indications (e.g. room flooding, high area temperatures, reports of steam in the secondary containment, an unexpected rise in feedwater flowrate, or unexpected main turbine control valve closure) may indicate that a primary system is discharging into the secondary containment.

Generic

EOP-SC temperature isolation setpoints or area radiation alarm setpoints in the areas of the main steam line tunnel, main turbine generator, RCIC, etc., indicate a direct path from the RCS to areas outside Primary Containment.

The indicators reaching the threshold barriers and confirmed to be caused by RCS leakage warrant an Alert classification. An UNISOLABLE leak which is indicated by a high alarm setpoint escalates to a Site Area Emergency when combined with Containment Barrier Loss threshold C.5 (after a containment isolation) and a General Emergency when the Fuel Clad Barrier criteria is also exceeded.

NMP2 Basis Reference(s):

1. N2-EOP-SC Secondary Containment Control
2. NER-2M-039, NMP2 Emergency Operating Procedures (EOP) Basis Document
3. NEI 99-01 RCS Potential Loss 3B

Barrier: Reactor Coolant System

Category: D. Rad

Degradation Threat: Potential Loss

Threshold:

None

Barrier: Reactor Coolant System

Category: E. Judgment

Degradation Threat: Potential Loss

Threshold:

2. **ANY** condition in the opinion of the Emergency Director that indicates potential loss of the RCS barrier

Basis:

Plant-Specific

The Emergency Director judgment threshold addresses any other factors relevant to determining if the RCS barrier is potentially lost. Such a determination should include IMMINENT barrier degradation, barrier monitoring capability and dominant accident sequences.

- IMMINENT barrier degradation exists if the degradation will likely occur within two hours based on a projection of current safety system performance. The term "IMMINENT" refers to the inability to reach final safety acceptance criteria before completing all checks.
- Barrier monitoring capability is decreased if there is a loss or lack of reliable indicators. This assessment should include instrumentation operability concerns, readings from portable instrumentation and consideration of offsite monitoring results.
- Dominant accident sequences lead to degradation of all fission product barriers and likely entry to the EOPs. The Emergency Director should be mindful of the Loss of AC power (Station Blackout) and ATWS EALs to assure timely emergency classification declarations.

Generic

This threshold addresses any other factors that are to be used by the Emergency Director in determining whether the RCS barrier is potentially lost. In addition, the inability to monitor the barrier should also be incorporated in this threshold as a factor in Emergency Director judgment that the barrier may be considered potentially lost.

NMP2 Basis Reference(s):

1. NEI 99-01 RCS Potential Loss 6

Barrier: Containment

Category: A. RPV Level

Degradation Threat: Loss

Threshold:

None

Barrier: Containment

Category: B. Primary Containment Pressure / Temperature

Degradation Threat: Loss

Threshold:

1. Primary Containment pressure rise followed by a rapid UNPLANNED drop in Primary Containment pressure

Basis:

Plant-Specific

None

Generic

Rapid UNPLANNED loss of pressure (i.e., not attributable to drywell spray or condensation effects) following an initial pressure increase from a high energy line break indicates a loss of containment integrity. Primary Containment pressure should increase as a result of mass and energy release into containment from a LOCA. Thus, Primary Containment pressure not increasing under these conditions indicates a loss of containment integrity.

This indicator relies on operator recognition of an unexpected response for the condition and therefore does not have a specific value associated with it. The unexpected response is important because it is the indicator for a containment bypass condition.

NMP2 Basis Reference(s):

1. NEI 99-01 CMT Loss 1A

Barrier: Containment

Category: B. Primary Containment Pressure / Temperature

Degradation Threat: Loss

Threshold:

2. Primary Containment pressure response **not** consistent with LOCA conditions

Basis:

Plant-Specific

USAR Section 6.2.1 provides a summary of Primary Containment pressure response for several postulated accident conditions resulting in the release of RCS inventory to the containment. These accidents include:

- Rupture of a recirculation line
- Rupture of a main steam line
- Intermediate size liquid line rupture
- Small size steam line rupture

The containment response to the main steam line, intermediate liquid line and small size steam line breaks were bounded by the recirculation line break. (ref. 1)

USAR Figures 6.2-2 and 6.2-3 illustrate the containment pressure response due to a recirculation line break (ref. 2, 3). The maximum calculated drywell pressure is 39.75 psig and is well below the design allowable pressure of 45 psig. (ref. 4, 5)

Due to conservatisms in LOCA analyses, actual pressure response is expected to be less than the analyzed response. For example, blowdown mass flowrate may be only 60-80% of the analyzed rate, initial containment pressure may be less than 0.75 psig, etc.

LOCA conditions are manifested on Control Room instrumentation by drywell pressure rising with suppression chamber pressure following in a manner similar to that shown in USAR Figures 6.2-2 and 6.2-3. A broken SRV tailpipe could infer this threshold if suppression chamber pressure is higher than drywell pressure; however, if the SRV is closed, the condition would no longer exist.

Generic

Rapid UNPLANNED loss of pressure (i.e., not attributable to drywell spray or condensation effects) following an initial pressure increase from a high energy line break indicates a loss of containment integrity. Primary Containment pressure should increase as a result of mass and energy release into containment from a LOCA. Thus, Primary Containment pressure not increasing under these conditions indicates a loss of containment integrity.

This indicator relies on operator recognition of an unexpected response for the condition and therefore does not have a specific value associated with it. The unexpected response is important because it is the indicator for a containment bypass condition.

NMP2 Basis Reference(s):

1. USAR Section 6.2.1
2. USAR Figure 6.2-2
3. USAR Figure 6.2-3
4. USAR Table 6.2-18
5. USAR Section 6.2.1.1.2
6. NEI 99-01 CMT Loss 1B

Barrier: Containment

Category: C. Isolation

Degradation Threat: Loss

Threshold:

3. Failure of **all** Primary Containment isolation valves in **ANY** one line to close following auto or manual initiation **AND** Direct downstream pathway outside Primary Containment and to the environment exists

Basis:

Plant-Specific

This threshold addresses failure of open isolation devices which should close upon receipt of a manual or automatic Primary Containment isolation signal resulting in a significant radiological release pathway directly to the environment. The concern is the UNISOLABLE open pathway to the environment. A failure of the ability to isolate any one line indicates a breach of Primary Containment integrity. Technical Specifications Table 3.6.1.3-1 provides a list of applicable isolation valves (ref. 1).

As stated above, the adjective “Direct” modifies “pathway” to discriminate against release paths through interfacing liquid systems. Leakage into a closed system is to be considered only if the closed system is breached and thereby creates a significant pathway to the environment. Examples include UNISOLABLE Main steam line or RCIC steam line breaks, UNISOLABLE RWCU system breaks, and unisloable Primary Containment atmosphere vent paths. If the main condenser is available with an UNISOLABLE main steam line, there may be releases through the steam jet air ejectors and gland seal exhausters. These pathways are monitored, however, and do not meet the intent of a nonisolable release path to the environment. These minor releases are assessed using the Category R EALs.

The existence of an in–line charcoal filter (GTS) does not make a release path indirect since the filter is not effective at removing fission noble gases. Typical filters have an efficiency of 95-99% removal of iodine. Given the magnitude of the core inventory of iodine, significant releases could still occur. In addition, since the fission product release would be driven by boiling in the reactor vessel, the high humidity in the release stream can be expected to render the filters ineffective in a short period.

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

N2-EOP-PC, Primary Containment Control may specify Primary Containment venting and intentional bypassing of the containment isolation valve logic even if offsite radioactivity release rate limits are exceeded (ref. 2). Under these conditions with a VALID containment isolation signal, the Containment barrier should be considered lost.

Generic

These thresholds address incomplete containment isolation that allows direct release to the environment.

The use of the modifier "direct" in defining the release path discriminates against release paths through interfacing liquid systems. The existence of an in–line charcoal filter does not make a release path indirect since the filter is not effective at removing fission product noble gases. Typical filters have an efficiency of 95-99% removal of iodine. Given the magnitude of the core inventory of iodine, significant releases could still occur. In addition, since the fission product release would be driven by boiling in the reactor vessel, the high humidity in the release stream can be expected to render the filters ineffective in a short period.

NMP2 Basis Reference(s):

1. Improved Technical Specifications Nine Mile Point Nuclear Station, Unit No. 2, Table 3.6.1.3-1
2. N2-EOP-PC Primary Containment Control
3. NEI 99-01 CMT Loss 3A

Barrier: Containment

Category: C. Isolation

Degradation Threat: Loss

Threshold:

4. Intentional Primary Containment venting per EOPs

Basis:

Plant-Specific

N2-EOP-PC, Primary Containment Control, and N2-EOP-PCH, Hydrogen Control, may specify Primary Containment venting and intentional bypassing of the containment isolation valve logic, even if offsite radioactivity release rate limits are exceeded (ref. 1, 2). The threshold is met when the operator begins venting the Primary Containment in accordance with EOP-6, Support Procedures (Attachment 21 or 25), not when actions are taken to bypass interlocks prior to opening the vent valves (ref. 3). Purge and vent actions specified in N2-EOP-PC Step PCP-1 to control Primary Containment pressure below the drywell high pressure scram setpoint or EOP-PCH Step 31 or 34 to lower hydrogen concentration does not meet this threshold because such action is only permitted if offsite radioactivity release rates will remain below the ODCM limits (ref. 1, 2).

Generic

These thresholds address incomplete containment isolation that allows direct release to the environment.

Site specific EOPs may direct containment isolation valve logic(s) to be intentionally bypassed, regardless of radioactivity release rates. Under these conditions with a VALID containment isolation signal, the containment should also be considered lost if containment venting is actually performed.

Intentional venting of Primary Containment for Primary Containment pressure or combustible gas control per EOPs to the secondary containment and/or the environment is considered a loss of containment. Containment venting for pressure when not in an accident situation should not be considered.

NMP2 Basis Reference(s):

1. N2-EOP-PC Primary Containment Control
2. N2-EOP-PCH Hydrogen Control

3. EOP-6 NMP2 EOP Support Procedure
4. NEI 99-01 CMT Loss 3B

Barrier: Containment

Category: C. Isolation

Degradation Threat: Loss

Threshold:

5. UNISOLABLE primary system leakage outside Primary Containment as indicated by exceeding **EITHER**:

   RB area maximum safe temperature value (N2-EOP-SC Detail S)

   OR

   RB area radiation > 8.00E+3 mR/hr

Basis:

Plant-Specific

The presence of elevated general area temperatures or radiation levels in the Reactor Building (RB) may be indicative of UNISOLABLE primary system leakage outside the Primary Containment. The EOP maximum safe values define this Containment barrier threshold because they are indicative of problems in the secondary containment that are spreading and pose a threat to achieving a safe plant shutdown. This threshold addresses problematic discharges outside Primary Containment that may not originate from a high-energy line break. The locations into which the primary system discharge is of concern correspond to the areas addressed in N2-EOP-SC Detail S (ref. 1). See Figure F-2.

A "Maximum Safe Value" is the highest value at which equipment necessary for the safe shutdown of the plant will operate and personnel can perform any actions necessary for the safe shutdown of the plant.

The maximum safe value for temperature is dependent on whether access is needed to areas within the reactor building to perform actions required by other EOP steps. Only areas in which the actions must be taken (and there is no other alternative) qualify as "areas" when determining the number of affected areas. (ref. 2)

The maximum safe value for radiation is 8.00E+3 mR/hr.

In general, multiple indications should be used to determine if a primary system is discharging outside Primary Containment. For example, a high area radiation condition does not necessarily indicate that a primary system is discharging into the secondary containment since this may be caused by radiation shine from nearby steam lines or the movement of radioactive materials. Conversely, a high area radiation condition in conjunction with other indications (e.g. room flooding, high area temperatures, reports of steam in the secondary containment, an unexpected rise in feedwater flowrate, or unexpected main turbine control valve closure) may indicate that a primary system is discharging into the secondary containment.

Generic

This threshold addresses incomplete containment isolation that allows direct release to the environment.

In addition, The presence of area radiation or temperature Maximum Safe Values indicating UNISOLABLE primary system leakage outside the Primary Containment are addressed after a containment isolation. The indicators should be confirmed to be caused by RCS leakage.

There is no Potential Loss threshold associated with this item.

NMP2 Reference(s):

1. N2-EOP-SC Secondary Containment Control
2. NER-2M-039, NMP2 Emergency Operating Procedures (EOP) Basis Document
3. NEI 99-01 CMT Loss 3C

Figure F-2: N2-EOP-SC Detail S

S	Maximum Safe Values	
Parameter	**Location**	**Maximum Safe Value**
Area Temperature (EOP-6 Att 28)	All areas	212°F
	Areas when access is required for support of EOP actions.	135°F
Area Radiation	All areas	8.00E+3 mR/hr
Area Water Level	All areas	Flooding alarm

Barrier: Containment

Category: D. Rad

Degradation Threat: Loss

Threshold:

None

Barrier: Containment

Category: E. Judgment

Degradation Threat: Loss

Threshold:

6. **ANY** condition in the opinion of the Emergency Director that indicates loss of the Containment barrier

Basis:

Plant-Specific

The Emergency Director judgment threshold addresses any other factors relevant to determining if the Containment barrier is lost. Such a determination should include IMMINENT barrier degradation, barrier monitoring capability and dominant accident sequences.

- IMMINENT barrier degradation exists if the degradation will likely occur within two hours based on a projection of current safety system performance. The term "IMMINENT" refers to recognition of the inability to reach safety acceptance criteria before completion of all checks.
- Barrier monitoring capability is decreased if there is a loss or lack of reliable indicators. This assessment should include instrumentation operability concerns, readings from portable instrumentation and consideration of offsite monitoring results.
- Dominant accident sequences lead to degradation of all fission product barriers and likely entry to the EOPs. The Emergency Director should be mindful of the Loss of AC power (Station Blackout) and ATWS EALs to assure timely emergency classification declarations.

Generic

This threshold addresses any other factors that are to be used by the Emergency Director in determining whether the Containment barrier is lost. In addition, the inability to monitor the barrier should also be incorporated in this threshold as a factor in Emergency Director judgment that the barrier may be considered lost.

The Containment barrier should not be declared lost based on exceeding Technical Specification action statement criteria, unless there is an event in progress requiring mitigation by the Containment barrier. When no event is in progress (Loss or Potential Loss of either Fuel Clad and/or RCS) the Containment barrier status is addressed by Technical Specifications.

NMP2 Basis Reference(s):

1. NEI 99-01 CMT Loss 6

Barrier: Containment

Category: A. RPV Level

Degradation Threat: Potential Loss

Threshold:

1. Primary Containment Flooding is required

Basis:

Plant-Specific

Requirements for Primary Containment Flooding are established in EOP-RPV Step L-16; EOP-C5 Steps L-8, L-10 and L-18; and EOP-C4 Override 1. These EOPs provide instructions to ensure adequate core cooling by maintaining RPV water level above prescribed limits or operating sufficient RPV injection sources when level cannot be determined. SAP entry is required when (ref. 1):

- RPV water level cannot be restored and maintained above -39 in. with insufficient Core Spray Cooling: The Minimum Steam Cooling RPV Water Level (MSCRWL) is the lowest RPV water level at which the covered portion of the reactor core will generate sufficient steam to preclude any clad temperature in the uncovered portion of the core from exceeding 1500°F. Core Spray Cooling is insufficient if RPV water level cannot be restored and maintained at or above -62 in. with at least 6350 gpm core spray loop flow. Consistent with the EOP definition of "cannot be restored and maintained," the determination that the parameter cannot be restored and maintained above the limit may be made at, before, or after the parameter actually decreases to this point.
- RPV water level cannot be determined and it is determined that core damage is occurring: When RPV water level cannot be determined, EOPs require RPV flooding strategies. RPV water level indication provides the primary means of knowing if adequate core cooling is being maintained. When all means of determining RPV water level are unavailable, reliance on alternate means of assuring adequate core cooling must be attempted. The instructions in EOP-C4 specify these means, which include emergency depressurization of the RPV and

injection into the RPV at a rate needed to flood to the elevation of the main steam lines or hold RPV pressure above the Minimum Steam Cooling Pressure (in ATWS events)

This threshold is also a Loss of the Fuel Clad barrier (FC Loss A.1). Since Primary Containment Flooding occurs after core uncovery has occurred a Loss of the RCS barrier exists (RCS Loss A.1). Primary Containment Flooding (SAP entry), therefore, represents a Loss of two barriers and a Potential Loss of a third, which requires a General Emergency classification.

Generic

There is no Loss threshold associated with this item.

The potential loss requirement for drywell flooding indicates adequate core cooling cannot be established and maintained and that core melt is possible. Entry into Primary Containment Flooding procedures (SAPs) is a logical escalation in response to the inability to maintain adequate core cooling.

The condition in this potential loss threshold represents a potential core melt sequence which, if not corrected, could lead to vessel failure and increased potential for containment failure. In conjunction with Reactor Vessel water level "Loss" thresholds in the Fuel Clad and RCS barrier columns, this threshold will result in the declaration of a General Emergency -- loss of two barriers and the potential loss of a third.

NMP2 Basis Reference(s):

1. NER-2M-039, NMP2 Emergency Operating Procedures (EOP) Basis Document
2. N2-EOP-C4 RPV Flooding
3. NEI 99-01 CMT Potential Loss 2

Barrier: Containment

Category: B. Primary Containment Pressure / Temperature

Degradation Threat: Potential Loss

Threshold:

2. Primary Containment pressure > 45 psig and rising

Basis:

Plant-Specific

If this threshold is exceeded, a challenge to the Primary Containment structure has occurred because assumptions used in the accident analysis are no longer VALID and an unanalyzed condition exists (ref. 1). This constitutes a Potential Loss of the Containment barrier even if a containment breach has not occurred.

Generic

The Primary Containment pressure of 45 psig is based on the Primary Containment design pressure.

NMP2 Basis Reference(s):

1. USAR Section 6.2.1.1.2
2. NEI 99-01 CMT Potential Loss 1A

Barrier: Containment
Category: B. Primary Containment Pressure / Temperature
Degradation Threat: Potential Loss
Threshold:

3. Explosive mixture exists inside Primary Containment ($\geq$ 6% H_2 and $\geq$ 5% O_2)

Basis:

Plant-Specific

Explosive (deflagration) mixtures in the Primary Containment are assumed to be elevated concentrations of hydrogen and oxygen. BWR industry evaluation of hydrogen generation for development of EOPs/SAPs indicates that any hydrogen concentration above minimum detectable is not to be expected within the short term. Post-LOCA hydrogen generation primarily caused by radiolysis is a slowly evolving, long-term condition. Hydrogen concentrations that rapidly develop are most likely caused by metal-water reaction. A metal-water reaction is indicative of an accident more severe than accidents considered in the plant design basis and would be indicative, therefore, of a potential threat to Primary Containment integrity. Hydrogen concentration of approximately 6% is considered the global deflagration concentration limit (ref. 1).

Except for brief periods during plant startup and shutdown, oxygen concentration in the Primary Containment is maintained at insignificant levels by nitrogen inertion. The specified values for this Potential Loss threshold are the minimum global deflagration concentration limits (6% hydrogen and 5% oxygen, ref. 1) and readily recognizable because 6% hydrogen is well above the N2-EOP-PCH entry condition (ref. 2). The minimum global deflagration hydrogen/oxygen concentrations (6%/5%, respectively) require intentional Primary Containment venting, which is defined to be a Loss of Containment (PC Loss C.4).

The USAR requires the $H_2/0_2$ analyzers to be able to provide and record combustible gas concentration in the Primary Containment within 90 minutes following a LOCA with safety system injection. The $H_2/0_2$ analyzers are normally in standby and require a 30 minute warrn-up/self-test period before they start providing data. (ref. 1)

If the hydrogen or oxygen monitor is unavailable, sampling and analysis may determine gas concentrations. The validity of sample results must be judged based upon plant conditions, since drawing and analyzing samples may take some time. If sample results cannot be relied upon and hydrogen concentrations cannot be determined by any other means, the concentrations must be considered "unknown." The monitors should not be considered "unavailable" until an attempt has been made to place them in service. (ref. 1)

Generic

BWRs specifically define the limits associated with explosive mixtures in terms of deflagration concentrations of hydrogen and oxygen.

NMP2 Basis Reference(s):

1. NER-2M-039, NMP2 Emergency Operating Procedures (EOP) Basis Document
2. N2-EOP-PCH Hydrogen Control
3. NEI 99-01 CMT Potential Loss 1B

Barrier: Containment

Category: B. Primary Containment Pressure / Temperature

Degradation Threat: Potential Loss

Threshold:

4. Suppression pool temperature and RPV pressure **cannot** be maintained below the Heat Capacity Temperature Limit (N2-EOP-PC Figure M)

Basis:

Plant-Specific

The Heat Capacity Temperature Limit (HCTL) is given in EOP Figure M. This threshold is met when N2-EOP-PC Step SPT-6 is reached (ref. 1).

Generic

The Heat Capacity Temperature Limit (HCTL) is the highest suppression pool temperature from which Emergency RPV Depressurization will not raise:

- Suppression chamber temperature above the design value (270°F),

  OR

- Suppression chamber pressure above Primary Containment Pressure Limit, before the rate of energy transfer from the RPV to the containment is greater than the capacity of the containment vent.

The HCTL is a function of RPV pressure and suppression pool water level. It is utilized to preclude failure of the containment and equipment in the containment necessary for the safe shutdown of the plant and therefore, the inability to maintain plant parameters below the limit constitutes a potential loss of Containment.

NMP2 Basis Reference(s):

1. N2-EOP-PC Primary Containment Control
2. NEI 99-01 CMT Potential Loss 1C

Barrier: Containment

Category: D. Rad

Degradation Threat: Potential Loss

Threshold:

5. Drywell area radiation ≥ 6.0 E4 R/hr (6.0 E7 mRem/hr)

Basis:

Plant-Specific

It is important to recognize that the radiation monitor may be sensitive to shine from the RPV or RCS piping (caused by lower than normal RPV water level for example). The Drywell High Range Radiation Monitors are the following (ref. 1):

- 2CEC*PNL880D: DRMS 2RMS*RE1B/D

  RMS*RUZ1A

  RMS*RUZ1B

- 2CEC*PNL880B: DRMS 2RMS*RE1A/C

  RMS*RUZ1C

  RMS*RUZ1D

Figure F-1 illustrates the location of the following four detectors inside the drywell (ref. 1):

- 2RMS*RE1A P.C. 268 170EAZ
- 2RMS*RE1C P.C. 267 024EAZ
- 2RMS*RE1B P.C. 268 245EAZ
- 2RMS*RE1D P.C. 268 353EAZ

The threshold value was calculated assuming the instantaneous release and dispersal of the reactor coolant noble gas and iodine inventory associated with 20% fuel clad damage into the drywell atmosphere (ref. 2, 3). The referenced calculation yields a value of 5.6 E4 R/hr. This has been rounded to 6.0 E4 R/hr because it is observable on existing instrumentation.

Generic

The 6.0 E4 R/hr reading is a value that indicates significant fuel damage well in excess of that required for loss of RCS and Fuel Clad.

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.

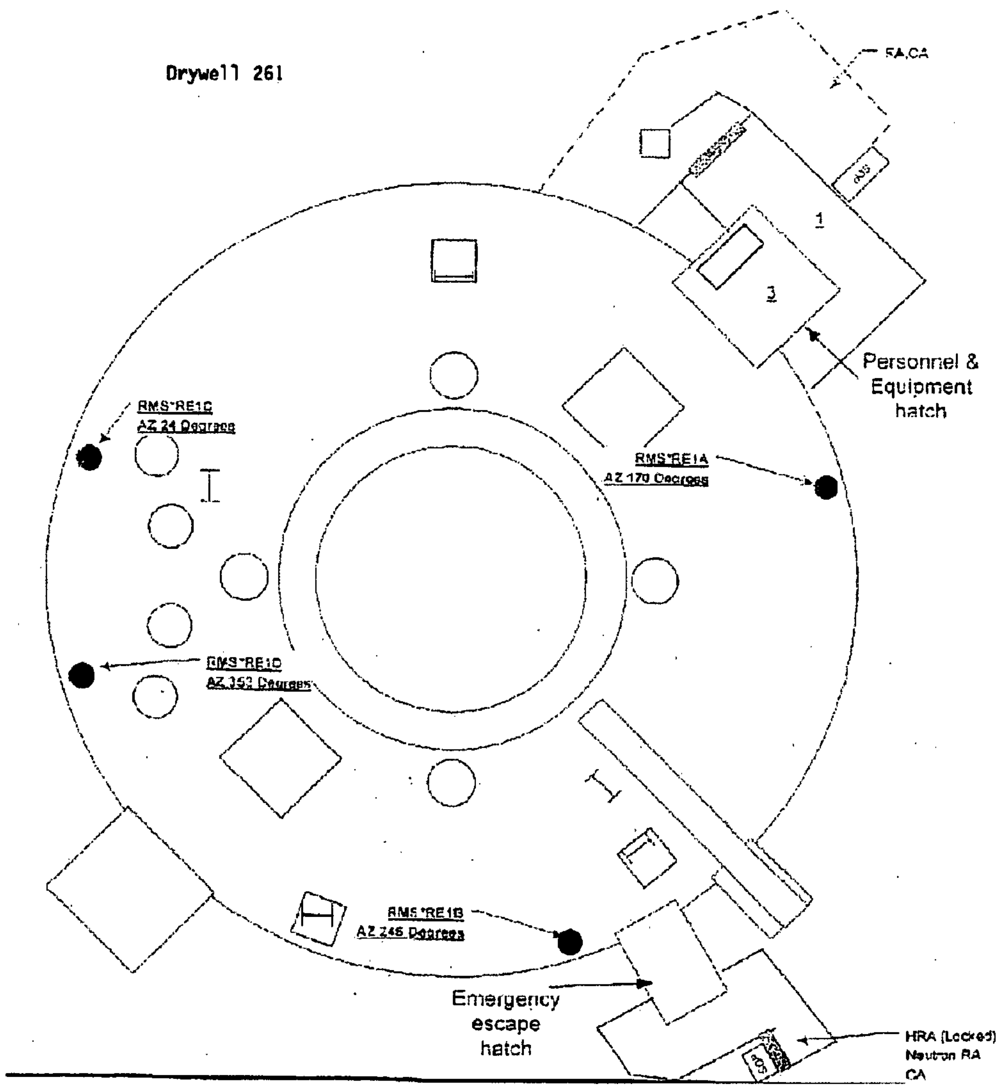
There is no Loss threshold associated with this item.

NMP2 Basis Reference(s):

1. N2-RSP-RMS-R106 Channel Calibration Test of the Drywell High Range Area Radiation Monitors
2. Calculation PR-C-24-0
3. CCN No. 009718 Calculation of Drywell Radiation General Emergency EAL
4. NEI 99-01 CMT Potential Loss 4

Figure F-1: Drywell High Range Radiation Monitor Detector Locations (ref. 1)

Drywell 261



Barrier: Containment
Category: E. Judgment
Degradation Threat: Potential Loss
Threshold:

6. **ANY** condition in the opinion of the Emergency Director that indicates potential loss of the Containment barrier

Basis:

Plant-Specific

The Emergency Director judgment threshold addresses any other factors relevant to determining if the Containment barrier is potentially lost. Such a determination should include IMMINENT barrier degradation, barrier monitoring capability and dominant accident sequences.

- IMMINENT barrier degradation exists if the degradation will likely occur within two hours based on a projection of current safety system performance. The term "IMMINENT" refers to recognition of the inability to reach safety acceptance criteria before completion of all checks.
- Barrier monitoring capability is decreased if there is a loss or lack of reliable indicators. This assessment should include instrumentation operability concerns, readings from portable instrumentation and consideration of offsite monitoring results.
- Dominant accident sequences lead to degradation of all fission product barriers and likely entry to the EOPs. The Emergency Director should be mindful of the Loss of AC power (Station Blackout) and ATWS EALs to assure timely emergency classification declarations.

Generic

This threshold addresses any other factors that are to be used by the Emergency Director in determining whether the Containment barrier is potentially lost. In addition, the inability to monitor the barrier should also be incorporated in this threshold as a factor in Emergency Director judgment that the barrier may be considered potentially lost.

The Containment barrier should not be declared potentially lost based on exceeding Technical Specification action statement criteria, unless there is an event in progress requiring mitigation by the Containment barrier. When no event is in progress (Loss or Potential Loss of either Fuel Clad and/or RCS) the Containment barrier status is addressed by Technical Specifications.

NMP2 Basis Reference(s):

1. NEI 99-01 CMT Potential Loss 6