DUKE ENERGY®

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Serial: BSEP 15-0092

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

Subject: Brunswick Steam Electric Plant, Unit Nos. 1 and 2
Renewed Facility Operating License Nos. DPR-71 and DPR-62
Docket Nos. 50-325 and 50-324
Response to Request for Additional Information Regarding Adoption of Emergency Action Level Scheme Pursuant to NEI 99-01, Revision 6, "Development of Emergency Action Levels for Non-Passive Reactors" (NRC TAC Nos. MF5766 and MF5767)

References:

1. Letter from William R. Gideon (Duke Energy) to U.S. Nuclear Regulatory Commission (Serial: BSEP 15-0010), *License Amendment Request to Adopt Emergency Action Level Scheme Pursuant to NEI 99-01, Revision 6, "Development of Emergency Action Levels for Non-Passive Reactors,"* dated January 30, 2015, ADAMS Accession Number ML15044A245

2. Electronic Mail from Peter S. Tam (NRC) to Michael K. Braden (Duke Energy), *Draft RAI on proposed amendment re. EAL scheme change (TAC MF5766, MF5767),* dated September 29, 2015, ADAMS Accession Number ML15272A426

3. Letter from Annette H. Pope (Duke Energy) to U.S. Nuclear Regulatory Commission (Serial: BSEP 15-0096), *Submittal of Response to Request for Additional Information – Adoption of Emergency Action Level Scheme Pursuant to NEI 99-01, Revision 6, " Development of Emergency Action Levels for Non-Passive Reactors," (NRC TAC Nos. MF5766 and MF5767),* dated November 12, 2015

Ladies and Gentlemen:

By letter dated January 30, 2015 (i.e., Reference 1), Duke Energy Progress, Inc., submitted a license amendment request (LAR) to adopt new emergency action levels (EALs) for use at the Brunswick Steam Electric Plant (BSEP), Unit Nos. 1 and 2.

On September 29, 2015 (i.e., Reference 2), the NRC provided an electronic mail with requests for additional information (RAIs) regarding the LAR submittal. The response to the RAIs is provided in Enclosure 1 of this letter. Enclosure 2 contains the redline version of the Technical Bases Document, 0PEP-02.2.1, *Emergency Action Level Technical Bases.* Enclosure 3 contains the updated clean version of that same document. Lastly, Enclosure 4 provides the revised EAL wall charts, 0PEP-02.1, *Initial Emergency Actions,* for information.

AX45
NRR

This document contains no regulatory commitments.

Please refer any questions regarding this submittal to Mr. Lee Grzeck, Manager – Regulatory Affairs, at (910) 457-2487.

I declare, under penalty of perjury, that the foregoing is true and correct. Executed on November 23, 2015.

Sincerely,

William R. Gideon

WRG/mkb

Enclosures:

1. Response to Request for Additional Information (RAI)
2. Revised BSEP Technical Bases Document, 0PEP-02.2.1, *Emergency Action Level Technical Bases* (Redline Version)
3. Revised BSEP Technical Bases Document, 0PEP-02.2.1, *Emergency Action Level Technical Bases* (Clean Version)
4. Revised BSEP EAL Wall Charts, 0PEP-02.1, *Initial Emergency Actions*

cc (with enclosures):

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Response to Request for Additional Information Regarding Adoption of Emergency Action Level Scheme Pursuant to NEI 99-01, Revision 6, "Development of Emergency Action Levels for Non-Passive Reactors"

By letter dated January 30, 2015, Duke Energy Progress, Inc., submitted a license amendment request (LAR) to adopt new emergency action levels (EALs) used at the Brunswick Steam Electric Plant (BSEP), Unit Nos. 1 and 2.

On September 29, 2015, the NRC provided an electronic mail with requests for additional information (RAIs) regarding the LAR submittal.

Duke Energy's responses to the RAIs are provided below.[3]

RAI-01

Section 4.3, "Instrumentation Used for EALs," to NEI 99-01, Revision 6, states in part, *"Scheme developers should ensure that specific values used as EAL setpoints are within the calibrated range of the referenced instrumentation."* Please confirm that all setpoints and indications used in the proposed EAL scheme are within the calibrated range(s) of the stated instrumentation and that the resolution of the instrumentation is appropriate for the setpoint/indication.

Response

Duke Energy has confirmed the setpoints and indications used in the proposed EAL scheme are within the calibrated range(s) of the stated instrumentation and that the resolution of the instrumentation is appropriate for the setpoint/indication.

RAI-02

Section 2.1, "Background," and Section 4.0, "References," use ADAMS Accession Number ML110240324 as the document the NRC endorsed when in fact it is ML12326A805, which is part of the endorsement package contained within ML13091A209. Please revise to reference either one of these ADAMS numbers as the NRC endorsed EAL scheme development guidance

Response

Duke Energy has revised the referenced ADAMS Accession Number to ML12326A805.

RAI-03

Section 2.5, "Technical Bases Information," states, *"A Plant-specific basis section that provides BNP [BSNP]-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. 6."* Due to the high probability that EAL decision-makers will be confused between these two sections when the information appears to be inconsistent, please explain the basis for two sections rather than one basis section that is specific to the plant, or revise accordingly.

Response

The BSEP site-specific and NEI 99-01 generic bases sections have been combined into a single bases section for each EAL. Section 2.5, *Technical Bases Information*, has been revised accordingly.

RAI-04

Section 5.0, "Definitions," does not include definitions for the following. Please incorporate these definitions into the document or provide justification for not including definitions consistent with NRC endorsed guidance:

- Alert,
- Notification of Unusual Event,
- Site Area Emergency,
- General Emergency,
- Emergency Action Level,
- Emergency Classification Level,
- Fission Product Barrier Threshold, and
- Initiating Condition.

Response

Duke Energy has added the following definitions to Section 5.0 consistent with NEI 99-01, Revision 6:

- Alert,
- Unusual Event (UE),
- Site Area Emergency (SAE),
- General Emergency (GE),
- Emergency Action Level (EAL),
- Emergency Classification Level (ECL),
- Fission Product Barrier Threshold, and
- Initiating Condition

RAI-05

For the following EALs, please explain why the listed NOTEs were included, or revise accordingly.

- RA1.2 – NOTE-3
- RS1.2 – NOTE-3
- RG1.2 – NOTE-3

Response

Duke Energy has deleted Note 3 from EALs RA1.2, RS1.2, and RG1.2

RAI-06

For EAL RA2.2, please explain why the information in the NEI 99-01 Basis section does not contain all of the actual information from NEI 99-01 as it is germane to this particular EAL, or revise accordingly.

Response

Duke Energy has re-instated the following deleted generic bases:

> *This IC applies to irradiated fuel that is licensed for dry storage up to the point that the loaded storage cask is sealed. Once sealed, damage to a loaded cask causing loss of the CONFINEMENT BOUNDARY is classified in accordance with EAL EU1.1.*

RAI-07

For EAL RS2.1, please explain why the escalation path apparently incorrectly lists EAL AG1 instead of EAL RG1, or revise accordingly.

Response

Duke Energy has corrected the EAL number citation to RG1.

RAI-08

For EALs CU2.1, SA1.1 and SU1.1, please add a list of the AC power sources to the EAL to ensure consistent and timely recognition of the event, or provide justification for not including this list. [NOTE: It is not necessary to have this list for EALs CA2.1, SG1.1, SG1.2, and SS1.1 as these EALs are concerned with a loss of all AC power sources.]

Response

Duke Energy has added new Tables C-6 and S-5 listing AC power sources to CU2.1, SA1.1 and SU1.1.

RAI-09

For EAL CG1.2 and the Fission Barrier Matrix, please confirm that the "Max Normal" and "Max Safe" values can be promptly obtained from within the Control Room, or revise accordingly to support timely event declaration.

Response

The 0EOP-03-SCCP Table 1 Secondary Containment area temperature "Max Normal" and "Max Safe" Operating Limits can be promptly obtained from within the Control Room.

The 0EOP-03-SCCP Table 3 Secondary Containment area radiation "Max Normal" Operating Limits can be promptly obtained from within the Control Room.

None of the 0EOP-03-SCCP Table 3 Secondary Containment area radiation "Max Safe" Operating Limits can be obtained from within the Control Room. These values must be determined by local area survey.

Duke Energy has deleted 0EOP-03-SCCP Table 3 Secondary Containment area radiation "Max Safe" Operating Limits from Containment Loss threshold B.1 and EAL CG1.2 Table C-2. 0EOP-03-SCCP Table 3 Secondary Containment area radiation "Max Safe" Operating Limits are adequately bounded by 0EOP-03-SCCP Table 1 Secondary Containment area temperature "Max Safe" Operating Limits.

RAI-10

For EALs CU4.1, SG1.2 and SS2.1, please explain the impact on timely event declaration, since the BNP Basis states that battery voltage must be read locally.

Response

As stated in the loss of vital DC EAL bases:

> *Note that the Control Room DC voltage indicator only reads battery charger output voltage and not battery voltage unless the charger output breaker is closed.* ***However ERFIS does provide DC battery voltage,*** *otherwise battery voltage must be read locally.*

The Emergency Response Facility Information System (ERFIS) provides timely DC vital battery voltage from within the Control Room.

Duke Energy has added the following bases for CU4.1, SG1.2 and SS2.1 to read:

> *In the event that DC battery voltage indication is not available via ERFIS, local voltage indication is available for each bus based on dispatching a field operator in accordance with 0AOP-39.0 Loss of DC Power. In this case the 15 minute classification clock begins upon receipt of the low voltage alarm in the Control Room. If battery voltage cannot be verified to be greater than or equal to 105 VDC within the 15 minutes, emergency classification must be made under this EAL.*

RAI-11

For EALs CU5.1 and SU7.1, please explain how the Corporate Telephone Communications System can work for in-plant/onsite communications, or revise accordingly.

Response

Duke Energy has replaced the Corporate Telephone Communications System with DEMNET in Tables S-3 and C-4 and deleted DEMNET's applicability as an onsite communications system.

RAI-12

For EAL HU2.1, please provide additional details on the process used to determine if the seismic activity exceeded the Operating Basis Earthquake (OBE) threshold and availability to support timely event declaration. If the OBE threshold is not recognized, in a timely fashion, from indications in or near the Control Room, then explain why the alternative EAL was not developed in accordance with NEI 99-01, Revision 6, or revise accordingly.

Response

OBE exceedance can be promptly determined at the Seismic Monitoring Panel LCD display (2-ENV-XU-823) located in the Control Room. OBE exceedance is indicated by a red OBE EXCEEDANCE indicator.

Alternatively, OBE exceedance can also be promptly determined at 2-ENV-XU-823 by either indicating lights located on the Seismic Monitoring Panel Alarm and Interconnect Panel or Seismic Monitoring Panel Recorder Panel.

In all cases, OBE exceedance can be determined promptly from the Control Room following receipt of the Seismic Event alarm or felt ground motion.

RAI-13

For EAL HU3.2, the proposed basis information includes the statement:
"Refer to Updated FSAR section 3.4.2, Protection from Internal Flooding, to identify susceptible internal flooding areas." If the intent of this statement is to limit the areas of consideration, then provide a list of areas in the EAL for the staff to review, or otherwise explain the purpose for this statement.

Response

The intent of the cited statement was not to limit the areas of consideration. However, Duke Energy has deleted the cited bases statement from HU3.2 bases section to preclude confusion.

RAI-14

For EAL HU3.4, please explain where the BSNP Basis information related to when the 15-minute clock starts came from.

Response

The 15-minute clock statement was included in the bases to clarify the time when the EAL threshold was met. However, such a statement is not necessary as the Shift Manager makes the determination of when the EAL threshold is met based on their assessment. As such, Duke Energy has deleted the cited BSEP bases statement related to the 15-minute clock.

RAI-15

For EALs HU4.1 and HU4.2, the areas listed in Table H-1 seem to be vague or too all-encompassing. Please explain if the listed areas are all the areas that contain equipment needed for safe operation, safe shutdown and safe cool-down, and if these areas can be fine-tuned to limit consideration for these EALs.

Response

Table H-1, *Fire Areas*, is based on BNP-E-9.010, *NFPA 805 Nuclear Safety Capability Assessment (NSCA)*, and 0PFP-PBAA, *Power Block Auxiliary Areas Prefire Plan*. Table H-1 Fire Areas include all structures containing functions and systems required for safe operation, safe shutdown, and safe cool-down of the plant (i.e., Safety Systems).

A balance must be established between defining major plant structures containing safe shutdown equipment as fire areas versus a detailed list of areas for every safety system component location. The Table H-1 list of fire areas achieves that balance in support of timely and accurate emergency classification for the end-user. Therefore, fine tuning of these areas would not be beneficial.

RAI-16

For EAL HU4.2, please explain why all the required information related to Appendix R was not carried over into this EAL, or revise accordingly.

Response

Duke Energy has re-instated the generic bases related to Appendix R to HU4.2 and added clarification that BSEP is a NFPA-805 plant to preclude confusion by the end-user.

RAI-17

For EALs HU4.2 and HU4.4, please confirm that the Independent Spent Fuel Storage Installation (ISFSI) would be an area applicable to these EALs, or revise accordingly.

Response

The BSEP ISFSI is contained wholly within the plant Protected Area. Therefore, the ISFSI would be applicable to EALs HU4.3 and HU4.4 for fires within the plant Protected Area.

RAI-18

For EAL HS6.1, please explain why reactivity control is not required at the Remote Shutdown Panel, or revise accordingly. In addition, please consider incorporating operating mode specificity to the key safety functions listed in the EAL.

Response

Duke Energy has deleted the bases statement that no further actions are required for reactivity control following reactor scram from the Control Room.

Duke Energy has revised HS6.1 mode applicability from ALL to Modes 1 through 5.

Duke Energy has revised HS6.1, based upon an assessment of applicable modes for each of the listed safety function, as follows:

> *An event has resulted in plant control being transferred from the Control Room to the Remote Shutdown Panels*
>
> ***AND***
>
> *Control of **any** of the following key safety functions is not reestablished within 22.5 min. (Note 1):*
>
> - *Reactivity (Modes 1 and 2 **only**)*
> - *RPV water level*
> - *RPV pressure (Modes 1, 2, 3 and 4 **only**)*

<u>RAI-19</u>

For EAL SU4.1, please explain the intent of the 15-minute qualifier, and basis for consideration in this EAL.

Response

Duke Energy has deleted the 15-minute threshold criteria from SU4.1.

<u>RAI-20</u>

For EAL SU4.2, please explain why the timing note is not part of this EAL, or revise accordingly.

Response

Duke Energy has added the Timing note (i.e., Note 1) to SU4.2.

<u>RAI-21</u>

For EALs SU6.1, SU6.2, SA6.1 and SS6.1, please explain why the power level (<2%) was added as it is not the intent of the EALs to explain to licensed operators what constitutes reactor shutdown. Industry experience with Boiling Water Reactor high power scrams shows that, for a short period of time, the power may exceed the APRM downscale value thus making it difficult for EAL decision-makers to adhere to the requirements of the EAL (i.e., an event could be declared based upon EAL wording when the plant is actually shutdown and no event is warranted.) Please justify or revise accordingly.

Response

The method used to determine if the reactor is shutdown following a reactor scram, for the purposes of emergency classification, is reactor power indicating < 2% (Average Power Range Monitor downscale). As specified in the generic developer's guidance;

> Developers may include site-specific EOP criteria indicative of a successful reactor shutdown in an EAL statement, the Basis or both (e.g., a reactor power level).

Reactor power < 2% is, therefore, the site-specific indication of a successful reactor scram.

<u>RAI-22</u>

Category E – Independent Spent Fuel Storage Installation (ISFSI) guidance: The statement: *"Formal offsite planning is not required because the postulated worst-case accident involving an ISFSI has insignificant consequences to the public health and safety,"* is not applicable to this proposed EAL scheme. Please remove, or provide further justification for inclusion.

In addition, please incorporate guidance related to the fact that EALs HU1 and HA1 are also considered for events that occur at the ISFSI.

Response

Duke Energy has deleted the cited statement from the ISFSI category introduction.

Duke Energy has revised the ISFSI category introduction:

> *The BNP ISFSI is contained wholly within the plant Protected Area. Therefore a security event related to the ISFSI would be applicable to EALs HU1.1, HA1.1 and HS1.1.*

RAI-23

Fission Product Barrier Matrix: The cited NEI 99-01 Basis sections for several of the Fission Product Barrier criteria are not from NRC-endorsed NEI 99-01, Revision 6. Please either revise to what has actually been endorsed, or (depending on respond to RAI-03) consider unifying the basis sections into one.

Response

As per the response to RAI-3, the BSEP site-specific and NEI 99-01, Revision 6, bases have been unified.

Brunswick Steam Electric Plant, Unit Nos. 1 and 2
Renewed Facility Operating License Nos. DPR-71 and DPR-62
Docket Nos. 50-325 and 50-324
Response to Request for Additional Information Regarding Adoption of Emergency Action Level Scheme Pursuant to NEI 99-01, Revision 6, "Development of Emergency Action Levels for Non-Passive Reactors"

Revised BSEP Technical Bases Document, 0PEP-02.2.1, "Emergency Action Level Technical Bases" (Redline Version)

DUKE ENERGY	BRUNSWICK NUCLEAR PLANT	R Reference Use

PLANT OPERATING MANUAL

VOLUME XIII

PLANT EMERGENCY PROCEDURE

0PEP-02.2.1

EMERGENCY ACTION LEVEL TECHNICAL BASES

REVISION 6 (RAI Response)

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1.0 PURPOSE

This document provides an explanation and rationale for each Emergency Action Level (EAL) included in the EAL Upgrade Project for Brunswick Nuclear Plant (BNP). It should be used to facilitate review of the BNP EALs and provide historical documentation for future reference. Decision-makers responsible for implementation of 0PEP-02.1 Initial Emergency Actions, may use this document as a technical reference in support of EAL interpretation. This information may assist the Site Emergency Coordinator in making classifications, particularly those involving judgment or multiple events. The basis information may also be useful in training and for explaining event classifications to offsite officials.

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 or less 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 BNP 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) Revisions 4 and 5 were 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 6 of NEI 99-01 has been issued which incorporates resolutions to numerous implementation issues including the NRC EAL Frequently Asked Questions (FAQs). Using NEI 99-01 Revision 6, "Methodology for the Development of Emergency Action Levels for Non-Passive Reactors," November 2012 (ADAMS Accession Number ML12326A805) (ref. 4.1.1), BNP conducted an EAL implementation upgrade project that produced the EALs discussed herein

Deleted: ML110240324

2.2 Fission Product Barriers

Fission product barrier thresholds represent threats to the defense in depth design concept that precludes the release of 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.

Many of the EALs derived from the NEI methodology are fission product barrier threshold based. That is, the conditions that define the EALs are based upon thresholds that represent the 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. A "Loss" threshold means the barrier no longer assures containment of radioactive materials. A "Potential Loss" threshold implies an increased probability of barrier loss and decreased certainty of maintaining the barrier.

The primary fission product barriers are:

A. <u>Fuel Clad (FC):</u> The Fuel Clad Barrier consists of the cladding material that contains the fuel pellets.

B. <u>Reactor Coolant System (RCS):</u> The RCS Barrier is the reactor coolant system pressure boundary and includes the RPV and all reactor coolant system piping up to and including the isolation valves.

C. <u>Primary Containment (PC):</u> The Primary Containment Barrier includes the drywell, the wetwell, their respective interconnecting paths, and other connections up to and including the outermost containment isolation valves. Primary Containment Barrier thresholds are used as criteria for escalation of the ECL from Alert to a Site Area Emergency or a General Emergency.

2.3 Fission Product Barrier Classification Criteria

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

<u>Alert:</u>

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

<u>Site Area Emergency:</u>

Loss or potential loss of any two barriers

<u>General Emergency:</u>

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

2.4 EAL Organization

The BNP 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, Startup, 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, Refueling 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 group, assignment of EALs to categories and subcategories:

Category and subcategory titles are selected to represent conditions that are operationally significant to the EAL-user. The BNP EAL categories are aligned to and represent the NEI 99-01"Recognition Categories." Subcategories are used in the BNP scheme as necessary to further divide the EALs of a category into logical sets of possible emergency classification thresholds. The BNP EAL categories and subcategories are listed below.

EAL Groups, Categories and Subcategories

EAL Group/Category	EAL Subcategory
Any Operating Mode:	
R – Abnormal **R**ad Levels / Rad Effluent	1 – Radiological Effluent 2 – Irradiated Fuel Event 3 – Area Radiation Levels
H – **H**azards and Other Conditions Affecting Plant Safety	1 – Security 2 – Seismic Event 3 – Natural or Technological Hazard 4 – Fire 5 – Hazardous Gas 6 – Control Room Evacuation 7 – SEC Judgment
E – Independent Spent Fuel Storage Installation (ISFSI)	1 – Confinement Boundary
Hot Conditions:	
S – **S**ystem Malfunction	1 – Loss of Emergency AC Power 2 – Loss of Vital DC Power 3 – Loss of Control Room Indications 4 – RCS Activity 5 – RCS Leakage 6 – RPS Failure 7 – Loss of Communications 8 – Hazardous Event Affecting Safety Systems
F – **F**ission Product Barrier Degradation	None
Cold Conditions:	
C – **C**old Shutdown / Refueling System Malfunction	1 – RPV Level 2 – Loss of Emergency AC Power 3 – RCS Temperature 4 – Loss of Vital DC Power 5 – Loss of Communications 6 – Hazardous Event Affecting Safety Systems

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 Section 3.0 and Attachments 1 & 2 of this document for such information.

2.5 Technical Bases Information

EAL technical bases are provided in Attachment 1 for each EAL according to EAL group (Any, Hot, Cold), EAL category (R, C, H, S, F and E) 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. 6.

<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, C, H, S, F or E)
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).

<u>Classification (enclosed in rectangle):</u>

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

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

Exact wording of the EAL as it appears in the EAL Classification Matrix

Mode Applicability

One or more of the following plant operating conditions comprise the mode to which each EAL is applicable: 1 - Power Operations, 2 - Startup, 3 - Hot Shutdown, 4 - Cold Shutdown, 5 - Refueling, D - Defueled, or All. (See Section 2.6 for operating mode definitions)

Definitions:

If the EAL wording contains a defined term, the definition of the term is included in this section. These definitions can also be found in Section 5.1.

Basis:

A basis section that provides BNP-relevant information concerning the EAL as well as a description of the rationale for the EAL as provided in NEI 99-01 Rev. 6.

Deleted: Plant-Specific

Deleted: . This is followed by a

Deleted: Generic basis section that provides

BNP Basis Reference(s):

Site-specific source documentation from which the EAL is derived

2.6 Operating Mode Applicability (ref. 4.1.7)

1 Power Operations

Reactor is critical and the mode switch is in RUN

2 Startup

The mode switch is in STARTUP/HOT STANDBY

3 Hot Shutdown

The mode switch is in SHUTDOWN, all reactor vessel head closure bolts are fully tensioned, and reactor coolant temperature is >212°F

4 Cold Shutdown

The mode switch is in SHUTDOWN, all reactor vessel head closure bolts are fully tensioned, and reactor coolant temperature is ≤ 212°F

5 Refuel

The mode switch is in REFUEL or SHUTDOWN and one or more reactor vessel head closure bolts are less than fully tensioned

D Defueled

RPV contains no irradiated fuel

The plant operating mode that exists at the time that the event occurs (prior to any protective system or operator action being 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.

3.0 GUIDANCE ON MAKING EMERGENCY CLASSIFICATIONS

3.1 General Considerations

When making an emergency classification, the Site Emergency Coordinator must consider all information having a bearing on the proper assessment of an Initiating Condition (IC). This includes the Emergency Action Level (EAL) plus the associated Operating Mode Applicability, Notes, and the informing basis information. In the Recognition Category F matrices, EALs are based on loss or potential loss of Fission Product Barrier Thresholds.

3.1.1 Classification Timeliness

NRC regulations require the licensee to establish and maintain the capability to assess, classify, and declare an emergency condition within 15 minutes after the availability of indications to plant operators that an emergency action level has been exceeded and to promptly declare the emergency condition as soon as possible following identification of the appropriate emergency classification level. The NRC staff has provided guidance on implementing this requirement in NSIR/DPR-ISG-01, "Interim Staff Guidance, Emergency Planning for Nuclear Power Plants" (ref. 4.1.12).

3.1.2 Valid Indications

All emergency classification assessments shall be based upon valid indications, reports or conditions. A valid indication, report, or condition, is one that has been verified through appropriate means such that there is no doubt regarding the indicator's operability, the condition's existence, or the report's accuracy. For example, verification could be accomplished through an instrument channel check, response on related or redundant indicators, or direct observation by plant personnel.

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.

3.1.3 Imminent Conditions

For ICs and EALs that have a stipulated time duration (e.g., 15 minutes, 30 minutes, etc.), the Site Emergency Coordinator 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. If an ongoing radiological release is detected and the release start time is unknown, it should be assumed that the release duration specified in the IC/EAL has been exceeded, absent data to the contrary.

3.1.4 Planned vs. Unplanned Events

A planned work activity that results in an expected event or condition which meets or exceeds an EAL does not warrant an emergency declaration provided that: 1) the activity proceeds as planned, and 2) the plant remains within the limits imposed by the operating license. Such activities include planned work to test, manipulate, repair, maintain or modify a system or component. In these cases, the controls associated with the planning, preparation and execution of the work will ensure that compliance is maintained with all aspects of the operating license provided that the activity proceeds and concludes as expected. Events or conditions of this type may be subject to the reporting requirements of 10 § CFR 50.72 (ref. 4.1.4).

3.1.5 Classification Based on Analysis

The assessment of some EALs is based on the results of analyses that are necessary to ascertain whether a specific EAL threshold has been exceeded (e.g., dose assessments, chemistry sampling, RCS leak rate calculation, etc.). For these EALs, the EAL wording or the associated basis discussion will identify the necessary analysis. In these cases, the 15-minute declaration period starts with the availability of the analysis results that show the threshold to be exceeded (i.e., this is the time that the EAL information is first available). The NRC expects licensees to establish the capability to initiate and complete EAL-related analyses within a reasonable period of time (e.g., maintain the necessary expertise on-shift).

3.1.6 Site Emergency Coordinator Judgment

While the EALs have been developed to address a full spectrum of possible events and conditions which may warrant emergency classification, a provision for classification based on operator/management experience and judgment is still necessary. The NEI 99-01 EAL scheme provides the Site Emergency Coordinator with the ability to classify events and conditions based upon judgment using EALs that are consistent with the Emergency Classification Level (ECL) definitions (refer to Category H). The Site Emergency Coordinator will need to determine if the effects or consequences of the event or condition reasonably meet or exceed a particular ECL definition. A similar provision is incorporated in the Fission Product Barrier Tables; judgment may be used to determine the status of a fission product barrier.

3.2 Classification Methodology

To make an emergency classification, the user will compare an event or condition (i.e., the relevant plant indications and reports) to an EAL(s) and determine if the EAL has been met or exceeded. The evaluation of an EAL must be consistent with the related Operating Mode Applicability and Notes. If an EAL has been met or exceeded, the associated IC is likewise met, the emergency classification process "clock" starts, and the ECL must be declared in accordance with plant procedures no later than fifteen minutes after the process "clock" started.

When assessing an EAL that specifies a time duration for the off-normal condition, the "clock" for the EAL time duration runs concurrently with the emergency classification process "clock." For a full discussion of this timing requirement, refer to NSIR/DPR-ISG-01 (ref. 4.1.14).

3.2.1 Classification of Multiple Events and Conditions

When multiple emergency events or conditions are present, the user will identify all met or exceeded EALs. The highest applicable ECL identified during this review is declared. For example:

- If an Alert EAL and a Site Area Emergency EAL are met, whether at one unit or at two different units, a Site Area Emergency should be declared.

There is no "additive" effect from multiple EALs meeting the same ECL. For example:

- If two Alert EALs are met, whether at one unit or at two different units, an Alert should be declared.

Related guidance concerning classification of rapidly escalating events or conditions is provided in Regulatory Issue Summary (RIS) 2007-02, *Clarification of NRC Guidance for Emergency Notifications During Quickly Changing Events* (ref. 4.1.2).

3.2.2 Consideration of Mode Changes During Classification

The mode in effect at the time that an event or condition occurred, and prior to any plant or operator response, is the mode that determines whether or not an IC is applicable. If an event or condition occurs, and results in a mode change before the emergency is declared, the emergency classification level is still based on the mode that existed at the time that the event or condition was initiated (and not when it was declared). Once a different mode is reached, any new event or condition, not related to the original event or condition, requiring emergency classification should be evaluated against the ICs and EALs applicable to the operating mode at the time of the new event or condition.

For events that occur in Cold Shutdown or Refueling, escalation is via EALs that are applicable in the Cold Shutdown or Refueling modes, even if Hot Shutdown (or a higher mode) is entered during the subsequent plant response. In particular, the fission product barrier EALs are applicable only to events that initiate in the Hot Shutdown mode or higher.

3.2.3 Classification of Imminent Conditions

Although EALs provide specific thresholds, the Site Emergency Coordinator must remain alert to events or conditions that could lead to meeting or exceeding an EAL within a relatively short period of time (i.e., a change in the ECL is IMMINENT). If, in the judgment of the Site Emergency Coordinator, meeting an EAL is IMMINENT, the emergency classification should be made as if the EAL has been met. While applicable to all emergency classification levels, this approach is particularly important at the higher emergency classification levels since it provides additional time for implementation of protective measures.

3.2.4 Emergency Classification Level Upgrading and Downgrading

An ECL may be downgraded when the event or condition that meets the highest IC and EAL no longer exists, and other site-specific downgrading requirements are met. If downgrading the ECL is deemed appropriate, the new ECL would then be based on a lower applicable IC(s) and EAL(s). The ECL may also simply be terminated.

As noted above, guidance concerning classification of rapidly escalating events or conditions is provided in RIS 2007-02 (ref. 4.1.2).

3.2.5 Classification of Short-Lived Events

Event-based ICs and EALs define a variety of specific occurrences that have potential or actual safety significance. By their nature, some of these events may be short-lived and, thus, over before the emergency classification assessment can be completed. If an event occurs that meets or exceeds an EAL, the associated ECL must be declared regardless of its continued presence at the time of declaration. Examples of such events include an earthquake or a failure of the reactor protection system to automatically scram the reactor followed by a successful manual scram.

3.2.6 Classification of Transient Conditions

Many of the ICs and/or EALs employ time-based criteria. These criteria will require that the IC/EAL conditions be present for a defined period of time before an emergency declaration is warranted. In cases where no time-based criterion is specified, it is recognized that some transient conditions may cause an EAL to be met for a brief period of time (e.g., a few seconds to a few minutes). The following guidance should be applied to the classification of these conditions.

EAL momentarily met during expected plant response - In instances where an EAL is briefly met during an expected (normal) plant response, an emergency declaration is not warranted provided that associated systems and components are operating as expected, and operator actions are performed in accordance with procedures.

EAL momentarily met but the condition is corrected prior to an emergency declaration – If an operator takes prompt manual action to address a condition, and the action is successful in correcting the condition prior to the emergency declaration, then the applicable EAL is not considered met and the associated emergency declaration is not required. For illustrative purposes, consider the following example:

> An ATWS occurs and the high pressure ECCS systems fail to automatically start. RPV level rapidly decreases and the plant enters an inadequate core cooling condition (a potential loss of both the fuel clad and RCS barriers). If an operator manually starts a high pressure ECCS system in accordance with an EOP step and clears the inadequate core cooling condition prior to an emergency declaration, then the classification should be based on the ATWS only.

It is important to stress that the 15-minute emergency classification assessment period (process clock) is not a "grace period" during which a classification may be delayed to allow the performance of a corrective action that would obviate the need to classify the event. Emergency classification assessments must be deliberate and timely, with no undue delays. The provision discussed above addresses only those rapidly evolving situations when an operator is able to take a successful corrective action prior to the Site Emergency Coordinator completing the review and steps necessary to make the emergency declaration. This provision is included to ensure that any public protective actions resulting from the emergency classification are truly warranted by the plant conditions.

3.2.7 After-the-Fact Discovery of an Emergency Event or Condition

In some cases, an EAL may be met but the emergency classification was not made at the time of the event or condition. This situation can occur when personnel discover that an event or condition existed which met an EAL, but no emergency was declared, and the event or condition no longer exists at the time of discovery. This may be due to the event or condition not being recognized at the time or an error that was made in the emergency classification process.

In these cases, no emergency declaration is warranted; however, the guidance contained in NUREG-1022 (ref. 4.1.3) is applicable. Specifically, the event should be reported to the NRC in accordance with 10 CFR § 50.72 (ref. 4.1.4) within one hour of the discovery of the undeclared event or condition. The licensee should also notify appropriate State and local agencies in accordance with the agreed upon arrangements.

3.2.8 Retraction of an Emergency Declaration

Guidance on the retraction of an emergency declaration reported to the NRC is discussed in NUREG-1022 (ref. 4.1.3).

4.0 REFERENCES

4.1 Developmental

4.1.1 NEI 99-01 Revision 6, Methodology for the Development of Emergency Action Levels for Non-Passive Reactors, ADAMS Accession Number ML12326A805

Deleted: ML110240324

4.1.2 RIS 2007-02 Clarification of NRC Guidance for Emergency Notifications During Quickly Changing Events, February 2, 2007.

4.1.3 NUREG-1022 Event Reporting Guidelines: 10CFR50.72 and 50.73

4.1.4 10 § CFR 50.72 Immediate Notification Requirements for Operating Nuclear Power Reactors

4.1.5 10 § CFR 50.73 License Event Report System

4.1.6 BNP Radiological Emergency Response Plan Figure 1-1.1 Brunswick Site Plan

4.1.7 BNP Radiological Emergency Response Plan Figure 1-1.3 Brunswick Site Building and Onsite Emergency Facility Locations

4.1.8 Technical Specifications Table 1.1-1 Modes

4.1.9 Technical Specifications Section 3.6 Containment Systems

4.1.10 PRO-NGGC-0201 NGG Procedure Writers Guide

4.1.11 NSIR/DPR-ISG-01 Interim Staff Guidance, Emergency Planning for Nuclear Power Plants

4.1.12 NGGM-PM-0028 Transnuclear NUHOMS Dry Fuel Storage Program Manual

4.2 Implementing

4.2.1 0PEP-02.1 Initial Emergency Actions

4.2.2 NEI 99-01 Rev. 6 to BNP EAL Comparison Matrix

4.2.3 BNP EAL Matrix

5.0 DEFINITIONS, ACRONYMS & ABBREVIATIONS

5.1 Definitions (ref. 4.1.1 except as noted)

Selected terms used in Initiating Condition and Emergency Action Level statements are set in all capital letters (e.g., ALL CAPS). These words are defined terms that have specific meanings as used in this document. The definitions of these terms are provided below.

Alert

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 PAG exposure levels.

Can/Cannot Be Maintained Above/Below

The value of an identified parameter is/is not able to be held within the specified limit. The determination requires an evaluation of system performance and availability in relation to parameter values and trends. An instruction prescribing action when a parameter cannot be maintained above or below a specified limit neither requires nor prohibits anticipatory action-depending upon plant conditions, the action may be taken as soon as it is determined that the limit will ultimately be exceeded, or delayed until the limit is actually reached. Once the parameter does exceed the limit, however, the action must be performed; it may not be delayed while attempts are made to restore the parameter to within the desired control band.

Can/Cannot Be Restored Above/Below

The value of an identified parameter is/is not able to be brought within the specified limit. The determination requires an evaluation of system performance and availability in relation to parameter values and trends. An instruction prescribing action when a value cannot be restored and maintained above or below a specified limit does not require immediate action simply because the current values is outside the range, but does not permit extended operation beyond the limit; the action must be taken as soon as it is apparent that the specified range cannot be attained.

Confinement Boundary

The barrier(s) between spent fuel and the environment once the spent fuel is processed for dry storage. As related to the BNP ISFSI, Confinement Boundary is defined as the Dry Shielded Canister (DSC) (Ref. 4.1.12).

Containment Closure

The procedurally defined conditions or actions taken to secure Primary or Secondary Containment and its associated structures, systems, and components as a functional barrier to fission product release under shutdown conditions.

As applied to BNP, Containment Closure is established when either Primary Containment is Operable per Section 3.6.1.1 of Technical Specifications or Secondary Containment is considered functional per the requirements of 0AP-022, BNP Outage Risk Management.

Emergency Action Level (EAL)

A pre-determined, site-specific, observable threshold for an Initiating Condition that, when met or exceeded, places the plant in a given emergency classification level.

Emergency Classification Level (ECL)

One of a set of names or titles established by the US Nuclear Regulatory Commission (NRC) for grouping off-normal events or conditions according to (1) potential or actual effects or consequences, and (2) resulting onsite and offsite response actions. The emergency classification levels, in ascending order of severity, are:

- Unusual Event (UE)
- Alert
- Site Area Emergency (SAE)
- General Emergency (GE)

EPA PAGs

Environment Protection Agency Protective Action Guidelines. The EPA PAGs are expressed in terms of dose commitment: 1 Rem TEDE or 5 Rem CDE Thyroid. Actual or projected offsite exposures in excess of the EPA PAGs requires BNP to recommend protective actions for the general public to offsite planning agencies.

Explosion

A rapid, violent and catastrophic failure of a piece of equipment due to combustion, chemical reaction or overpressurization. A release of steam (from high energy lines or components) or an electrical component failure (caused by short circuits, grounding, arcing, etc.) should not automatically be considered an explosion. Such events require a post-event inspection to determine if the attributes of an explosion are present.

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.

Fission Product Barrier Threshold

A pre-determined, site-specific, observable threshold indicating the loss or potential loss of a fission product barrier.

Flooding

A condition where water is entering a room or area faster than installed equipment is capable of removal, resulting in a rise of water level within the room or area.

General Emergency

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 PAG exposure levels offsite for more than the immediate site area.

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 BNP 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 BNP. Non-terrorism-based EALs should be used to address such activities (i.e., this may include 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

The trajectory of events or conditions is such that an EAL will be met within a relatively short period of time regardless of mitigation or corrective actions.

Impede(d)

Personnel access to a room or area is hindered to an extent that extraordinary measures are necessary to facilitate entry of personnel into the affected room/area (e.g., requiring use of protective equipment, such as SCBAs, that is not routinely employed).

Initiating Condition (IC)

An event or condition that aligns with the definition of one of the four emergency classification levels by virtue of the potential or actual effects or consequences.

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.

Maintain

Take appropriate action to hold the value of an identified parameter within specified limits.

Normal Levels

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

Owner Controlled Area

Area depicted as the property boundary in BNP Radiological Emergency Response Plan Figure 1-1.1 Brunswick Site Plan (ref. 4.1.6).

Projectile

An object directed toward a Nuclear Power Plant that could cause concern for its continued operability, reliability, or personnel safety.

Protected Area

The double-fenced security area with intrusion detection devices immediately surrounding the plant structures. The Protected Area is depicted in BNP Radiological Emergency Response Plan Figure 1-1.3 Brunswick Site Building and Onsite Emergency Facility Locations (ref. 4.1.7).

RCS 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).

Refueling Pathway

The reactor refueling cavity, spent fuel pool and fuel transfer canal (cattle chute) comprise the refueling pathway.

Restore

Take the appropriate action required to return the value of an identified parameter to the applicable limits

Safety System

A system required for safe plant operation, cooling down the plant and/or placing it in the cold shutdown condition, including the ECCS. These are typically systems classified as safety-related (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 Area Emergency

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 PAG exposure levels beyond the SITE BOUNDARY.

Site Boundary

Area as depicted in BNP Radiological Emergency Response Plan Figure 1-1.1 Brunswick Site Plan (ref. 4.1.6).

Unisolable

An open or breached system line that cannot be isolated, remotely or locally.

Unplanned

A parameter change or an event that is not 1) the result of an intended evolution or 2) an expected plant response to a transient. The cause of the parameter change or event may be known or unknown.

Unusual Event

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.

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 a component or structure that is readily observable without measurements, testing, or analysis. The visual impact of the damage is sufficient to cause concern regarding the operability or reliability of the affected component or structure.

5.2 Abbreviations/Acronyms

°F .... Degrees Fahrenheit
° .... Degrees
AC .... Alternating Current
AOP .... Abnormal Operating Procedure
APRM .... Average Power Range Meter
ATWS .... Anticipated Transient Without Scram
BNP .... Brunswick Nuclear Plant
BWR .... Boiling Water Reactor
BWROG .... Boiling Water Reactor Owners Group
CDE .... Committed Dose Equivalent
CFR .... Code of Federal Regulations
CS .... Core Spray
DBA .... Design Basis Accident
DC .... Direct Current
EAL .... Emergency Action Level
ECCS .... Emergency Core Cooling System
ECL .... Emergency Classification Level
EOF .... Emergency Operations Facility
EOP .... Emergency Operating Procedure
EPA .... Environmental Protection Agency
EPG .... Emergency Procedure Guideline
EPIP .... Emergency Plan Implementing Procedure
ESF .... Engineered Safety Feature
FAA .... Federal Aviation Administration
FBI .... Federal Bureau of Investigation
FEMA .... Federal Emergency Management Agency
FSAR .... Final Safety Analysis Report
GE .... General Emergency
HCTL .... Heat Capacity Temperature Limit
HPCI .... High Pressure Coolant Injection
IC .... Initiating Condition
IPEEE .... Individual Plant Examination of External Events (Generic Letter 88-20)
ISFSI .... Independent Spent Fuel Storage Installation

K_{eff} .... Effective Neutron Multiplication Factor
LCO .... Limiting Condition of Operation
LER .... Licensee Event Report
LOCA .... Loss of Coolant Accident
LPSI .... Low Pressure Safety Injection
LWR .... Light Water Reactor
MPC .... Maximum Permissible Concentration/Multi-Purpose Canister
MPH .... Miles Per Hour
MSIV .... Main Steam Isolation Valve
MSL .... Main Steam Line
mR, mRem, mrem, mREM .... milli-Roentgen Equivalent Man
MW .... Megawatt
NEI .... Nuclear Energy Institute
NESP .... National Environmental Studies Project
NPP .... Nuclear Power Plant
NRC .... Nuclear Regulatory Commission
NSSS .... Nuclear Steam Supply System
NORAD .... North American Aerospace Defense Command
(NO)UE .... Notification of Unusual Event
OBE .... Operating Basis Earthquake
OCA .... Owner Controlled Area
ODCM/ODAM .... Offsite Dose Calculation (Assessment) Manual
ORO .... Offsite Response Organization
PA .... Protected Area
PRA/PSA .... Probabilistic Risk Assessment / Probabilistic Safety Assessment
PWR .... Pressurized Water Reactor
PSIG .... Pounds per Square Inch Gauge
R .... Roentgen
RB .... Reactor Building
RCIC .... Reactor Core Isolation Cooling
RCS .... Reactor Coolant System
Rem, rem, REM .... Roentgen Equivalent Man
RETS .... Radiological Effluent Technical Specifications
RPS .... Reactor Protection System

RPV .... Reactor Pressure Vessel
RWCU .... Reactor Water Cleanup
SAR .... Safety Analysis Report
SBGTS .... Stand-By Gas Treatment System
SBO .... Station Blackout
SCBA .... Self-Contained Breathing Apparatus
SEC .... Site Emergency Coordinator
SPDS .... Safety Parameter Display System
SRO .... Senior Reactor Operator
TEDE .... Total Effective Dose Equivalent
TAF .... Top of Active Fuel
TSC .... Technical Support Center

6.0 BNP-TO-NEI 99-01 Rev. 6 EAL CROSS-REFERENCE

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

BNP	NEI 99-01 Rev. 6	
EAL	IC	Example EAL
RU1.1	AU1	1, 2
RU1.2	AU1	3
RU2.1	AU2	1
RA1.1	AA1	1
RA1.2	AA1	2
RA1.3	AA1	3
RA1.4	AA1	4
RA2.1	AA2	1
RA2.2	AA2	2
RA2.3	AA2	3
RA3.1	AA3	1
RA3.2	AA3	2
RS1.1	AS1	1
RS1.2	AS1	2
RS1.3	AS1	3
RS2.1	AS2	1
RG1.1	AG1	1
RG1.2	AG1	2
RG1.3	AG1	3
RG2.1	AG2	1

BNP	NEI 99-01 Rev. 6	
EAL	IC	Example EAL
CU1.1	CU1	1
CU1.2	CU1	2
CU2.1	CU2	1
CU3.1	CU3	1
CU3.2	CU3	2
CU4.1	CU4	1
CU5.1	CU5	1, 2, 3
CA1.1	CA1	1
CA1.2	CA1	2
CA2.1	CA2	1
CA3.1	CA3	1, 2
CA6.1	CA6	1
CS1.1	CS1	1
CS1.2	CS1	2
CS1.3	CS1	3
CG1.1	CG1	1
CG1.2	CG1	2
FA1.1	FA1	1
FS1.1	FS1	1
FG1.1	FG1	1
HU1.1	HU1	1
HU1.2	HU1	2
HU1.3	HU1	3
HU2.1	HU2	1
HU3.1	HU3	1

BNP	NEI 99-01 Rev. 6	
EAL	**IC**	**Example EAL**
HU3.2	HU3	2
HU3.3	HU3	3
HU3.4	HU3	4
HU3.5	HU3	5
HU4.1	HU4	1
HU4.2	HU4	2
HU4.3	HU4	3
HU4.4	HU4	4
HU7.1	HU7	1
HA1.1	HA1	1
HA1.2	HA1	2
HA5.1	HA5	1
HA6.1	HA6	1
HA7.1	HA7	1
HS1.1	HS1	1
HS6.1	HS6	1
HS7.1	HS7	1
HG1.1	HG1	1
HG7.1	HG7	1
SU1.1	SU1	1
SU3.1	SU2	1
SU4.1	SU3	1
SU4.2	SU3	2
SU5.1	SU4	1, 2, 3
SU6.1	SU5	1

BNP	NEI 99-01 Rev. 6	
EAL	IC	Example EAL
SU6.2	SU5	2
SU7.1	SU6	1, 2, 3
SA1.1	SA1	1
SA3.1	SA2	1
SA6.1	SA5	1
SA8.1	SA9	1
SS1.1	SS1	1
SS2.1	SS8	1
SS6.1	SS5	1
SG1.1	SG1	1
SG1.2	SG8	1
EU1.1	E-HU1	1

7.0 ATTACHMENTS

7.1 Attachment 1, Emergency Action Level Technical Bases

7.2 Attachment 2, Fission Product Barrier Matrix and Basis

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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. Radiological Effluent

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. Irradiated Fuel Event

Conditions indicative of a loss of adequate shielding or damage to irradiated fuel may preclude access to vital plant areas or result in radiological releases that warrant emergency classification.

3. Area Radiation Levels

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

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Category: R – Abnormal Rad Levels / Rad Effluent

Subcategory: 1 – Radiological Effluent

Initiating Condition: Release of gaseous or liquid radioactivity > 2 times the ODCM limits for 60 minutes or longer

EAL:

RU1.1 Unusual Event

Reading on **any** Table R-1 effluent radiation monitor > column "UE" for ≥ 60 min. (Notes 1, 2, 3)

Note 1: The SEC should declare the event promptly upon determining that time limit has been exceeded, or will likely be exceeded.

Note 2: If an ongoing release is detected and the release start time is unknown, assume that the release duration has exceeded the specified time limit.

Note 3: If the effluent flow past an effluent monitor is known to have stopped, indicating that the release path is isolated, the effluent monitor reading is no longer VALID for classification purposes.

Table R-1 Effluent Monitor Classification Thresholds

	Release Point	Monitor	GE	SAE	Alert	UE
Gaseous	Main Stack Rad	D12-RM-23S	2.13E+09 μCi/sec	2.13E+08 μCi/sec	2.13E+07 μCi/sec	1.80E+06 μCi/sec
Gaseous	Reactor Bldg Vent Noble Gas	CAC-AQH-1264-3	---	---	---	6.14E+04 cpm
Gaseous	Turbine Bldg Vent	D12-RM-23	1.07E+08 μCi/sec	1.07E+07 μCi/sec	1.07E+06 μCi/sec	1.13E+04 μCi/sec
Liquid	Service Water Effluent Rad	D12-RM-K605	---	---	---	2 x hi alarm
Liquid	Radwaste Effluent Rad	D12-RM-K604	---	---	---	2 x hi-hi alarm

Mode Applicability:

All

Deleted: 204

Definition(s):

None

Basis:

Gaseous Releases

The column "UE" gaseous release values in Table R-1 represent two times the appropriate ODCM release rate limits associated with the specified monitors (ref. 2, 3).

Instrumentation that may be used to assess this EAL is listed below (ref. 1):

- Main Stack Monitoring System Noble Gas Activity Monitor 2-D12-RM-23S (1/2-D12-RR-4599-4)
- Reactor Building Ventilation Monitoring System Noble Gas Activity Monitor 1(2)-CAC-AQH-1264-3
- Turbine Building Ventilation Monitoring System Noble Gas Activity Monitor 1(2)-D12-RM-23 (1/2-DL12-RR-4548-4)

Liquid Releases

Instrumentation that may be used to assess this EAL is listed below:

- Liquid Radwaste Radioactivity Monitor 2-D12-RM-K604 (batch release)
- Main Service Water Effluent Radioactivity Monitor 1(2)-D12-RM-K605 (continuous release)

The Liquid Radwaste Radioactivity Monitor Hi-Hi alarm automatically closes Radwaste Liquid Effluent Discharge Valves D12-V27A and 27B. The Hi-Hi alarm setpoint is set in accordance with the ODCM and includes a conservative reduction factor of 20 to the ODCM release rate limit (ref. 1, 2).

The Main Service Water Effluent Radioactivity Monitor High alarm setpoint is set in accordance with the ODCM and ensures continuous liquid releases do not exceed ODCM Section 7.3.3 limits.

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This IC addresses a potential decrease in the level of safety of the plant as indicated by a low-level radiological release that exceeds regulatory commitments for an extended period of time (e.g., an uncontrolled release). It includes any gaseous or liquid radiological release, monitored or un-monitored, including those for which a radioactivity discharge permit is normally prepared.

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

Radiological effluent EALs are also included to provide a basis for classifying events and conditions that cannot be readily or appropriately classified on the basis of plant conditions alone. The inclusion of both plant condition and radiological effluent EALs more fully addresses the spectrum of possible accident events and conditions.

Classification based on effluent monitor readings assumes that a release path to the environment is established. If the effluent flow past an effluent monitor is known to have stopped due to actions to isolate the release path, then the effluent monitor reading is no longer valid for classification purposes.

Releases should not be prorated or averaged. For example, a release exceeding 4 times release limits for 30 minutes does not meet the EAL.

This EAL addresses normally occurring continuous radioactivity releases from monitored gaseous or liquid effluent pathways.

Escalation of the emergency classification level would be via IC RA1.

BNP Basis Reference(s):

1. BNP ODCM Appendix E Radioactive Liquid and Gaseous Effluent Monitoring Instrumentation Numbers
2. BNP Offsite Dose Calculation Manual
3. EP-EALCALC-BNP-0801 Radiological Gaseous Effluent Values (EALs RG1, RS1, RA1 and RU1)
4. NEI 99-01 AU1

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Category: R – Abnormal Rad Levels / Rad Effluent

Subcategory: 1 – Radiological Effluent

Initiating Condition: Release of gaseous or liquid radioactivity greater than 2 times the ODCM limits for 60 minutes or longer.

EAL:

RU1.2	**Unusual Event**
Sample analysis for a gaseous or liquid release indicates a concentration or release rate > 2 x ODCM limits for ≥ 60 min. (Notes 1, 2)	

Note 1: The SEC should declare the event promptly upon determining that time limit has been exceeded, or will likely be exceeded.

Note 2: If an ongoing release is detected and the release start time is unknown, assume that the release duration has exceeded the specified time limit.

Mode Applicability:

All

Definition(s):

None

Basis:

This IC addresses a potential decrease in the level of safety of the plant as indicated by a low-level radiological release that exceeds regulatory commitments for an extended period of time (e.g., an uncontrolled release). It includes any gaseous or liquid radiological release, monitored or un-monitored, including those for which a radioactivity discharge permit is normally prepared.

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

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Radiological effluent EALs are also included to provide a basis for classifying events and conditions that cannot be readily or appropriately classified on the basis of plant conditions alone. The inclusion of both plant condition and radiological effluent EALs more fully addresses the spectrum of possible accident events and conditions.

Releases should not be prorated or averaged. For example, a release exceeding 4 times release limits for 30 minutes does not meet the EAL.

This EAL addresses uncontrolled gaseous or liquid releases that are detected by sample analyses or environmental surveys, particularly on unmonitored pathways (e.g., spills of radioactive liquids into storm drains, heat exchanger leakage in river water systems, etc.).

Escalation of the emergency classification level would be via IC RA1.

BNP Basis Reference(s):

1. BNP ODCM Appendix E Radioactive Liquid and Gaseous Effluent Monitoring Instrumentation Numbers
2. BNP Offsite Dose Calculation Manual
3. NEI 99-01 AU1

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Category: R – Abnormal Rad Levels / Rad Effluent

Subcategory: 1 – Radiological Effluent

Initiating Condition: Release of gaseous or liquid radioactivity resulting in offsite dose greater than 10 mrem TEDE or 50 mrem thyroid CDE

EAL:

RA1.1 Alert

In the absence of real-time dose assessment, reading on **any** Table R-1 effluent radiation monitor > column "ALERT" for ≥ 15 min. (Notes 1, 2, 3, 4)

Note 1: The SEC should declare the event promptly upon determining that time limit has been exceeded, or will likely be exceeded.

Note 2: If an ongoing release is detected and the release start time is unknown, assume that the release duration has exceeded the specified time limit.

Note 3: If the effluent flow past an effluent monitor is known to have stopped, indicating that the release path is isolated, the effluent monitor reading is no longer VALID for classification purposes.

Note 4 The pre-calculated effluent monitor values presented in EALs RA1.1, RS1.1 and RG1.1 should be used for emergency classification assessments until the results from a dose assessment using actual meteorology are available.

Table R-1 Effluent Monitor Classification Thresholds

	Release Point	Monitor	GE	SAE	Alert	UE
Gaseous	Main Stack Rad	D12-RM-23S	2.13E+09 μCi/sec	2.13E+08 μCi/sec	2.13E+07 μCi/sec	1.80E+06 μCi/sec
Gaseous	Reactor Bldg Vent Noble Gas	CAC-AQH-1264-3	---	---	---	6.14E+04 cpm
Gaseous	Turbine Bldg Vent	D12-RM-23	1.07E+08 μCi/sec	1.07E+07 μCi/sec	1.07E+06 μCi/sec	1.13E+04 μCi/sec
Liquid	Service Water Effluent Rad	D12-RM-K605	---	---	---	2 x hi alarm
Liquid	Radwaste Effluent Rad	D12-RM-K604	---	---	---	2 x hi-hi alarm

Mode Applicability:

All

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Definition(s):

None

Basis:

This EAL address gaseous radioactivity releases, that for whatever reason, cause effluent radiation monitor readings corresponding to site boundary doses that exceed either (ref. 2, 3):

- 10 mRem TEDE
- 50 mRem CDE Thyroid

The column "ALERT" gaseous effluent release values in Table R-1 correspond to calculated doses of 1% (10% of the SAE thresholds) of the EPA Protective Action Guidelines (TEDE or CDE Thyroid).

Instrumentation that may be used to assess this EAL is listed below (ref 1):

- Main Stack Monitoring System Noble Gas Activity Monitor 2-D12-RM-23S (1/2-D12-RR-4599-4)
- Turbine Building Ventilation Monitoring System Noble Gas Activity Monitor 1(2)-D12-RM-23 (1/2-DL12-RR-4548-4)

This IC addresses a release of gaseous or liquid radioactivity that results in projected or actual offsite doses greater than or equal to 1% of the EPA Protective Action Guides (PAGs). It includes both monitored and un-monitored releases. Releases of this magnitude represent an actual or potential substantial degradation of the level of safety of the plant as indicated by a radiological release that significantly exceeds regulatory limits (e.g., a significant uncontrolled release).

Radiological effluent EALs are also included to provide a basis for classifying events and conditions that cannot be readily or appropriately classified on the basis of plant conditions alone. The inclusion of both plant condition and radiological effluent EALs more fully addresses the spectrum of possible accident events and conditions.

The TEDE dose is set at 1% of the EPA PAG of 1,000 mrem while the 50 mrem thyroid CDE was established in consideration of the 1:5 ratio of the EPA PAG for TEDE and thyroid CDE.

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Classification based on effluent monitor readings assumes that a release path to the environment is established. If the effluent flow past an effluent monitor is known to have stopped due to actions to isolate the release path, then the effluent monitor reading is no longer valid for classification purposes.

Escalation of the emergency classification level would be via IC RS1.

BNP Basis Reference(s):

1. BNP ODCM Appendix E Radioactive Liquid and Gaseous Effluent Monitoring Instrumentation Numbers
2. BNP Offsite Dose Calculation Manual
3. EP-EALCALC-BNP-0801 Radiological Gaseous Effluent Values (EALs RG1, RS1, RA1 and RU1)
4. NEI 99-01 AA1

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Category: R – Abnormal Rad Levels / Rad Effluent

Subcategory: 1 – Radiological Effluent

Initiating Condition: Release of gaseous or liquid radioactivity resulting in offsite dose greater than 10 mrem TEDE or 50 mrem thyroid CDE

EAL:

RA1.2	**Alert**
Dose assessment using actual meteorology indicates doses > 10 mrem TEDE or 50 mrem thyroid CDE at or beyond the SITE BOUNDARY (Note 4)	

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Note 4: The pre-calculated effluent monitor values presented in EALs RA1.1, RS1.1 and RG1.1 should be used for emergency classification assessments until the results from a dose assessment using actual meteorology are available.

Deleted: Note 3: . If the effluent flow past an effluent monitor is known to have stopped, indicating that the release path is isolated, the effluent monitor reading is no longer VALID for classification purposes.¶

Mode Applicability:

All

Definition(s):

SITE BOUNDARY - Area as depicted in BNP Radiological Emergency Response Plan Figure 1-1.1 Brunswick Site Plan

Basis:

Dose assessments are performed by computer-based methods (ref. 1, 2, 3)

This IC addresses a release of gaseous or liquid radioactivity that results in projected or actual offsite doses greater than or equal to 1% of the EPA Protective Action Guides (PAGs). It includes both monitored and un-monitored releases. Releases of this magnitude represent an actual or potential substantial degradation of the level of safety of the plant as indicated by a radiological release that significantly exceeds regulatory limits (e.g., a significant uncontrolled release).

Radiological effluent EALs are also included to provide a basis for classifying events and conditions that cannot be readily or appropriately classified on the basis of plant conditions alone. The inclusion of both plant condition and radiological effluent EALs more fully addresses the spectrum of possible accident events and conditions.

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The TEDE dose is set at 1% of the EPA PAG of 1,000 mrem while the 50 mrem thyroid CDE was established in consideration of the 1:5 ratio of the EPA PAG for TEDE and thyroid CDE.

Classification based on effluent monitor readings assumes that a release path to the environment is established. If the effluent flow past an effluent monitor is known to have stopped due to actions to isolate the release path, then the effluent monitor reading is no longer valid for classification purposes.

Escalation of the emergency classification level would be via IC RS1.

BNP Basis Reference(s):

1. EMG-NGGC-0002 Off-site Dose Assessment
2. 0PEP-03.4.7 Automation of Off-Site Dose Projections
3. 0E&RC-03.4.8, Offsite Dose Projections for Monitored Releases
3. NEI 99-01 AA1

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Category: R – Abnormal Rad Levels / Rad Effluent

Subcategory: 1 – Radiological Effluent

Initiating Condition: Release of gaseous or liquid radioactivity resulting in offsite dose greater than 10 mrem TEDE or 50 mrem thyroid CDE

EAL:

RA1.3	**Alert**
Analysis of a liquid effluent sample indicates a concentration or release rate that would result in doses > 10 mrem TEDE or 50 mrem thyroid CDE at or beyond the SITE BOUNDARY for 60 min. of exposure (Notes 1, 2)	

Note 1: The SEC should declare the event promptly upon determining that time limit has been exceeded, or will likely be exceeded.

Note 2: If an ongoing release is detected and the release start time is unknown, assume that the release duration has exceeded the specified time limit.

Mode Applicability:

All

Definition(s):

SITE BOUNDARY - Area as depicted in BNP Radiological Emergency Response Plan Figure 1-1.1 Brunswick Site Plan

Basis:

Dose assessments based on liquid releases are performed per Offsite Dose Calculation Manual (ref. 1).

This IC addresses a release of gaseous or liquid radioactivity that results in projected or actual offsite doses greater than or equal to 1% of the EPA Protective Action Guides (PAGs). It includes both monitored and un-monitored releases. Releases of this magnitude represent an actual or potential substantial degradation of the level of safety of the plant as indicated by a radiological release that significantly exceeds regulatory limits (e.g., a significant uncontrolled release).

Radiological effluent EALs are also included to provide a basis for classifying events and conditions that cannot be readily or appropriately classified on the basis of plant conditions alone. The inclusion of both plant condition and radiological effluent EALs more fully addresses the spectrum of possible accident events and conditions.

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The TEDE dose is set at 1% of the EPA PAG of 1,000 mrem while the 50 mrem thyroid CDE was established in consideration of the 1:5 ratio of the EPA PAG for TEDE and thyroid CDE.

Classification based on effluent monitor readings assumes that a release path to the environment is established. If the effluent flow past an effluent monitor is known to have stopped due to actions to isolate the release path, then the effluent monitor reading is no longer valid for classification purposes.

Escalation of the emergency classification level would be via IC RS1.

BNP Basis Reference(s):

1. BNP Offsite Dose Calculation Manual
2. NEI 99-01 AA1

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Category: R – Abnormal Rad Levels / Rad Effluent

Subcategory: 1 – Radiological Effluent

Initiating Condition: Release of gaseous or liquid radioactivity resulting in offsite dose greater than 10 mrem TEDE or 50 mrem thyroid CDE

EAL:

RA1.4 Alert

Field survey results indicate **EITHER** of the following at or beyond the SITE BOUNDARY:

- Closed window dose rates > 10 mR/hr expected to continue for ≥ 60 min.
- Analyses of field survey samples indicate thyroid CDE > 50 mrem for 60 min. of inhalation.

(Notes 1, 2)

Note 1: The SEC should declare the event promptly upon determining that time limit has been exceeded, or will likely be exceeded.

Note 2: If an ongoing release is detected and the release start time is unknown, assume that the release duration has exceeded the specified time limit.

Mode Applicability:

All

Definition(s):

SITE BOUNDARY - Area as depicted in BNP Radiological Emergency Response Plan Figure 1-1.1 Brunswick Site Plan

Basis:

0PEP-02.6.6, Environmental Monitoring Team Leader and 0PEP-03.5.5 Environmental Monitoring and Plume Tracking provide guidance for emergency or post-accident radiological environmental monitoring (ref. 1, 2).

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This IC addresses a release of gaseous or liquid radioactivity that results in projected or actual offsite doses greater than or equal to 1% of the EPA Protective Action Guides (PAGs). It includes both monitored and un-monitored releases. Releases of this magnitude represent an actual or potential substantial degradation of the level of safety of the plant as indicated by a radiological release that significantly exceeds regulatory limits (e.g., a significant uncontrolled release).

Radiological effluent EALs are also included to provide a basis for classifying events and conditions that cannot be readily or appropriately classified on the basis of plant conditions alone. The inclusion of both plant condition and radiological effluent EALs more fully addresses the spectrum of possible accident events and conditions.

The TEDE dose is set at 1% of the EPA PAG of 1,000 mrem while the 50 mrem thyroid CDE was established in consideration of the 1:5 ratio of the EPA PAG for TEDE and thyroid CDE.

Classification based on effluent monitor readings assumes that a release path to the environment is established. If the effluent flow past an effluent monitor is known to have stopped due to actions to isolate the release path, then the effluent monitor reading is no longer valid for classification purposes.

Escalation of the emergency classification level would be via IC RS1.

BNP Basis Reference(s):

1. 0PEP-02.6.6, Environmental Monitoring Team Leader
2. 0PEP-03.5.5 Environmental Monitoring and Plume Tracking
3. NEI 99-01 AA1

Deleted: 17
Deleted: 204

Category: R – Abnormal Rad Levels / Rad Effluent

Subcategory: 1 – Radiological Effluent

Initiating Condition: Release of gaseous radioactivity resulting in offsite dose greater than 100 mrem TEDE or 500 mrem thyroid CDE

EAL:

RS1.1 Site Area Emergency

In the absence of real-time dose assessment, reading on **any** Table R-1 effluent radiation monitor > column "SAE" for ≥ 15 min.
(Notes 1, 2, 3, 4)

Note 1: The SEC should declare the event promptly upon determining that time limit has been exceeded, or will likely be exceeded.

Note 2: If an ongoing release is detected and the release start time is unknown, assume that the release duration has exceeded the specified time limit.

Note 3: If the effluent flow past an effluent monitor is known to have stopped, indicating that the release path is isolated, the effluent monitor reading is no longer VALID for classification purposes.

Note 4: The pre-calculated effluent monitor values presented in EALs RA1.1, RS1.1 and RG1.1 should be used for emergency classification assessments until the results from a dose assessment using actual meteorology are available.

Table R-1 Effluent Monitor Classification Thresholds

	Release Point	Monitor	GE	SAE	Alert	UE
Gaseous	Main Stack Rad	D12-RM-23S	2.13E+09 μCi/sec	2.13E+08 μCi/sec	2.13E+07 μCi/sec	1.80E+06 μCi/sec
Gaseous	Reactor Bldg Vent Noble Gas	CAC-AQH-1264-3	----	----	----	6.14E+04 cpm
Gaseous	Turbine Bldg Vent	D12-RM-23	1.07E+08 μCi/sec	1.07E+07 μCi/sec	1.07E+06 μCi/sec	1.13E+04 μCi/sec
Liquid	Service Water Effluent Rad	D12-RM-K605	----	----	----	2 x hi alarm
Liquid	Radwaste Effluent Rad	D12-RM-K604	----	----	----	2 x hi-hi alarm

Mode Applicability:

All

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Definition(s):

None

Basis:

This EAL address gaseous radioactivity releases, that for whatever reason, cause effluent radiation monitor readings corresponding to site boundary doses that exceed either (ref. 2):

- 100 mRem TEDE
- 500 mRem CDE Thyroid

The column "SAE" gaseous effluent release values in Table R-1 correspond to calculated doses of 10% of the EPA Protective Action Guidelines (TEDE or CDE Thyroid).

Instrumentation that may be used to assess this EAL is listed below (ref 1):

- Main Stack Monitoring System Noble Gas Activity Monitor 2-D12-RM-23S (1/2-D12-RR-4599-4)
- Turbine Building Ventilation Monitoring System Noble Gas Activity Monitor 1(2)-D12-RM-23 (1/2-DL12-RR-4548-4)

This IC addresses a release of gaseous radioactivity that results in projected or actual offsite doses greater than or equal to 10% of the EPA Protective Action Guides (PAGs). It includes both monitored and un-monitored releases. Releases of this magnitude are associated with the failure of plant systems needed for the protection of the public.

Radiological effluent EALs are also included to provide a basis for classifying events and conditions that cannot be readily or appropriately classified on the basis of plant conditions alone. The inclusion of both plant condition and radiological effluent EALs more fully addresses the spectrum of possible accident events and conditions.

The TEDE dose is set at 10% of the EPA PAG of 1,000 mrem while the 500 mrem thyroid CDE was established in consideration of the 1:5 ratio of the EPA PAG for TEDE and thyroid CDE.

Classification based on effluent monitor readings assumes that a release path to the environment is established. If the effluent flow past an effluent monitor is known to have stopped due to actions to isolate the release path, then the effluent monitor reading is no longer valid for classification purposes.

Escalation of the emergency classification level would be via IC RG1.

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BNP Basis Reference(s):

1. BNP ODCM Appendix E Radioactive Liquid and Gaseous Effluent Monitoring Instrumentation Numbers
2. EP-EALCALC-BNP-0801 Radiological Gaseous Effluent Values (EALs RG1, RS1, RA1 and RU1)
3. NEI 99-01 AS1

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Deleted: 204

Category: R – Abnormal Rad Levels / Rad Effluent

Subcategory: 1 – Radiological Effluent

Initiating Condition: Release of gaseous radioactivity resulting in offsite dose greater than 100 mrem TEDE or 500 mrem thyroid CDE

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 (Note 4)	

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Note 4: The pre-calculated effluent monitor values presented in EALs RA1.1, RS1.1 and RG1.1 should be used for emergency classification assessments until the results from a dose assessment using actual meteorology are available.

Deleted: Note 3: . If the effluent flow past an effluent monitor is known to have stopped, indicating that the release path is isolated, the effluent monitor reading is no longer VALID for classification purposes.¶

Mode Applicability:

All

Definition(s):

SITE BOUNDARY - Area as depicted in BNP Radiological Emergency Response Plan Figure 1-1.1 Brunswick Site Plan

Basis:

Dose assessments are performed by computer-based methods (ref. 1, 2, 3)

This IC addresses a release of gaseous radioactivity that results in projected or actual offsite doses greater than or equal to 10% of the EPA Protective Action Guides (PAGs). It includes both monitored and un-monitored releases. Releases of this magnitude are associated with the failure of plant systems needed for the protection of the public.

Radiological effluent EALs are also included to provide a basis for classifying events and conditions that cannot be readily or appropriately classified on the basis of plant conditions alone. The inclusion of both plant condition and radiological effluent EALs more fully addresses the spectrum of possible accident events and conditions.

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The TEDE dose is set at 10% of the EPA PAG of 1,000 mrem while the 500 mrem thyroid CDE was established in consideration of the 1:5 ratio of the EPA PAG for TEDE and thyroid CDE.

Classification based on effluent monitor readings assumes that a release path to the environment is established. If the effluent flow past an effluent monitor is known to have stopped due to actions to isolate the release path, then the effluent monitor reading is no longer valid for classification purposes.

Escalation of the emergency classification level would be via IC RG1.

BNP Basis Reference(s):

1. EMG-NGGC-0002 Off-site Dose Assessment
2. 0PEP-03.4.7 Automation of Off-Site Dose Projections
3. 0E&RC-03.4.8, Offsite Dose Projections for Monitored Releases
4. NEI 99-01 AS1

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Category: R – Abnormal Rad Levels / Rad Effluent

Subcategory: 1 – Radiological Effluent

Initiating Condition: Release of gaseous radioactivity resulting in offsite dose greater than 100 mrem TEDE or 500 mrem thyroid CDE

EAL:

RS1.3 Site Area Emergency

Field survey results indicate EITHER of the following at or beyond the SITE BOUNDARY:

- Closed window dose rates > 100 mR/hr expected to continue for ≥ 60 min.
- Analyses of field survey samples indicate thyroid CDE > 500 mrem for 60 min. of inhalation.

(Notes 1, 2)

Note 1: The SEC should declare the event promptly upon determining that time limit has been exceeded, or will likely be exceeded.

Note 2: If an ongoing release is detected and the release start time is unknown, assume that the release duration has exceeded the specified time limit.

Mode Applicability:

All

Definition(s):

SITE BOUNDARY - Area as depicted in BNP Radiological Emergency Response Plan Figure 1-1.1 Brunswick Site Plan

Basis:

0PEP-02.6.6, Environmental Monitoring Team Leader and 0PEP-03.5.5 Environmental Monitoring and Plume Tracking provide guidance for emergency or post-accident radiological environmental monitoring (ref. 1, 2).

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This IC addresses a release of gaseous radioactivity that results in projected or actual offsite doses greater than or equal to 10% of the EPA Protective Action Guides (PAGs). It includes both monitored and un-monitored releases. Releases of this magnitude are associated with the failure of plant systems needed for the protection of the public.

Radiological effluent EALs are also included to provide a basis for classifying events and conditions that cannot be readily or appropriately classified on the basis of plant conditions alone. The inclusion of both plant condition and radiological effluent EALs more fully addresses the spectrum of possible accident events and conditions.

The TEDE dose is set at 10% of the EPA PAG of 1,000 mrem while the 500 mrem thyroid CDE was established in consideration of the 1:5 ratio of the EPA PAG for TEDE and thyroid CDE.

Escalation of the emergency classification level would be via IC RG1.

BNP Basis Reference(s):

1. 0PEP-02.6.6, Environmental Monitoring Team Leader
2. 0PEP-03.5.5 Environmental Monitoring and Plume Tracking
3. NEI 99-01 AS1

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Category: R – Abnormal Rad Levels / Rad Effluent

Subcategory: 1 – Radiological Effluent

Initiating Condition: Release of gaseous radioactivity resulting in offsite dose greater than 1,000 mrem TEDE or 5,000 mrem thyroid CDE

EAL:

RG1.1 General Emergency

In the absence of real-time dose assessment, reading on **any** Table R-1 effluent radiation monitor > column "GE" for ≥ 15 min.
(Notes 1, 2, 3, 4)

Note 1: The SEC should declare the event promptly upon determining that time limit has been exceeded, or will likely be exceeded.

Note 2: If an ongoing release is detected and the release start time is unknown, assume that the release duration has exceeded the specified time limit.

Note 3: If the effluent flow past an effluent monitor is known to have stopped, indicating that the release path is isolated, the effluent monitor reading is no longer VALID for classification purposes.

Note 4: The pre-calculated effluent monitor values presented in EALs RA1.1, RS1.1 and RG1.1 should be used for emergency classification assessments until the results from a dose assessment using actual meteorology are available.

Table R-1 Effluent Monitor Classification Thresholds

	Release Point	Monitor	GE	SAE	Alert	UE
Gaseous	Main Stack Rad	D12-RM-23S	2.13E+09 μCi/sec	2.13E+08 μCi/sec	2.13E+07 μCi/sec	1.80E+06 μCi/sec
Gaseous	Reactor Bldg Vent Noble Gas	CAC-AQH-1264-3	---	---	---	6.14E+04 cpm
Gaseous	Turbine Bldg Vent	D12-RM-23	1.07E+08 μCi/sec	1.07E+07 μCi/sec	1.07E+06 μCi/sec	1.13E+04 μCi/sec
Liquid	Service Water Effluent Rad	D12-RM-K605	---	---	---	2 x hi alarm
Liquid	Radwaste Effluent Rad	D12-RM-K604	---	---	---	2 x hi-hi alarm

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Mode Applicability:

All

Definition(s):

None

Basis:

This EAL address gaseous radioactivity releases, that for whatever reason, cause effluent radiation monitor readings corresponding to site boundary doses that exceed either (ref. 2):

- 1000 mRem TEDE
- 5000 mRem CDE Thyroid

The column "SAE" gaseous effluent release values in Table R-1 correspond to calculated doses of 100% of the EPA Protective Action Guidelines (TEDE or CDE Thyroid).

Instrumentation that may be used to assess this EAL is listed below (ref 1):

- Main Stack Monitoring System Noble Gas Activity Monitor 2-D12-RM-23S (1/2-D12-RR-4599-4)
- Turbine Building Ventilation Monitoring System Noble Gas Activity Monitor 1(2)-D12-RM-23 (1/2-DL12-RR-4548-4)

This IC addresses a release of gaseous radioactivity that results in projected or actual offsite doses greater than or equal to the EPA Protective Action Guides (PAGs). It includes both monitored and un-monitored releases. Releases of this magnitude will require implementation of protective actions for the public.

Radiological effluent EALs are also included to provide a basis for classifying events and conditions that cannot be readily or appropriately classified on the basis of plant conditions alone. The inclusion of both plant condition and radiological effluent EALs more fully addresses the spectrum of possible accident events and conditions.

The TEDE dose is set at the EPA PAG of 1,000 mrem 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.

Classification based on effluent monitor readings assumes that a release path to the environment is established. If the effluent flow past an effluent monitor is known to have stopped due to actions to isolate the release path, then the effluent monitor reading is no longer valid for classification purposes.

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BNP Basis Reference(s):

1. BNP ODCM Appendix E Radioactive Liquid and Gaseous Effluent Monitoring Instrumentation Numbers
2. EP-EALCALC-BNP-0801 Radiological Gaseous Effluent Values (EALs RG1, RS1, RA1 and RU1)
3. NEI 99-01 AG1

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Category: R – Abnormal Rad Levels / Rad Effluent

Subcategory: 1 – Radiological Effluent

Initiating Condition: Release of gaseous radioactivity resulting in offsite dose greater than 1,000 mrem TEDE or 5,000 mrem thyroid CDE

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 (Note 4)

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Note 4: The pre-calculated effluent monitor values presented in EALs RA1.1, RS1.1 and RG1.1 should be used for emergency classification assessments until the results from a dose assessment using actual meteorology are available.

Deleted: Note 3: . If the effluent flow past an effluent monitor is known to have stopped, indicating that the release path is isolated, the effluent monitor reading is no longer VALID for classification purposes.¶

Applicability:

All

Definition(s):

SITE BOUNDARY - Area as depicted in BNP Radiological Emergency Response Plan Figure 1-1.1 Brunswick Site Plan

BNP Basis:

Dose assessments are performed by computer-based methods (ref. 1, 2, 3)

This IC addresses a release of gaseous radioactivity that results in projected or actual offsite doses greater than or equal to the EPA Protective Action Guides (PAGs). It includes both monitored and un-monitored releases. Releases of this magnitude will require implementation of protective actions for the public.

Radiological effluent EALs are also included to provide a basis for classifying events and conditions that cannot be readily or appropriately classified on the basis of plant conditions alone. The inclusion of both plant condition and radiological effluent EALs more fully addresses the spectrum of possible accident events and conditions.

The TEDE dose is set at the EPA PAG of 1,000 mrem 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.

Classification based on effluent monitor readings assumes that a release path to the environment is established. If the effluent flow past an effluent monitor is known to have stopped due to actions to isolate the release path, then the effluent monitor reading is no longer valid for classification purposes.

BNP Basis Reference(s):

1. EMG-NGGC-0002 Off-site Dose Assessment
2. 0PEP-03.4.7 Automation of Off-Site Dose Projections
3. 0E&RC-03.4.8, Offsite Dose Projections for Monitored Releases
4. NEI 99-01 AG1

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Category: R – Abnormal Rad Levels / Rad Effluent

Subcategory: 1 – Radiological Effluent

Initiating Condition: Release of gaseous radioactivity resulting in offsite dose greater than 1,000 mrem TEDE or 5,000 mrem thyroid CDE

EAL:

RG1.3 General Emergency

Field survey results indicate EITHER of the following at or beyond the SITE BOUNDARY:

- Closed window dose rates > 1,000 mR/hr expected to continue for ≥ 60 min.
- Analyses of field survey samples indicate thyroid CDE > 5,000 mrem for 60 min. of inhalation.

(Notes 1, 2)

Note 1: The SEC should declare the event promptly upon determining that time limit has been exceeded, or will likely be exceeded.

Note 2: If an ongoing release is detected and the release start time is unknown, assume that the release duration has exceeded the specified time limit.

Mode Applicability:

All

Definition(s):

SITE BOUNDARY - Area as depicted in BNP Radiological Emergency Response Plan Figure 1-1.1 Brunswick Site Plan

Basis:

0PEP-02.6.6, Environmental Monitoring Team Leader and 0PEP-03.5.5 Environmental Monitoring and Plume Tracking provide guidance for emergency or post-accident radiological environmental monitoring (ref. 1, 2).

This IC addresses a release of gaseous radioactivity that results in projected or actual offsite doses greater than or equal to the EPA Protective Action Guides (PAGs). It includes both monitored and un-monitored releases. Releases of this magnitude will require implementation of protective actions for the public.

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Radiological effluent EALs are also included to provide a basis for classifying events and conditions that cannot be readily or appropriately classified on the basis of plant conditions alone. The inclusion of both plant condition and radiological effluent EALs more fully addresses the spectrum of possible accident events and conditions.

The TEDE dose is set at the EPA PAG of 1,000 mrem 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.

BNP Basis Reference(s):

1. 0PEP-02.6.6, Environmental Monitoring Team Leader
2. 0PEP-03.5.5 Environmental Monitoring and Plume Tracking
3. NEI 99-01 AG1

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Category: R – Abnormal Rad Levels / Rad Effluent

Subcategory: 2 – Irradiated Fuel Event

Initiating Condition: Unplanned loss of water level above irradiated fuel

EAL:

RU2.1 Unusual Event

UNPLANNED water level drop in the REFUELING PATHWAY as indicated by low water level alarm (A-04 6-6) or indication

AND

UNPLANNED rise in area radiation levels as indicated by **any** of the following radiation monitors:

- ARM Channel 26 New Fuel Vault
- ARM Channel 27 North of Fuel Pool
- ARM Channel 28 Between Reactor and Fuel Pool
- ARM Channel 29 Cask Wash Area

Mode Applicability:

All

Definition(s):

UNPLANNED-. A parameter change or an event that is not 1) the result of an intended evolution or 2) an expected plant response to a transient. The cause of the parameter change or event may be known or unknown.

REFUELING PATHWAY-. The reactor refueling cavity, spent fuel pool and fuel transfer canal (cattle chute) comprise the refueling pathway.

Basis:

The spent fuel pool low water level alarm setpoint is actuated by level switch G410-LSHL-N001 at a setpoint of 37' 5". Water level restoration instructions are performed in accordance with 1(2)APP A-04 6-6 Fuel Pool Level Low (ref. 1).

When the spent fuel pool and reactor cavity are connected, there could exist the possibility of uncovering irradiated fuel. Therefore, this EAL is applicable for conditions in which irradiated fuel is being transferred to and from the RPV and spent fuel pool.

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This IC addresses a decrease in water level above irradiated fuel sufficient to cause elevated radiation levels. This condition could be a precursor to a more serious event and is also indicative of a minor loss in the ability to control radiation levels within the plant. It is therefore a potential degradation in the level of safety of the plant.

A water level decrease will be primarily determined by indications from available level instrumentation. Other sources of level indications may include reports from plant personnel (e.g., from a refueling crew) or video camera observations (if available). A significant drop in the water level may also cause an increase in the radiation levels of adjacent areas that can be detected by monitors in those locations.

The effects of planned evolutions should be considered. For example, a refueling bridge area radiation monitor reading may increase due to planned evolutions such as lifting of the reactor vessel head or movement of a fuel assembly. Note that this EAL is applicable only in cases where the elevated reading is due to an unplanned loss of water level.

A drop in water level above irradiated fuel within the reactor vessel may be classified in accordance Recognition Category C during the Cold Shutdown and Refueling modes.

Escalation of the emergency classification level would be via IC RA2.

BNP Basis Reference(s):

1. 1(2)APP-A-04 6-6 Fuel Pool Level Low
2. DBD-11 Radiation Monitoring System
3. NEI 99-01 AU2

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Category: R – Abnormal Rad Levels / Rad Effluent

Subcategory: 2 – Irradiated Fuel Event

Initiating Condition: Significant lowering of water level above, or damage to, irradiated fuel

EAL:

RA2.1	**Unusual Event**
Uncovery of irradiated fuel in the REFUELING PATHWAY	

Mode Applicability:

All

Definition(s):

REFUELING PATHWAY-. The reactor refueling cavity, spent fuel pool and fuel transfer canal (cattle chute) comprise the refueling pathway.

Basis:

None.

This IC addresses events that have caused imminent or actual damage to an irradiated fuel assembly, or a significant lowering of water level within the spent fuel pool. These events present radiological safety challenges to plant personnel and are precursors to a release of radioactivity to the environment. As such, they represent an actual or potential substantial degradation of the level of safety of the plant.

This EAL escalates from RU2.1 in that the loss of level, in the affected portion of the REFUELING PATHWAY, is of sufficient magnitude to have resulted in uncovery of irradiated fuel. Indications of irradiated fuel uncovery may include direct or indirect visual observation (e.g., reports from personnel or camera images), as well as significant changes in water and radiation levels, or other plant parameters. Computational aids may also be used (e.g., a boil-off curve). Classification of an event using this EAL should be based on the totality of available indications, reports and observations.

While an area radiation monitor could detect an increase in a dose rate due to a lowering of water level in some portion of the REFUELING PATHWAY, the reading may not be a reliable indication of whether or not the fuel is actually uncovered. To the degree possible, readings should be considered in combination with other available indications of inventory loss.

A drop in water level above irradiated fuel within the reactor vessel may be classified in accordance Recognition Category C during the Cold Shutdown and Refueling modes.

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Escalation of the emergency classification level would be via IC RS1.

BNP Basis Reference(s):

1. 1(2)APP-A4 6-6 (Fuel pool Level Low)
2. 1(2)APP-A7 2-2 (Reactor Water Level Hi/Low)
3. NEI 99-01 AA2

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Category: R – Abnormal Rad Levels / Rad Effluent

Subcategory: 2 – Irradiated Fuel Event

Initiating Condition: Significant lowering of water level above, or damage to, irradiated fuel

EAL:

RA2.2 Alert

Damage to irradiated fuel resulting in a release of radioactivity

AND

Any of the following radiation monitor indications:

- Reactor Bldg Vent Rad Monitor Channel A or B (> 3 mR/hr)
- ARM Channel 26 New Fuel Vault (> 6 mR/hr)
- ARM Channel 27 North of Fuel Pool (>10 mR/hr)
- ARM Channel 28 Between Reactor and Fuel Pool (> 1000 mR/hr)
- ARM Channel 29 Cask Wash Area (>40 mR/hr)

Mode Applicability:

All

Definition(s):

None

Basis:

The high alarm setpoints for the radiation monitors are (ref. 1, 2, 3 4):

- Reactor Building Exhaust Plenum Rad Monitor Channel A or B > 3 mR/hr
- ARM Channel 26 New Fuel Vault > 6 mR/hr
- ARM Channel 27 North of Fuel Pool > 10 mR/hr
- ARM Channel 28 Between Reactor and Fuel Pool > 1000 mR/hr
- ARM Channel 29 Cask Wash Area > 40mR/hr

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This IC addresses events that have caused imminent or actual damage to an irradiated fuel assembly, or a significant lowering of water level within the spent fuel pool. These events present radiological safety challenges to plant personnel and are precursors to a release of radioactivity to the environment. As such, they represent an actual or potential substantial degradation of the level of safety of the plant.

This IC applies to irradiated fuel that is licensed for dry storage up to the point that the loaded storage cask is sealed. Once sealed, damage to a loaded cask causing loss of the CONFINEMENT BOUNDARY is classified in accordance with EAL EU1.1.

Escalation of the emergency would be based on either Recognition Category R or C ICs.

This EAL addresses a release of radioactive material caused by mechanical damage to irradiated fuel. Damaging events may include the dropping, bumping or binding of an assembly, or dropping a heavy load onto an assembly. A rise in readings on radiation monitors should be considered in conjunction with in-plant reports or observations of a potential fuel damaging event (e.g., a fuel handling accident).

Escalation of the emergency classification level would be via IC RS1.

BNP Basis Reference(s):

1. 1(2)APP-UA-03 3-7
2. 1(2)APP-UA-03 4-5
3. 1(2)APP-UA-03 4-7
4. DBD-11 Radiation Monitoring System
5. NEI 99-01 AA2

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Category: R – Abnormal Rad Levels / Rad Effluent

Subcategory: 2 – Irradiated Fuel Event

Initiating Condition: Significant lowering of water level above, or damage to, irradiated fuel

EAL:

RA2.3	**Alert**
Lowering of spent fuel pool level to ≤ 105 ft. 3 in. ele.	

Mode Applicability:

All

Definition(s):

None

Basis:

Post-Fukushima order EA-12-051 required the installation of reliable SFP level indication capable of identifying normal level (Level 1 – 116 ft. 1 in. ele.), SFP level 10 ft. above the top of the fuel racks (Level 2 – 105 ft. 3 in. ele.) and SFP level at the top of the fuel racks (Level 3 – 95 ft. 3 in. ele.).

An indicated level of 105 ft. 3 in. corresponds to the Level 2 setpoint (Ref. 1).

This IC addresses events that have caused imminent or actual damage to an irradiated fuel assembly, or a significant lowering of water level within the spent fuel pool. These events present radiological safety challenges to plant personnel and are precursors to a release of radioactivity to the environment. As such, they represent an actual or potential substantial degradation of the level of safety of the plant.

Escalation of the emergency would be based on either Recognition Category R or C ICs.

Spent fuel pool water level at this value is within the lower end of the level range necessary to prevent significant dose consequences from direct gamma radiation to personnel performing operations in the vicinity of the spent fuel pool. This condition reflects a significant loss of spent fuel pool water inventory and thus it is also a precursor to a loss of the ability to adequately cool the irradiated fuel assembles stored in the pool.

Escalation of the emergency classification level would be via IC RS1.

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BNP Basis Reference(s):

1. PCHG-DESG Engineering Change 0000089578R0
2. NEI 99-01 AA2

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Category: R – Abnormal Rad Levels / Rad Effluent

Subcategory: 2 – Irradiated Fuel Event

Initiating Condition: Spent fuel pool level at the top of the fuel racks

EAL:

RS2.1	**Site Area Emergency**
Lowering of spent fuel pool level to ≤ 95 ft. 3 in. ele.	

Mode Applicability:

All

Definition(s):

None

Basis:

Post-Fukushima order EA-12-051 required the installation of reliable SFP level indication capable of identifying normal level (Level 1 – 116 ft. 1 in. ele.), SFP level 10 ft. above the top of the fuel racks (Level 2 – 105 ft. 3 in. ele.) and SFP level at the top of the fuel racks (Level 3 – 95 ft. 3 in. ele.).

An indicated level of 95 ft. 3 in. corresponds to the Level 3 setpoint (Ref. 1).

This EAL addresses a significant loss of spent fuel pool inventory control and makeup capability leading to IMMINENT fuel damage. This condition entails major failures of plant functions needed for protection of the public and thus warrant a Site Area Emergency declaration.

It is recognized that this IC would likely not be met until well after another Site Area Emergency IC was met; however, it is included to provide classification diversity.

Escalation of the emergency classification level would be via IC RG1 or RG2.

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BNP Basis Reference(s):

1. PCHG-DESG Engineering Change 0000089578R0
2. NEI 99-01 AS2

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Category: R – Abnormal Rad Levels / Rad Effluent

Subcategory: 2 – Irradiated Fuel Event

Initiating Condition: Spent fuel pool level cannot be restored to at least the top of the fuel racks for 60 minutes or longer

EAL:

RG2.1	**General Emergency**
Spent fuel pool level cannot be restored ≥ 95 ft. 3 in. ele. for ≥ 60 min. (Note 1)	

Note 1: The SEC should declare the event promptly upon determining that time limit has been exceeded, or will likely be exceeded.

Mode Applicability:

All

Definition(s):

None

Basis:

Post-Fukushima order EA-12-051 required the installation of reliable SFP level indication capable of identifying normal level (Level 1 – 116 ft. 1 in. ele.), SFP level 10 ft. above the top of the fuel racks (Level 2 – 105 ft. 3 in. ele.) and SFP level at the top of the fuel racks (Level 3 – 95 ft. 3 in. ele.).

An indicated level of 95 ft. 3 in. corresponds to the Level 3 setpoint (Ref. 1).

This EAL addresses a significant loss of spent fuel pool inventory control and makeup capability leading to a prolonged uncovery of spent fuel. This condition will lead to fuel damage and a radiological release to the environment.

It is recognized that this IC would likely not be met until well after another General Emergency IC was met; however, it is included to provide classification diversity.

BNP Basis Reference(s):

1. PCHG-DESG Engineering Change 0000089578R0
2. NEI 99-01 AG2

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Category: R – Abnormal Rad Levels / Rad Effluent

Subcategory: 3 – Area Radiation Levels

Initiating Condition: Radiation levels that IMPEDE access to equipment necessary for normal plant operations, cooldown or shutdown

EAL:

RA3.1	**Alert**

Dose rates > 15 mR/hr in **EITHER** of the following areas:

Control Room (ARM Channel 1-1)

OR

Central Alarm Station (by survey)

Mode Applicability:

All

Definition(s):

None

Basis:

Areas that meet this threshold include the Control Room and the Central Alarm Station (CAS). ARM Channel 1-1 monitors the Control room for area radiation (ref. 1). The CAS is included in this EAL because of its' importance to permitting access to areas required to assure safe plant operations.

There is no permanently installed CAS area radiation monitors that may be used to assess this EAL threshold. Therefore this threshold must be assessed via local radiation survey for the CAS.

This IC addresses elevated radiation levels in certain plant rooms/areas sufficient to preclude or impede personnel from performing actions necessary to maintain normal plant operation, or to perform a normal plant cooldown and shutdown. As such, it represents an actual or potential substantial degradation of the level of safety of the plant. The Site Emergency Coordinator should consider the cause of the increased radiation levels and determine if another IC may be applicable.

Escalation of the emergency classification level would be via Recognition Category R, C or F ICs.

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BNP Basis Reference(s):

1. 1(2)APP-UA-03 6-7 (Area RAD Control Room Hi)
2. NEI 99-01 AA3

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Category: R – Abnormal Rad Levels / Rad Effluent

Subcategory: 3 – Area Radiation Levels

Initiating Condition: Radiation levels that IMPEDE access to equipment necessary for normal plant operations, cooldown or shutdown

EAL:

RA3.2 Alert

An UNPLANNED event results in radiation levels that prohibit or IMPEDE access to **any** Table R-2 rooms or areas (Note 5)

Note 5: If the equipment in the listed room or area was already inoperable or out-of-service before the event occurred, then no emergency classification is warranted.

Table R-2 Safe Operation & Shutdown Areas	
Room/Area	**Mode Applicability**
Reactor Building -17' North RHR Unit-1 & 2	3, 4, 5
Reactor Building -17' South RHR Unit-1 & 2	3, 4, 5
Reactor Building 20' East & West MCC Areas Unit-1 & 2	3, 4, 5
Reactor Building 20' Pipe Tunnel Unit-1 & 2	3, 4, 5

Mode Applicability:

All

Definition(s):

IMPEDE(D)- Personnel access to a room or area is hindered to an extent that extraordinary measures are necessary to facilitate entry of personnel into the affected room/area (e.g., requiring use of protective equipment, such as SCBAs, that is not routinely employed).

UNPLANNED- A parameter change or an event that is not 1) the result of an intended evolution or 2) an expected plant response to a transient. The cause of the parameter change or event may be known or unknown.

Basis:

If the equipment in the listed room or area was already inoperable, or out-of-service, before the event occurred, then no emergency should be declared since the event will have no adverse impact beyond that already allowed by Technical Specifications at the time of the event.

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The list of plant rooms or areas with entry-related mode applicability identified specify those rooms or areas that contain equipment which require a manual/local action as specified in operating procedures used for normal plant operation, cooldown and shutdown. Rooms or areas in which actions of a contingent or emergency nature would be performed (e.g., an action to address an off-normal or emergency condition such as emergency repairs, corrective measures or emergency operations) are not included. In addition, the list specifies the plant mode(s) during which entry would be required for each room or area (ref. 1).

This IC addresses elevated radiation levels in certain plant rooms/areas sufficient to preclude or impede personnel from performing actions necessary to maintain normal plant operation, or to perform a normal plant cooldown and shutdown. As such, it represents an actual or potential substantial degradation of the level of safety of the plant. The Site Emergency Coordinator should consider the cause of the increased radiation levels and determine if another IC may be applicable.

For RA3.2, an Alert declaration is warranted if entry into the affected room/area is, or may be, procedurally required during the plant operating mode in effect at the time of the elevated radiation levels. The emergency classification is not contingent upon whether entry is actually necessary at the time of the increased radiation levels. Access should be considered as impeded if extraordinary measures are necessary to facilitate entry of personnel into the affected room/area (e.g., installing temporary shielding, requiring use of non-routine protective equipment, requesting an extension in dose limits beyond normal administrative limits).

An emergency declaration is not warranted if any of the following conditions apply:

- The plant is in an operating mode different than the mode specified for the affected room/area (i.e., entry is not required during the operating mode in effect at the time of the elevated radiation levels). For example, the plant is in Mode 1 when the radiation increase occurs, and the procedures used for normal operation, cooldown and shutdown do not require entry into the affected room until Mode 4.
- The increased radiation levels are a result of a planned activity that includes compensatory measures which address the temporary inaccessibility of a room or area (e.g., radiography, spent filter or resin transfer, etc.).
- The action for which room/area entry is required is of an administrative or record keeping nature (e.g., normal rounds or routine inspections).
- The access control measures are of a conservative or precautionary nature, and would not actually prevent or impede a required action.

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- If the equipment in the listed room or area was already inoperable, or out-of-service, before the event occurred, then no emergency should be declared since the event will have no adverse impact beyond that already allowed by Technical Specifications at the time of the event.

Escalation of the emergency classification level would be via Recognition Category R, C or F ICs.

BNP Basis Reference(s):

1. Attachment 3 Safe Operation & Shutdown Areas Tables R-2 & H-2 Bases
2. NEI 99-01 AA3

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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 - Refueling, D – Defueled).

The events of this category pertain to the following subcategories:

1. RPV Level

Reactor Pressure Vessel water level is directly related to the status of adequate core cooling and, therefore, fuel clad integrity.

2. Loss of Emergency AC Power

Loss of essential 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 4160 VAC emergency buses.

3. RCS Temperature

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

4. Loss of Vital 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 or degraded voltage on the 125 VDC vital buses.

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5. Loss of Communications

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

6. Hazardous Event Affecting Safety Systems

Certain hazardous natural and technological events may result in visible damage to or degraded performance of safety systems warranting classification.

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Category: C – Cold Shutdown / Refueling System Malfunction

Subcategory: 1 – RPV Level

Initiating Condition: UNPLANNED loss of RPV inventory for 15 minutes or longer

EAL:

CU1.1	**Unusual Event**
UNPLANNED loss of reactor coolant results in RPV water level less than a required lower limit for ≥ 15 min. (Note 1)	

Note 1: The SEC should declare the event promptly upon determining that time limit has been exceeded, or will likely be exceeded.

Mode Applicability:

4 - Cold Shutdown, 5 - Refueling

Definition(s):

UNPLANNED-. A parameter change or an event that is not 1) the result of an intended evolution or 2) an expected plant response to a transient. The cause of the parameter change or event may be known or unknown.

Basis:

Figure C-1 illustrates the elevations of the RPV level instrument ranges (ref. 2).

With the plant in Cold Shutdown, RPV water level is normally maintained above the RPV low level scram setpoint of 166 in. above TAF (ref. 1, 3). However, if RPV level is being controlled below the RPV low level scram setpoint, or if level is being maintained in a designated band in the reactor vessel it is the inability to maintain level above the low end of the designated control band due to a loss of inventory resulting from a leak in the RCS that is the concern.

With the plant in Refueling mode, RPV water level is normally maintained at or above the reactor vessel flange (Technical Specification LCO 3.9.6 requires at least 23 ft of water above the top of the reactor vessel flange in the refueling cavity during refueling operations). The RPV flange is at an indicated level of 355 in. as indicated on the red scale of B21-LI-R605A/B Shutdown Range Reactor Water Level Indication (ref. 4).

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This IC addresses the inability to restore and maintain water level to a required minimum level (or the lower limit of a level band), or a loss of the ability to monitor RPV level concurrent with indications of coolant leakage. Either of these conditions is considered to be a potential degradation of the level of safety of the plant.

Refueling evolutions that decrease RCS water inventory are carefully planned and controlled. An UNPLANNED event that results in water level decreasing below a procedurally required limit warrants the declaration of an Unusual Event due to the reduced water inventory that is available to keep the core covered.

This EAL recognizes that the minimum required RPV level can change several times during the course of a refueling outage as different plant configurations and system lineups are implemented. This EAL is met if the minimum level, specified for the current plant conditions, cannot be maintained for 15 minutes or longer. The minimum level is typically specified in the applicable operating procedure but may be specified in another controlling document.

The 15-minute threshold duration allows sufficient time for prompt operator actions to restore and maintain the expected water level. This criterion excludes transient conditions causing a brief lowering of water level.

Continued loss of RCS inventory may result in escalation to the Alert emergency classification level via either IC CA1 or CA3.

BNP Basis Reference(s):

1. 0EOP-01-NL EOP/SAMG NUMERICAL LIMITS AND VALUES, Table 1E
2. SD-01.2 Reactor Vessel Instrumentation Figure 01.2-1 Reactor Water Level Instrument Ranges
3. 1(2) APP A7 2-2 (Reactor Water Level Hi/Low)
4. 0GP-06 Cold Shutdown to Refueling (Head Unbolted) step 5.1.14
5. NEI 99-01 CU1

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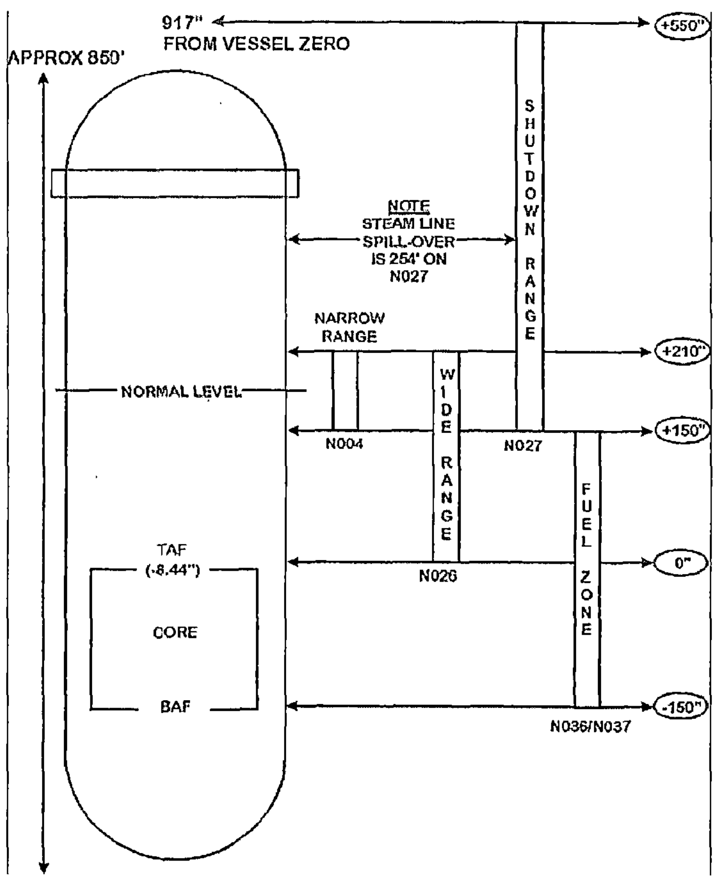
Figure C-1 RPV Levels (ref. 2)

Reactor Water Level Instrument Ranges



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Category: C – Cold Shutdown / Refueling System Malfunction

Subcategory: 1 – RPV Level

Initiating Condition: UNPLANNED loss of RPV inventory

EAL:

CU1.2 Unusual Event
RPV water level cannot be monitored AND UNPLANNED increase in **any** Table C-1 sump or tank levels due to a loss of RPV inventory

Table C-1 Sumps & Tanks
• Drywell Floor Drain Sump
• Drywell Equipment Drain Sump
• RB Floor Drain Sump
• RB Equipment Drain Sump
• Torus
• Visual Observation

Mode Applicability:

4 - Cold Shutdown, 5 – Refueling

Definition(s):

UNPLANNED-. A parameter change or an event that is not 1) the result of an intended evolution or 2) an expected plant response to a transient. The cause of the parameter change or event may be known or unknown.

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Basis:

In Cold Shutdown mode, the RCS will normally be intact and standard RPV level monitoring means are available. RPV level in the Refuel mode is normally monitored using the red scale of B21-LI-R605A/B Shutdown Range Reactor Water Level Indication.

In this EAL, all water level indication is unavailable and the RPV inventory loss must be detected by the leakage indications listed in Table C-1. 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 (ref. 1). A Reactor Building equipment or floor drain sump level rise may also be indicative of RCS inventory losses external to the Primary Containment from systems connected to the RPV. With RHR System operating in the Shutdown Cooling mode, an unexplained rise in torus level could be indicative of RHR valve misalignment or leakage (ref. 5). 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 that cannot be isolated could be indicative of a loss of RPV inventory.

This IC addresses the inability to restore and maintain water level to a required minimum level (or the lower limit of a level band), or a loss of the ability to monitor RPV level concurrent with indications of coolant leakage. Either of these conditions is considered to be a potential degradation of the level of safety of the plant.

Refueling evolutions that decrease RCS water inventory are carefully planned and controlled. An UNPLANNED event that results in water level decreasing below a procedurally required limit warrants the declaration of an Unusual Event due to the reduced water inventory that is available to keep the core covered.

This EAL addresses a condition where all means to determine RPV level have been lost. In this condition, operators may determine that an inventory loss is occurring by observing changes in sump and/or tank levels. Sump and/or tank level changes must be evaluated against other potential sources of water flow to ensure they are indicative of leakage from the RCS.

Continued loss of RCS inventory may result in escalation to the Alert emergency classification level via either IC CA1 or CA3.

BNP Basis Reference(s):

1. 0OP-47 Floor and Equipment Drain System Operating Procedure
2. 1OI-03.1 Control Room Operator Daily Surveillance Report
3. 2OI-03.2 Control Room Operator Daily Surveillance Report
4. 0AOP-14.0 Abnormal Primary Containment Conditions
5. 1(2)OP-17 Residual Heat Removal System Operating Procedure
6. NEI 99-01 CU1

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Category: C – Cold Shutdown / Refueling System Malfunction

Subcategory: 1 – RPV Level

Initiating Condition: Loss of RPV inventory

EAL:

CA1.1	**Alert**
Loss of RPV inventory as indicated by RPV water level < 105 in. above TAF (Level 2)	

Mode Applicability:

4 - Cold Shutdown, 5 – Refuel

Definition(s):

None

Basis:

The threshold RPV level of 105 in. is the low-low ECCS actuation setpoint (ref. 1). RPV level is normally monitored using the instruments in Figure C-1 (ref. 2).

When reactor vessel water level drops to 105 in. above TAF high pressure steam-driven injection sources HPCI (ECCS) and RCIC receive an initiation signal (ref. 1). Although these systems cannot restore RCS inventory in the cold condition, the Low-Low (Level 2) ECCS actuation setpoint is operationally significant and is indicative of a loss of RCS inventory significantly below the low level scram setpoint specified in CU1.1.

This IC addresses conditions that are precursors to a loss of the ability to adequately cool irradiated fuel (i.e., a precursor to a challenge to the fuel clad barrier). This condition represents a potential substantial reduction in the level of plant safety.

For this EAL, a lowering of water level below 105 in above TAF indicates that operator actions have not been successful in restoring and maintaining RPV water level. The heat-up rate of the coolant will increase as the available water inventory is reduced. A continuing decrease in water level will lead to core uncovery.

Although related, this EAL is concerned with the loss of RCS inventory and not the potential concurrent effects on systems needed for decay heat removal (e.g., loss of a Decay Heat Removal suction point). An increase in RCS temperature caused by a loss of decay heat removal capability is evaluated under IC CA3.

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If RPV water level continues to lower, then escalation to Site Area Emergency would be via IC CS1.

BNP Basis Reference(s):

1. 0EOP-01-NL EOP/SAMG NUMERICAL LIMITS AND VALUES, Table 1E
2. SD-01.2 Reactor Vessel Instrumentation Figure 01.2-1 Reactor Water Level Instrument Ranges

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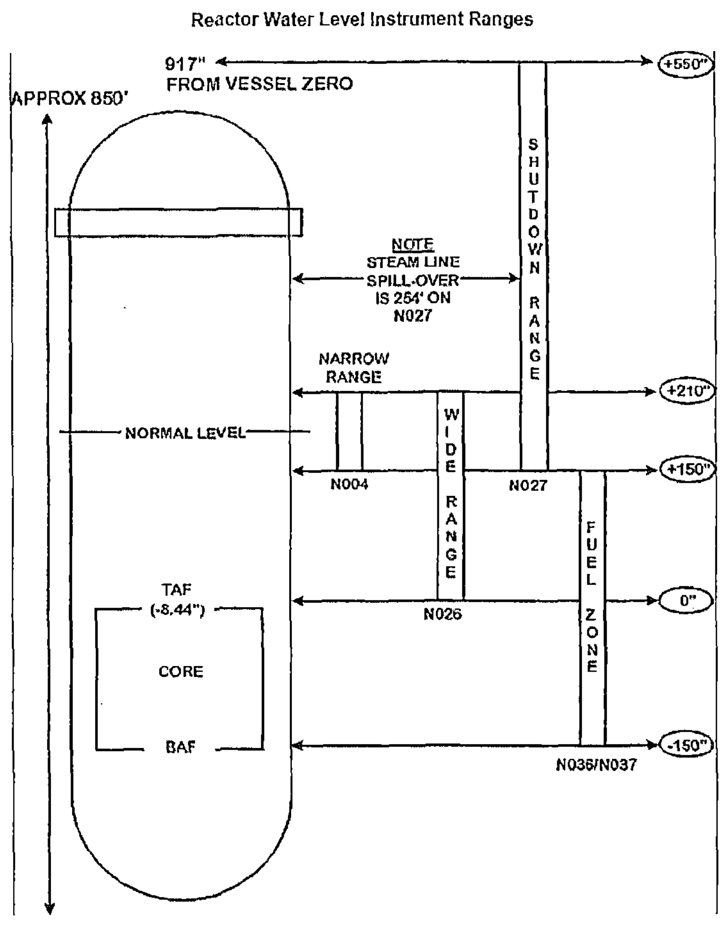
Figure C-1 RPV Levels (ref. 2)



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Category: C – Cold Shutdown / Refueling System Malfunction

Subcategory: 1 – RPV Level

Initiating Condition: Loss of RPV inventory

EAL:

CA1.2 Alert

RPV water level cannot be monitored for ≥ 15 min. (Note 1)

AND

UNPLANNED increase in **any** Table C-1 sump or tank levels due to a loss of RPV inventory

Table C-1 Sumps & Tanks
• Drywell Floor Drain Sump • Drywell Equipment Drain Sump • RB Floor Drain Sump • RB Equipment Drain Sump • Torus • Visual Observation

Mode Applicability:

4 - Cold Shutdown, 5 – Refuel

Definition(s):

UNPLANNED-. A parameter change or an event that is not 1) the result of an intended evolution or 2) an expected plant response to a transient. The cause of the parameter change or event may be known or unknown.

Basis:

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

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In this EAL, all water level indication would be unavailable, and the RPV inventory loss must be detected by the leakage indications listed in Table C-1. Sump level increases must be evaluated against other potential sources of leakage. Drywell equipment and floor drain sump level rise is the normal method of monitoring and calculating leakage from the RPV (ref. 1 thru 4). A Reactor Building equipment or floor drain sump level rise may also be indicative of RCS inventory losses external to the Primary Containment from systems connected to the RPV. With RHR System operating in the Shutdown Cooling mode, an unexplained rise in torus level could be indicative of RHR valve misalignment or leakage (ref. 5). 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 that cannot be isolated could be indicative of a loss of RPV inventory.

This IC addresses conditions that are precursors to a loss of the ability to adequately cool irradiated fuel (i.e., a precursor to a challenge to the fuel clad barrier). This condition represents a potential substantial reduction in the level of plant safety.

For this EAL, the inability to monitor RPV level may be caused by instrumentation and/or power failures, or water level dropping below the range of available instrumentation. If water level cannot be monitored, operators may determine that an inventory loss is occurring by observing changes in sump and/or tank levels. Sump and/or tank level changes must be evaluated against other potential sources of water flow to ensure they are indicative of leakage from the RCS.

The 15-minute duration for the loss of level indication was chosen because it is half of the EAL duration specified in IC CS1.

If the RCS inventory level continues to lower, then escalation to Site Area Emergency would be via IC CS1.

BNP Basis Reference(s):

1. 0OP-47 Floor and Equipment Drain System Operating Procedure
2. 1OI-03.1 Control Room Operator Daily Surveillance Report
3. 2OI-03.2 Control Room Operator Daily Surveillance Report
4. 0AOP-14.0 Abnormal Primary Containment Conditions
5. 1(2)OP-17 Residual Heat Removal System Operating Procedure
6. NEI 99-01 CA1

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Category: C – Cold Shutdown / Refueling System Malfunction

Subcategory: 1 – RPV Level

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

EAL:

CS1.1	**Site Area Emergency**
CONTAINMENT CLOSURE **not** established	
AND	
RPV level < 45 in. (Level 3)	

Mode Applicability:

4 - Cold Shutdown, 5 - Refuel

Definition(s):

CONTAINMENT CLOSURE - The procedurally defined conditions or actions taken to secure Primary or Secondary Containment and its associated structures, systems, and components as a functional barrier to fission product release under shutdown conditions.

As applied to BNP, Containment Closure is established when either Primary Containment is Operable per Section 3.6.1.1 of Technical Specifications or Secondary Containment is considered functional per the requirements of 0AP-022, BNP Outage Risk Management.

Basis:

RPV level is normally monitored using the instruments in Figure C-1 (ref. 2).

When RPV level decreases to 45 in., RPV water level is below the low-low-low ECCS actuation setpoint (Level 3) (ref. 1).

The magnitude of this loss of water indicates that makeup systems have not been effective and may not be capable of preventing further RPV water level decrease and potential core uncovery. The inability to restore and maintain level after reaching this setpoint infers a failure of the RCS barrier and Potential Loss of the Fuel Clad barrier.

This IC addresses a significant and prolonged loss of RCS inventory control and makeup capability leading to IMMINENT fuel damage. The lost inventory may be due to a RCS component failure, a loss of configuration control or prolonged boiling of reactor coolant.

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These conditions entail major failures of plant functions needed for protection of the public and thus warrant a Site Area Emergency declaration.

Following an extended loss of core decay heat removal and inventory makeup, decay heat will cause reactor coolant boiling and a further reduction in reactor vessel level. If RPV level cannot be restored, fuel damage is probable.

Outage/shutdown contingency plans typically provide for re-establishing or verifying CONTAINMENT CLOSURE following a loss of heat removal or RCS inventory control functions. The difference in the specified RPV levels of CS1.1 and CS1.2 reflect the fact that with CONTAINMENT CLOSURE established, there is a lower probability of a fission product release to the environment.

This EAL addresses concerns raised by Generic Letter 88-17, Loss of Decay Heat Removal; SECY 91-283, Evaluation of Shutdown and Low Power Risk Issues; NUREG-1449, Shutdown and Low-Power Operation at Commercial Nuclear Power Plants in the United States; and NUMARC 91-06, Guidelines for Industry Actions to Assess Shutdown Management.

Escalation of the emergency classification level would be via IC CG1 or RG1.

BNP Basis Reference(s):

1. 0EOP-01-NL EOP/SAMG NUMERICAL LIMITS AND VALUES, Table 1E
2. BNP Technical Specifications, Sections 3.6.1.1
3. 0AP-022, BNP Outage Risk Management, Section 6.5
4. SD-01.2 Reactor Vessel Instrumentation Figure 01.2-1 Reactor Water Level Instrument Ranges
5. NEI 99-01 CS1

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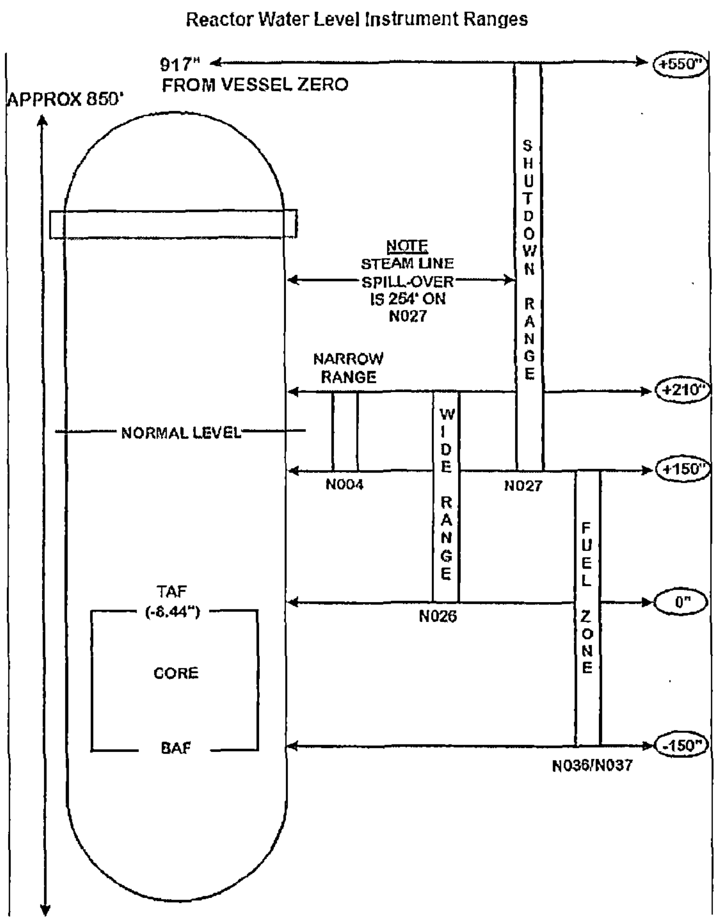
Figure C-1 RPV Levels (ref. 4)



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Category: C – Cold Shutdown / Refueling System Malfunction

Subcategory: 1 – RPV Level

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

EAL:

CS1.2	**Site Area Emergency**
CONTAINMENT CLOSURE established	
AND	
RPV level < TAF	

Mode Applicability:

4 - Cold Shutdown, 5 - Refuel

Definition(s):

CONTAINMENT CLOSURE - The procedurally defined conditions or actions taken to secure Primary or Secondary Containment and its associated structures, systems, and components as a functional barrier to fission product release under shutdown conditions.

As applied to BNP, Containment Closure is established when either Primary Containment is Operable per Section 3.6.1.1 of Technical Specifications or Secondary Containment is considered functional per the requirements of 0AP-022, BNP Outage Risk Management.

Basis:

When RPV level drops to the top of active fuel (TAF) (an indicated RPV level of ~ -8 in.), core uncovery starts to occur (ref. 1).

This IC addresses a significant and prolonged loss of RPV level control and makeup capability leading to IMMINENT fuel damage. The lost inventory may be due to a RCS component failure, a loss of configuration control or prolonged boiling of reactor coolant. These conditions entail major failures of plant functions needed for protection of the public and thus warrant a Site Area Emergency declaration.

Following an extended loss of core decay heat removal and inventory makeup, decay heat will cause reactor coolant boiling and a further reduction in reactor vessel level. If RPV level cannot be restored, fuel damage is probable.

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Outage/shutdown contingency plans typically provide for re-establishing or verifying CONTAINMENT CLOSURE following a loss of heat removal or RCS inventory control functions. The difference in the specified RCS/reactor vessel levels of CS1.1 and CS1.2 reflect the fact that with CONTAINMENT CLOSURE established, there is a lower probability of a fission product release to the environment.

This EAL addresses concerns raised by Generic Letter 88-17, Loss of Decay Heat Removal; SECY 91-283, Evaluation of Shutdown and Low Power Risk Issues; NUREG-1449, Shutdown and Low-Power Operation at Commercial Nuclear Power Plants in the United States; and NUMARC 91-06, Guidelines for Industry Actions to Assess Shutdown Management.

Escalation of the emergency classification level would be via IC CG1 or RG1.

BNP Basis Reference(s):

1. 0EOP-01-NL EOP/SAMG NUMERICAL LIMITS AND VALUES
2. BNP Technical Specifications, Sections 3.6.1.1 and 3.6.4.1
3. 0AP-022, BNP Outage Risk Management, Section 6.5

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Category: C – Cold Shutdown / Refueling System Malfunction

Subcategory: 1 – RPV Level

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

EAL:

CS1.3 Site Area Emergency

RPV water level cannot be monitored for ≥ 30 min. (Note 1)
AND
Core uncovery is indicated by **EITHER** of the following:

- UNPLANNED increase in **any** Table C-1 sump or tank levels due to a loss of RPV inventory
- UNPLANNED increase in ARM Channel 28 Between Reactor and Fuel Pool > 1000 mR/hr

Note 1: The SEC should declare the event promptly upon determining that time limit has been exceeded, or will likely be exceeded.

Table C-1 Sumps & Tanks
• Drywell Floor Drain Sump
• Drywell Equipment Drain Sump
• RB Floor Drain Sump
• RB Equipment Drain Sump
• Torus
• Visual Observation

Mode Applicability:

4 – Cold Shutdown, 5 – Refuel

Definition(s):

UNPLANNED-. A parameter change or an event that is not 1) the result of an intended evolution or 2) an expected plant response to a transient. The cause of the parameter change or event may be known or unknown.

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Basis:

If RPV level monitoring capability is unavailable, the RPV inventory loss must be detected by the leakage indications listed in Table C-1. Sump level increases must be evaluated against other potential sources of leakage. Drywell equipment and floor drain sump level rise is the normal method of monitoring and calculating leakage from the RPV (ref. 1 thru 4). A Reactor Building equipment or floor drain sump level rise may also be indicative of RCS inventory losses external to the Primary Containment from systems connected to the RPV. With RHR System operating in the Shutdown Cooling mode, an unexplained rise in torus level could be indicative of RHR valve misalignment or leakage (ref. 5). 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.

In the Refueling Mode, as water level in the reactor vessel lowers, the dose rate above the core will increase. The dose rate due to this core shine should result in indications on installed area radiation monitors. ARM Channel 28 Between Reactor and Fuel Pool is located on the Refuel Floor in the Reactor Building and is designed to provide monitoring of radiation due to a fuel handling event or loss of shielding during refueling operations. If this radiation monitor reaches and exceeds the alarm setpoint of 1,000 mr/hr, a loss of inventory with potential to uncover the core is likely to have occurred (Ref. 6, 7).

This IC addresses a significant and prolonged loss of reactor vessel/RCS inventory control and makeup capability leading to IMMINENT fuel damage. The lost inventory may be due to a RCS component failure, a loss of configuration control or prolonged boiling of reactor coolant. These conditions entail major failures of plant functions needed for protection of the public and thus warrant a Site Area Emergency declaration.

Following an extended loss of core decay heat removal and inventory makeup, decay heat will cause reactor coolant boiling and a further reduction in reactor vessel level. If RPV level cannot be restored, fuel damage is probable.

The 30-minute criterion is tied to a readily recognizable event start time (i.e., the total loss of ability to monitor level), and allows sufficient time to monitor, assess and correlate reactor and plant conditions to determine if core uncovery has actually occurred (i.e., to account for various accident progression and instrumentation uncertainties). It also allows sufficient time for performance of actions to terminate leakage, recover inventory control/makeup equipment and/or restore level monitoring.

The inability to monitor RPV level may be caused by instrumentation and/or power failures, or water level dropping below the range of available instrumentation. If water level cannot be monitored, operators may determine that an inventory loss is occurring by observing changes in sump and/or tank levels. Sump and/or tank level changes must be evaluated against other potential sources of water flow to ensure they are indicative of leakage from the RCS.

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This EAL addresses concerns raised by Generic Letter 88-17, Loss of Decay Heat Removal; SECY 91-283, Evaluation of Shutdown and Low Power Risk Issues; NUREG-1449, Shutdown and Low-Power Operation at Commercial Nuclear Power Plants in the United States; and NUMARC 91-06, Guidelines for Industry Actions to Assess Shutdown Management.

Escalation of the emergency classification level would be via IC CG1 or RG1

BNP Basis Reference(s):

1. 0OP-47 Floor and Equipment Drain System Operating Procedure
2. 1OI-03.1 Control Room Operator Daily Surveillance Report
3. 2OI-03.2 Control Room Operator Daily Surveillance Report
4. 0AOP-14.0 Abnormal Primary Containment Conditions
5. 1(2)OP-17 Residual Heat Removal System Operating Procedure
6. 1(2)APP-UA-03 4-7
7. DBD-11 Radiation Monitoring System
8. NEI 99-01 CS1

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Category: C – Cold Shutdown / Refueling System Malfunction

Subcategory: 1 – RPV Level

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

EAL:

CG1.1 General Emergency
RPV level < TAF for ≥ 30 min. (Note 1)
AND
Any Containment Challenge indication, Table C-2

Note 1: The SEC should declare the event promptly upon determining that time limit has been exceeded, or will likely be exceeded.

Note 6: If CONTAINMENT CLOSURE is re-established prior to exceeding the 30-minute time limit, declaration of a General Emergency is **not** required.

Table C-2 Containment Challenge Indications
• CONTAINMENT CLOSURE **not** established (Note 6)
• Primary Containment hydrogen concentration > 6%
• UNPLANNED rise in PC pressure
•

Deleted: Exceeding one or more Secondary Containment Control Maximum Safe Operating Area Radiation Levels . (0EOP-03-SCCP Table 3)

Mode Applicability:

4 - Cold Shutdown, 5 – Refuel

Definition(s):

CONTAINMENT CLOSURE - The procedurally defined conditions or actions taken to secure Primary or Secondary Containment and its associated structures, systems, and components as a functional barrier to fission product release under shutdown conditions.

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As applied to BNP, Containment Closure is established when either Primary Containment is Operable per Section 3.6.1.1 of Technical Specifications or Secondary Containment is considered functional per the requirements of 0AP-022, BNP Outage Risk Management.

Basis:

When RPV level drops below ~ -8 in., the top of active fuel, core uncovery starts to occur (ref. 5).

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Three conditions are associated with a challenge to Primary Containment (PC) integrity:

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- CONTAINMENT CLOSURE is not established (Ref. 6).

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- In the early stages of a core uncovery event, it is unlikely that hydrogen buildup due to a core uncovery could result in an explosive mixture of dissolved gases in the Primary Containment. However, Primary Containment monitoring and/or sampling should be performed to verify this assumption and a General Emergency declared if it is determined that an explosive mixture exists. An explosive mixture can be formed when hydrogen gas concentration in the Primary Containment atmosphere is greater than 6% by volume in the presence of oxygen (>5%) (ref. 2, 3). In Cold Shutdown and Refueling modes it is assumed that the Primary Containment is de-inerted.

  Monitors CAC-AT-4409 and 4410 monitor hydrogen and oxygen concentration in the drywell during normal operation. These monitors are isolated by accident isolation signals. However, monitors CAC-AT-4409 and 4410 will be realigned to the primary containment for post-accident monitoring via an operator actuated isolation signal override circuit when directed by the EOPs. (ref. 4).

- Any unplanned increase in PC pressure in the Cold Shutdown or Refueling mode indicates a potential loss of containment closure capability. Unplanned Primary Containment pressure increases indicates containment closure cannot be assured and the Primary Containment cannot be relied upon as a barrier to fission product release.

This IC addresses the inability to restore and maintain reactor vessel level above the top of active fuel with containment challenged. This condition represents actual or IMMINENT substantial core degradation or melting with potential for loss of containment integrity. Releases can be reasonably expected to exceed EPA PAG exposure levels offsite for more than the immediate site area.

Deleted: <#>Secondary Containment radiation monitors should provide indication of increased release that may be indicative of a challenge to Containment Closure. The Maximum Safe Operating Values are 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 0EOP-03-SCCP, Secondary Containment Control, (ref. 5).¶

Following an extended loss of core decay heat removal and inventory makeup, decay heat will cause reactor coolant boiling and a further reduction in reactor vessel level. If RPV level cannot be restored, fuel damage is probable.

ATTACHMENT 1
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EAL Bases

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With CONTAINMENT CLOSURE not established, there is a high potential for a direct and unmonitored release of radioactivity to the environment. If CONTAINMENT CLOSURE is re-established prior to exceeding the 30-minute time limit, then declaration of a General

Emergency is not required.

The existence of an explosive mixture means, at a minimum, that the containment atmospheric hydrogen concentration is sufficient to support a hydrogen burn (i.e., at the lower deflagration limit). A hydrogen burn will raise containment pressure and could result in collateral equipment damage leading to a loss of containment integrity. It therefore represents a challenge to Containment integrity.

In the early stages of a core uncovery event, it is unlikely that hydrogen buildup due to a core uncovery could result in an explosive gas mixture in containment. If all installed hydrogen gas monitors are out-of-service during an event leading to fuel cladding damage, it may not be possible to obtain a containment hydrogen gas concentration reading as ambient conditions within the containment will preclude personnel access. During periods when installed containment hydrogen gas monitors are out-of-service, operators may use the other listed indications to assess whether or not containment is challenged.

The 30-minute criterion is tied to a readily recognizable event start time (i.e., the total loss of ability to monitor level), and allows sufficient time to monitor, assess and correlate reactor and plant conditions to determine if core uncovery has actually occurred (i.e., to account for various accident progression and instrumentation uncertainties). It also allows sufficient time for performance of actions to terminate leakage, recover inventory control/makeup equipment and/or restore level monitoring.

This EAL addresses concerns raised by Generic Letter 88-17, *Loss of Decay Heat Removal*; SECY 91-283, *Evaluation of Shutdown and Low Power Risk Issues*; NUREG-1449, *Shutdown and Low-Power Operation at Commercial Nuclear Power Plants in the United States*; and NUMARC 91-06, *Guidelines for Industry Actions to Assess Shutdown Management*.

BNP Basis Reference(s):

1. BNP Technical Specifications Sections 3.6.1.1 and 3.6.4.1
2. BWROG EPG/SAG Revision 2, Sections PC/G
3. 0EOP-02-PCCP, Primary Containment Control
4. Updated FSAR section 6.2.5.2.2
5. 0EOP-01-NL EOP/SAMG NUMERICAL LIMITS AND VALUES
6. 0AP-022 BNP Outage Risk Management, Section 6.5
7. NEI 99-01 CG1

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Category: C – Cold Shutdown / Refueling System Malfunction

Subcategory: 1 – RPV Level

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

EAL:

CG1.2 General Emergency

RPV level **cannot** be monitored for ≥ 30 min. (Note 1)

AND

Core uncovery is indicated by **EITHER** of the following:

- UNPLANNED increase in **any** Table C-1 sump or tank levels due to a loss of RPV inventory
- UNPLANNED increase in ARM Channel 28 Between Reactor and Fuel Pool > 1000 mR/hr

AND

Any Containment Challenge indication, Table C-2

Note 1: The SEC should declare the event promptly upon determining that time limit has been exceeded, or will likely be exceeded.

Note 6: If CONTAINMENT CLOSURE is re-established prior to exceeding the 30-minute time limit, declaration of a General Emergency is not required.

Table C-1 Sumps & Tanks
• Drywell Floor Drain Sump
• Drywell Equipment Drain Sump
• RB Floor Drain Sump
• RB Equipment Drain Sump
• Torus
• Visual Observation

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Table C-2 Containment Challenge Indications
• CONTAINMENT CLOSURE **not** established (Note 6)
• Primary Containment hydrogen concentration > 6%
• UNPLANNED rise in PC pressure
•

Deleted: Exceeding one or more Secondary Containment Control Maximum Safe Operating Area Radiation Levels (0EOP-03-SCCP Table 3)

Mode Applicability:

4 - Cold Shutdown, 5 – Refuel

Definition(s):

CONTAINMENT CLOSURE - The procedurally defined conditions or actions taken to secure Primary or Secondary Containment and its associated structures, systems, and components as a functional barrier to fission product release under shutdown conditions.

As applied to BNP, Containment Closure is established when either Primary Containment is Operable per Section 3.6.1.1 of Technical Specifications or Secondary Containment is considered functional per the requirements of 0AP-022, BNP Outage Risk Management.

UNPLANNED-. A parameter change or an event that is not 1) the result of an intended evolution or 2) an expected plant response to a transient. The cause of the parameter change or event may be known or unknown.

Basis:

If RPV level monitoring capability is unavailable, the RPV inventory loss must be detected by the leakage indications listed in Table C-1. Sump level increases must be evaluated against other potential sources of leakage. Drywell equipment and floor drain sump level rise is the normal method of monitoring and calculating leakage from the RPV (ref. 1 thru 4). A Reactor Building equipment or floor drain sump level rise may also be indicative of RCS inventory losses external to the Primary Containment from systems connected to the RPV. With RHR System operating in the Shutdown Cooling mode, an unexplained rise in torus level could be indicative of RHR valve misalignment or leakage (ref. 5). 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 that cannot be isolated could be indicative of a loss of RPV inventory.

In the Refueling Mode, as water level in the reactor vessel lowers, the dose rate above the core will increase. The dose rate due to this core shine should result in indications on installed area radiation monitors. ARM Channel 28 Between Reactor and Fuel Pool is located on the

Refuel Floor in the Reactor Building and is designed to provide monitoring of radiation due to a fuel handling event or loss of shielding during refueling operations. If this radiation monitor reaches and exceeds the alarm setpoint of 1,000 mr/hr, a loss of inventory with potential to uncover the core is likely to have occurred (Ref. 6, 7).

Three conditions are associated with a challenge to Primary Containment (PC) integrity:

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- CONTAINMENT CLOSURE is not established (Ref. 12).

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- In the early stages of a core uncovery event, it is unlikely that hydrogen buildup due to a core uncovery could result in an explosive mixture of dissolved gases in the Primary Containment. However, Primary Containment monitoring and/or sampling should be performed to verify this assumption and a General Emergency declared if it is determined that an explosive mixture exists. An explosive mixture can be formed when hydrogen gas concentration in the Primary Containment atmosphere is greater than 6% by volume in the presence of oxygen (>5%) (ref. 9, 10). In Cold Shutdown and Refueling modes it is assumed that the Primary Containment is de-inerted.

  Monitors CAC-AT-4409 and 4410 monitor hydrogen and oxygen concentration in the drywell during normal operation. These monitors are isolated by accident isolation signals. However, monitors CAC-AT-4409 and 4410 will be realigned to the primary containment for post-accident monitoring via an operator actuated isolation signal override circuit when directed by the EOPs. (ref. 11).

- Any unplanned increase in PC pressure in the Cold Shutdown or Refueling mode indicates a potential loss of containment closure capability. Unplanned Primary Containment pressure increases indicates containment closure cannot be assured and the Primary Containment cannot be relied upon as a barrier to fission product release.

This IC addresses the inability to restore and maintain reactor vessel level above the top of active fuel with containment challenged. This condition represents actual or IMMINENT substantial core degradation or melting with potential for loss of containment integrity. Releases can be reasonably expected to exceed EPA PAG exposure levels offsite for more than the immediate site area.

Deleted: <#>Secondary Containment radiation monitors should provide indication of increased release that may be indicative of a challenge to Containment Closure. The Maximum Safe Operating Values are 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 0EOP-03-SCCP, Secondary Containment Control, (ref. 12).¶

Following an extended loss of core decay heat removal and inventory makeup, decay heat will cause reactor coolant boiling and a further reduction in reactor vessel level. If RPV level cannot be restored, fuel damage is probable.

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With CONTAINMENT CLOSURE not established, there is a high potential for a direct and unmonitored release of radioactivity to the environment. If CONTAINMENT CLOSURE is re-established prior to exceeding the 30-minute time limit, then declaration of a General Emergency is not required.

The existence of an explosive mixture means, at a minimum, that the containment atmospheric hydrogen concentration is sufficient to support a hydrogen burn (i.e., at the lower deflagration limit). A hydrogen burn will raise containment pressure and could result in collateral equipment damage leading to a loss of containment integrity. It therefore represents a challenge to Containment integrity.

In the early stages of a core uncovery event, it is unlikely that hydrogen buildup due to a core uncovery could result in an explosive gas mixture in containment. If all installed hydrogen gas monitors are out-of-service during an event leading to fuel cladding damage, it may not be possible to obtain a containment hydrogen gas concentration reading as ambient conditions within the containment will preclude personnel access. During periods when installed containment hydrogen gas monitors are out-of-service, operators may use the other listed indications to assess whether or not containment is challenged.

The 30-minute criterion is tied to a readily recognizable event start time (i.e., the total loss of ability to monitor level), and allows sufficient time to monitor, assess and correlate reactor and plant conditions to determine if core uncovery has actually occurred (i.e., to account for various accident progression and instrumentation uncertainties). It also allows sufficient time for performance of actions to terminate leakage, recover inventory control/makeup equipment and/or restore level monitoring.

The inability to monitor RPV level may be caused by instrumentation and/or power failures, or water level dropping below the range of available instrumentation. If water level cannot be monitored, operators may determine that an inventory loss is occurring by observing changes in sump and/or tank levels. Sump and/or tank level changes must be evaluated against other potential sources of water flow to ensure they are indicative of leakage from theRCS.

This EAL addresses concerns raised by Generic Letter 88-17, *Loss of Decay Heat Removal*; SECY 91-283, *Evaluation of Shutdown and Low Power Risk Issues*; NUREG-1449, *Shutdown and Low-Power Operation at Commercial Nuclear Power Plants in the United States*; and NUMARC 91-06, *Guidelines for Industry Actions to Assess Shutdown Management.*

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BNP Basis Reference(s):

1. 0OP-47 Floor and Equipment Drain System Operating Procedure
2. 1OI-03.1 Control Room Operator Daily Surveillance Report
3. 2OI-03.2 Control Room Operator Daily Surveillance Report
4. 0AOP-14.0 Abnormal Primary Containment Conditions
5. 1(2)OP-17 Residual Heat Removal System Operating Procedure
6. 1(2)APP-UA-03 4-7
7. DBD-11 Radiation Monitoring System
8. BNP Technical Specifications Section 3.6.1.1 and 3.6.4.1
9. BWROG EPG/SAG Revision 2, Sections PC/G
10. 0EOP-02-PCCP Primary Containment Control
11. Updated FSAR section 6.2.5.2.2
12. 0AP-022 BNP Outage Risk Management, Section 6.5
14. NEI 99-01 CG1

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Category: C – Cold Shutdown / Refueling System Malfunction

Subcategory: 2 – Loss of Emergency AC Power

Initiating Condition: Loss of all but one AC power source to emergency buses for 15 minutes or longer

EAL:

CU2.1 Unusual Event

AC power capability to Emergency 4 KV Buses E1(E3) and E2(E4) reduced to a single power source, Table C-6, for ≥ 15 min. (Note 1)

AND

Any additional single power source failure will result in loss of **all** unit-specific AC power to SAFETY SYSTEMS

Note 1: The SEC should declare the event promptly upon determining that time limit has been exceeded, or will likely be exceeded.

Table C-6 AC Power Sources
Offsite:
• SAT
• UAT backfed through MPT (**only** if already aligned)
Onsite:
• UAT via Main Generator
• EDG1(3)
• EDG2(4)

Mode Applicability:

4 - Cold Shutdown, 5 – Refuel, D - Defueled

Definition(s):

SAFETY SYSTEM - A system required for safe plant operation, cooling down the plant and/or placing it in the cold shutdown condition, including the ECCS. These are typically systems classified as safety-related (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.

Basis:

The 4160 VAC System provides the power requirements for operation and safe shutdown of the plant. 4160 VAC power is divided into Balance of Plant (BOP Bus), and Emergency (E-Bus) distribution. The BOP Buses consist of the Conventional Buses (Common A/B Buses; Buses 1B/2B, 1C/2C, 1D/2D) and Caswell Beach Buses A/B. The Emergency switchgear are Buses E1/E3 (Division I) and E2/E4 (Division II) (Ref. 1, 2).

The E-Buses are normally powered through the respective BOP Buses (1D to E1, 1C to E2, 2D to E3, 2C to E4) via a master/slave breaker arrangement. Each E-Bus has a dedicated Diesel Generator to supply an emergency source of power to safe shutdown loads in the event of a loss of the normal power source or loss of offsite power. The DGs will automatically start and tie onto the E-Buses if the normal power source or offsite power is lost. The DGs can provide power to the E-Buses only. In the event the diesel generator is unavailable for an E-Bus, crosstie capability exists for each E-Bus from the same division of the opposite unit (E1 to E3, E2 to E4). Although the E-Buses within the unit also have crosstie capability, this alignment is not permitted by plant procedures, with the exception of E1 to E2 during specific Alternate Safe Shutdown (ASSD) conditions.

During periods of unit shutdown, when the Startup Auxiliary Transformer (SAT) would be the only normal source of offsite power, the Unit Auxiliary Transformer (UAT) can be made available by establishing a UAT backfeed. Backfeed from the UAT will require the use of keys for the control selector switches and opening of the respective generator's manual no-load disconnect (Ref. 3).

Because 2 RHR pumps on each unit are powered from the unaffected unit, the words "unit-specific" have been added to clarify that the cross-connected RHR pump power cannot be credited as an AC power source relative to this EAL.

This cold condition EAL is equivalent to the hot condition EAL SA1.1.

This IC describes a significant degradation of offsite and onsite AC power sources such that any additional single failure would result in a loss of all AC power to SAFETY SYSTEMS. In this condition, the sole AC power source may be powering one, or more than one, train of safety-related equipment.

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When in the cold shutdown, refueling, or defueled mode, this condition is not classified as an Alert because of the increased time available to restore another power source to service. Additional time is available due to the reduced core decay heat load, and the lower temperatures and pressures in various plant systems. Thus, when in these modes, this condition is considered to be a potential degradation of the level of safety of the plant.

An "AC power source" is a source recognized in AOPs, and capable of supplying required power to an emergency bus. Some examples of this condition are presented below.

- A loss of all offsite power with a concurrent failure of one division of emergency power sources (e.g., onsite diesel generators).
- A loss of all offsite power and loss of all emergency power sources (e.g., onsite diesel generators) with a single train of emergency buses being back-fed from the unit main generator.
- A loss of emergency power sources (e.g., onsite diesel generators) with a single division of emergency buses being back-fed from an offsite power source.

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

The subsequent loss of the remaining single power source would escalate the event to an Alert in accordance with IC CA2.

BNP Basis Reference(s):

1. Drawing BN-50.0.01 Electrical Distribution
2. BNP Updated FSAR Chapter 8
3. 1(2)OP-50 Plant Electric System Operating Procedure
4. 1(2)APP-UA15 2-1 (Bus E-1 Undervoltage)
5. 1(2)APP-UA16 2-1 (Bus E-2 Undervoltage)
6. 1(2)APP-UA17 2-1 (Bus E-3 Undervoltage)
7. 1(2)APP-UA18 2-1 (Bus E-4 Undervoltage)
8. NEI 99-01 CU2

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Category: C – Cold Shutdown / Refueling System Malfunction

Subcategory: 2 – Loss of Emergency AC Power

Initiating Condition: Loss of **all** offsite and **all** onsite AC power to emergency buses for 15 minutes or longer

EAL:

CA2.1	**Alert**
Loss of **all** offsite and **all** onsite AC power capability to Emergency 4 KV Buses E1(E3) and E2(E4) for ≥ 15 min. (Note 1)	

Note 1: The SEC should declare the event promptly upon determining that time limit has been exceeded, or will likely be exceeded.

Mode Applicability:

4 - Cold Shutdown, 5 - Refuel, D - Defueled

Basis:

The 4160 VAC System provides the power requirements for operation and safe shutdown of the plant. 4160 VAC power is divided into Balance of Plant (BOP Bus), and Emergency (E-Bus) distribution. The BOP Buses consist of the Conventional Buses (Common A/B Buses; Buses 1B/2B, 1C/2C, 1D/2D) and Caswell Beach Buses A/B. The Emergency switchgear are Buses E1/E3 (Division I) and E2/E4 (Division II) (Ref. 1, 2).

The E-Buses are normally powered through the respective BOP Buses (1D to E1, 1C to E2, 2D to E3, 2C to E4) via a master/slave breaker arrangement. Each E-Bus has a dedicated Diesel Generator to supply an emergency source of power to safe shutdown loads in the event of a loss of the normal power source or loss of offsite power. The DGs will automatically start and tie onto the E-Buses if the normal power source or offsite power is lost. The DGs can provide power to the E-Buses only. In the event the diesel generator is unavailable for an E-Bus, crosstie capability exists for each E-Bus from the same division of the opposite unit (E1 to E3, E2 to E4). Although the E-Buses within the unit also have crosstie capability, this alignment is not permitted by plant procedures, with the exception of E1 to E2 during specific Alternate Safe Shutdown (ASSD) conditions.

During periods of unit shutdown, when the Startup Auxiliary Transformer (SAT) would be the only normal source of offsite power, the Unit Auxiliary Transformer (UAT) can be made available by establishing a UAT backfeed. Backfeed from the UAT will require the use of keys for the control selector switches and opening of the respective generator's manual no-load disconnect (Ref. 3).

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

This IC addresses a total loss of AC power that compromises the performance of all SAFETY SYSTEMS requiring electric power including those necessary for emergency core cooling, containment heat removal/pressure control, spent fuel heat removal and the ultimate heat sink.

When in the cold shutdown, refueling, or defueled mode, this condition is not classified as a Site Area Emergency because of the increased time available to restore an emergency bus to service. Additional time is available due to the reduced core decay heat load, and the lower temperatures and pressures in various plant systems. Thus, when in these modes, this condition represents an actual or potential substantial degradation of the level of safety of the plant.

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

Escalation of the emergency classification level would be via IC CS1 or RS1.

BNP Basis Reference(s):

1. Drawing BN-50.0.01 Electrical Distribution
2. BNP Updated FSAR Chapter 8
3. 1(2)OP-50 Plant Electric System Operating Procedure
4. 1(2)APP-UA15 2-1 (Bus E-1 Undervoltage)
5. 1(2)APP-UA16 2-1 (Bus E-2 Undervoltage)
6. 1(2)APP-UA17 2-1 (Bus E-3 Undervoltage)
7. 1(2)APP-UA18 2-1 (Bus E-4 Undervoltage)
8. NEI 99-01 CA2

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Category: C – Cold Shutdown / Refueling System Malfunction

Subcategory: 3 – RCS Temperature

Initiating Condition: UNPLANNED increase in RCS temperature

EAL:

CU3.1 Unusual Event

UNPLANNED increase in RCS temperature to > 212°F due to loss of decay heat removal capability

Mode Applicability:

4 - Cold Shutdown, 5 - Refuel

Definition(s):

UNPLANNED-. A parameter change or an event that is not 1) the result of an intended evolution or 2) an expected plant response to a transient. The cause of the parameter change or event may be known or unknown.

Basis:

Several instruments are capable of providing indication of RCS temperature with respect to the Technical Specification cold shutdown temperature limit (212°F, ref. 1). These include (ref. 2):

- Recirculation Suction Temperatures read on B32-TR-R650 located on panel P-603 (if recirculation loop is in operation)
- RHR HX 2A(B) Inlet Temperature as read on E41-TR-R605 Point 1(2), on Panel H12-P614 (RHR HX in service)
- RHR HX 2A(B) Outlet Temperature as read on E41-TR-R605 Point 3(4), on Panel H12-P614 (RHR HX **not** in service)
- PPC Display 815, RPV HEATUP/COOLDOWN MONITOR (natural circulation)

This IC addresses an UNPLANNED increase in RCS temperature above the Technical Specification cold shutdown temperature limitand represents a potential degradation of the level of safety of the plant. If the RCS is not intact and CONTAINMENT CLOSURE is not established during this event, the Site Emergency Coordinator should also refer to IC CA3.

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A momentary UNPLANNED excursion above the Technical Specification cold shutdown temperature limit when the heat removal function is available does not warrant a classification.

This EAL involves a loss of decay heat removal capability, or an addition of heat to the RCS in excess of that which can currently be removed, such that reactor coolant temperature cannot be maintained below the cold shutdown temperature limit specified in Technical Specifications. During this condition, there is no immediate threat of fuel damage because the core decay heat load has been reduced since the cessation of power operation.

During an outage, the level in the reactor vessel will normally be maintained at or above the reactor vessel flange. Refueling evolutions that lower water level below the reactor vessel flange are carefully planned and controlled. A loss of forced decay heat removal at reduced inventory may result in a rapid increase in reactor coolant temperature depending on the time after shutdown.

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

Escalation to Alert would be via IC CA1 based on an inventory loss or IC CA3 based on exceeding plant configuration-specific time criteria.

BNP Basis Reference(s):

1. BNP Technical Specifications Table 1.1-1
2. 1(2)PT-01.7 Heatup/Cooldown Monitoring
3. NEI 99-01 CU3

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Category: C – Cold Shutdown / Refueling System Malfunction

Subcategory: 3 – RCS Temperature

Initiating Condition: UNPLANNED increase in RCS temperature

EAL:

CU3.2	**Unusual Event**
Loss of **all** RCS temperature and RPV level indication for ≥ 15 min. (Note 1)	

Note 1: The SEC should declare the event promptly upon determining that time limit has been exceeded, or will likely be exceeded.

Mode Applicability:

4 - Cold Shutdown, 5 - Refuel

Definition(s):

None

Basis:

RPV water level is normally monitored using the instruments in Figure C-1 (ref. 1).

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

- Recirculation Suction Temperatures read on B32-TR-R650 located on panel P-603 (if recirculation loop is in operation)
- RHR HX 2A(B) Inlet Temperature as read on E41-TR-R605 Point 1(2), on Panel H12-P614 (RHR HX in service)
- RHR HX 2A(B) Outlet Temperature as read on E41-TR-R605 Point 3(4), on Panel H12-P614 (RHR HX **not** in service)
- PPC Display 815, RPV HEATUP/COOLDOWN MONITOR (natural circulation)

This EAL addresses the inability to determine RCS temperature and RPV level, and represents a potential degradation of the level of safety of the plant. If the RCS is not intact and CONTAINMENT CLOSURE is not established during this event, the Site Emergency Coordinator should also refer to IC CA3.

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This EAL reflects a condition where there has been a significant loss of instrumentation capability necessary to monitor RCS conditions and operators would be unable to monitor key parameters necessary to assure core decay heat removal. During this condition, there is no immediate threat of fuel damage because the core decay heat load has been reduced since the cessation of power operation.

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

Escalation to Alert would be via IC CA1 based on an inventory loss or IC CA3 based on exceeding plant configuration-specific time criteria.

BNP Basis Reference(s):

1. SD-01.2 Reactor Vessel Instrumentation Figure 01.2-1 Reactor Water Level Instrument Ranges
2. Technical Specifications Table 1.1-1
3. 1(2)PT-01.7 Heatup/Cooldown Monitoring
4. NEI 99-01 CU3

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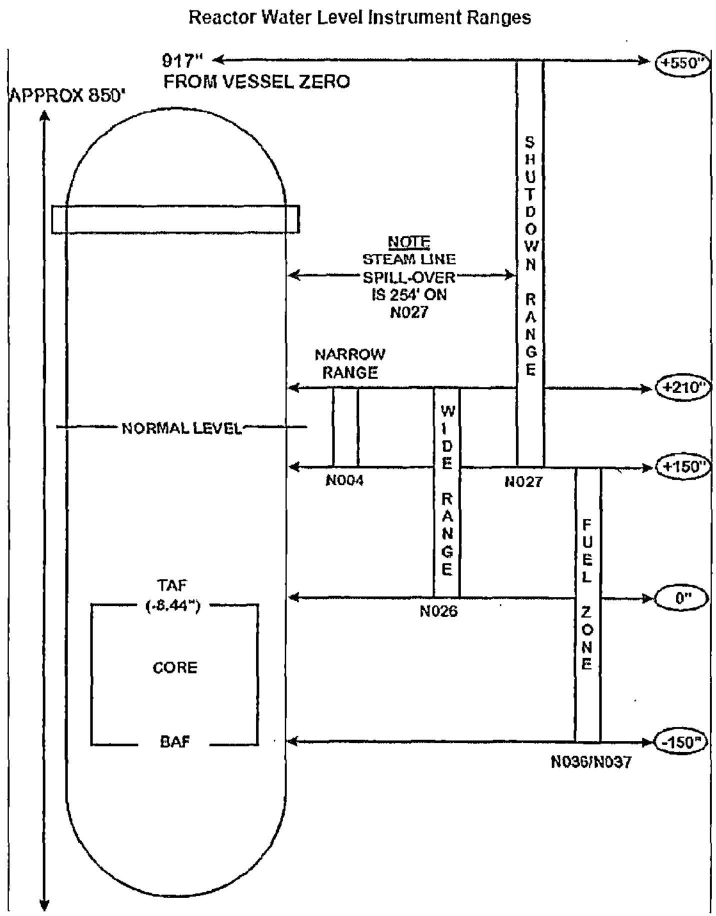
Figure C-1 RPV Levels (ref. 1)



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Category: C – Cold Shutdown / Refueling System Malfunction

Subcategory: 3 – RCS Temperature

Initiating Condition: Inability to maintain plant in cold shutdown

EAL:

CA3.1 Alert

UNPLANNED increase in RCS temperature to > 212°F for > Table C-3 duration (Note 1)

OR

UNPLANNED RPV pressure increase > 10 psig due to a loss of RCS cooling

Note 1: The SEC should declare the event promptly upon determining that the applicable time has been exceeded, or will likely be exceeded.

Table C-3: RCS Heat-up Duration Thresholds		
RCS Status	**Containment Closure Status**	**Heat-up 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

Definition(s):

CONTAINMENT CLOSURE - The procedurally defined conditions or actions taken to secure Primary or Secondary Containment and its associated structures, systems, and components as a functional barrier to fission product release under shutdown conditions.

As applied to BNP, Containment Closure is established when either Primary Containment is Operable per Section 3.6.1.1 of Technical Specifications or Secondary Containment is considered functional per the requirements of 0AP-022, BNP Outage Risk Management.

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UNPLANNED -. A parameter change or an event that is not 1) the result of an intended evolution or 2) an expected plant response to a transient. The cause of the parameter change or event may be known or unknown.

Basis:

A 10 psig RPV pressure increase can be read on (ref. 1):

- Indicator PI-R605A located on Panel P603
- Indicator PI-R605B located on Panel P601
- Recorder LPR-R608 located on P603
- Indicator C32-PI-3332 located on the Remote Shutdown Panel

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

- Recirculation Suction Temperatures read on B32-TR-R650 located on panel P-603 (if recirculation loop is in operation)
- RHR HX 2A(B) Inlet Temperature as read on E41-TR-R605 Point 1(2), on Panel H12-P614 (RHR HX in service)
- RHR HX 2A(B) Outlet Temperature as read on E41-TR-R605 Point 3(4), on Panel H12-P614 (RHR HX **not** in service)
- PPC Display 815, RPV HEATUP/COOLDOWN MONITOR (natural circulation)

This IC addresses conditions involving a loss of decay heat removal capability or an addition of heat to the RCS in excess of that which can currently be removed. Either condition represents an actual or potential substantial degradation of the level of safety of the plant.

A momentary UNPLANNED excursion above the Technical Specification cold shutdown temperature limit when the heat removal function is available does not warrant a classification.

The RCS Heat-up Duration Thresholds table addresses an increase in RCS temperature when CONTAINMENT CLOSURE is established but the RCS is not intact.. The 20-minute criterion was included to allow time for operator action to address the temperature increase.

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The RCS Heat-up Duration Thresholds table also addresses an increase in RCS temperature with the RCS intact. The status of CONTAINMENT CLOSURE is not crucial in this condition since the intact RCS is providing a high pressure barrier to a fission product release. The 60-minute time frame should allow sufficient time to address the temperature increase without a substantial degradation in plant safety.

Finally, in the case where there is an increase in RCS temperature, the RCS is not intact , and CONTAINMENT CLOSURE is not established, no heat-up duration is allowed (i.e., 0 minutes). This is because 1) the evaporated reactor coolant may be released directly into the Primary Containment or Reactor Building atmosphere and subsequently to the environment, and 2) there is reduced reactor coolant inventory above the top of irradiated fuel.

The RCS pressure increase threshold provides a pressure-based indication of RCS heat-up in the absence of RCS temperature monitoring capability.

Escalation of the emergency classification level would be via IC CS1 or RS1.

BNP Basis Reference(s):

1. Reactor Vessel Instrumentation System Description SD-01.2
2. BNP Technical Specifications Table 1.1-1
3. 1(2)PT-01.7 Heatup/Cooldown Monitoring
4. Technical Specifications Sections 3.6.1.1 and 3.6.4.1
5. 0AP-022, BNP Outage Risk Management
6. NEI 99-01 CA3

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Category: C – Cold Shutdown / Refueling System Malfunction

Subcategory: 4 – Loss of Vital DC Power

Initiating Condition: Loss of Vital DC power for 15 minutes or longer

EAL:

CU4.1	**Unusual Event**
< 105 VDC bus voltage indications on Technical Specification **required** 125 VDC buses for ≥ 15 min. (Note 1)	

Note 1: The SEC should declare the event promptly upon determining that time limit has been exceeded, or will likely be exceeded.

Mode Applicability:

4 - Cold Shutdown, 5 - Refuel

Definition(s):

None

Basis:

There are two independent divisions per unit, designated Division I and Division II (Batteries 1(2)A-1 and 1(2)A-2 for Division I and Batteries 1(2)B-1 and 1(2)B-2 for Division II) . Each division consists of a 250 VDC battery center tapped to form two 125 VDC batteries. Each 125 VDC battery has an associated full capacity battery charger. The chargers are supplied from the same AC load groups for which the associated DC subsystem supplies the control power. During normal operation, the DC loads are powered from the battery chargers with the batteries floating on the system. In case of loss of normal power to the battery charger, the DC loads are automatically powered from the station batteries.

105 VDC is the minimum design voltage limit (ref. 1).

Note that the Control Room DC voltage indicator only reads battery charger output voltage and not battery voltage unless the charger output breaker is closed. However ERFIS does provide DC battery voltage, otherwise battery voltage must be read locally.

In the event that DC battery voltage indication is not available via ERFIS, local voltage indication is available for each bus based on dispatching a field operator in accordance with 0AOP-39.0 Loss of DC Power. In this case the 15 minute classification clock begins upon receipt of the low voltage alarm in the Control Room. If battery voltage cannot be verified to be greater than or equal to 105 VDC within the 15 minutes, emergency classification must be made under this EAL.

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This EAL is the cold condition equivalent of the hot condition loss of DC power EAL SS7.1.

This IC addresses a loss of vital DC power which compromises the ability to monitor and control operable SAFETY SYSTEMS when the plant is in the cold shutdown or refueling mode. In these modes, the core decay heat load has been significantly reduced, and coolant system temperatures and pressures are lower; these conditions increase the time available to restore a vital DC bus to service. Thus, this condition is considered to be a potential degradation of the level of safety of the plant.

As used in this EAL, "required" means the vital DC buses necessary to support operation of the in-service, or operable, train or trains of SAFETY SYSTEM equipment. For example, if Division I is out-of-service (inoperable) for scheduled outage maintenance work and Division II is in-service (operable), then a loss of Vital DC power affecting Division II would require the declaration of an Unusual Event. A loss of Vital DC power to Division I would not warrant an emergency classification.

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

Depending upon the event, escalation of the emergency classification level would be via IC CA1 or CA3, or an IC in Recognition Category R.

BNP Basis Reference(s):

1. BNP Technical Specification Bases B.3.8.4
2. 0AOP-39.0 LOSS OF DC POWER
3. NEI 99-01 CU4

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Category: C – Cold Shutdown / Refueling System Malfunction

Subcategory: 5 – Loss of Communications

Initiating Condition: Loss of **all** onsite or offsite communications capabilities

EAL:

CU5.1 Unusual Event

Loss of **all** Table C-4 onsite communication methods

OR

Loss of **all** Table C-4 Offsite communication methods

OR

Loss of **all** Table C-4 NRC communication methods

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Table C-4 Communication Methods

System	Onsite	Offsite	NRC
Public Address System	X		
PBX Telephone System	X	X	X
DEMNET		X	X
Commercial Telephones	X	X	X
Satellite Phones		X	X
Cellular Phones		X	X
NRC Emergency Telecommunications System			X

Deleted: Corporate Telephone Communications System

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Mode Applicability:

4 - Cold Shutdown, 5 - Refuel, D – Defueled

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Definition(s):

None

Basis:

Onsite/offsite/NRC communications include one or more of the systems listed in Table C-4 (ref. 1).

Public Address System

The Brunswick Plant public address system provides paging and party line communications between stations located throughout the plant. Inside and outside type wall and desk-mounted stations are used to communicate between roaming personnel and fixed work locations. Plant-wide instructions are issued using the paging feature. This system is powered from the plant uninterruptible power supply which employs battery reserve as well as diesel generator emergency supply.

PBX Telephone System

The Brunswick Site PBX telephone system provides communication capability between telephone stations located within the plant by dialing the four-digit telephone station code. The PBX telephone system also provides for outside communications. The PBX switch located in the TSC/EOF building is also backed up by a battery UPS capable of supplying power for a minimum of 8 hours and is augmented by a Diesel Generator capable of supplying power to the TSC/EOF building for at least 5 days.

DEMNET

Deleted: Corporate Telephone Communications System (Voicenet and/or

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DEMNET is the primary means of offsite communication. This circuit allows intercommunication among the EOF, TSC, control room, counties, and states. DEMNET operates as an internet based (VoIP) communications system with a satellite back-up. Should the internet transfer rate become slow or unavailable, the DEMNET will automatically transfer to satellite mode.

Deleted: Interconnected through the site PBX and the emergency telephone system, the Voicenet system provides a means to communicate with other corporate locations with which the plant has a need to communicate. This system bypasses external commercial telephone lines and switching equipment. Corporate transmission facilities provide fiber optic, copper-wire, and microwave radio to ensure a high degree of system reliability. In addition to the redundancy provided by the three system options, backup power is provided for the systems.

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Commercial Telephones

Commercial telephone lines, which supply public telephone communications, are employed by Duke Energy in four ways: (1) tie-ins through the PBX to any other plant location, (2) lines to plant emergency facilities, (3) lines to the Joint Information Center for public information purposes, and (4) lines to the AEF. The local service provider provides primary and secondary power for their lines at the Central Office.

Satellite Phones

A total of three portable satellite telephones are available which enable communication when all other phone systems are inoperable, e.g. following a major hurricane. These portable systems can be powered by internal batteries, external DC sources as well as external AC sources. Two of these phones require outside use, while one phone may used either outside or in the EOF with a permanently mounted external antenna.

Cellular Phones

Selected plant personnel are provided with cellular telephones. These phones may be used during emergencies if other communications means are not readily available or are inoperable. These phones are not expected to be used in the Control Room or Power Block due to interference with plant equipment and loss of signal to the phone.

NRC Emergency Telecommunications System

The NRC uses a Duke Energy dedicated telephone line which allows direct telephone communications from the plant to NRC regional and national offices. The Duke Energy communications line provides a link independent of the local public telephone network. Telephones connected to this network are located in the Brunswick Control Room, Technical Support Center, and Emergency Operations Facility. These lines will not function if the PBX Telephone System fails.

This EAL is the cold condition equivalent of the hot condition EAL SU7.1.

This IC addresses a significant loss of onsite or offsite communications capabilities. While not a direct challenge to plant or personnel safety, this event warrants prompt notifications to OROs and the NRC.

This IC should be assessed only when extraordinary means are being utilized to make communications possible (e.g., use of non-plant, privately owned equipment, relaying of onsite information via individuals or multiple radio transmission points, individuals being sent to offsite locations, etc.).

The first EAL condition addresses a total loss of the communications methods used in support of routine plant operations.

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The second EAL condition addresses a total loss of the communications methods used to notify all OROs of an emergency declaration. The OROs referred to here are the State, Brunswick and New Hanover County EOCs

The third EAL addresses a total loss of the communications methods used to notify the NRC of an emergency declaration.

BNP Basis Reference(s):

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1. 0ERP Radiological Emergency Response Plan Appendix A
2. SD-48 Communication Systems
3. NEI 99-01 CU5

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Category: C – Cold Shutdown / Refueling System Malfunction

Subcategory: 6 – Hazardous Event Affecting Safety Systems

Initiating Condition: Hazardous event affecting a SAFETY SYSTEM needed for the current operating mode

EAL:

CA6.1 Alert

The occurrence of **any** Table C-5 hazardous event

AND

EITHER of the following:

- Event damage has caused indications of degraded performance in at least one train of a SAFETY SYSTEM needed for the current operating mode
- The event has caused VISIBLE DAMAGE to a SAFETY SYSTEM component or structure needed for the current operating mode

Table C-5 Hazardous Events
• Seismic event (earthquake) • Internal or external FLOODING event • High winds or tornado strike • FIRE • EXPLOSION • Other events with similar hazard characteristics as determined by the Shift Manager

Mode Applicability:

4 - Cold Shutdown, 5 - Refuel

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Definition(s):

EXPLOSION - A rapid, violent and catastrophic failure of a piece of equipment due to combustion, chemical reaction or overpressurization. A release of steam (from high energy lines or components) or an electrical component failure (caused by short circuits, grounding, arcing, etc.) should not automatically be considered an explosion. Such events require a post-event inspection to determine if the attributes of an explosion are present.

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.

FLOODING - A condition where water is entering a room or area faster than installed equipment is capable of removal, resulting in a rise of water level within the room or area.

SAFETY SYSTEM - A system required for safe plant operation, cooling down the plant and/or placing it in the cold shutdown condition, including the ECCS. These are typically systems classified as safety-related (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.

VISIBLE DAMAGE - Damage to a component or structure that is readily observable without measurements, testing, or analysis. The visual impact of the damage is sufficient to cause concern regarding the operability or reliability of the affected component or structure.

Basis:

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- The significance of seismic events are discussed under EAL HU2.1 (ref. 1).
- Internal FLOODING may be caused by events such as component failures, equipment misalignment, or outage activity mishaps (ref. 2, 3).
- Seismic Category I structures are analyzed to withstand a sustained, design wind velocity of at least 135 mph. (ref. 4).

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- Areas containing functions and systems required for safe shutdown of the plant are identified by Fire Zone in the fire response procedure (ref. 5, 6).
- An explosion that degrades the performance of a SAFETY SYSTEM train or visibly damages a SAFETY SYSTEM component or structure would be classified under this EAL.

This IC addresses a hazardous event that causes damage to a SAFETY SYSTEM, or a structure containing SAFETY SYSTEM components, needed for the current operating mode. This condition significantly reduces the margin to a loss or potential loss of a fission product barrier, and therefore represents an actual or potential substantial degradation of the level of safety of the plant.

The first conditional addresses damage to a SAFETY SYSTEM train that is in service/operation since indications for it will be readily available. The indications of degraded performance should be significant enough to cause concern regarding the operability or reliability of the SAFETY SYSTEM train.

The second conditional addresses damage to a SAFETY SYSTEM component that is not in service/operation or readily apparent through indications alone, or to a structure containing SAFETY SYSTEM components. Operators will make this determination based on the totality of available event and damage report information. This is intended to be a brief assessment not requiring lengthy analysis or quantification of the damage.

Escalation of the emergency classification level would be via IC CS1 or RS1.

BNP Basis Reference(s):

1. 1(2)APP-UA-28 6-4 Seismic Event
2. 0AOP-13.0 Operations During Hurricane, Flood Conditions, Tornado or Earthquake
3. Updated FSAR section 3.4.2 Protection From Internal Flooding
4. Updated FSAR Section 2.3.1.2.7
5. BNP-E-9. 010 NFPA 805 Nuclear Safety Capability Assessment (NSCA)
6. 0PFP-PBAA Power Block Auxiliary Areas Prefire Plan
7. NEI 99-01 CA6

Deleted: BNP-E-9.004 Safe Shutdown Analysis Report

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.

1. Security

Unauthorized entry attempts into the Protected Area, bomb threats, sabotage attempts, and actual security compromises threatening loss of physical control of the plant.

2. Seismic Event

Natural events such as earthquakes have potential to cause plant structure or equipment damage of sufficient magnitude to threaten personnel or plant safety.

3. Natural or Technology Hazard

Other natural and non-naturally occurring events that can cause damage to plant facilities include tornados, FLOODING, hazardous material releases and events restricting site access warranting classification.

4. Fire

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

5. Hazardous Gas

Toxic, corrosive, asphyxiant or flammable gas leaks can affect normal plant operations or preclude access to plant areas required to safely shutdown the plant.

6. 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.

7. SEC 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 Site Emergency Coordinator the latitude to classify emergency conditions consistent with the established classification criteria based upon Site Emergency Coordinator judgment.

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Category: H – Hazards

Subcategory: 1 – Security

Initiating Condition: Confirmed SECURITY CONDITION or threat

EAL:

HU1.1 **Unusual Event**
A SECURITY CONDITION that does **not** involve a HOSTILE ACTION as reported by the Security Shift Supervision

Mode Applicability:

All

Definition(s):

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.

HOSTILE ACTION - An act toward BNP 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 BNP. Non-terrorism-based EALs should be used to address such activities (i.e., this may include violent acts between individuals in the owner controlled area).

Basis:

The Security Shift Supervision is defined as either the Security Shift Lieutenant or the Security Shift Sergeant.

This EAL is based on the BNP Physical Security Plan (ref. 1).

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This IC addresses events that pose a threat to plant personnel or SAFETY SYSTEM equipment, and thus represent a potential degradation in the level of plant safety. Security events which do not meet one of these EALs are adequately addressed by the requirements of 10 CFR § 73.71 or 10 CFR § 50.72. Security events assessed as HOSTILE ACTIONS are classifiable under ICs HA1, HS1 and HG1.

Timely and accurate communications between Security Shift Supervision and the Control Room is essential for proper classification of a security-related event. Classification of these events will initiate appropriate threat-related notifications to plant personnel and Offsite Response Organizations.

Security plans and terminology are based on the guidance provided by NEI 03-12, *Template for the Security Plan, Training and Qualification Plan, Safeguards Contingency Plan.*

This EAL references the Security Shift Supervison because these are the individuals trained to confirm that a security event is occurring or has occurred. Training on security event confirmation and classification is controlled due to the nature of Safeguards and 10 CFR § 2.39 information.

Emergency plans and implementing procedures are public documents; therefore, EALs should not incorporate Security-sensitive information. This includes information that may be advantageous to a potential adversary, such as the particulars concerning a specific threat or threat location. Security-sensitive information should be contained in non-public documents such as the BNP Physical Security Plan (ref. 1).

Escalation of the emergency classification level would be via IC HA1.

BNP Basis Reference(s):

1. BNP Physical Security Plan
2. SEC-NGGC-2170 Security Event Procedures
3. 0AOP-40.0 Security Events
4. NEI 99-01 HU1

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Category: H – Hazards

Subcategory: 1 – Security

Initiating Condition: Confirmed SECURITY CONDITION or threat

EAL:

HU1.2	**Unusual Event**
Notification of a credible security threat directed at the site	

Mode Applicability:

All

Definition(s):

None

Basis:

The Security Shift Supervision is defined as either the Security Shift Lieutenant or the Security Shift Sergeant.

This EAL is based on the BNP Physical Security Plan (ref. 1).

This IC addresses events that pose a threat to plant personnel or SAFETY SYSTEM equipment, and thus represent a potential degradation in the level of plant safety. Security events which do not meet one of these EALs are adequately addressed by the requirements of 10 CFR § 73.71 or 10 CFR § 50.72. Security events assessed as HOSTILE ACTIONS are classifiable under ICs HA1, HS1 and HG1.

Timely and accurate communications between Security Shift Supervision and the Control Room is essential for proper classification of a security-related event. Classification of these events will initiate appropriate threat-related notifications to plant personnel and Offsite Response Organizations.

Security plans and terminology are based on the guidance provided by NEI 03-12, *Template for the Security Plan, Training and Qualification Plan, Safeguards Contingency Plan.*

This EAL addresses the receipt of a credible security threat. The credibility of the threat is assessed in accordance with the BNP Physical Security Plan (ref. 1).

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Emergency plans and implementing procedures are public documents; therefore, EALs should not incorporate Security-sensitive information. This includes information that may be advantageous to a potential adversary, such as the particulars concerning a specific threat or threat location. Security-sensitive information should be contained in non-public documents such as the BNP Physical Security Plan (ref. 1).

Escalation of the emergency classification level would be via IC HA1.

BNP Basis Reference(s):

1. BNP Physical Security Plan
2. SEC-NGGC-2170 Security Event Procedures
3. 0AOP-40.0 Security Events
4. NEI 99-01 HU1

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Category: H – Hazards

Subcategory: 1 – Security

Initiating Condition: Confirmed SECURITY CONDITION or threat

EAL:

HU1.3	**Unusual Event**
A validated notification from the NRC providing information of an aircraft threat	

Mode Applicability:

All

Definition(s):

None

Basis:

The Security Shift Supervision is defined as either the Security Shift Lieutenant or the Security Shift Sergeant.

This EAL is based on the BNP Physical Security Plan (ref. 1).

This IC addresses events that pose a threat to plant personnel or SAFETY SYSTEM equipment, and thus represent a potential degradation in the level of plant safety. Security events which do not meet one of these EALs are adequately addressed by the requirements of 10 CFR § 73.71 or 10 CFR § 50.72. Security events assessed as HOSTILE ACTIONS are classifiable under ICs HA1, HS1 and HG1.

Timely and accurate communications between Security Shift Supervision and the Control Room is essential for proper classification of a security-related event. Classification of these events will initiate appropriate threat-related notifications to plant personnel and Offsite Response Organizations.

Security plans and terminology are based on the guidance provided by NEI 03-12, *Template for the Security Plan, Training and Qualification Plan, Safeguards Contingency Plan.*

This EAL addresses the threat from the impact of an aircraft on the plant. The NRC Headquarters Operations Officer (HOO) will communicate to the licensee if the threat involves an aircraft. The status and size of the plane may also be provided by NORAD through the NRC. Validation of the threat is performed in accordance with the BNP Physical Security Plan (ref. 1).

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Emergency plans and implementing procedures are public documents; therefore, EALs should not incorporate Security-sensitive information. This includes information that may be advantageous to a potential adversary, such as the particulars concerning a specific threat or threat location. Security-sensitive information should be contained in non-public documents such as the BNP Physical Security Plan (ref. 1).

Escalation of the emergency classification level would be via IC HA1.

BNP Basis Reference(s):

1. BNP Physical Security Plan
2. SEC-NGGC-2170 Security Event Procedures
3. 0AOP-40.0 Security Events
4. NEI 99-01 HU1

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Category: H – Hazards

Subcategory: 1 – Security

Initiating Condition: Hostile action within the owner controlled area or airborne attack threat within 30 minutes

EAL:

HA1.1	**Alert**
A HOSTILE ACTION is occurring or has occurred within the OWNER CONTROLLED AREA as reported by the Security Shift Supervision	

Mode Applicability:

All

Definition(s):

HOSTILE ACTION - An act toward BNP 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 BNP. Non-terrorism-based EALs should be used to address such activities (i.e., this may include violent acts between individuals in the owner controlled area).

OWNER CONTROLLED AREA - Area depicted as the property boundary in BNP Radiological Emergency Response Plan Figure 1-1.1 Brunswick Site Plan.

Basis:

The Security Shift Supervision is defined as either the Security Shift Lieutenant or the Security Shift Sergeant.

This IC addresses the occurrence of a HOSTILE ACTION within the OWNER CONTROLLED AREA or notification of an aircraft attack threat. This event will require rapid response and assistance due to the possibility of the attack progressing to the PROTECTED AREA, or the need to prepare the plant and staff for a potential aircraft impact.

Timely and accurate communications between the Security Shift Supervision and the Control Room is essential for proper classification of a security-related event.

Security plans and terminology are based on the guidance provided by NEI 03-12, *Template for the Security Plan, Training and Qualification Plan, Safeguards Contingency Plan.*

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As time and conditions allow, these events require a heightened state of readiness by the plant staff and implementation of onsite protective measures (e.g., evacuation, dispersal or sheltering). The Alert declaration will also heighten the awareness of Offsite Response Organizations (OROs), allowing them to be better prepared should it be necessary to consider further actions.

This IC does not apply to incidents that are accidental events, acts of civil disobedience, or otherwise are not a HOSTILE ACTION perpetrated by a HOSTILE FORCE. Examples include the crash of a small aircraft, shots from hunters, physical disputes between employees, etc. Reporting of these types of events is adequately addressed by other EALs, or the requirements of 10 CFR § 73.71 or 10 CFR § 50.72.

This EAL is applicable for any HOSTILE ACTION occurring, or that has occurred, in the OWNER CONTROLLED AREA.

Emergency plans and implementing procedures are public documents; therefore, EALs should not incorporate Security-sensitive information. This includes information that may be advantageous to a potential adversary, such as the particulars concerning a specific threat or threat location. Security-sensitive information should be contained in non-public documents such as the BNP Physical Security Plan (ref. 1).

BNP Basis Reference(s):

1. BNP Physical Security Plan
2. SEC-NGGC-2170 Security Event Procedures
3. 0AOP-40.0 Security Events
4. NEI 99-01 HA1

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Category: H – Hazards

Subcategory: 1 – Security

Initiating Condition: Hostile action within the owner controlled area or airborne attack threat within 30 minutes

EAL:

HA1.2	**Alert**
A validated notification from NRC of an aircraft attack threat within 30 min. of the site	

Mode Applicability:

All

Definition(s):

None

Basis:

The Security Shift Supervision is defined as either the Security Shift Lieutenant or the Security Shift Sergeant.

This IC addresses the occurrence of a HOSTILE ACTION within the OWNER CONTROLLED AREA or notification of an aircraft attack threat. This event will require rapid response and assistance due to the possibility of the attack progressing to the PROTECTED AREA, or the need to prepare the plant and staff for a potential aircraft impact.

Timely and accurate communications between the Security Shift Supervision and the Control Room is essential for proper classification of a security-related event.

Security plans and terminology are based on the guidance provided by NEI 03-12, *Template for the Security Plan, Training and Qualification Plan, Safeguards Contingency Plan*.

As time and conditions allow, these events require a heightened state of readiness by the plant staff and implementation of onsite protective measures (e.g., evacuation, dispersal or sheltering). The Alert declaration will also heighten the awareness of Offsite Response Organizations (OROs), allowing them to be better prepared should it be necessary to consider further actions.

This IC does not apply to incidents that are accidental events, acts of civil disobedience, or otherwise are not a HOSTILE ACTION perpetrated by a HOSTILE FORCE. Examples include the crash of a small aircraft, shots from hunters, physical disputes between employees, etc.

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Reporting of these types of events is adequately addressed by other EALs, or the requirements of 10 CFR § 73.71 or 10 CFR § 50.72.

This EAL addresses the threat from the impact of an aircraft on the plant, and the anticipated arrival time is within 30 minutes. The intent of this EAL is to ensure that threat-related notifications are made in a timely manner so that plant personnel and OROs are in a heightened state of readiness. This EAL is met when the threat-related information has been validated in accordance with site-specific security procedures.

The NRC Headquarters Operations Officer (HOO) will communicate to the licensee if the threat involves an aircraft. The status and size of the plane may be provided by NORAD through the NRC.

In some cases, it may not be readily apparent if an aircraft impact within the OWNER CONTROLLED AREA was intentional (i.e., a HOSTILE ACTION). It is expected, although not certain, that notification by an appropriate Federal agency to the site would clarify this point. In this case, the appropriate federal agency is intended to be NORAD, FBI, FAA or NRC. The emergency declaration, including one based on other ICs/EALs, should not be unduly delayed while awaiting notification by a Federal agency.

Emergency plans and implementing procedures are public documents; therefore, EALs should not incorporate Security-sensitive information. This includes information that may be advantageous to a potential adversary, such as the particulars concerning a specific threat or threat location. Security-sensitive information should be contained in non-public documents such as the BNP Physical Security Plan (ref. 1).

BNP Basis Reference(s):

1. BNP Physical Security Plan
2. SEC-NGGC-2170 Security Event Procedures
3. 0AOP-40.0 Security Events
4. NEI 99-01 HA1

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Category: H – Hazards

Subcategory: 1 – Security

Initiating Condition: Hostile Action within the Protected Area

EAL:

HS1.1 **Site Area Emergency**
A HOSTILE ACTION is occurring or has occurred within the PROTECTED AREA as reported by the Security Shift Supervision

Mode Applicability:

All

Definition(s):

HOSTILE ACTION - An act toward BNP 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 BNP. Non-terrorism-based EALs should be used to address such activities (i.e., this may include violent acts between individuals in the owner controlled area).

PROTECTED AREA - The double-fenced security area with intrusion detection devices immediately surrounding the plant structures. The Protected Area is depicted in BNP Radiological Emergency Response Plan Figure 1-1.3 Brunswick Site Building and Onsite Emergency Facility Locations.

Basis:

The Security Shift Supervision is defined as either the Security Shift Lieutenant or the Security Shift Sergeant.

These individuals are the designated onsite personnel 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 BNP Physical Security Plan (Safeguards) information. (ref. 1)

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This IC addresses the occurrence of a HOSTILE ACTION within the PROTECTED AREA. This event will require rapid response and assistance due to the possibility for damage to plant equipment.

Timely and accurate communications between Security Shift Supervision and the Control Room is essential for proper classification of a security-related event.

Security plans and terminology are based on the guidance provided by NEI 03-12, *Template for the Security Plan, Training and Qualification Plan, Safeguards Contingency Plan.*

As time and conditions allow, these events require a heightened state of readiness by the plant staff and implementation of onsite protective measures (e.g., evacuation, dispersal or sheltering). The Site Area Emergency declaration will mobilize Offsite Response Organization (ORO) resources and have them available to develop and implement public protective actions in the unlikely event that the attack is successful in impairing multiple safety functions.

This IC does not apply to incidents that are accidental events, acts of civil disobedience, or otherwise are not a HOSTILE ACTION perpetrated by a HOSTILE FORCE. Examples include the crash of a small aircraft, shots from hunters, physical disputes between employees, etc. Reporting of these types of events is adequately addressed by other EALs, or the requirements of 10 CFR § 73.71 or 10 CFR § 50.72.

Emergency plans and implementing procedures are public documents; therefore, EALs should not incorporate Security-sensitive information. This includes information that may be advantageous to a potential adversary, such as the particulars concerning a specific threat or threat location. Security-sensitive information should be contained in non-public documents such as the BNP Physical Security Plan (ref. 1).

Escalation of the emergency classification level would be via IC HG1.

BNP Basis Reference(s):

1. BNP Physical Security Plan
2. SEC-NGGC-2170 Security Event Procedures
3. 0AOP-40.0 Security Events
4. NEI 99-01 HS1

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Category: H – Hazards

Subcategory: 1 – Security

Initiating Condition: Hostile Action resulting in loss of physical control of the facility

EAL:

HG1.1 General Emergency

A HOSTILE ACTION is occurring or has occurred within the PROTECTED AREA as reported by the Security Shift Supervision

AND EITHER of the following has occurred:

Any of the following safety functions cannot be controlled or maintained

- Reactivity
- RPV water level
- RCS heat removal

OR

Damage to spent fuel has occurred or is IMMINENT

Mode Applicability:

All

Definition(s):

HOSTILE ACTION - An act toward BNP 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 BNP. Non-terrorism-based EALs should be used to address such activities (i.e., this may include violent acts between individuals in the owner controlled area).

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IMMINENT - The trajectory of events or conditions is such that an EAL will be met within a relatively short period of time regardless of mitigation or corrective actions.

PROTECTED AREA - The double-fenced security area with intrusion detection devices immediately surrounding the plant structures. The Protected Area is depicted in BNP Radiological Emergency Response Plan Figure 1-1.3 Brunswick Site Building and Onsite Emergency Facility Locations.

Basis:

The Security Shift Supervision is defined as either the Security Shift Lieutenant or the Security Shift Sergeant.

This IC addresses an event in which a HOSTILE FORCE has taken physical control of the facility to the extent that the plant staff can no longer operate equipment necessary to maintain key safety functions. It also addresses a HOSTILE ACTION leading to a loss of physical control that results in actual or IMMINENT damage to spent fuel due to 1) damage to a spent fuel pool cooling system (e.g., pumps, heat exchangers, controls, etc.) or, 2) loss of spent fuel pool integrity such that sufficient water level cannot be maintained.

Timely and accurate communications between Security Shift Supervision and the Control Room is essential for proper classification of a security-related event.

Security plans and terminology are based on the guidance provided by NEI 03-12, *Template for the Security Plan, Training and Qualification Plan, Safeguards Contingency Plan.*

Emergency plans and implementing procedures are public documents; therefore, EALs should not incorporate Security-sensitive information. This includes information that may be advantageous to a potential adversary, such as the particulars concerning a specific threat or threat location. Security-sensitive information should be contained in non-public documents such as the BNP Physical Security Plan (ref.1).

BNP Basis Reference(s):

1. BNP Physical Security Plan
2. SEC-NGGC-2170 Security Event Procedures
3. 0AOP-40.0 Security Events
4. NEI 99-01 HG1

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Category: H – Hazards and Other Conditions Affecting Plant Safety

Subcategory: 2 – Seismic Event

Initiating Condition: Seismic event greater than OBE levels

EAL:

HU2.1	**Unusual Event**
Seismic event > OBE per 0AOP-13.0	

Mode Applicability:

All

Definition(s):

None

Basis:

Ground motion acceleration of 0.08g is the Operating Basis Earthquake for BNP (ref. 1).

Unit 2 has an active Kinematrics Condor Seismic Monitoring System with the following components used for seismic detection for the Brunswick Site:

> The system will detect and digitally record the response to actual earthquake loading in terms of acceleration time history from the existing accelerometers mounted in the Unit 2 -17ft. elevation (basement) of the Reactor Building and also at +89 foot elevation mounted on the Reactor Containment structure. The system will automatically evaluate the recorded acceleration time history in order to determine the response spectra of the events and compare those to the Operating Basis Earthquake (OBE) parameters graphically. It will also determine the exceedance of the OBE, and provides a hard copy of this comparison. The system will provide an immediate Event Alarm output signal at a trigger threshold value of 0.01g to alarm the existing Annunciator 1(2)UA-28 6-4 *SEISMIC EVENT* in the Control Room back to alert the Operators to a seismic event. (ref. 1, 2)

The BNP seismic instrumentation supports readily assessable OBE indications (> 0.08g acceleration) within the Control Room at panel 2-ENV-XU-823. 0AOP-13.0 provides the guidance for determining if the OBE earthquake threshold is exceeded. (ref. 3).

The Shift Manager or Site Emergency Coordinator may seek external verification if deemed appropriate (e.g., a call to the USGS, check internet news sources, etc.); however, the verification action must not preclude a timely emergency declaration.

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To avoid inappropriate emergency classification resulting from spurious actuation of the seismic instrumentation or felt motion not attributable to seismic activity, an offsite agency (USGS, National Earthquake Information Center) can confirm that an earthquake has occurred in the area of the plant. As stated above, such confirmation should not, however, preclude a timely emergency declaration. The NEIC can be contacted by calling **(303) 273-8500**. Select **option #1** and inform the analyst you wish to confirm recent seismic activity in the vicinity of BNP. Provide the analyst with the following BNP coordinates: **33° 57' 30" north latitude, 78° 00' 30" west longitude** (ref. 4). Alternatively, near real-time seismic activity can be accessed via the NEIC website:

http://earthquake.usgs.gov/eqcenter/

This IC addresses a seismic event that results in accelerations at the plant site greater than those specified for an Operating Basis Earthquake (OBE). An earthquake greater than an OBE but less than a Safe Shutdown Earthquake (SSE) should have no significant impact on safety-related systems, structures and components; however, some time may be required for the plant staff to ascertain the actual post-event condition of the plant (e.g., performs walk-downs and post-event inspections). Given the time necessary to perform walk-downs and inspections, and fully understand any impacts, this event represents a potential degradation of the level of safety of the plant.

Event verification with external sources should not be necessary during or following an OBE. Earthquakes of this magnitude should be readily felt by onsite personnel and recognized as a seismic event (e.g., lateral accelerations in excess of 0.08g). The Shift Manager or Site Emergency Coordinator may seek external verification if deemed appropriate (e.g., a call to the USGS, check internet news sources, etc.); however, the verification action must not preclude a timely emergency declaration.

Depending upon the plant mode at the time of the event, escalation of the emergency classification level would be via IC CA6 or SA8.

BNP Basis Reference(s):

1. Updated FSAR section 2.5.2.6
2. 1(2)APP-UA-28 6-4 Seismic Event
3. 0AOP-13.0 Operations During Hurricane, Flood Conditions, Tornado or Earthquake
4. Updated FSAR section 2.1.1.1
5. NEI 99-01 HU2

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Category: H – Hazards and Other Conditions Affecting Plant Safety

Subcategory: 3 – Natural or Technological Hazard

Initiating Condition: Hazardous event

EAL:

HU3.1	**Unusual Event**
A tornado strike within the PROTECTED AREA	

Mode Applicability:

All

Definition(s):

PROTECTED AREA - The double-fenced security area with intrusion detection devices immediately surrounding the plant structures. The Protected Area is depicted in BNP Radiological Emergency Response Plan Figure 1-1.3 Brunswick Site Building and Onsite Emergency Facility Locations.

Basis:

If damage is confirmed visually or by other in-plant indications, the event may be escalated to an Alert under EAL CA6.1 or SA8.1.

A tornado striking (touching down) within the Protected Area warrants declaration of an Unusual Event regardless of the measured wind speed at the meteorological tower. A tornado is defined as a violently rotating column of air in contact with the ground and extending from the base of a thunderstorm.

This IC addresses hazardous events that are considered to represent a potential degradation of the level of safety of the plant.

EAL HU3.1 addresses a tornado striking (touching down) within the PROTECTED AREA.

Escalation of the emergency classification level would be based on ICs in Recognition Categories R, F, S or C.

BNP Basis Reference(s):

1. NEI 99-01 HU3

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Category: H – Hazards and Other Conditions Affecting Plant Safety

Subcategory: 3 – Natural or Technological Hazard

Initiating Condition: Hazardous event

EAL:

HU3.2	**Unusual Event**
Internal room or area FLOODING of a magnitude sufficient to require manual or automatic electrical isolation of a SAFETY SYSTEM component needed for the current operating mode	

Mode Applicability:

All

Definition(s):

FLOODING - A condition where water is entering a room or area faster than installed equipment is capable of removal, resulting in a rise of water level within the room or area.

SAFETY SYSTEM - A system required for safe plant operation, cooling down the plant and/or placing it in the cold shutdown condition, including the ECCS. These are typically systems classified as safety-related (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.

Basis:

Deleted: Refer to Updated FSAR section 3.4.2 Protection From Internal Flooding to identify susceptible internal flooding areas (ref. 1).¶

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This IC addresses hazardous events that are considered to represent a potential degradation of the level of safety of the plant.

This EAL addresses FLOODING of a building room or area that results in operators isolating power to a SAFETY SYSTEM component due to water level or other wetting concerns. Classification is also required if the water level or related wetting causes an automatic isolation of a SAFETY SYSTEM component from its power source (e.g., a breaker or relay trip). To warrant classification, operability of the affected component must be required by Technical Specifications for the current operating mode.

Escalation of the emergency classification level would be based on ICs in Recognition Categories R, F, S or C.

BNP Basis Reference(s):

1. Updated FSAR section 3.4.2 Protection From Internal Flooding
2. NEI 99-01 HU3

Category: H – Hazards and Other Conditions Affecting Plant Safety

Subcategory: 3 – Natural or Technological Hazard

Initiating Condition: Hazardous event

EAL:

HU3.3	**Unusual Event**
Movement of personnel within the PROTECTED AREA is IMPEDED due to an event external to the PROTECTED AREA involving hazardous materials (e.g., an offsite chemical spill or toxic gas release)	

Mode Applicability:

All

Definition(s):

IMPEDE(D)- Personnel access to a room or area is hindered to an extent that extraordinary measures are necessary to facilitate entry of personnel into the affected room/area (e.g., requiring use of protective equipment, such as SCBAs, that is not routinely employed).

PROTECTED AREA - The double-fenced security area with intrusion detection devices immediately surrounding the plant structures. The Protected Area is depicted in BNP Radiological Emergency Response Plan Figure 1-1.3 Brunswick Site Building and Onsite Emergency Facility Locations.

Basis:

The following documents provide additional information on hazardous substances and spills.

- 0AOP-34.0 Chlorine Emergencies (Ref. 1)
- 0AOP-44.0 Sodium Hypochlorite or Acti-Brom Leak Response (Ref. 2)
- 0AOP-43.0 Hydrogen Emergency (Ref. 3)
- 0AOP-05.0 Radioactive Spills, High Radiation, and Airborne Activity (Ref. 4)
- Regulatory Guide 1.78, Evaluating the Habitability of a Nuclear Power Plant Control Room During a Postulated Hazardous Chemical Release, Table 1, Toxicity Limits (IDLH Limits) for Some Hazardous Chemicals (Ref. 5)

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This IC addresses hazardous events that are considered to represent a potential degradation of the level of safety of the plant.

This EAL addresses a hazardous materials event originating external to the PROTECTED AREA and of sufficient magnitude to impede the movement of personnel within the PROTECTED AREA.

Escalation of the emergency classification level would be based on ICs in Recognition Categories R, F, S or C.

BNP Basis Reference(s):

1. 0AOP-34.0 Chlorine Emergencies
2. 0AOP-44.0 Sodium Hypochlorite or Acti-Brom Leak Response
3. 0AOP-43.0 Hydrogen Emergency
4. 0AOP-05.0 Radioactive Spills, High Radiation, and Airborne Activity
5. Regulatory Guide 1.78, Evaluating the Habitability of a Nuclear Power Plant Control Room During a Postulated Hazardous Chemical Release, Table 1, Toxicity Limits (IDLH Limits) for Some Hazardous Chemicals
6. NEI 99-01 HU3

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Category: H – Hazards and Other Conditions Affecting Plant Safety

Subcategory: 3 – Natural or Technological Hazard

Initiating Condition: Hazardous event

EAL:

HU3.4	**Unusual Event**
A hazardous event that results in onsite conditions sufficient to prohibit the plant staff from accessing the site via personal vehicles (Note 7)	

Note 7: This EAL does not apply to routine traffic impediments such as fog, snow, ice, or vehicle breakdowns or accidents.

Mode Applicability:

All

Definition(s):

None

Basis:

Deleted: The 15 minute clocks starts when the ORO Director of Emergency Services (Brunswick and New Hanover Counties) and the Shift Manager agree that Onsite/Offsite conditions are sufficient to prohibit the plant staff from accessing the site via personal vehicles.

This IC addresses hazardous events that are considered to represent a potential degradation of the level of safety of the plant.

This EAL addresses a hazardous event, either onsite or offsite, that causes an onsite impediment to vehicle movement and significant enough to prohibit the plant staff from accessing the site using personal vehicles. Examples of such an event include site FLOODING caused by a hurricane, heavy rains, up-river water releases, dam failure, etc., or an onsite train derailment blocking the access road.

This EAL is not intended apply to routine impediments such as fog, snow, ice, or vehicle breakdowns or accidents, but rather to more significant conditions such as the Hurricane Andrew strike on Turkey Point in 1992, the flooding around the Cooper Station during the Midwest floods of 1993, or the flooding around Ft. Calhoun Station in 2011.

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Escalation of the emergency classification level would be based on ICs in Recognition Categories R, F, S or C.

BNP Basis Reference(s):

1. NEI 99-01 HU3

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Category: H – Hazards and Other Conditions Affecting Plant Safety

Subcategory: 3 – Natural or Technological Hazard

Initiating Condition: Hazardous event

EAL:

HU3.5 **Unusual Event**
Intake Canal water level > +19 ft Mean Sea Level
OR
Intake Canal water level < -7.75 ft Mean Sea Level

Mode Applicability:

All

BNP Basis:

The high Intake Canal level is the highest remotely measurable Intake Canal water level. Otherwise it would have been based based the plant design that Class I structures and engineered safety features systems are protected against still water flooding (elevation 22.0 feet). BNP is geographically located in close proximity to the Atlantic coastal storm track and has an approximate grade elevation of 20 feet above Mean Sea Level. Hurricanes and tropical storms are therefore, the most extreme weather phenomena that affect the site area. Potential subsequent flooding should be considered even though the plant structures were designed to compensate, via installed sump pumps, for a maximum site flooding depth of 22 feet above Mean Sea Level during the Maximum Probable Hurricane. (ref. 1).

The minimum water level predicted for the Maximum Probable Hurricane is -7.5 feet Mean Sea Level under special case circumstances. The abnormal operating procedure for a hurricane requires that each unit be shutdown prior to arrival of hurricane conditions at the site. The SW System has been analyzed in modes 4 and 5 for an intake canal water level of -7.75 feet Mean Sea Level corresponding to -8.63 feet Mean Sea Level in the pump suction bay for the maximum pressure drop, 0.88 feet, across the traveling screens. (ref. 2, 3).

This IC addresses hazardous events that are considered to represent a potential degradation of the level of safety of the plant.

This EAL addresses high and low external water levels as a result of a hurricane.

Escalation of the emergency classification level would be based on ICs in Recognition Categories R, F, S or C.

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BNP Basis Reference(s):

1. Updated FSAR section 2.4.10.2
2. Updated FSAR section 9.2.1.2.3
3. 0AOP-13.0 Operations During Hurricane, Flood Conditions, Tornado or Earthquake

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Category: H – Hazards and Other Conditions Affecting Plant Safety

Subcategory: 4 – Fire

Initiating Condition: FIRE potentially degrading the level of safety of the plant

EAL:

HU4.1 **Unusual Event**
A FIRE is not extinguished within 15 min. of **any** of the following FIRE detection indications (Note 1): • Report from the field (i.e., visual observation) • Receipt of multiple (more than 1) fire alarms or indications • Field verification of a single fire alarm **AND** The FIRE is located within **any** Table H-1 area

Note 1: The SEC should declare the event promptly upon determining that time limit has been exceeded, or will likely be exceeded.

Table H-1 **Fire Areas**
• Reactor Building • Diesel Generator Building • Diesel 4-Day Tank Rooms • Service Water Building • Turbine Building • Control Building • CSTs • Diesel Fuel Oil Storage Tank

Mode Applicability:

All

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Definition(s):

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.

Basis:

The 15 minute requirement begins with a credible notification that a fire is occurring, or receipt of multiple valid fire detection system alarms or field validation of a single fire alarm. The alarm is to be validated using available Control Room indications or alarms to prove that it is not spurious, or by reports from the field. Actual field reports must be made within the 15 minute time limit or a classification must be made. If a fire is verified to be occurring by field report, the 15 minute time limit is from the original receipt of the fire detection alarm.

Table H-1 Fire Areas are based on BNP-E-9. 010 NFPA 805 Nuclear Safety Capability Assessment (NSCA), and 0PFP-PBAA Power Block Auxiliary Areas Prefire Plan. Table H-1 Fire Areas include those structures containing functions and systems required for safe shutdown of the plant (SAFETY SYSTEMS) (ref. 1, 2).

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This IC addresses the magnitude and extent of FIRES that may be indicative of a potential degradation of the level of safety of the plant.

For EAL HU4.1 the intent of the 15-minute duration is to size the FIRE and to discriminate against small FIRES that are readily extinguished (e.g., smoldering waste paper basket). In addition to alarms, other indications of a FIRE could be a drop in fire main pressure, automatic activation of a suppression system, etc.

Upon receipt, operators will take prompt actions to confirm the validity of an initial fire alarm, indication, or report. For EAL assessment purposes, the emergency declaration clock starts at the time that the initial alarm, indication, or report was received, and not the time that a subsequent verification action was performed. Similarly, the fire duration clock also starts at the time of receipt of the initial alarm, indication or report.

Depending upon the plant mode at the time of the event, escalation of the emergency classification level would be via IC CA6 or SA8.

BNP Basis Reference(s):

1. BNP-E-9. 010 NFPA 805 Nuclear Safety Capability Assessment (NSCA)
2. 0PFP-PBAA Power Block Auxiliary Areas Prefire Plan
3. NEI 99-01 HU4

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Category: H – Hazards and Other Conditions Affecting Plant Safety

Subcategory: 4 – Fire

Initiating Condition: FIRE potentially degrading the level of safety of the plant

EAL:

HU4.2 Unusual Event

Receipt of a single fire alarm (i.e., no other indications of a FIRE)

AND

The fire alarm is indicating a FIRE within **any** Table H-1 area

AND

The existence of a FIRE is not verified within 30 min. of alarm receipt (Note 1)

Note 1: The SEC should declare the event promptly upon determining that time limit has been exceeded, or will likely be exceeded.

Table H-1 Fire Areas
• Reactor Building • Diesel Generator Building • Diesel 4-Day Tank Rooms • Service Water Building • Turbine Building • Control Building • CSTs • Diesel Fuel Oil Storage Tank

Mode Applicability:

All

Deleted: 122
Deleted: 204

Definition(s):

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.

Basis:

The 30 minute requirement begins upon receipt of a single valid fire detection system alarm. The alarm is to be validated using available Control Room indications or alarms to prove that it is not spurious, or by reports from the field. Actual field reports must be made within the 30 minute time limit or a classification must be made. If a fire is verified to be occurring by field report, classification shall be made based on EAL HU4.1.

Table H-1 Fire Areas are based on BNP-E-9. 010 NFPA 805 Nuclear Safety Capability Assessment (NSCA) and 0PFP-PBAA Power Block Auxiliary Areas Prefire Plan. Table H-1 Fire Areas include those structures containing functions and systems required for safe shutdown of the plant (SAFETY SYSTEMS) (ref. 1, 2).

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This IC addresses the magnitude and extent of FIRES that may be indicative of a potential degradation of the level of safety of the plant.

This EAL addresses receipt of a single fire alarm, and the existence of a FIRE is not verified (i.e., proved or disproved) within 30-minutes of the alarm. Upon receipt, operators will take prompt actions to confirm the validity of a single fire alarm. For EAL assessment purposes, the 30-minute clock starts at the time that the initial alarm was received, and not the time that a subsequent verification action was performed.

A single fire alarm, absent other indication(s) of a FIRE, may be indicative of equipment failure or a spurious activation, and not an actual FIRE. For this reason, additional time is allowed to verify the validity of the alarm. The 30-minute period is a reasonable amount of time to determine if an actual FIRE exists; however, after that time, and absent information to the contrary, it is assumed that an actual FIRE is in progress.

If an actual FIRE is verified by a report from the field, then HU4.1 is immediately applicable, and the emergency must be declared if the FIRE is not extinguished within 15-minutes of the report. If the alarm is verified to be due to an equipment failure or a spurious activation, and this verification occurs within 30-minutes of the receipt of the alarm, then this EAL is not applicable and no emergency declaration is warranted.

Appendix R to 10 CFR 50, states in part:

> Criterion 3 of Appendix A to this part specifies that "Structures, systems, and components important to safety shall be designed and located to minimize, consistent with other safety requirements, the probability and effect of fires and explosions."

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When considering the effects of fire, those systems associated with achieving and maintaining safe shutdown conditions assume major importance to safety because damage to them can lead to core damage resulting from loss of coolant through boil-off.

Because fire may affect safe shutdown systems and because the loss of function of systems used to mitigate the consequences of design basis accidents under post-fire conditions does not per se impact public safety, the need to limit fire damage to systems required to achieve and maintain safe shutdown conditions is greater than the need to limit fire damage to those systems required to mitigate the consequences of design basis accidents.

In addition, Appendix R to 10 CFR 50, requires, among other considerations, the use of 1-hour fire barriers for the enclosure of cable and equipment and associated non-safety circuits of one redundant train (G.2.c). As used in this EAL, the 30-minutes to verify a single alarm is well within this worst-case 1-hour time period.

It should be noted however, BNP is not an Appendix R plant but rather falls under the requirements of NFPA-805 for fire protection.

Depending upon the plant mode at the time of the event, escalation of the emergency classification level would be via IC CA6 or SA8.

BNP Basis Reference(s):

1. BNP-E-9. 010 NFPA 805 Nuclear Safety Capability Assessment (NSCA)
2. 0PFP-PBAA Power Block Auxiliary Areas Prefire Plan
3. NEI 99-01 HU4

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Category: H – Hazards and Other Conditions Affecting Plant Safety

Subcategory: 4 – Fire

Initiating Condition: FIRE potentially degrading the level of safety of the plant

EAL:

HU4.3	**Unusual Event**
A FIRE within the plant PROTECTED AREA not extinguished within 60 min. of the initial report, alarm or indication (Note 1)	

Note 1: The SEC should declare the event promptly upon determining that time limit has been exceeded, or will likely be exceeded.

Mode Applicability:

All

Definition(s):

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.

PROTECTED AREA - The double-fenced security area with intrusion detection devices immediately surrounding the plant structures. The Protected Area is depicted in BNP Radiological Emergency Response Plan Figure 1-1.3 Brunswick Site Building and Onsite Emergency Facility Locations.

Basis:

This IC addresses the magnitude and extent of FIRES that may be indicative of a potential degradation of the level of safety of the plant.

In addition to a FIRE addressed by EAL HU4.1 or HU4.2, a FIRE within the plant PROTECTED AREA not extinguished within 60-minutes may also potentially degrade the level of plant safety.

Depending upon the plant mode at the time of the event, escalation of the emergency classification level would be via IC CA6 or SA8.

BNP Basis Reference(s):

1. NEI 99-01 HU4

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Category: H – Hazards and Other Conditions Affecting Plant Safety

Subcategory: 4 – Fire

Initiating Condition: FIRE potentially degrading the level of safety of the plant

EAL:

HU4.4	**Unusual Event**
A FIRE within the plant PROTECTED AREA that requires firefighting support by an offsite fire response agency to extinguish	

Mode Applicability:

All

Definition(s):

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.

PROTECTED AREA - The double-fenced security area with intrusion detection devices immediately surrounding the plant structures. The Protected Area is depicted in BNP Radiological Emergency Response Plan Figure 1-1.3 Brunswick Site Building and Onsite Emergency Facility Locations.

Basis:

The dispatch of an offsite firefighting agency to the site requires an emergency declaration only if it is needed to actively support firefighting efforts because the fire is beyond the capability of the Fire Brigade to extinguish.

This IC addresses the magnitude and extent of FIRES that may be indicative of a potential degradation of the level of safety of the plant.

If a FIRE within the plant PROTECTED AREA is of sufficient size to require a response by an offsite firefighting agency (e.g., a local town Fire Department), then the level of plant safety is potentially degraded. The dispatch of an offsite firefighting agency to the site requires an emergency declaration only if it is needed to actively support firefighting efforts because the fire is beyond the capability of the Fire Brigade to extinguish. Declaration is not necessary if the agency resources are placed on stand-by, or supporting post-extinguishment recovery or investigation actions.

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Depending upon the plant mode at the time of the event, escalation of the emergency classification level would be via IC CA6 or SA8.

BNP Basis Reference(s):

1. NEI 99-01 HU4

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Category: H – Hazards and Other Conditions Affecting Plant Safety

Subcategory: 5 – Hazardous Gases

Initiating Condition: Gaseous release IMPEDING access to equipment necessary for normal plant operations, cooldown or shutdown

EAL:

HA5.1 Alert

Release of a toxic, corrosive, asphyxiant or flammable gas into **any** Table H-2 rooms or areas

AND

Entry into the room or area is prohibited or IMPEDED (Note 5)

Note 5: If the equipment in the listed room or area was already inoperable or out-of-service before the event occurred, then no emergency classification is warranted.

Table H-2 Safe Operation & Shutdown Areas

Room/Area	Mode Applicability
Reactor Building -17' North RHR Unit-1 & 2	3, 4, 5
Reactor Building -17' South RHR Unit-1 & 2	3, 4, 5
Reactor Building 20' East & West MCC Areas Unit-1 & 2	3, 4, 5
Reactor Building 20' Pipe Tunnel Unit-1 & 2	3, 4, 5

Mode Applicability:

All

Definition(s):

IMPEDE(D)- Personnel access to a room or area is hindered to an extent that extraordinary measures are necessary to facilitate entry of personnel into the affected room/area (e.g., requiring use of protective equipment, such as SCBAs, that is not routinely employed).

Basis:

If the equipment in the listed room or area was already inoperable, or out-of-service, before the event occurred, then no emergency should be declared since the event will have no adverse impact beyond that already allowed by Technical Specifications at the time of the event.

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The list of plant rooms or areas with entry-related mode applicability identified specify those rooms or areas that contain equipment which require a manual/local action as specified in operating procedures used for normal plant operation, cooldown and shutdown. Rooms or areas in which actions of a contingent or emergency nature would be performed (e.g., an action to address an off-normal or emergency condition such as emergency repairs, corrective measures or emergency operations) are not included. In addition, the list specifies the plant mode(s) during which entry would be required for each room or area (ref. 1).

The following documents provide additional information on hazardous substances and spills.

- 0AOP-34.0 Chlorine Emergencies (Ref. 2)
- 0AOP-44.0 Sodium Hypochlorite or Acti-Brom Leak Response (Ref. 3)
- 0AOP-43.0 Hydrogen Emergency (Ref. 4)

This IC addresses an event involving a release of a hazardous gas that precludes or impedes access to equipment necessary to maintain normal plant operation, or required for a normal plant cooldown and shutdown. This condition represents an actual or potential substantial degradation of the level of safety of the plant.

An Alert declaration is warranted if entry into the affected room/area is, or may be, procedurally required during the plant operating mode in effect at the time of the gaseous release. The emergency classification is not contingent upon whether entry is actually necessary at the time of the release.

Evaluation of the IC and EAL do not require atmospheric sampling; it only requires the Site Emergency Coordinator's judgment that the gas concentration in the affected room/area is sufficient to preclude or significantly impede procedurally required access. This judgment may be based on a variety of factors including an existing job hazard analysis, report of ill effects on personnel, advice from a subject matter expert or operating experience with the same or similar hazards. Access should be considered as impeded if extraordinary measures are necessary to facilitate entry of personnel into the affected room/area (e.g., requiring use of protective equipment, such as SCBAs, that is not routinely employed).

An emergency declaration is not warranted if any of the following conditions apply:

- The plant is in an operating mode different than the mode specified for the affected room/area (i.e., entry is not required during the operating mode in effect at the time of the gaseous release). For example, the plant is in Mode 1 when the gaseous release occurs, and the procedures used for normal operation, cooldown and shutdown do not require entry into the affected room until Mode 4.
- The gas release is a planned activity that includes compensatory measures which address the temporary inaccessibility of a room or area (e.g., fire suppression system testing).

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- The action for which room/area entry is required is of an administrative or record keeping nature (e.g., normal rounds or routine inspections).
- The access control measures are of a conservative or precautionary nature, and would not actually prevent or impede a required action.
- If the equipment in the listed room or area was already inoperable, or out-of-service, before the event occurred, then no emergency should be declared since the event will have no adverse impact 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.

This EAL does not apply to firefighting activities that automatically or manually activate a fire suppression system in an area, or to intentional inerting of containment.

Escalation of the emergency classification level would be via Recognition Category R, C or F ICs.

BNP Basis Reference(s):

1. Attachment 3 Safe Operation & Shutdown Areas Tables R-2 & H-2 Bases
2. 0AOP-34.0 Chlorine Emergencies
3. 0AOP-44.0 Sodium Hypochlorite or Acti-Brom Leak Response
4. 0AOP-43.0 Hydrogen Emergency
5. NEI 99-01 HA5

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Category: H – Hazards and Other Conditions Affecting Plant Safety

Subcategory: 6 – Control Room Evacuation

Initiating Condition: Control Room evacuation resulting in transfer of plant control to alternate locations

EAL:

HA6.1 **Alert**
An event has resulted in plant control being transferred from the Control Room to the Remote Shutdown Panels

Mode Applicability:

All

Definition(s):

None

Basis:

The Shift Manager (SM) determines if the Control Room is inoperable and requires evacuation. Control Room inhabitability may be caused by fire, dense smoke, noxious fumes, bomb threat in or adjacent to the Control Room, or other life threatening conditions (Ref. 1, 2).

Inability to establish plant control from outside the Control Room escalates this event to a Site Area Emergency per EAL HS6.1.

This IC addresses an evacuation of the Control Room that results in transfer of plant control to alternate locations outside the Control Room. The loss of the ability to control the plant from the Control Room is considered to be a potential substantial degradation in the level of plant safety.

Following a Control Room evacuation, control of the plant will be transferred to alternate shutdown locations. The necessity to control a plant shutdown from outside the Control Room, in addition to responding to the event that required the evacuation of the Control Room, will present challenges to plant operators and other on-shift personnel. Activation of the ERO and emergency response facilities will assist in responding to these challenges.

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Escalation of the emergency classification level would be via IC HS6.

BNP Basis Reference(s):

1. 0AOP-32.0, Plant Shutdown from Outside Control Room
2. 0PLP-01.5 Alternate Shutdown Capability Controls
3. NEI 99-01 HA6

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Category: H – Hazards and Other Conditions Affecting Plant Safety

Subcategory: 6 – Control Room Evacuation

Initiating Condition: Inability to control a key safety function from outside the Control Room

EAL:

HS6.1 Site Area Emergency

An event has resulted in plant control being transferred from the Control Room to the Remote Shutdown Panels

AND

Control of **any** of the following key safety functions is not reestablished within 22.5 min. (Note 1):

- Reactivity (Modes 1 and 2 **only**)
- RPV water level
- RPV pressure (Modes 1, 2, 3 and 4 **only**)

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Note 1: The SEC should declare the event promptly upon determining that time limit has been exceeded, or will likely be exceeded.

Mode Applicability:

1 - Power Operations, 2 - Startup, 3 - Hot Shutdown, 4 - Cold Shutdown, 5 - Refueling

Deleted: All

Definition(s):

None

Basis:

The Shift Manager determines if the Control Room is inoperable and requires evacuation. Control Room inhabitability may be caused by fire, dense smoke, noxious fumes, bomb threat in or adjacent to the Control Room, or other life threatening conditions (Ref. 1, 2).

The immediate actions of 0AOP-32 direct a reactor scram prior to evacuating the Control Room. Local control of high pressure injection sources and Safety Relief Valves (SRVs) establishes control of RPV water level and pressure.

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This IC addresses an evacuation of the Control Room that results in transfer of plant control to alternate locations, and the control of a key safety function cannot be reestablished in a timely manner. The failure to gain control of a key safety function following a transfer of plant control to alternate locations is a precursor to a challenge to one or more fission product barriers within a relatively short period of time.

The determination of whether or not "control" is established at the remote safe shutdown location(s) is based on Site Emergency Coordinator judgment. The Site Emergency Coordinator is expected to make a reasonable, informed judgment within 22.5 minutes whether or not the operating staff has control of key safety functions from the remote safe shutdown location(s) (ref. 3).

Escalation of the emergency classification level would be via IC FG1 or CG1

BNP Basis Reference(s):

1. 0AOP-32.0, Plant Shutdown from Outside Control Room
2. 0PLP-01.5 Alternate Shutdown Capability Controls
3. Calculation No. BNP-E-9.007 ASSD Manual Action Feasibility
4. NEI 99-01 HS6

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Category: H – Hazards and Other Conditions Affecting Plant Safety

Subcategory: 7 – SEC Judgment

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

EAL:

HU7.1	**Unusual Event**
Other conditions exist which in the judgment of the Site Emergency Coordinator 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

Definition(s):

None

Basis:

The Site Emergency Coordinator is the designated onsite individual having the responsibility and authority for implementing the BNP Emergency Response Plan. The Operations Shift Manager (SM) initially acts in the capacity of the Site Emergency Coordinator and takes actions as outlined in the Emergency Plan implementing procedures. If required by the emergency classification or if deemed appropriate by the Site Emergency Coordinator, emergency response personnel are notified and instructed to report to their emergency response locations. In this manner, the individual usually in charge of activities in the Control Room is responsible for initiating the necessary emergency response, but Plant Management is expected to manage the emergency response as soon as available to do so in anticipation of the possible wide-ranging responsibilities associated with managing a major emergency (ref. 1).

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

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BNP Basis Reference(s):

1. 0ERP BNP Radiological Emergency Response Plan section 3.0 Emergency Response Organization
2. NEI 99-01 HU7

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Category: H – Hazards and Other Conditions Affecting Plant Safety

Subcategory: 7 – SEC Judgment

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

EAL:

HA7.1	**Alert**

Other conditions exist which, in the judgment of the Site Emergency Coordinator, 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.

Mode Applicability:

All

Definition(s):

HOSTILE ACTION - An act toward BNP 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 BNP. Non-terrorism-based EALs should be used to address such activities (i.e., this may include violent acts between individuals in the owner controlled area).

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Basis:

The Site Emergency Coordinator is the designated onsite individual having the responsibility and authority for implementing the BNP Emergency Response Plan. The Operations Shift Manager (SM) initially acts in the capacity of the Site Emergency Coordinator and takes actions as outlined in the Emergency Plan implementing procedures. If required by the emergency classification or if deemed appropriate by the Site Emergency Coordinator, emergency response personnel are notified and instructed to report to their emergency response locations. In this manner, the individual usually in charge of activities in the Control Room is responsible for initiating the necessary emergency response, but Plant Management is expected to manage the emergency response as soon as available to do so in anticipation of the possible wide-ranging responsibilities associated with managing a major emergency (ref.1).

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

BNP Basis Reference(s):

1. 0ERP BNP Radiological Emergency Response Plan section 3.0 Emergency Response Organization
2. NEI 99-01 HA7

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Category: H – Hazards and Other Conditions Affecting Plant Safety

Subcategory: 7 – SEC Judgment

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

EAL:

HS7.1 **Site Area Emergency**
Other conditions exist which in the judgment of the Site Emergency Coordinator 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 beyond the site boundary

Mode Applicability:

All

Definition(s):

HOSTILE ACTION - An act toward BNP 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 BNP. Non-terrorism-based EALs should be used to address such activities (i.e., this may include violent acts between individuals in the owner controlled area)

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Basis:

The Site Emergency Coordinator is the designated onsite individual having the responsibility and authority for implementing the BNP Emergency Response Plan. The Operations Shift Manager (SM) initially acts in the capacity of the Site Emergency Coordinator and takes actions as outlined in the Emergency Plan implementing procedures. If required by the emergency classification or if deemed appropriate by the Site Emergency Coordinator, emergency response personnel are notified and instructed to report to their emergency response locations. In this manner, the individual usually in charge of activities in the Control Room is responsible for initiating the necessary emergency response, but Plant Management is expected to manage the emergency response as soon as available to do so in anticipation of the possible wide-ranging responsibilities associated with managing a major emergency (ref. 1).

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

BNP Basis Reference(s):

1. 0ERP BNP Radiological Emergency Response Plan section 3.0 Emergency Response Organization
2. NEI 99-01 HS7

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Category: H – Hazards and Other Conditions Affecting Plant Safety

Subcategory: 7 – SEC Judgment

Initiating Condition: Other conditions exist which in the judgment of the Site Emergency Coordinator warrant declaration of a General Emergency

EAL:

HG7.1	**General Emergency**
Other conditions exist which in the judgment of the Site Emergency Coordinator 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 offsite for more than the immediate site area	

Mode Applicability:

All

Definition(s):

HOSTILE ACTION - An act toward BNP 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 BNP. Non-terrorism-based EALs should be used to address such activities (i.e., this may include violent acts between individuals in the owner controlled area).

IMMINENT - The trajectory of events or conditions is such that an EAL will be met within a relatively short period of time regardless of mitigation or corrective actions.

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Basis:

The Site Emergency Coordinator is the designated onsite individual having the responsibility and authority for implementing the BNP Emergency Response Plan. The Operations Shift Manager(SM) initially acts in the capacity of the Site Emergency Coordinator and takes actions as outlined in the Emergency Plan implementing procedures. If required by the emergency classification or if deemed appropriate by the Site Emergency Coordinator, emergency response personnel are notified and instructed to report to their emergency response locations. In this manner, the individual usually in charge of activities in the Control Room is responsible for initiating the necessary emergency response, but Plant Management is expected to manage the emergency response as soon as available to do so in anticipation of the possible wide-ranging responsibilities associated with managing a major emergency (ref. 1).

Releases can reasonably be expected to exceed EPA PAG plume exposure levels outside the Site Boundary.

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

BNP Basis Reference(s):

1. 0ERP BNP Radiological Emergency Response Plan section 3.0 Emergency Response Organization
2. NEI 99-01 HG7

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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 Emergency AC Power

Loss of emergency 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 sources for 4160 V emergency buses.

2. Loss of Vital DC Power

Loss of emergency 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 vital plant 125 VDC power sources.

3. Loss of Control Room Indications

Certain events that degrade plant operator ability to effectively assess plant conditions within the plant warrant emergency classification. Losses of indicators are in this subcategory.

4. RCS Activity

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 (2% - 5% clad failures) is indicative of fuel failures and is covered under the Fission Product Barrier Degradation category. 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.

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5. RCS Leakage

The reactor pressure vessel provides a volume for the coolant that covers the reactor core. The reactor pressure 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 indicates potential pipe cracks that may propagate to an extent threatening fuel clad, RCS and Primary Containment integrity.

6. RPS Failure

This subcategory includes 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 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 Primary Containment integrity.

7. Loss of Communications

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

8. Hazardous Event Affecting Safety Systems

Various natural and technological events that result in degraded plant safety system performance or significant visible damage warrant emergency classification under this subcategory.

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Category: S – System Malfunction

Subcategory: 1 – Loss of Emergency AC Power

Initiating Condition: Loss of **all** offsite AC power capability to emergency buses for 15 minutes or longer

EAL:

SU1.1	**Unusual Event**
Loss of **all** offsite AC power capability, Table S-5, to Emergency 4 KV Buses E1(E3) and E2(E4) for ≥ 15 min. (Note 1)	

Note 1: The SEC should declare the event promptly upon determining that time limit has been exceeded, or will likely be exceeded.

Table S-5 AC Power Sources
Offsite:
• SAT
• UAT backfed through MPT (**only if** already aligned)
Onsite:
• UAT via Main Generator
• EDG1(3)
• EDG2(4)

Mode Applicability:

1 - Power Operations, 2 - Startup, 3 - Hot Shutdown

Definition(s):

None

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Basis:

The 4160 VAC System provides the power requirements for operation and safe shutdown of the plant. 4160 VAC power is divided into Balance of Plant (BOP Bus), and Emergency (E-Bus) distribution. The BOP Buses consist of the Conventional Buses (Common A/B Buses; Buses 1B/2B, 1C/2C, 1D/2D) and Caswell Beach Buses A/B. The Emergency switchgear are Buses E1/E3 (Division I) and E2/E4 (Division II).

The E-Buses are normally powered through the respective BOP Buses (1D to E1, 1C to E2, 2D to E3, 2C to E4) via a master/slave breaker arrangement. Each E-Bus has a dedicated Diesel Generator to supply an emergency source of power to safe shutdown loads in the event of a loss of the normal power source or loss of offsite power. The DGs will automatically start and tie onto the E-Buses if the normal power source or offsite power is lost. The DGs can provide power to the E-Buses only. In the event the diesel generator is unavailable for an E-Bus, crosstie capability exists for each E-Bus from the same division of the opposite unit (E1 to E3, E2 to E4). Although the E-Buses within the unit also have crosstie capability, this alignment is not permitted by plant procedures, with the exception of E1 to E2 during specific Alternate Safe Shutdown (ASSD) conditions.

During periods of unit shutdown, when the Startup Auxiliary Transformer (SAT) would be the only normal source of offsite power, the Unit Auxiliary Transformer (UAT) can be made available by establishing a UAT backfeed. Backfeed from the UAT will require the use of keys for the control selector switches and opening of the respective generator's manual no-load disconnect. (Ref. 1, 2)

The 15-minute interval was selected as a threshold to exclude transient or momentary power losses.

This IC addresses a prolonged loss of offsite power. The loss of offsite power sources renders the plant more vulnerable to a complete loss of power to AC emergency buses. This condition represents a potential reduction in the level of safety of the plant.

For emergency classification purposes, "capability" means that an offsite AC power source(s) is available to the emergency buses, whether or not the buses are powered from it.

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

Escalation of the emergency classification level would be via IC SA1.

BNP Basis Reference(s):

1. BNP Updated FSAR Chapter 8
2. 1(2)OP-50 Plant Electric System Operating Procedure
3. NEI 99-01 SU1

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Category: S – System Malfunction

Subcategory: 1 – Loss of Emergency AC Power

Initiating Condition: Loss of **all but one** AC power source to emergency buses for 15 minutes or longer

EAL:

SA1.1 Alert

AC power capability to Emergency 4 KV Buses E1(E3) and E2(E4) reduced to a single power source, Table S-5, for ≥ 15 min. (Note 1)

AND

Any additional single power source failure will result in loss of **all** unit-specific AC power to SAFETY SYSTEMS

Note 1: The SEC should declare the event promptly upon determining that time limit has been exceeded, or will likely be exceeded.

Table S-5 AC Power Sources
Offsite:
• SAT
• UAT backfed through MPT **(only if** already aligned)
Onsite:
• UAT via Main Generator
• EDG1(3)
• EDG2(4)

Mode Applicability:

1 - Power Operations, 2 - Startup, 3 - Hot Shutdown

Definition(s):

SAFETY SYSTEM - A system required for safe plant operation, cooling down the plant and/or placing it in the cold shutdown condition, including the ECCS. These are typically systems classified as safety-related (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.

Basis:

The 4160 VAC System provides the power requirements for operation and safe shutdown of the plant. 4160 VAC power is divided into Balance Of Plant (BOP Bus), and Emergency (E-Bus) distribution. The BOP Buses consist of the Conventional Buses (Common A/B Buses; Buses 1B/2B, 1C/2C, 1D/2D) and Caswell Beach Buses A/B. The Emergency switchgear are Buses E1/E3 (Division I) and E2/E4 (Division II).

The E-Buses are normally powered through the respective BOP Buses (1D to E1, 1C to E2, 2D to E3, 2C to E4) via a master/slave breaker arrangement. Each E-Bus has a dedicated Diesel Generator to supply an emergency source of power to safe shutdown loads in the event of a loss of the normal power source or loss of offsite power. The DGs will automatically start and tie onto the E-Buses if the normal power source or offsite power is lost. The DGs can provide power to the E-Buses only. In the event the diesel generator is unavailable for an E-Bus, crosstie capability exists for each E-Bus from the same division of the opposite unit (E1 to E3, E2 to E4). Although the E-Buses within the unit also have crosstie capability, this alignment is not permitted by plant procedures, with the exception of E1 to E2 during specific Alternate Safe Shutdown (ASSD) conditions.

During periods of unit shutdown, when the Startup Auxiliary Transformer (SAT) would be the only normal source of offsite power, the Unit Auxiliary Transformer (UAT) can be made available by establishing a UAT backfeed. Backfeed from the UAT will require the use of keys for the control selector switches and opening of the respective generator's manual no-load disconnect (Ref. 1, 2).

The 15-minute interval was selected as a threshold to exclude transient or momentary power losses. If the capability of a second source of emergency bus power is not restored within 15 minutes, an Alert is declared under this EAL.

Because 2 RHR pumps on each unit are powered from the unaffected unit, the words "unit-specific" have been added to clarify that the cross-connected RHR pump power cannot be credited as an AC power source relative to this EAL.

This IC describes a significant degradation of offsite and onsite AC power sources such that any additional single failure would result in a loss of all AC power to SAFETY SYSTEMS. In this condition, the sole AC power source may be powering one, or more than one, train of safety-related equipment. This IC provides an escalation path from IC SU1.

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An "AC power source" is a source recognized in AOPs, and capable of supplying required power to an emergency bus. Some examples of this condition are presented below.

- A loss of all offsite power with a concurrent failure of all but one emergency power source (e.g., an onsite diesel generator).
- A loss of all offsite power and loss of all emergency power sources (e.g., onsite diesel generators) with a single train of emergency buses being back-fed from the unit main generator.
- A loss of emergency power sources (e.g., onsite diesel generators) with a single train of emergency buses being back-fed from an offsite power source.

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

Escalation of the emergency classification level would be via IC SS1.

BNP Basis Reference(s):

1. BNP Updated FSAR Chapter 8
2. 1(2)OP-50 Plant Electric System Operating Procedure
3. NEI 99-01 SA1

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Category: S – System Malfunction

Subcategory: 1 – Loss of Emergency AC Power

Initiating Condition: Loss of **all** offsite and **all** onsite AC power to emergency buses for 15 minutes or longer

EAL:

SS1.1	**Site Area Emergency**
Loss of **all** offsite and **all** onsite AC power capability to Emergency 4 KV Buses E1(E3) and E2(E4) for ≥ 15 min. (Note 1)	

Note 1: The SEC should declare the event promptly upon determining that time limit has been exceeded, or will likely be exceeded.

Mode Applicability:

1 - Power Operations, 2 - Startup, 3 - Hot Shutdown

Definition(s):

None

Basis:

This EAL is indicated by the loss of all offsite and onsite AC power capability to Emergency 4 KV Buses E1(E3) and E2(E4) for greater than or equal to 15 minutes.

The 4160 VAC System provides the power requirements for operation and safe shutdown of the plant. 4160 VAC power is divided into Balance Of Plant (BOP Bus), and Emergency (E-Bus) distribution. The BOP Buses consist of the Conventional Buses (Common A/B Buses; Buses 1B/2B, 1C/2C, 1D/2D) and Caswell Beach Buses A/B. The Emergency switchgear are Buses E1/E3 (Division I) and E2/E4 (Division II).

The E-Buses are normally powered through the respective BOP Buses (1D to E1, 1C to E2, 2D to E3, 2C to E4) via a master/slave breaker arrangement. Each E-Bus has a dedicated Diesel Generator to supply an emergency source of power to safe shutdown loads in the event of a loss of the normal power source or loss of offsite power. The DGs will automatically start and tie onto the E-Buses if the normal power source or offsite power is lost. The DGs can provide power to the E-Buses only. In the event the diesel generator is unavailable for an E-Bus, crosstie capability exists for each E-Bus from the same division of the opposite unit (E1 to E3, E2 to E4). Although the E-Buses within the unit also have crosstie capability, this alignment is not permitted by plant procedures, with the exception of E1 to E2 during specific Alternate Safe Shutdown (ASSD) conditions.

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During periods of unit shutdown, when the Startup Auxiliary Transformer (SAT) would be the only normal source of offsite power, the Unit Auxiliary Transformer (UAT) can be made available by establishing a UAT backfeed. Backfeed from the UAT will require the use of keys for the control selector switches and opening of the respective generator's manual no-load disconnect (Ref. 1, 2)

The 15-minute interval was selected as a threshold to exclude transient or momentary power losses. The interval begins when both offsite and onsite AC power are lost.

This IC addresses a total loss of AC power that compromises the performance of all SAFETY SYSTEMS requiring electric power including those necessary for emergency core cooling, containment heat removal/pressure control, spent fuel heat removal and the ultimate heat sink. In addition, fission product barrier monitoring capabilities may be degraded under these conditions. This IC represents a condition that involves actual or likely major failures of plant functions needed for the protection of the public.

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

Escalation of the emergency classification level would be via ICs RG1, FG1 or SG1.

BNP Basis Reference(s):

1. BNP Updated FSAR Chapter 8
2. 1(2)OP-50 Plant Electric System Operating Procedure
3. 0AOP-36.2 Station Blackout
4. NEI 99-01 SS1

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Category: S –System Malfunction

Subcategory: 1 – Loss of Emergency AC Power

Initiating Condition: Prolonged loss of **all** offsite and **all** onsite AC power to emergency buses **OR** loss of **all** emergency AC and vital DC power sources for 15 minutes or longer

EAL:

SG1.1	**General Emergency**

Loss of **all** offsite and **all** onsite AC power capability to Emergency 4 KV Buses E1(E3) and E2(E4)

AND EITHER:

- Restoration of at least one emergency bus in < 4 hours is **not** likely (Note 1)
- RPV water level **cannot** be restored and maintained > MSCRWL (LL-4)

Note 1: The SEC should declare the event promptly upon determining that time limit has been exceeded, or will likely be exceeded.

Mode Applicability:

1 - Power Operations, 2 - Startup, 3 - Hot Shutdown

Definition(s):

None

Basis:

This EAL is indicated by the extended loss of all offsite and onsite AC power capability to 4 KV emergency buses E1(E3) and E2(E4) either for greater then the BNP Station Blackout (SBO) coping analysis time (4 hrs.) (ref. 1) or that has resulted in indications of an actual loss of adequate core cooling.

Indication of continuing core cooling degradation is manifested by the inability to restore and maintain RPV water level above the Minimum Steam Cooling Reactor Water Level (LL-4) (ref. 4, 5).

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The 4160 VAC System provides the power requirements for operation and safe shutdown of the plant. 4160 VAC power is divided into Balance of Plant (BOP Bus), and Emergency (E-Bus) distribution. The BOP Buses consist of the Conventional Buses (Common A/B Buses; Buses 1B/2B, 1C/2C, 1D/2D) and Caswell Beach Buses A/B. The Emergency switchgear are Buses E1/E3 (Division I) and E2/E4 (Division II).

The E-Buses are normally powered through the respective BOP Buses (1D to E1, 1C to E2, 2D to E3, 2C to E4) via a master/slave breaker arrangement. Each E-Bus has a dedicated Diesel Generator to supply an emergency source of power to safe shutdown loads in the event of a loss of the normal power source or loss of offsite power. The DGs will automatically start and tie onto the E-Buses if the normal power source or offsite power is lost. The DGs can provide power to the E-Buses only. In the event the diesel generator is unavailable for an E-Bus, crosstie capability exists for each E-Bus from the same division of the opposite unit (E1 to E3, E2 to E4). Although the E-Buses within the unit also have crosstie capability, this alignment is not permitted by plant procedures, with the exception of E1 to E2 during specific Alternate Safe Shutdown (ASSD) conditions.

During periods of unit shutdown, when the Startup Auxiliary Transformer (SAT) would be the only normal source of offsite power, the Unit Auxiliary Transformer (UAT) can be made available by establishing a UAT backfeed. Backfeed from the UAT will require the use of keys for the control selector switches and opening of the respective generator's manual no-load disconnect (Ref. 2, 3).

Four hours is the station blackout coping time (ref 1).

Indication of continuing core cooling degradation must be based on fission product barrier monitoring with particular emphasis on Site Emergency Coordinator judgment as it relates to imminent Loss or Potential Loss of fission product barriers and degraded ability to monitor fission product barriers. Indication of continuing core cooling degradation is manifested by an RPV level that cannot be restored and maintained > MSCRWL (LL-4) (ref. 4, 5). Core submergence is the most desirable means of core cooling, however when RPV level is below TAF, the uncovered portion of the core can be cooled by less reliable means (i.e., steam cooling or spray cooling).

This IC addresses a prolonged loss of all power sources to AC emergency buses. A loss of all AC power compromises the performance of all SAFETY SYSTEMS requiring electric power including those necessary for emergency core cooling, containment heat removal/pressure control, spent fuel heat removal and the ultimate heat sink. A prolonged loss of these buses will lead to a loss of one or more fission product barriers. In addition, fission product barrier monitoring capabilities may be degraded under these conditions.

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The EAL should require declaration of a General Emergency prior to meeting the thresholds for IC FG1. This will allow additional time for implementation of offsite protective actions.

Escalation of the emergency classification from Site Area Emergency will occur if it is projected that power cannot be restored to at least one AC emergency bus by the end of the analyzed station blackout coping period. Beyond this time, plant responses and event trajectory are subject to greater uncertainty, and there is an increased likelihood of challenges to multiple fission product barriers.

The estimate for restoring at least one emergency bus should be based on a realistic appraisal of the situation. Mitigation actions with a low probability of success should not be used as a basis for delaying a classification upgrade. The goal is to maximize the time available to prepare for, and implement, protective actions for the public.

The EAL will also require a General Emergency declaration if the loss of AC power results in parameters that indicate an inability to adequately remove decay heat from the core.

BNP Basis Reference(s):

1. 0AOP-36.2 STATION BLACKOUT, Section 4.0
2. BNP Updated FSAR Chapter 8
3. 1(2)OP-50 Plant Electric System Operating Procedure
4. 1(2)EOP-01-RVCP, Reactor Vessel Control
5. 0EOP-01-NL, EOP/SAMG NUMERICAL LIMITS AND VALUES

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Category: S –System Malfunction

Subcategory: 1 – Loss of Emergency AC Power

Initiating Condition: Prolonged loss of **all** offsite and **all** onsite AC power to emergency buses **OR** loss of **all** emergency AC and vital DC power sources for 15 minutes or longer

EAL:

SG1.2 **General Emergency**
Loss of **all** offsite and **all** onsite AC power capability to Emergency 4 KV Buses E1(E3) and E2(E4) for ≥ 15 min.
AND
Loss of **all** 125 VDC power based on battery bus voltage indications < 105 VDC on **all** vital DC buses 1(2)A-1, A-2, B-1 and B-2 for ≥ 15 min.
(Note 1)

Note 1: The SEC should declare the event promptly upon determining that time limit has been exceeded, or will likely be exceeded.

Mode Applicability:

1 - Power Operations, 2 - Startup, 3 - Hot Shutdown

Definition(s):

None

Basis:

This EAL is indicated by the loss of all offsite and onsite emergency AC power capability to 4 KV emergency buses E1(E3) and E2(E4) for greater than 15 minutes in combination with degraded vital DC power voltage. This EAL addresses operating experience from the March 2011 accident at Fukushima Daiichi.

The 4160 VAC System provides the power requirements for operation and safe shutdown of the plant. 4160 VAC power is divided into Balance of Plant (BOP Bus), and Emergency (E-Bus) distribution. The BOP Buses consist of the Conventional Buses (Common A/B Buses; Buses 1B/2B, 1C/2C, 1D/2D) and Caswell Beach Buses A/B. The Emergency switchgear are Buses E1/E3 (Division I) and E2/E4 (Division II).

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The E-Buses are normally powered through the respective BOP Buses (1D to E1, 1C to E2, 2D to E3, 2C to E4) via a master/slave breaker arrangement. Each E-Bus has a dedicated Diesel Generator to supply an emergency source of power to safe shutdown loads in the event of a loss of the normal power source or loss of offsite power. The DGs will automatically start and tie onto the E-Buses if the normal power source or offsite power is lost. The DGs can provide power to the E-Buses only. In the event the diesel generator is unavailable for an E-Bus, crosstie capability exists for each E-Bus from the same division of the opposite unit (E1 to E3, E2 to E4). Although the E-Buses within the unit also have crosstie capability, this alignment is not permitted by plant procedures, with the exception of E1 to E2 during specific Alternate Safe Shutdown (ASSD) conditions.

During periods of unit shutdown, when the Startup Auxiliary Transformer (SAT) would be the only normal source of offsite power, the Unit Auxiliary Transformer (UAT) can be made available by establishing a UAT backfeed. Backfeed from the UAT will require the use of keys for the control selector switches and opening of the respective generator's manual no-load disconnect (Ref. 2, 3).

There are two independent vital 125 VDC divisions per unit, designated Division I and Division II (Batteries 1(2)A-1 and 1(2)A-2 for Division I and Batteries 1(2)B-1 and 1(2)B-2 for Division II). Each division consists of a 250 VDC battery center tapped to form two 125 VDC batteries. Each 125 VDC battery has an associated full capacity battery charger. The chargers are supplied from the same AC load groups for which the associated DC subsystem supplies the control power. During normal operation, the DC loads are powered from the battery chargers with the batteries floating on the system. In case of loss of normal power to the battery charger, the DC loads are automatically powered from the station batteries.

105 VDC is the minimum design voltage limit (ref. 4, 5).

In the event that DC battery voltage indication is not available via ERFIS, local voltage indication is available for each bus based on dispatching a field operator in accordance with 0AOP-39.0 Loss of DC Power. In this case the 15 minute classification clock begins upon receipt of the low voltage alarm in the Control Room. If battery voltage cannot be verified to be greater than or equal to 105 VDC within the 15 minutes, emergency classification must be made under this EAL.

This IC addresses a concurrent and prolonged loss of both emergency AC and Vital DC power. A loss of all emergency AC power compromises the performance of all SAFETY SYSTEMS requiring electric power including those necessary for emergency core cooling, containment heat removal/pressure control, spent fuel heat removal and the ultimate heat sink. A loss of vital DC power compromises the ability to monitor and control SAFETY SYSTEMS. A sustained loss of both emergency AC and vital DC power will lead to multiple challenges to fission product barriers.

Fifteen minutes was selected as a threshold to exclude transient or momentary power losses. The 15-minute emergency declaration clock begins at the point when both EAL thresholds are met.

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BNP Basis Reference(s):

1. 0AOP-36.2 STATION BLACKOUT, Section 4.0
2. BNP Updated FSAR Chapter 8
3. 1(2)OP-50 Plant Electric System Operating Procedure
4. BNP Technical Specification Bases B.3.8.4
5. 0AOP-39.0 LOSS OF DC POWER

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Category: S – System Malfunction

Subcategory: 2 – Loss of Vital DC Power

Initiating Condition: Loss of all vital DC power for 15 minutes or longer

EAL:

SS2.1 **Site Area Emergency**
Loss of **all** 125 VDC power based on battery bus voltage indications < 105 VDC on **all** vital DC buses 1(2)A-1, A-2, B-1 and B-2 for ≥ 15 min. (Note 1)

Note 1: The SEC should declare the event promptly upon determining that time limit has been exceeded, or will likely be exceeded.

Mode Applicability:

1 - Power Operations, 2 - Startup, 3 - Hot Shutdown

Definition(s):

None

Basis:

There are two independent vital 125 VDC divisions per unit, designated Division I and Division II (Batteries 1(2)A-1 and 1(2)A-2 for Division I and Batteries 1(2)B-1 and 1(2)B-2 for Division II). Each division consists of a 250 VDC battery center tapped to form two 125 VDC batteries. Each 125 VDC battery has an associated full capacity battery charger. The chargers are supplied from the same AC load groups for which the associated DC subsystem supplies the control power. During normal operation, the DC loads are powered from the battery chargers with the batteries floating on the system. In case of loss of normal power to the battery charger, the DC loads are automatically powered from the station batteries.

Note that the Control Room DC voltage indicator only reads battery charger output voltage and not battery voltage unless the charger output breaker is closed. However ERFIS does provide DC battery voltage, otherwise battery voltage must be read locally.

In the event that DC battery voltage indication is not available via ERFIS, local voltage indication is available for each bus based on dispatching a field operator in accordance with 0AOP-39.0 Loss of DC Power. In this case the 15 minute classification clock begins upon receipt of the low voltage alarm in the Control Room. If battery voltage cannot be verified to be greater than or equal to 105 VDC within the 15 minutes, emergency classification must be made under this EAL.

105 VDC is the minimum design voltage limit (ref. 1).

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This IC addresses a loss of vital DC power which compromises the ability to monitor and control SAFETY SYSTEMS. In modes above Cold Shutdown, this condition involves a major failure of plant functions needed for the protection of the public.

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

Escalation of the emergency classification level would be via ICs RG1, FG1 or SG1.

BNP Basis Reference(s):

1. BNP Technical Specification Bases B.3.8.4
2. 0AOP-39.0 LOSS OF DC POWER
3. NEI 99-01 SS8

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Category: S – System Malfunction

Subcategory: 3 – Loss of Control Room Indications

Initiating Condition: UNPLANNED loss of Control Room indications for 15 minutes or longer

EAL:

SU3.1	**Unusual Event**
An UNPLANNED event results in the inability to monitor one or more Table S-1 parameters from within the Control Room for ≥ 15 min. (Note 1)	

Note 1: The SEC should declare the event promptly upon determining that time limit has been exceeded, or will likely be exceeded.

Table S-1 Safety System Parameters
• Reactor power • RPV water level • RPV pressure • Primary containment pressure • Torus water level • Torus temperature

Mode Applicability:

1 - Power Operations, 2 - Startup, 3 - Hot Shutdown

Definition(s):

UNPLANNED - A parameter change or an event that is not 1) the result of an intended evolution or 2) an expected plant response to a transient. The cause of the parameter change or event may be known or unknown.

Basis:

SAFETY SYSTEM parameters listed in Table S-1 are monitored in the Control Room through a combination of hard control panel indicators as well as computer based information systems. The ERFIS, which displays SPDS required information, serves as a redundant compensatory indicator which may be utilized in lieu of normal Control Room indicators (ref. 1, 2).

As used in this EAL "within the Control Room" means any available indicator available within the Control Room boundary, including back panels.

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This IC addresses the difficulty associated with monitoring normal plant conditions without the ability to obtain SAFETY SYSTEM parameters from within the Control Room. This condition is a precursor to a more significant event and represents a potential degradation in the level of safety of the plant.

As used in this EAL, an "inability to monitor" means that values for one or more of the listed parameters cannot be determined from within the Control Room. This situation would require a loss of all of the Control Room sources for the given parameter(s). For example, the reactor power level cannot be determined from any analog, digital or recorder source within the Control Room.

An event involving a loss of plant indications, annunciators and/or display systems is evaluated in accordance with 10 CFR 50.72 (and associated guidance in NUREG-1022) to determine if an NRC event report is required. The event would be reported if it significantly impaired the capability to perform emergency assessments. In particular, emergency assessments necessary to implement abnormal operating procedures, emergency operating procedures, and emergency plan implementing procedures addressing emergency classification, accident assessment, or protective action decision-making.

This EAL is focused on a selected subset of plant parameters associated with the key safety functions of reactivity control, RPV level and RCS heat removal. The loss of the ability to determine one or more of these parameters from within the Control Room is considered to be more significant than simply a reportable condition. In addition, if all indication sources for one or more of the listed parameters are lost, then the ability to determine the values of other SAFETY SYSTEM parameters may be impacted as well. For example, if the value for RPV water level cannot be determined from the indications and recorders on a main control board, the SPDS or the plant computer, the availability of other parameter values may be compromised as well.

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

Escalation of the emergency classification level would be via IC SA3.

BNP Basis Reference(s):

1. Updated FSAR Update Section 7.7.1.9
2. 0OI-01.08 Control of Equipment and System Status
3. NEI 99-01 SU2

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Category: S – System Malfunction

Subcategory: 3 – Loss of Control Room Indications

Initiating Condition: UNPLANNED loss of Control Room indications for 15 minutes or longer with a significant transient in progress

EAL:

SA3.1 Alert

An UNPLANNED event results in the inability to monitor one or more Table S-1 parameters from within the Control Room for ≥ 15 min. (Note 1)

AND

Any significant transient is in progress, Table S-2

Note 1: The SEC should declare the event promptly upon determining that time limit has been exceeded, or will likely be exceeded.

Table S-1 Safety System Parameters
• Reactor power • RPV water level • RPV pressure • Primary containment pressure • Torus water level • Torus temperature

Table S-2 Significant Transients
• Reactor scram • Runback > 25% rated thermal power • Electrical load rejection > 25% electrical load • ECCS injection • Thermal power oscillations > 10% (peak to peak)

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Mode Applicability:

1 - Power Operations, 2 - Startup, 3 - Hot Shutdown

Definition(s):

UNPLANNED - A parameter change or an event that is not 1) the result of an intended evolution or 2) an expected plant response to a transient. The cause of the parameter change or event may be known or unknown.

Basis:

SAFETY SYSTEM parameters listed in Table S-1 are monitored in the Control Room through a combination of hard control panel indicators as well as computer based information systems. The ERFIS, which displays SPDS required information, serves as a redundant compensatory indicator which may be utilized in lieu of normal Control Room indicators (ref. 1, 2).

Significant transients are listed in Table S-2 and include response to automatic or manually initiated functions such as scrams, runbacks (Recirculation) involving greater than 25% thermal power change, electrical load rejections of greater than 25% full electrical load, ECCS injections, or thermal power oscillations of 10% (peak to peak) or greater.

As used in this EAL "within the Control Room" means any available indicator available within the Control Room boundary, including back panels.

This IC addresses the difficulty associated with monitoring rapidly changing plant conditions during a transient without the ability to obtain SAFETY SYSTEM parameters from within the Control Room. During this condition, the margin to a potential fission product barrier challenge is reduced. It thus represents a potential substantial degradation in the level of safety of the plant.

As used in this EAL, an "inability to monitor" means that values for one or more of the listed parameters cannot be determined from within the Control Room. This situation would require a loss of all of the Control Room sources for the given parameter(s). For example, the reactor power level cannot be determined from any analog, digital or recorder source within the Control Room.

An event involving a loss of plant indications, annunciators and/or display systems is evaluated in accordance with 10 CFR 50.72 (and associated guidance in NUREG-1022) to determine if an NRC event report is required. The event would be reported if it significantly impaired the capability to perform emergency assessments. In particular, emergency assessments necessary to implement abnormal operating procedures, emergency operating procedures, and emergency plan implementing procedures addressing emergency classification, accident assessment, or protective action decision-making.

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This EAL is focused on a selected subset of plant parameters associated with the key safety functions of reactivity control, RPV level and RCS heat removal. The loss of the ability to determine one or more of these parameters from within the Control Room is considered to be more significant than simply a reportable condition. In addition, if all indication sources for one or more of the listed parameters are lost, then the ability to determine the values of other SAFETY SYSTEM parameters may be impacted as well. For example, if the value for RPV water level cannot be determined from the indications and recorders on a main control board, the SPDS or the plant computer, the availability of other parameter values may be compromised as well.

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

Escalation of the emergency classification level would be via ICs FS1 or IC RS1

BNP Basis Reference(s):

1. Updated FSAR Update Section 7.7.1.9
2. 0OI-01.08 Control of Equipment and System Status
3. NEI 99-01 SA2

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Category: S – System Malfunction

Subcategory: 4 – RCS Activity

Initiating Condition: Reactor coolant activity greater than Technical Specification allowable limits

EAL:

SU4.1	**Unusual Event**
Steam Jet Air Ejector Radiation Monitor 1(2)D12-RM-K601A /B Hi-Hi alarm (Process Off-Gas Rad Hi-Hi alarm 1(2)UA-03 4-2	

Deleted:) ≥ 15 min. (Note 1)

Mode Applicability:

Deleted: Note 1: The SEC should declare the event promptly upon determining that time limit has been exceeded, or will likely be exceeded.¶

1 - Power Operations, 2 - Startup, 3 - Hot Shutdown

Definition(s):

None

Basis:

The Steam Jet Air Ejector radiation monitor setpoint provides reasonable assurance that the total body exposure to an individual at the exclusion area boundary will not exceed a small fraction of the limits of 10CFR100 in the event of an inadvertent release via the condenser air ejector (ref. 2, 3).

At the Hi-Hi alarm setpoint, the process Off-Gas timer is started. After the process Off-Gas timer has timed out (15 minutes), the Off-Gas system will isolate (ref. 1).

This IC addresses a reactor coolant activity value that exceeds an allowable limit specified in Technical Specifications. This condition is a precursor to a more significant event and represents a potential degradation of the level of safety of the plant.

Escalation of the emergency classification level would be via ICs FA1 or the Recognition Category R ICs.

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BNP Basis Reference(s):

1. ARP 1(2)APP-UA-03 4-2 Process Off-Gas Rad Hi-Hi
2. BNP Offsite Dose Calculation Manual section 3.1.3
3. BNP Technical Specifications section 3.7.5
4. NEI 99-01 SU3

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Category: S – System Malfunction

Subcategory: 4 – RCS Activity

Initiating Condition: Reactor coolant activity greater than Technical Specification allowable limits

EAL:

SU4.2	**Unusual Event**
Coolant activity > 0.2 μCi/gm I-131 dose equivalent for > 48 hours (Note 1)	
OR	
Coolant activity > 4.0 μCi/gm I-131 dose equivalent instantaneous	

Note 1: The SEC should declare the event promptly upon determining that time limit has been exceeded, or will likely be exceeded.

Mode Applicability:

1 - Power Operations, 2 - Startup, 3 - Hot Shutdown

Definition(s):

None

Basis:

The specific iodine activity is limited to ≤ 0.2 μCi/gm Dose Equivalent I-131. This limit ensures the source term assumed in the safety analysis for the Main Steam Line Break (MSLB) is not exceeded, so any release of radioactivity to the environment during an MSLB is less than a small fraction of the 10 CFR 50.67 limits (ref. 1).

The upper limit of 4.0 μCi/gm Dose Equivalent I-131 ensures that the thyroid dose from an MSLB will not exceed the dose guidelines of 10 CFR 50.67 or Control Room operator dose limits specified in GDC 19 of 10 CFR 50, Appendix A (ref. 1).

This IC addresses a reactor coolant activity value that exceeds an allowable limit specified in Technical Specifications. This condition is a precursor to a more significant event and represents a potential degradation of the level of safety of the plant.

Escalation of the emergency classification level would be via ICs FA1 or the Recognition Category R ICs.

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BNP Basis Reference(s):

1. BNP Technical Specifications section 3.4.6
2. NEI 99-01 SU3

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Category: S – System Malfunction

Subcategory: 5 – RCS Leakage

Initiating Condition: RCS leakage for 15 minutes or longer

EAL:

SU5.1 Unusual Event

RCS unidentified or pressure boundary leakage > 10 gpm for ≥ 15 min.

OR

RCS identified leakage > 25 gpm for ≥ 15 min.

OR

Leakage from the RCS to a location outside Primary Containment > 25 gpm for ≥ 15 min. (Note 1)

Note 1: The SEC should declare the event promptly upon determining that time limit has been exceeded, or will likely be exceeded.

Mode Applicability:

1 - Power Operations, 2 - Startup, 3 - Hot Shutdown

Definition(s):

None

Basis:

Leakage is monitored by utilizing the following techniques:

- Sensing excess flow in piping systems
- Sensing pressure and temperature changes in the primary containment
- Monitoring for high flow and temperature through selected drains,
- Sampling airborne particulate and gaseous radioactivity.
- Drywell floor and equipment drain sump leak rate system

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Identified leakage is leakage into the drywell, such as that from pump seals or valve packing, that is captured and conducted to a sump or collecting tank; or leakage into the drywell atmosphere from sources that are both specifically located and known either not to interfere with the operation of leakage detection systems or not to be pressure boundary leakage. (ref. 1, 2)

Unidentified leakage is all leakage into the drywell that is not identified leakage. (ref. 1, 2)

Pressure boundary leakage is leakage through a nonisolable fault in a Reactor Coolant System (RCS) component body, pipe wall, or vessel wall. (ref. 1, 2)

The drywell floor drain sump flow monitoring system monitors the leakage collected in the floor drain sump. This unidentified leakage consists of leakage from control rod drives, valve flanges, floor drains, the Reactor Building Closed Cooling Water System, and drywell cooler drains, and any leakage not collected in the drywell equipment drain sump. The drywell floor drain sump is provided with two sump pumps. A flow transmitter in the common discharge line of the drywell floor drain sump pumps inputs to a flow integrator. In addition to the required instrumentation, the starting frequency and run duration of a sump pump motor are monitored by timer circuitry to provide a signal (alarm) in the Control Room indicating that leakage has reached a specified limit. (ref. 2)

RCS leakage outside of the Primary Containment that is not considered identified or unidentified leakage per Technical Specifications includes leakage via interfacing systems such as RCS to the Reactor Building Closed Cooling Water (RBCCW system), or systems that directly see RCS pressure outside primary containment such as Reactor Water Cleanup (RWCU), reactor water sampling system and Residual Heat Removal (RHR) system (when in the shutdown cooling mode) (ref. 3)

Escalation of this EAL to the Alert level is via Category F, Fission Product Barrier Degradation, EAL FA1.1. The note has been added to remind the EAL-user to review Table F-1 for possible escalation to higher emergency classifications.

This IC addresses RCS leakage which may be a precursor to a more significant event. In this case, RCS leakage has been detected and operators, following applicable procedures, have been unable to promptly isolate the leak. This condition is considered to be a potential degradation of the level of safety of the plant.

The first and second EAL conditions are focused on a loss of mass from the RCS due to "unidentified leakage", "pressure boundary leakage" or "identified leakage" (as these leakage types are defined in the plant Technical Specifications). The third condition addresses an RCS mass loss caused by an UNISOLABLE leak through an interfacing system. These conditions thus apply to leakage into the Primary Containment, or a location outside of Primary Containment.

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The leak rate values for each condition were selected because they are usually observable with normal Control Room indications. Lesser values typically require time-consuming calculations to determine (e.g., a mass balance calculation). The first condition uses a lower value that reflects the greater significance of unidentified or pressure boundary leakage.

A stuck-open Safety Relief Valve (SRV) or SRV leakage is not considered either identified or unidentified leakage by Technical Specifications and, therefore, is not applicable to this EAL.

The 15-minute threshold duration allows sufficient time for prompt operator actions to isolate the leakage, if possible.

Escalation of the emergency classification level would be via ICs of Recognition Category R or F.

BNP Basis Reference(s):

1. BNP Technical Specifications Definitions section 1.1
2. BNP Technical Specifications Bases 3.4.5
3. BNP UFSAR section 5.1 Reactor Coolant System and Connected Systems
4. NEI 99-01 SU4

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Category: S – System Malfunction

Subcategory: 6 – RPS Failure

Initiating Condition: Automatic or manual scram fails to shut down the reactor

EAL:

SU6.1 Unusual Event

An automatic scram did **not** reduce reactor power to < 2% (APRM downscale) after **any** RPS setpoint is exceeded

AND

A subsequent automatic scram or manual scram action taken at the reactor control console (Manual PBs, Mode Switch, ARI) is successful in shutting down the reactor as indicated by reactor power < 2% (APRM downscale) (Note 8)

Note 8: A manual scram action is any operator action, or set of actions, which causes the control rods to be rapidly inserted into the core, and **does not** include manually driving in control rods or implementation of boron injection strategies.

Mode Applicability:

1 - Power Operations, 2 - Startup

Definition(s):

None

Basis:

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 (ref. 1).

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 below the APRM downscale setpoint of 2%.

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For the purposes of emergency classification, successful manual scram actions are those which can be quickly performed from the reactor control console (i.e., manual scram pushbuttons, mode switch, or ARI actuation). Reactor shutdown achieved by use of the 0EOP-01-LEP-02 actions does not constitute a successful manual scram (ref. 4).

Following any automatic RPS scram signal, 1(2)EOP-01-RSP (ref. 2) and 1(2)EOP-01-ATWS (ref. 3) 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 Unusual Event.

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Taking the mode switch to shutdown is a manual scram action. When the Mode Switch is taken out of the Run position, however, the nuclear instrumentation scram setpoint is lowered. If reactor power remains above the lowered setpoint, an automatic scram is initiated.

For the purposes of this EAL, a successful automatic initiation of ARI that reduces reactor power below 2% is not considered 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. However, a successful automatic or manual initiation of ARI is an acceptable means of establishing reactor shutdown conditions relative to the EAL threshold in the absence of any required subsequent manual scram actions.

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 below 2% (ref. 2, 3), the event escalates to the Alert under EAL SA6.1.

If by procedure, operator actions include the initiation of an immediate manual scram following receipt of an automatic scram signal and there are no clear indications that the automatic scram failed (such as a time delay following indications that a scram setpoint was exceeded), it may be difficult to determine if the reactor was shut down because of automatic scram or manual actions. If a subsequent review of the scram actuation indications reveals that the automatic scram did not cause the reactor to be shut down, then consideration should be given to evaluating the fuel for potential damage, and the reporting requirements of 50.72 should be considered for the transient event.

This IC addresses a failure of the RPS to initiate or complete an automatic or manual reactor scram that results in a reactor shutdown, and either a subsequent operator manual action taken at the reactor control consoles or an automatic scram is successful in shutting down the reactor. This event is a precursor to a more significant condition and thus represents a potential degradation of the level of safety of the plant.

Following the failure on an automatic reactor scram, operators will promptly initiate manual actions at the reactor control consoles to shutdown the reactor (e.g., initiate a manual reactor scram). If these manual actions are successful in shutting down the reactor, core heat generation will quickly fall to a level within the capabilities of the plant's decay heat removal systems.

If an initial manual reactor scram is unsuccessful, operators will promptly take manual action at another location(s) on the reactor control consoles to shutdown the reactor (e.g., initiate a manual reactor scram) using a different switch). Depending upon several factors, the initial or subsequent effort to manually scram the reactor, or a concurrent plant condition, may lead to the generation of an automatic reactor scram signal. If a subsequent manual or automatic scram is successful in shutting down the reactor, core heat generation will quickly fall to a level within the capabilities of the plant's decay heat removal systems.

A manual action at the reactor control consoles is any operator action, or set of actions, which causes the control rods to be rapidly inserted into the core (e.g., initiating a manual reactor scram). This action does not include manually driving in control rods or implementation of boron injection strategies. Actions taken at back-panels or other locations within the Control Room, or any location outside the Control Room, are not considered to be "at the reactor control consoles".

Taking the Reactor Mode Switch to SHUTDOWN is considered to be a manual scram action.

The plant response to the failure of an automatic or manual reactor scram will vary based upon several factors including the reactor power level prior to the event, availability of the condenser, performance of mitigation equipment and actions, other concurrent plant conditions, etc. If subsequent operator manual actions taken at the reactor control consoles are also unsuccessful in shutting down the reactor, then the emergency classification level will escalate to an Alert via IC SA6. Depending upon the plant response, escalation is also possible via IC FA1. Absent the plant conditions needed to meet either IC SA6 or FA1, an Unusual Event declaration is appropriate for this event.

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Should a reactor scram signal be generated as a result of plant work (e.g., RPS setpoint testing), the following classification guidance should be applied.

- If the signal causes a plant transient that should have included an automatic reactor scram and the RPS fails to automatically shutdown the reactor, then this IC and the EALs are applicable, and should be evaluated.
- If the signal does not cause a plant transient and the scram failure is determined through other means (e.g., assessment of test results), then this IC and the EALs are not applicable and no classification is warranted.

BNP Basis Reference(s):

1. BNP Technical Specifications section 3.3.1.1 RPS Instrumentation
2. 1(2) EOP-01-RSP, Reactor Scram
3. 1(2) EOP-01-ATWS, ATWS
4. 0EOP-01-LEP-02 Alternate Control Rod Insertion
5. NEI 99-01 SU5

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Category: S – System Malfunction

Subcategory: 6 – RPS Failure

Initiating Condition: Automatic or manual scram fails to shut down the reactor

EAL:

SU6.2	**Unusual Event**

A manual scram did **not** reduce reactor power to < 2% (APRM downscale) after **any** manual scram action was initiated

AND

A subsequent automatic scram or manual scram action taken at the reactor control console (Manual PBs, Mode Switch, ARI) is successful in shutting down the reactor as indicated by reactor power < 2% (APRM downscale) (Note 8)

Note 8: A manual scram action is any operator action, or set of actions, which causes the control rods to be rapidly inserted into the core, and **does not** include manually driving in control rods or implementation of boron injection strategies.

Mode Applicability:

1 - Power Operations, 2 - Startup

Definition(s):

None

Basis:

This EAL addresses a failure of a manually initiated scram in the absence of having exceeded an automatic RPS trip setpoint and a subsequent automatic or manual scram is successful in shutting down the reactor (reactor power < 2%). (ref. 1).

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 a manual 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 below the APRM downscale setpoint of 2%.

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For the purposes of emergency classification, successful manual scram actions are those which can be quickly performed from the reactor control console (i.e., manual scram pushbuttons, mode switch, or ARI actuation). Reactor shutdown achieved by use of the 0EOP-01-LEP-02 actions does not constitute a successful manual scram (ref. 2, 3).

Taking the mode switch to shutdown is a manual scram action. When the Mode Switch is taken out of the Run position, however, the nuclear instrumentation scram setpoint is lowered. If reactor power remains above the lowered setpoint, an automatic scram is initiated.

Successful automatic or manual initiation of ARI is an acceptable means of establishing reactor shutdown conditions relative to the EAL threshold in the absence of any required subsequent manual scram actions.

If both subsequent automatic and subsequent manual reactor scram actions in the Control Room fail to reduce reactor power below the power associated with the safety system design (< 2%) following a failure of an initial manual scram, the event escalates to an Alert under EAL SA6.1

This IC addresses a failure of the RPS to initiate or complete an automatic or manual reactor scram that results in a reactor shutdown, and either a subsequent operator manual action taken at the reactor control consoles or an automatic scram is successful in shutting down the reactor. This event is a precursor to a more significant condition and thus represents a potential degradation of the level of safety of the plant.

Following the failure on an automatic reactor scram, operators will promptly initiate manual actions at the reactor control consoles to shutdown the reactor (e.g., initiate a manual reactor scram). If these manual actions are successful in shutting down the reactor, core heat generation will quickly fall to a level within the capabilities of the plant's decay heat removal systems.

If an initial manual reactor scram is unsuccessful, operators will promptly take manual action at another location(s) on the reactor control consoles to shutdown the reactor (e.g., initiate a manual reactor scram) using a different switch). Depending upon several factors, the initial or subsequent effort to manually scram the reactor, or a concurrent plant condition, may lead to the generation of an automatic reactor scram signal. If a subsequent manual or automatic scram is successful in shutting down the reactor, core heat generation will quickly fall to a level within the capabilities of the plant's decay heat removal systems.

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A manual action at the reactor control consoles is any operator action, or set of actions, which causes the control rods to be rapidly inserted into the core (e.g., initiating a manual reactor scram). This action does not include manually driving in control rods or implementation of boron injection strategies. Actions taken at back-panels or other locations within the Control Room, or any location outside the Control Room, are not considered to be "at the reactor control consoles".

Taking the Reactor Mode Switch to SHUTDOWN is considered to be a manual scram action.

The plant response to the failure of an automatic or manual reactor scram will vary based upon several factors including the reactor power level prior to the event, availability of the condenser, performance of mitigation equipment and actions, other concurrent plant conditions, etc. If subsequent operator manual actions taken at the reactor control consoles are also unsuccessful in shutting down the reactor, then the emergency classification level will escalate to an Alert via IC SA6. Depending upon the plant response, escalation is also possible via IC FA1. Absent the plant conditions needed to meet either IC SA6 or FA1, an Unusual Event declaration is appropriate for this event.

Should a reactor scram signal be generated as a result of plant work (e.g., RPS setpoint testing), the following classification guidance should be applied.

- If the signal causes a plant transient that should have included an automatic reactor scram and the RPS fails to automatically shutdown the reactor, then this IC and the EALs are applicable, and should be evaluated.
- If the signal does not cause a plant transient and the scram failure is determined through other means (e.g., assessment of test results), then this IC and the EALs are not applicable and no classification is warranted.

BNP Basis Reference(s):

1. BNP Technical Specifications section 3.3.1.1 RPS Instrumentation
2. 1(2) EOP-01-RSP, Reactor Scram
3. 1(2) EOP-01-ATWS, ATWS
4. NEI 99-01 SU5

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Category: S – System Malfunction

Subcategory: 2 – RPS Failure

Initiating Condition: Automatic or manual scram fails to shut down the reactor and subsequent manual actions taken at the reactor control consoles are not successful in shutting down the reactor

EAL:

SA6.1	**Alert**
An automatic or manual scram fails to reduce reactor power to < 2% (APRM downscale) **AND** Manual scram actions taken at the reactor control console (Manual PBs, Mode Switch, ARI) are **not** successful in shutting down the reactor as indicated by reactor power $\geq$ 2% (Note 8)	

Note 8: A manual scram action is any operator action, or set of actions, which causes the control rods to be rapidly inserted into the core, and **does not** include manually driving in control rods or implementation of boron injection strategies.

Mode Applicability:

1 - Power Operations, 2 - Startup

Definition(s):

None

Basis:

This EAL addresses any automatic or manual reactor scram signal that fails to shut down the reactor followed by a subsequent 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.

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., manual scram pushbuttons, mode switch, or ARI actuation). Reactor shutdown achieved by use of the 0EOP-01-LEP-02 actions does not constitute a successful manual scram (ref. 1).

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For the purposes of this EAL, a successful automatic initiation of ARI that reduces reactor power below 2% is not considered 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. However, a successful automatic initiation of ARI is an acceptable means of establishing reactor shutdown conditions relative to the EAL threshold in the absence of any required subsequent manual scram actions.

The APRM downscale trip setpoint (2%) is a minimum reading on the power range scale that indicates power production. 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. Below the APRM downscale trip setpoint, plant response will be similar to that observed during a normal shutdown. Nuclear instrumentation (APRM) indications or other reactor parameters (steam flow, RPV pressure, torus temperature trend) can be used to determine if reactor power is greater than 2 % power (ref. 2, 3).

Escalation of this event to a Site Area Emergency would be under EAL SS6.

This IC addresses a failure of the RPS to initiate or complete an automatic or manual reactor scram that results in a reactor shutdown, and subsequent operator manual actions taken at the reactor control consoles to shutdown the reactor are also unsuccessful. This condition represents an actual or potential substantial degradation of the level of safety of the plant. An emergency declaration is required even if the reactor is subsequently shutdown by an action taken away from the reactor control consoles since this event entails a significant failure of the RPS.

A manual action at the reactor control console is any operator action, or set of actions, which causes the control rods to be rapidly inserted into the core (e.g., initiating a manual reactor scram). This action does not include manually driving in control rods or implementation of boron injection strategies. If this action(s) is unsuccessful, operators would immediately pursue additional manual actions at locations away from the reactor control console (e.g., locally opening breakers). Actions taken at backpanels or other locations within the Control Room, or any location outside the Control Room, are not considered to be "at the reactor control console".

Taking the Reactor Mode Switch to SHUTDOWN is considered to be a manual scram action.

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The plant response to the failure of an automatic or manual reactor scram will vary based upon several factors including the reactor power level prior to the event, availability of the condenser, performance of mitigation equipment and actions, other concurrent plant conditions, etc. If the failure to shut down the reactor is prolonged enough to cause a challenge to RPV water level or RCS heat removal safety functions, the emergency classification level will escalate to a Site Area Emergency via IC SS6. Depending upon plant responses and symptoms, escalation is also possible via IC FS1. Absent the plant conditions needed to meet either IC SS6 or FS1, an Alert declaration is appropriate for this event.

It is recognized that plant responses or symptoms may also require an Alert declaration in accordance with the Recognition Category F ICs; however, this IC and EAL are included to ensure a timely emergency declaration.

BNP Basis Reference(s):

1. 0EOP-01-LEP-02, Alternate Control Rod Insertion
2. 1(2) EOP-01-RSP, Reactor Scram
3. 1(2) EOP-01-ATWS, ATWS
4. NEI 99-01 SA5

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Deleted: LPC
Deleted: Level/Power Control

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Category: S – System Malfunction

Subcategory: 2 – RPS Failure

Initiating Condition: Inability to shut down the reactor causing a challenge to RPV water level or RCS heat removal

EAL:

SS6.1 Site Area Emergency

An automatic or manual scram fails to reduce reactor power to < 2% (APRM downscale)

AND

All actions to shut down the reactor are **not** successful as indicated by reactor power ≥ 2%

AND EITHER:

- RPV level **cannot** be restored and maintained > LL-4 or **cannot** be determined
- Suppression pool water temperature and RPV pressure **cannot** be maintained below the HCTL

Mode Applicability:

1 - Power Operations, 2 - Startup

Definition(s):

None

Basis:

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 SA6.1), and
- Indications that either core cooling is extremely challenged or heat removal is extremely challenged.

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Reactor shutdown achieved by use of 0EOP-01-LEP-02 Alternate Control Rod Insertion 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.

The APRM downscale trip setpoint (2%) is a minimum reading on the power range scale that indicates power production. 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. Below the APRM downscale trip setpoint, plant response will be similar to that observed during a normal shutdown. Nuclear instrumentation (APRM) indications or other reactor parameters (steam flow, RPV pressure, torus temperature trend) can be used to determine if reactor power is greater than 2% power (ref. 1, 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.

Indication that core cooling is extremely challenged is manifested by inability to restore and maintain RPV water level above LL-4. LL-4 is the Minimum Steam Cooling RPV Water Level (MSCRWL) (ref. 3). The MSCRWL is the lowest RPV level at which the covered portion of the reactor core will generate sufficient steam to prevent any clad temperature in the uncovered part of the core from exceeding 1500°F. This water level is utilized in the EOPs to preclude fuel damage when RPV level is below the top of active fuel. RPV level below the MSCRWL for an extended period of time without satisfactory core spray cooling could be a precursor of a core melt sequence. When RPV level cannot be determined, EOPs require entry to 0EOP-01-RXFP, Reactor Flooding. 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 0EOP-01-RXFP 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 Alternate Flooding Pressure (ref. 4).

The Heat Capacity Temperature Limit (HCTL) is the highest suppression pool water temperature from which Emergency RPV Depressurization will not raise suppression chamber pressure above Primary Containment Pressure Limit A (PCPL-A), while 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. This threshold is met when the final step SP/T-13 of section SP/T in 0EOP-02-PCCP, Primary Containment Control, is reached (ref. 5). This condition addresses loss of functions required for hot shutdown with the reactor at pressure and temperature.

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This IC addresses a failure of the RPS to initiate or complete an automatic or manual reactor scram that results in a reactor shutdown, all subsequent operator actions to manually shutdown the reactor are unsuccessful, and continued power generation is challenging the capability to adequately remove heat from the core and/or the RCS. This condition will lead to fuel damage if additional mitigation actions are unsuccessful and thus warrants the declaration of a Site Area Emergency.

In some instances, the emergency classification resulting from this IC/EAL may be higher than that resulting from an assessment of the plant responses and symptoms against the Recognition Category F ICs/EALs. This is appropriate in that the Recognition Category F ICs/EALs do not address the additional threat posed by a failure to shut down the reactor. The inclusion of this IC and EAL ensures the timely declaration of a Site Area Emergency in response to prolonged failure to shutdown the reactor.

Escalation of the emergency classification level would be via IC RG1 or FG1.

BNP Basis Reference(s):

1. 1(2) EOP-01-RSP, Reactor Scram
2. 1(2) EOP-01-ATWS, ATWS
3. 0EOP-01-NL EOP/SAMG Numerical Limits and Values, Attachment 1, pg 37-40, Figures 1-10 and 1-11
4. 0EOP-01-RXFP, Reactor Flooding
5. 0EOP-02-PCCP, Primary Containment Control
6. NEI 99-01 SS5

Deleted: Procedure

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Category: S – System Malfunction

Subcategory: 7 – Loss of Communications

Initiating Condition: Loss of **all** onsite or offsite communications capabilities

EAL:

SU7.1 Unusual Event

Loss of **all** Table S-3 onsite communication methods

OR

Loss of **all** Table S-3 offsite communication methods

OR

Loss of **all** Table S-3 NRC communication methods

Table S-3 Communication Methods			
System	**Onsite**	**Offsite**	**NRC**
Public Address System	X		
PBX Telephone System	X	X	X
DEMNET		X	X
Commercial Telephones	X	X	X
Satellite Phones		X	X
Cellular Phones		X	X
NRC Emergency Telecommunications System			X

Deleted: Corporate Telephone Communications System
Deleted: X

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Mode Applicability:

1 - Power Operations, 2 - Startup, 3 - Hot Shutdown

Definition(s):

None

Basis:

Onsite/offsite/NRC communications include one or more of the systems listed in Table S-3 (ref. 1).

Public Address System

The Brunswick Plant public address system provides paging and party line communications between stations located throughout the plant. Inside and outside type wall and desk-mounted stations are used to communicate between roaming personnel and fixed work locations. Plant-wide instructions are issued using the paging feature. This system is powered from the plant uninterruptible power supply which employs battery reserve as well as diesel generator emergency supply.

PBX Telephone System

The Brunswick Site PBX telephone system provides communication capability between telephone stations located within the plant by dialing the four-digit telephone station code. The PBX telephone system also provides for outside communications. The PBX switch located in the TSC/EOF building is also backed up by a battery UPS capable of supplying power for a minimum of 8 hours and is augmented by a Diesel Generator capable of supplying power to the TSC/EOF building for at least 5 days.

DEMNET

Deleted: Corporate Telephone Communications System (Voicenet and/or

Deleted:)

DEMNET is the primary means of offsite communication. This circuit allows intercommunication among the EOF, TSC, control room, counties, and states. DEMNET operates as an internet based (VoIP) communications system with a satellite back-up. Should the internet transfer rate become slow or unavailable, the DEMNET will automatically transfer to satellite mode.

Deleted: Interconnected through the site PBX and the emergency telephone system, the Voicenet system provides a means to communicate with other corporate locations with which the plant has a need to communicate. This system bypasses external commercial telephone lines and switching equipment. Corporate transmission facilities provide fiber optic, copper-wire, and microwave radio to ensure a high degree of system reliability. In addition to the redundancy provided by the three system options, backup power is provided for the systems.

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Commercial Telephones

Commercial telephone lines, which supply public telephone communications, are employed by Duke Energy in four ways: (1) tie-ins through the PBX to any other plant location, (2) lines to plant emergency facilities, (3) lines to the Joint Information Center for public information purposes, and (4) lines to the AEF. The local service provider provides primary and secondary power for their lines at the Central Office.

Satellite Phones

A total of three portable satellite telephones are available which enable communication when all other phone systems are inoperable, e.g. following a major hurricane. These portable systems can be powered by internal batteries, external DC sources as well as external AC sources. Two of these phones require outside use, while one phone may used either outside or in the EOF with a permanently mounted external antenna.

Cellular Phones

Selected plant personnel are provided with cellular telephones. These phones may be used during emergencies if other communications means are not readily available or are inoperable. These phones are not expected to be used in the Control Room or Power Block due to interference with plant equipment and loss of signal to the phone.

NRC Emergency Telecommunications System

The NRC uses a Duke Energy dedicated telephone line which allows direct telephone communications from the plant to NRC regional and national offices. The Duke Energy communications line provides a link independent of the local public telephone network. Telephones connected to this network are located in the Brunswick Control Room, Technical Support Center, and Emergency Operations Facility. These lines will not function if the PBX Telephone System fails.

This EAL is the hot condition equivalent of the cold condition EAL CU5.1.

This IC addresses a significant loss of onsite or offsite communications capabilities. While not a direct challenge to plant or personnel safety, this event warrants prompt notifications to OROs and the NRC.

This IC should be assessed only when extraordinary means are being utilized to make communications possible (e.g., use of non-plant, privately owned equipment, relaying of onsite information via individuals or multiple radio transmission points, individuals being sent to offsite locations, etc.).

The first EAL condition addresses a total loss of the communications methods used in support of routine plant operations.

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The second EAL condition addresses a total loss of the communications methods used to notify all OROs of an emergency declaration. The OROs referred to here are the State, Brunswick and New Hanover County EOCs

The third EAL addresses a total loss of the communications methods used to notify the NRC of an emergency declaration.

BNP Basis Reference(s):

1. 0ERP Radiological Emergency Response Plan Appendix A
2. SD-48 Communication Systems
3. NEI 99-01 SU6

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Category: S – System Malfunction

Subcategory: 8 – Hazardous Event Affecting Safety Systems

Initiating Condition: Hazardous event affecting a SAFETY SYSTEM needed for the current operating mode

EAL:

SA8.1 Alert

The occurrence of **any** Table S-4 hazardous event

AND EITHER:

- Event damage has caused indications of degraded performance in at least one train of a SAFETY SYSTEM needed for the current operating mode
- The event has caused VISIBLE DAMAGE to a SAFETY SYSTEM component or structure needed for the current operating mode

Table S-4 Hazardous Events
• Seismic event (earthquake) • Internal or external FLOODING event • High winds or tornado strike • FIRE • EXPLOSION • Other events with similar hazard characteristics as determined by the Shift Manager

Mode Applicability:

1 - Power Operations, 2 - Startup, 3 - Hot Shutdown

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Definition(s):

EXPLOSION - A rapid, violent and catastrophic failure of a piece of equipment due to combustion, chemical reaction or overpressurization. A release of steam (from high energy lines or components) or an electrical component failure (caused by short circuits, grounding, arcing, etc.) should not automatically be considered an explosion. Such events require a post-event inspection to determine if the attributes of an explosion are present.

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.

FLOODING - A condition where water is entering a room or area faster than installed equipment is capable of removal, resulting in a rise of water level within the room or area.

SAFETY SYSTEM - A system required for safe plant operation, cooling down the plant and/or placing it in the cold shutdown condition, including the ECCS. These are typically systems classified as safety-related (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.

VISIBLE DAMAGE - Damage to a component or structure that is readily observable without measurements, testing, or analysis. The visual impact of the damage is sufficient to cause concern regarding the operability or reliability of the affected component or structure.

Basis:

- The significance of seismic events are discussed under EAL HU2.1 (ref. 1).
- Internal FLOODING may be caused by events such as component failures, equipment misalignment, or outage activity mishaps (ref. 2, 3).
- Seismic Category I structures are analyzed to withstand a sustained, design wind velocity of at least 135 mph. (ref. 4).

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- Areas containing functions and systems required for safe shutdown of the plant are identified by Fire Zone in the fire response procedure (ref. 5, 6).
- An explosion that degrades the performance of a SAFETY SYSTEM train or visibly damages a SAFETY SYSTEM component or structure would be classified under this EAL.

This IC addresses a hazardous event that causes damage to a SAFETY SYSTEM, or a structure containing SAFETY SYSTEM components, needed for the current operating mode. This condition significantly reduces the margin to a loss or potential loss of a fission product barrier, and therefore represents an actual or potential substantial degradation of the level of safety of the plant.

The first condition addresses damage to a SAFETY SYSTEM train that is in service/operation since indications for it will be readily available. The indications of degraded performance should be significant enough to cause concern regarding the operability or reliability of the SAFETY SYSTEM train.

The second condition addresses damage to a SAFETY SYSTEM component that is not in service/operation or readily apparent through indications alone, or to a structure containing SAFETY SYSTEM components. Operators will make this determination based on the totality of available event and damage report information. This is intended to be a brief assessment not requiring lengthy analysis or quantification of the damage.

Escalation of the emergency classification level would be via IC FS1 or RS1.

BNP Basis Reference(s):

1. 1(2)APP-UA-28 6-4 Seismic Event
2. 0AOP-13.0 Operations During Hurricane, Flood Conditions, Tornado or Earthquake
3. Updated FSAR section 3.4.2 Protection From Internal Flooding
4. Updated FSAR Section 2.3.1.2.7
5. BNP-E-9. 010 NFPA 805 Nuclear Safety Capability Assessment (NSCA)
6. 0PFP-PBAA Power Block Auxiliary Areas Prefire Plan
7. NEI 99-01 SA9

Deleted: BNP-E-9.004 Safe Shutdown Analysis Report

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Category E – Independent Spent Fuel Storage Installation (ISFSI)

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

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.

Deleted: . Formal offsite planning is not required because the postulated worst-case accident involving an ISFSI has insignificant consequences to the public health and safety.

An 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.

The BNP ISFSI is contained wholly within the plant Protected Area. Therefore a security event related to the ISFSI would be applicable to EALs HU1.1, HA1.1 and HS1.1.

Deleted: 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 HS1.1

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

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Category: E - ISFSI

Sub-category: None

Initiating Condition: Damage to a loaded cask CONFINEMENT BOUNDARY

EAL:

EU1.1	**Notification of Unusual Event**
Damage to a loaded canister confinement boundary as indicated by an on-contact radiation reading on the surface of a loaded spent fuel cask > any of the following: • 1,400 mrem/hr on the HSM-H front surface • 10 mrem/hr on the HSM-H door centerline • 20 mrem/hr on the end shield wall exterior	

Mode Applicability:

All

Definition(s):

CONFINEMENT BOUNDARY-. The barrier(s) between spent fuel and the environment once the spent fuel is processed for dry storage. As related to the BNP ISFSI, Confinement Boundary is defined as the Dry Shielded Canister (DSC).

Basis:

The BNP ISFSI utilizes the NUHOMS® Type 2 -61BTH dry spent fuel storage (ref. 1, 2).

The NUHOMS® Type 2 61BTH spent fuel storage system is a modular canister based spent fuel storage and transfer system and consists of the following components:

- A 61BTH Dry Shielded Canister (DSC) provides confinement, an inert environment, structural support, and criticality control for 61 BWR fuel assemblies.
- A horizontal storage module (HSM-H) is provided for environmental protection, shielding, and heat rejection during storage.
- An OS197FC-B transfer cask that supports onsite transfer of the 61BTH DSC.

The NUHOMS® System confinement vessel is the DSC. The DSC is welded and designed to provide confinement of all radionuclides under normal, off-normal, and accident conditions.

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Confinement boundary is defined as the barrier(s) between areas containing radioactive substances and the environment. Therefore, damage to a confinement boundary must be a confirmed physical breach between the spent fuel and the environment for the Dry Shield Canister (DSC).

The values shown represent 2 times the limits specified in the ISFSI Certificate of Compliance (COC) Technical Specification for radiation external to a loaded (NUHOMS® Type 2 -61BTH) MPC (HSM-H) overpack (ref. 1, 2). The survey method(s) used to assess this EAL threshold shall be consistent with those used to ensure compliance with the COC Technical Specification limits (ref. 2).

This IC addresses an event that results in damage to the CONFINEMENT BOUNDARY of a storage cask containing spent fuel. It applies to irradiated fuel that is licensed for dry storage beginning at the point that the loaded storage cask is sealed. The issues of concern are the creation of a potential or actual release path to the environment, degradation of one or more fuel assemblies due to environmental factors, and configuration changes which could cause challenges in removing the cask or fuel from storage.

The existence of "damage" is determined by radiological survey. The technical specification multiple of "2 times", which is also used in Recognition Category R IC RU1, is used here to distinguish between non-emergency and emergency conditions. The emphasis for this classification is the degradation in the level of safety of the spent fuel cask and not the magnitude of the associated dose or dose rate. It is recognized that in the case of extreme damage to a loaded cask, the fact that the "on-contact" dose rate limit is exceeded may be determined based on measurement of a dose rate at some distance from the cask.

Security-related events for ISFSIs are covered under ICs HU1 and HA1.

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BNP Basis Reference(s):

1. 0PLP-36 BNP 10CFR50.72.212 Report
2. Transnuclear, Inc. Standardized NUHOMS Horizontal Modular Storage System Certificate of Compliance No. 1004, Ammendment 10 Enclosure 1
3. NGGM-PM-0028 Transnuclear NUHOMS Dry Fuel Storage Program Manual
4. NEI 99-01 E-HU1

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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): The Fuel Clad Barrier consists of the cladding material that contains the fuel pellets.
B. Reactor Coolant System (RCS): The RCS Barrier is the reactor coolant system pressure boundary and includes the RPV and all reactor coolant system piping up to and including the isolation valves.
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 PC barrier. Primary Containment Barrier thresholds are used as criteria for escalation of the ECL from Alert to a Site Area Emergency or a General Emergency.

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:

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

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The logic used for emergency classification based on fission product barrier monitoring should reflect the following considerations:

- The Fuel Clad Barrier and the RCS Barrier are weighted more heavily than the Primary Containment Barrier.
- Unusual Event ICs associated with RCS and Fuel Clad Barriers are addressed under System Malfunction ICs.
- For accident conditions involving a radiological release, evaluation of the fission product barrier thresholds will need to be performed in conjunction with dose assessments to ensure correct and timely escalation of the emergency classification. For example, an evaluation of the fission product barrier thresholds may result in a Site Area Emergency classification while a dose assessment may indicate that an EAL for General Emergency IC RG1 has been exceeded.
- The fission product barrier thresholds specified within a scheme reflect plant-specific BNP design and operating characteristics.
- As used in this category, the term RCS leakage encompasses not just those types defined in Technical Specifications but also includes the loss of RCS mass to any location– inside the primary containment, an interfacing system, or outside of the primary containment. The release of liquid or steam mass from the RCS due to the as-designed/expected operation of a relief valve is not considered to be RCS leakage.
- At the Site Area Emergency level, EAL users should maintain cognizance of how far present conditions are from meeting a threshold that would require a General Emergency declaration. For example, if the Fuel Clad and RCS fission product barriers were both lost, then there should be frequent assessments of containment radioactive inventory and integrity. Alternatively, if both the Fuel Clad and RCS fission product barriers were potentially lost, the SEC would have more assurance that there was no immediate need to escalate to a General Emergency.

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Category: Fission Product Barrier Degradation

Subcategory: N/A

Initiating Condition: **Any** loss or **any** potential loss of either Fuel Clad or RCS barrier

EAL:

FA1.1	**Alert**
Any loss or **any** potential loss of either Fuel Clad or RCS (Table F-1)	

Mode Applicability:

1 - Power Operations, 2 - Startup, 3 - Hot Shutdown

Definition(s):

None

Basis:

Fuel Clad, RCS and Primary 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 FS1.1

BNP Basis Reference(s):

1. NEI 99-01 FA1

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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 barriers (Table F-1)	

Mode Applicability:

1 - Power Operations, 2 - Startup, 3 - Hot Shutdown

Definition(s):

None

Basis:

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 Site Emergency Coordinator would have greater assurance that escalation to a General Emergency is less imminent.

BNP Basis Reference(s):

1. NEI 99-01 FS1

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Category: Fission Product Barrier Degradation

Subcategory: N/A

Initiating Condition: Loss of **any** two barriers and loss or potential loss of third barrier

EAL:

FG1.1	**General Emergency**
Loss of **any** two barriers	
AND	
Loss or potential loss of third barrier (Table F-1)	

Mode Applicability:

1 - Power Operations, 2 - Startup, 3 - Hot Shutdown

Definition(s):

None

Basis:

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

BNP Basis Reference(s):

1. NEI 99-01 FG1

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. RCS Leak Rate
B. Primary Containment Conditions
C. Primary Containment Radiation/RCS Activity
D. Primary Containment Integrity or Bypass
E. SEC Jugement

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 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, and FA1.1 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,..., F.

Table F-1 Fission Product Barrier Threshold Matrix

	Fuel Clad Barrier		Reactor Coolant System Barrier		Primary Containment Barrier	
Category	Loss	Potential Loss	Loss	Potential Loss	Loss	Potential Loss
A RPV Water Level	1. Entry to SAMG-01 required	1. RPV level **cannot** be restored and maintained > TAF **or cannot** be determined	1. RPV level **cannot** be restored and maintained > TAF **or cannot** be determined	None	None	1. Entry to SAMG-01 required
B RCS Leak Rate	None	None	1. UNISOLABLE break in any of the following: • Main steam • HPCI steam Line • RCIC steam Line • RWCU • Feedwater 2. Emergency Depressurization is required	1. UNISOLABLE primary system leakage that results in exceeding EITHER of the following: • One or more Secondary Containment area radiation Maximum Normal Operating Limits (0EOP-03-SCCP Table 3) • One or more Secondary Containment area temperature Maximum Normal Operating Limits (0EOP-03-SCCP Table 1)	1. UNISOLABLE primary system leakage that results in exceeding EITHER of the following: • One or more Secondary Containment area radiation Maximum Safe Operating Limits (0EOP-03-SCCP Table 3) • One or more Secondary Containment area temperature Maximum Safe Operating Limits (0EOP-03-SCCP Table 1)	None
C PC Conditions	None	None	1. Primary Containment pressure > 1.7 psig due to RCS leakage	None	1. UNPLANNED rapid drop in Primary Containment pressure following Primary Containment pressure rise 2. Primary Containment pressure response **not** consistent with LOCA conditions	1. Primary Containment pressure > 62 psig 2. Deflagration concentrations exist inside PC ($H_2 \geq 6\%$ AND $O_2 \geq 5\%$) 3. Heat Capacity Temperature Limit (HCTL) exceeded
D PC Rad / RCS Activity	1. Drywell radiation > 2,000 R/hr 2. Primary coolant activity > 300 µCi/gm I-131 dose equivalent	None	1. Drywell radiation > 27 R/hr with reactor shutdown	None	None	1. Drywell radiation > 20,000 R/hr
E PC Integrity or Bypass	None	None	None	None	1. UNISOLABLE direct downstream pathway to the environment exists after Primary Containment isolation signal 2. Intentional Primary Containment venting per EOPs	None
F ED Judgment	1. Any condition in the opinion of the SEC that indicates loss of the fuel clad barrier	1. Any condition in the opinion of the SEC that indicates potential loss of the fuel clad barrier	1. Any condition in the opinion of the SEC that indicates loss of the RCS barrier	1. Any condition in the opinion of the SEC that indicates potential loss of the RCS barrier	1. Any condition in the opinion of the SEC that indicates loss of the Primary Containment barrier	1. Any condition in the opinion of the SEC that indicates potential loss of the Primary Containment barrier

Barrier: Fuel Clad

Category: A. RPV Level

Degradation Threat: Loss

Threshold:

1. Entry to SAMG-01 required

Definition(s):

N/A

Basis:

1(2)EOP-01-RVCP, 0EOP-01-ATWS and 0EOP-01-RXFP specify the requirement for entry to SAMG-01 when core cooling is severely challenged. 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. SAMG-01 is required when (ref. 1):

Deleted: LPC

- Reactor water level cannot be restored and maintained above -57.5 inches (Jet Pump Suction) with at least one core spray pump injecting into the reactor vessel
- Reactor vessel water level cannot be restored and maintained above LL-4 (MSCRWL)
- The reactor vessel flooding conditions cannot be restored and maintained (5 SRVs open and reactor vessel pressure more than 50 psig above suppression chamber pressure)
- When at least 1 SRV cannot be opened and reactor vessel pressure cannot be restored and maintained above the minimum alternate reactor vessel flooding pressure (Table 1 values that are dependent on number of open SRVs)

The above EOP conditions represent a challenge to core cooling and are the minimum values to assure core cooling without further degradation of the clad.

This threshold is also a Potential Loss of the Containment barrier (PC P-Loss A.1). Since entry to SAMG-01 occurs after core uncovery has occurred a Loss of the RCS barrier exists (RCS Loss A.1). Entry to SAMG-01, therefore, represents a Loss of two barriers and a Potential Loss of a third, which requires a General Emergency classification.

The Loss threshold represents the EOP requirement for entry to SAMG-01. This is identified in the BWROG EPGs/SAGs when the phrase, "enter all Severe Accident Guidelines" appears. Since a site-specific RPV water level is not specified here, the Loss threshold phrase, "Entry to SAMG-01 required," also accommodates the EOP need to enter SAMG-01 when RPV water level cannot be determined and core damage due to inadequate core cooling is believed to be occurring.

BNP Basis Reference(s):

1. 0SAMG-06.0 SAMG Primary Containment Flooding Basis Document
2. NEI 99-01 RPV Water Level Fuel Clad Loss 2.A

Barrier: Fuel Clad

Category: A. RPV Level

Degradation Threat: Potential Loss

Threshold:

1. RPV level **cannot** be restored and maintained > TAF or **cannot** be determined

Definition(s):

N/A

Basis:

An RPV level instrument reading of -7.5 in. indicates RPV level is at the top of active fuel (TAF) (ref. 1). When RPV level is at or above the TAF, the core is completely submerged. Core submergence is the most desirable means of core cooling. When RPV level is below TAF, the uncovered portion of the core must be cooled by less reliable means (i.e., steam cooling or spray cooling). If core uncovery is threatened, the EOPs specify alternate, more extreme, RPV level control measures in order to restore and maintain adequate core cooling. Since core uncovery begins if RPV level drops to TAF, the level is indicative of a challenge to core cooling and the Fuel Clad barrier.

When RPV level cannot be determined, EOPs require entry to 0EOP-01-RXFP, Reactor Flooding. RPV water level indication provides the primary means of knowing if adequate core cooling is being maintained (ref. 2, 3). 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 0EOP-01-RXFP 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 Alternate Reactor Vessel Flooding Pressure (in scram-failure events) (ref. 4). 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 1(2)EOP-01-ATWS, ATWS, may require intentionally lowering RPV water level to TAF and control level between the LL-4, the Minimum Steam Cooling RPV Water Level (MSCRWL) and TAF (ref. 3). Under these conditions, a high-power ATWS event exists and requires at least an Alert classification in accordance with the System Malfunction - RPS Failure EALs, however under these conditions a potential loss of the fuel clad does not exist.

Deleted: LPC

Deleted: Level/Power Control

This water level corresponds to the top of the active fuel and is used in the EOPs to indicate a challenge to core cooling.

The RPV water level threshold is the same as RCS barrier Loss threshold 1.A. Thus, this threshold indicates a Potential Loss of the Fuel Clad barrier and a Loss of the 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, 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) or when procedural guidance or a lack of low pressure RPV injection sources preclude Emergency RPV depressurization. EOPs allow the operator a wide choice of RPV injection sources to consider when restoring RPV water level to within prescribed limits. EOPs also specify depressurization of the RPV in order to facilitate RPV water level control with low-pressure injection sources. In some events, elevated RPV pressure may prevent restoration of RPV water level until pressure drops below the shutoff heads of available injection sources. Therefore, this Fuel Clad barrier Potential Loss is met only after either: 1) the RPV has been depressurized, or required emergency RPV depressurization has been attempted, giving the operator an opportunity to assess the capability of low-pressure injection sources to restore RPV water level or 2) no low pressure RPV injection systems are available, precluding RPV depressurization in an attempt to minimize loss of RPV inventory.

The term "cannot be restored and maintained above" means the value of RPV water level is not able to be brought above the specified limit (top of active fuel). The determination requires an evaluation of system performance and availability in relation to the RPV water level value and trend. A threshold prescribing declaration when a threshold value *cannot* be restored and maintained above a specified limit does not require immediate action simply because the current value is below the top of active fuel, but does not permit extended operation below the limit; the threshold must be considered reached as soon as it is apparent that the top of active fuel cannot be attained.

In high-power ATWS/failure to scram events, EOPs may direct the operator to deliberately lower RPV water level to the top of active fuel in order to reduce reactor power. RPV water level is then controlled between the top of active fuel and the Minimum Steam Cooling RPV Water Level (MSCRWL). Although such action is a challenge to core cooling and the Fuel Clad barrier, the immediate need to reduce reactor power is the higher priority. For such events, ICs SA6 or SS6 will dictate the need for emergency classification.

Since the loss of ability to determine if adequate core cooling is being provided presents a significant challenge to the fuel clad barrier, a potential loss of the fuel clad barrier is specified.

BNP Basis Reference(s):

1. 0EOP-01-NL EOP/SAMG Numerical Limits and Values, Attachment 1
2. 1(2)EOP-01-RVCP Reactor Vessel Control
3. 1(2)EOP-01-ATWS, ATWS
4. 0EOP-01-RXFP, Reactor Flooding
5. NEI 99-01 RPV Water Level Fuel Clad Potential Loss 2.A

Deleted: -

Deleted: LPC

Deleted: Level/Power Control

Barrier: Fuel Clad

Category: B. RCS Leak Rate

Degradation Threat: Loss

Threshold:

None

Barrier: Fuel Clad

Category: B. RCS Leak Rate

Degradation Threat: Potential Loss

Threshold:

None

Barrier: Fuel Clad

Category: C. PC Conditions

Degradation Threat: Loss

Threshold:

None

Barrier: Fuel Clad

Category: C. PC Conditions

Degradation Threat: Potential Loss

Threshold:

None

Barrier: Fuel Clad

Category: D. PC Radiation / RCS Activity

Degradation Threat: Loss

Threshold:

1. Drywell radiation > 2,000 R/hr

Definition(s):

None

Basis:

The Drywell High-Range Radiation Monitor (1(2)D22-RI-4195, 1(2)D22-RI-4196, 1(2)D22-RI-4197, 1(2)D22-RI-4198) reading of 2,000 R/hr indicates the release of reactor coolant into the drywell with elevated activity indicative of fuel damage.

Based on 2% clad damage, a containment radiation level of 2000 R/hr is derived as follows:

Per 0PEP-03.6.3 Table 3, 100% Cladding Damage column 'No Spray' for 1 hour after shutdown is 100,000 R/hr. Per Step 7.2.2.1, 0.02 x 100,000 R/hr = 2000 R/hr containment radiation (ref. 1).

The radiation monitor reading corresponds to an instantaneous release of all reactor coolant mass into the primary containment, assuming that reactor coolant activity equals 300 μCi/gm dose equivalent I-131. Reactor coolant activity above this level is greater than that expected for iodine spikes and corresponds to an approximate range of 2% to 5% fuel clad damage. Since this condition indicates that a significant amount of fuel clad damage has occurred, it represents a loss of the Fuel Clad Barrier.

The radiation monitor reading in this threshold is higher than that specified for RCS Barrier Loss threshold D.1 since it indicates a loss of both the Fuel Clad Barrier and the RCS Barrier. Note that a combination of the two monitor readings appropriately escalates the emergency classification level to a Site Area Emergency.

There is no Potential Loss threshold associated with Primary Containment Radiation.

BNP Basis Reference(s):

1. 0PEP-03.6.3 Estimate of the Extent of Core Damage Under Accident Conditions
2. NEI 99-01 Primary Containment Radiation Fuel Clad Loss 4.A

Barrier: Fuel Clad

Category: D. PC Radiation / RCS Activity

Degradation Threat: Loss

Threshold:

2. Primary coolant activity > 300 μCi/gm I-131 dose equivalent

Basis:

This threshold indicates that RCS radioactivity concentration is greater than 300 μCi/gm dose equivalent I-131. Reactor coolant activity above this level is greater than that expected for iodine spikes and corresponds to an approximate range of 2% to 5% fuel clad damage. Since this condition indicates that a significant amount of fuel clad damage has occurred, it represents a loss of the Fuel Clad Barrier.

There is no Potential Loss threshold associated with RCS Activity.

There is no Potential Loss threshold associated with Primary Containment Radiation.

BNP Basis Reference(s):

1. NEI 99-01 RCS Activity Fuel Clad Loss 1.A

Barrier: Fuel Clad

Category: D. PC Radiation / RCS Activity

Degradation Threat: Potential Loss

Threshold:

None

Barrier: Fuel Clad

Category: E. PC Integrity or Bypass

Degradation Threat: Loss

Threshold:

None

Barrier: Fuel Clad

Category: E. PC Integrity or Bypass

Degradation Threat: Potential Loss

Threshold:

None

Barrier: Fuel Clad

Category: F. SEC Judgment

Degradation Threat: Loss

Threshold:

1. **Any** condition in the opinion of the Site Emergency Coordinator that indicates loss of the Fuel Clad barrier

Definition(s):

None

Basis:

The Site Emergency Coordinator 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 Site Emergency Coordinator should be mindful of the Loss of AC power (Station Blackout) and ATWS EALs to assure timely emergency classification declarations.

This threshold addresses any other factors that are to be used by the Site Emergency Coordinator in determining whether the Fuel Clad barrier is lost

BNP Basis Reference(s):

1. NEI 99-01 Emergency Director Judgment Fuel Clad Loss 6.A

Barrier: Fuel Clad

Category: F. SEC Judgment

Degradation Threat: Potential Loss

Threshold:

1. **Any** condition in the opinion of the Site Emergency Coordinator that indicates potential loss of the Fuel Clad barrier

Basis:

The Emergency Coordinator 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 Site Emergency Coordinator should be mindful of the Loss of AC power (Station Blackout) and ATWS EALs to assure timely emergency classification declarations.

This threshold addresses any other factors that are to be used by the Site Emergency Coordinator in determining whether the Fuel Clad barrier is potentially lost. The Site Emergency Coordinator should also consider whether or not to declare the barrier potentially lost in the event that barrier status cannot be monitored.

BNP Basis Reference(s):

1. NEI 99-01 Emergency Director Judgment Potential Fuel Clad Loss 6.A

Barrier: Reactor Coolant System

Category: A. RPV Water Level

Degradation Threat: Loss

Threshold:

1. RPV level **cannot** be restored and maintained > TAF or **cannot** be determined

Definition(s):

None

Basis:

An RPV level instrument reading of -7.5 in. indicates RPV level is at the top of active fuel (TAF) (ref. 1). TAF is significantly lower than the normal operating RPV level control band. To reach this level, RPV inventory loss would have previously required isolation of the RCS and Primary Containment barriers, and initiation of all ECCS. If RPV level cannot be maintained above TAF, ECCS and other sources of RPV injection have been ineffective or incapable of reversing the decreasing level trend. The cause of the loss of RPV inventory is therefore assumed to be a LOCA. By definition, a LOCA event is a Loss of the RCS barrier.

When RPV level cannot be determined, EOPs require entry to 0EOP-01-RXFP, Reactor Flooding (ref. 2). The instructions in 0EOP-01-RXFP specify emergency depressurization of the RPV, which is defined to be a Loss of the RCS barrier (RCS Loss C.4).

The conditions of this threshold are also a Potential Loss of the Fuel Clad barrier (FC P-Loss A.1). A Loss of the RCS barrier and Potential Loss of the Fuel Clad barrier requires a Site Area Emergency classification.

Note that 1(2)EOP-01-ATWS, ATWS, may require intentionally lowering RPV water level to TAF and control level between LL-4, the Minimum Steam Cooling RPV Water Level (MSCRWL), and TAF (ref. 3). Under these conditions, a high-power ATWS event exists and requires at least an Alert classification in accordance with the System Malfunction - RPS Failure EALs, however under these conditions a loss of the RCS does not exist.

Deleted: LPC

Deleted: Level/Power Control

This water level corresponds to the top of active fuel and is used in the EOPs to indicate challenge to core cooling.

The RPV water level threshold is the same as Fuel Clad barrier Potential Loss threshold 1.A. Thus, this threshold indicates a Loss of the RCS barrier and Potential Loss of the Fuel Clad barrier and 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, 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) or when procedural guidance or a lack of low pressure RPV injection sources preclude Emergency RPV depressurization. EOPs allow the operator a wide choice of RPV injection sources to consider when restoring RPV water level to within prescribed limits. EOPs also specify depressurization of the RPV in order to facilitate RPV water level control with low-pressure injection sources. In some events, elevated RPV pressure may prevent restoration of RPV water level until pressure drops below the shutoff heads of available injection sources. Therefore, this RCS barrier Loss is met only after either: 1) the RPV has been depressurized, or required emergency RPV depressurization has been attempted, giving the operator an opportunity to assess the capability of low-pressure injection sources to restore RPV water level or 2) no low pressure RPV injection systems are available, precluding RPV depressurization in an attempt to minimize loss of RPV inventory.

BNP Basis Reference(s):

1. 0EOP-01-NL EOP/SAMG Numerical Limits and Values, Attachment 1
2. 1(2)EOP-01-RVCP Reactor Vessel Control
3. 1(2)EOP-01ATWS, ATWS
4. 0EOP-01-RXFP, Reactor Flooding
5. NEI 99-01 RPV Water Level RCS Loss 2.A

Deleted: -

Deleted: -LPC

Deleted: Level/Power Control

Barrier: Reactor Coolant System

Category: A. RPV Water Level

Degradation Threat: Potential Loss

Threshold:

None

Barrier: Reactor Coolant System

Category: B. RCS Leak Rate

Degradation Threat: Loss

Threshold:

1. UNISOLABLE break outside Primary Containment in any of the following:
   - Main steam line
   - HPCI steam line
   - RCIC steam line
   - RWCU
   - Feedwater

Definition(s):

UNISOLABLE - An open or breached system line that cannot be isolated, remotely or locally.

Basis:

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. In the case of a failure of both isolation valves to close but in which no downstream flowpath exists, emergency declaration under this threshold would not be required. 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 Primary Containment (see PC Loss E.1) barriers and justifies declaration of a Site Area Emergency (i.e., Loss or Potential Loss of any two barriers).

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.

Large high-energy lines that rupture outside primary containment can discharge significant amounts of inventory and jeopardize the pressure-retaining capability of the RCS until they are isolated. If it is determined that the ruptured line cannot be promptly isolated from the Control Room, the RCS barrier Loss threshold is met.

BNP Basis Reference(s):

1. 1(2)OP-01 Nuclear Boiler System
2. 1(2)OP-25 Main Steam System Operating Procedure
3. 1(2)OP-19 High Pressure Coolant Injection System Operating Procedure
4. 1(2)OP-16 Reactor Core Isolation Cooling System Operating Procedure
5. 1(2)OP-14 Reactor Water Cleanup System Operating Procedure
6. 1(2)OP-32 Condensate and Feedwater System Operating Procedure
7. NEI 99-01 RCS Leak Rate RCS Loss 3.A

Barrier: Reactor Coolant System

Category: B. RCS Leak Rate

Degradation Threat: Loss

Threshold:

2. Emergency Depressurization is required

Definition(s):

N/A

Basis:

Plant symptoms requiring Emergency Depressurization per the EOPs are indicative of a loss of the RCS barrier. If Emergency depressurization is required, the plant operators are directed to open safety relief valves (SRVs) and keep them open regardless of any subsequent radiological release rate (ref. 1 – 6). 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.

Emergency RPV Depressurization in accordance with the EOPs is indicative of a loss of the RCS barrier. If Emergency RPV Depressurization is performed, 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 barrier exists due to the diminished effectiveness of the RCS to retain fission products within its boundary.

BNP Basis Reference(s):

1. 0EOP-01-UG User's Guide
2. 1(2)EOP-01-RVCP Reactor Vessel Control
3. 1(2)EOP-01-ATWS ATWS
4. 0EOP-02-PCCP Primary Containment Control
5. 0EOP-03-SCCP Secondary Containment Control
6. 0EOP-04-RRCP Radioactivity Release Control
7. 0EOP-01-RXFP Reactor Flooding
8. NEI 99-01 RCS Leak Rate RCS Loss 3.B

Deleted: LCP

Deleted: Level/Power Control

Barrier: Reactor Coolant System

Category: B. RCS Leak Rate

Degradation Threat: Potential Loss

Threshold:

1. UNISOLABLE primary system leakage that results in exceeding **EITHER** of the following:
   - One or more Secondary Containment area radiation Maximum Normal Operating Limits (0EOP-03-SCCP Table SC-3)
   - One or more Secondary Containment area temperature Maximum Normal Operating Limits (0EOP-03-SCCP Table SC-1)

Definition(s):

UNISOLABLE - An open or breached system line that cannot be isolated, remotely or locally.

Basis:

The presence of elevated general area temperatures or radiation levels in the secondary containment may be indicative of unisolable primary system leakage outside the primary containment. The Maximum Normal Operating Limit values define this RCS threshold because they are the maximum normal operating values and signify the onset of abnormal system operation. When parameters reach this 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. The locations into which the primary system discharge is of concern correspond to the areas addressed in 0EOP-03-SCCP, Secondary Containment Control Tables (ref. 1).

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.

Potential loss of RCS based on primary system leakage outside the primary containment is determined from EOP temperature or radiation Max Normal Operating values in areas such as main steam line tunnel, RCIC, HPCI, etc., which indicate a direct path from the RCS to areas outside primary containment.

A Max Normal Operating value is the highest value of the identified parameter expected to occur during normal plant operating conditions with all directly associated support and control systems functioning properly.

The indicators reaching the threshold barriers and confirmed to be caused by RCS leakage from a primary system warrant an Alert classification. A primary system is defined to be the pipes, valves, and other equipment which connect directly to the RPV such that a reduction in RPV pressure will effect a decrease in the steam or water being discharged through an unisolated break in the system.

An UNISOLABLE leak which is indicated by Max Normal Operating values escalates to a Site Area Emergency when combined with Containment Barrier Loss threshold 3.A (after a containment isolation) and a General Emergency when the Fuel Clad Barrier criteria is also exceeded.

BNP Basis Reference(s):

1. 0EOP-03-SCCP, Secondary Containment Control
2. NEI 99-01 RCS Leak Rate RCS Potential Loss 3.A

Barrier: Reactor Coolant System

Category: C. PC Conditions

Degradation Threat: Loss

Threshold:

1. Primary Containment pressure > 1.7 psig due to RCS leakage

Definition(s):

None

Basis:

The drywell high pressure scram setpoint is an entry condition to 1(2)EOP-01-RVCP Reactor Vessel Control, and 0EOP-02-PCCP, Primary Containment Control (ref. 1, 2, 3). Normal primary containment pressure control functions (e.g., operation of drywell coolers, vent through SBGT, etc.) are specified in 0EOP-02-PCCP in advance of less desirable but more effective functions (e.g., operation of drywell or suppression pool sprays, etc.).

In the BNP 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. 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. PC pressure greater than 1.7 psig with corollary indications (e.g., 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 1.7 psig should not be considered an RCS barrier Loss.

1.7 psig is the drywell high pressure setpoint which indicates a LOCA by automatically initiating the ECCS or equivalent makeup system.

There is no Potential Loss threshold associated with Primary Containment Pressure.

BNP Basis Reference(s):

1. 0EOP-01-NL EOP/SAMG Numerical Limits and Values, Attachment 3

Deleted: -

2. 1(2)EOP-01-RVCP Reactor Vessel Control
3. 0EOP-02-PCCP Primary Containment Control
4. BNP Updated FSAR Chapter 6 Emergency Core Cooling Systems
5. NEI 99-01 Primary Containment Pressure RCS Loss 1.A

Barrier: Reactor Coolant System

Category: C. PC Conditions

Degradation Threat: Potential Loss

Threshold:

None

Barrier: Reactor Coolant System

Category: D. PC Radiation / RCS Activity

Degradation Threat: Loss

Threshold:

1. Drywell radiation > 27 R/hr with reactor shutdown

Definition(s):

N/A

Basis:

The Drywell High-Range Radiation Monitor (1(2)D22-RI-4195, 1(2)D22-RI-4196, 1(2)D22-RI-4197, 1(2)D22-RI-4198) reading of 27 R/hr is based on coolant activity at the Technical Specification limit of 4 µCi/gm I-131).

The containment radiation level of 27 R/hr is derived as follows:

0PEP-03.6.3 Table 3 100% Cladding Damage column 'No Spray' for 1 hour after shutdown is 100,000 R/hr. Assuming that 300 µCi/gm I-131 is approximately 2% cladding failure, a coolant activity of 4 µCi/gm I-131 is ratioed to approximately 0.027% (0.00027) clad failure. Per Step 7.2.2.1, 0.00027 x 100,000 R/hr = 27 R/hr containment radiation corresponding to Technical Specification coolant activity. (ref. 1)

The threshold value is only applicable with the reactor shutdown as the high range detectors normally read as high as 100 R/hr during power operations due to shine from the reactor.

The radiation monitor reading corresponds to an instantaneous release of all reactor coolant mass into the primary containment, assuming that reactor coolant activity equals Technical Specification allowable limits. This value is lower than that specified for Fuel Clad Barrier Loss threshold D.1 since it indicates a loss of the RCS Barrier only.

There is no Potential Loss threshold associated with Primary Containment Radiation.

BNP Basis Reference(s):

1. 0PEP-03.6.3 Estimate of the Extent of Core Damage Under Accident Conditions
2. NEI 99-01 Primary Containment Radiation RCS Loss 4.A

Barrier: Reactor Coolant System

Category: D. PC Radiation / RCS Activity

Degradation Threat: Potential Loss

Threshold:

None

Barrier: Reactor Coolant System

Category: E. PC Integrity or Bypass

Degradation Threat: Loss

Threshold:

None

Barrier: Reactor Coolant System

Category: E. PC Integrity or Bypass

Degradation Threat: Potential Loss

Threshold:

None

Barrier: Reactor Coolant System

Category: F. SEC Judgment

Degradation Threat: Loss

Threshold:

1. **Any** condition in the opinion of the Site Emergency Coordinator that indicates loss of the RCS barrier

Definition(s):

None

Basis:

The Site Emergency Coordinator 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 Site Emergency Coordinator should be mindful of the Loss of AC power (Station Blackout) and ATWS EALs to assure timely emergency classification declarations.

This threshold addresses any other factors that may be used by the Site Emergency Coordinator in determining whether the RCS Barrier is lost.

BNP Basis Reference(s):

1. NEI 99-01 Emergency Director Judgment RCS Loss 6.A

Barrier: Reactor Coolant System

Category: F. SEC Judgment

Degradation Threat: Potential Loss

Threshold:

1. **Any** condition in the opinion of the Site Emergency Coordinator that indicates potential loss of the RCS barrier

Definition(s):

None

Basis:

The Site Emergency Coordinator 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 Site Emergency Coordinator should be mindful of the Loss of AC power (Station Blackout) and ATWS EALs to assure timely emergency classification declarations.

This threshold addresses any other factors that may be used by the Site Emergency Coordinator in determining whether the RCS Barrier is potentially lost. The Emergency Director should also consider whether or not to declare the barrier potentially lost in the event that barrier status cannot be monitored.

BNP Basis Reference(s):

1. NEI 99-01 Emergency Director Judgment RCS Potential Loss 6.A

Barrier: Primary Containment

Category: A. RPV Water Level

Degradation Threat: Loss

Threshold:

None

Barrier: Primary Containment

Category: A. RPV Water Level

Degradation Threat: Potential Loss

Threshold:

1. Entry to SAMG-01 required

Definition(s):

None

Basis:

1(2)EOP-01-RVCP, 1(2)EOP-01-ATWS and 0EOP-01-RXFP specify the requirement for entry to SAMG-01 when core cooling is severely challenged. 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. Entry to SAMG-01 is required when (ref. 1):

Deleted: LPC

- Reactor water level cannot be restored and maintained above -57.5 inches (Jet Pump Suction) with at least one core spray pump injecting into the reactor vessel
- Reactor vessel water level cannot be restored and maintained above LL-4 (MSCRWL)
- The reactor vessel flooding conditions cannot be restored and maintained (5 SRVs open and reactor vessel pressure more than 50 psig above suppression chamber pressure)
- When at least 1 SRV cannot be opened and reactor vessel pressure cannot be restored and maintained above the minimum alternate reactor vessel flooding pressure (0EOP-01-RXFP Table 1 values that are dependent on number of open SRVs)

The above EOP conditions represent a challenge to core cooling and are the minimum values to assure core cooling without further degradation of the clad.

This threshold is also a Loss of the Fuel Clad barrier (FC Loss A.1). Since entry to SAMG-01 occurs after core uncovery has occurred a Loss of the RCS barrier exists (RCS Loss A.1). Entry to SAMG-01, therefore, represents a Loss of two barriers and a Potential Loss of a third, which requires a General Emergency classification.

The Potential Loss threshold is identical to the Fuel Clad Loss RPV Water Level threshold A.1. The Potential Loss requirement for entry to SAMG-01 indicates adequate core cooling cannot be restored and maintained and that core damage is possible. BWR EPGs/SAGs specify the conditions that require entry to SAMG-01. When entry to SAMG-01 is required, the EPGs are exited and SAGs are entered. Entry into SAGs is a logical escalation in response to the inability to restore and maintain adequate core cooling.

PRA studies indicate that the condition of this Potential Loss threshold could be a core melt sequence which, if not corrected, could lead to RPV failure and increased potential for primary containment failure. In conjunction with the RPV water level Loss thresholds in the Fuel Clad and RCS barrier columns, this threshold results in the declaration of a General Emergency.

BNP Basis Reference(s):

1. 0SAMG-06.0 SAMG Primary Containment Flooding Basis Document
2. NEI 99-01 RPV Water Level PC Potential Loss 2.A

Barrier: Primary Containment

Category: B. RCS Leak Rate

Degradation Threat: Loss

Threshold:

1. UNISOLABLE primary system leakage that results in exceeding one or more Secondary Containment area temperature Maximum Safe Operating Limits (0EOP-03-SCCP Table SC-1)

Definition(s):

UNISOLABLE - An open or breached system line that cannot be isolated, remotely or locally.

Basis:

The presence of elevated general area temperatures in the secondary containment may be indicative of unisolable primary system leakage outside the primary containment. The Maximum Safe Operating Limit area temperature 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 0EOP-03-SCCP, Secondary Containment Control Table 1 (ref. 1) (see below).

In general, multiple indications should be used to determine if a primary system is discharging outside Primary Containment. A high area temperature condition in conjunction with other indications (e.g. room flooding, 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.

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Deleted: EITHER of the following:¶ One or more Secondary Containment area radiation Maximum Safe Operating Limits (0EOP-03-SCCP Table 3)¶

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Deleted: 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

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The Max Safe Operating Temperatures are each the highest value of these parameters at which neither: (1) equipment necessary for the safe shutdown of the plant will fail, nor (2) personnel access necessary for the safe shutdown of the plant will be precluded. EOPs utilize these temperatures to establish conditions under which RPV depressurization is required.

Deleted: and the Max Safe Operating Radiation Level

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The temperatures should be confirmed to be caused by RCS leakage from a primary system. A primary system is defined to be the pipes, valves, and other equipment which connect directly to the RPV such that a reduction in RPV pressure will effect a decrease in the steam or water being discharged through an unisolated break in the system.

In combination with RCS Potential Loss 2.A this threshold would result in a Site Area Emergency.

There is no Potential Loss threshold associated with Primary Containment Isolation Failure.

BNP Basis Reference(s):

1. 0EOP-03-SCCP Secondary Containment Control
2. NEI 99-01 RCS Leak Rate PC Loss 3.C

Barrier: Primary Containment

Category: B. RCS Leak Rate

Degradation Threat: Potential Loss

Threshold:

None

Barrier: Primary Containment

Category: C. PC Conditions

Degradation Threat: Loss

Threshold:

1. UNPLANNED rapid drop in Primary Containment pressure following Primary Containment pressure rise

Definition(s):

UNPLANNED - A parameter change or an event that is not 1) the result of an intended evolution or 2) an expected plant response to a transient. The cause of the parameter change or event may be known or unknown.

Basis:

Rapid UNPLANNED loss of primary containment pressure (i.e., not attributable to drywell spray or condensation effects) following an initial pressure increase indicates a loss of primary containment integrity.

This threshold relies on operator recognition of an unexpected response for the condition and therefore a specific value is not assigned. The unexpected (UNPLANNED) response is important because it is the indicator for a containment bypass condition.

BNP Basis Reference(s):

1. NEI 99-01 Primary Containment Conditions PC Loss 1.A

Barrier: Primary Containment

Category: C. PC Conditions

Degradation Threat: Loss

Threshold:

2. Primary Containment pressure response **not** consistent with LOCA conditions

Definition(s):

None

Basis:

The calculated pressure response of the containment is shown in Figure 6-11. Figure 6-11 shows that the maximum calculated drywell pressure is 48 psia (33 psig), which is well below the design allowable pressure of 62 psig (ref. 2). The primary containment pressure stabilizes at about 40 psia (25 psig), as shown on Figure 6-1.

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.

Primary containment pressure should increase as a result of mass and energy release into the primary containment from a LOCA. Thus, primary containment pressure not increasing under these conditions indicates a loss of primary containment integrity.

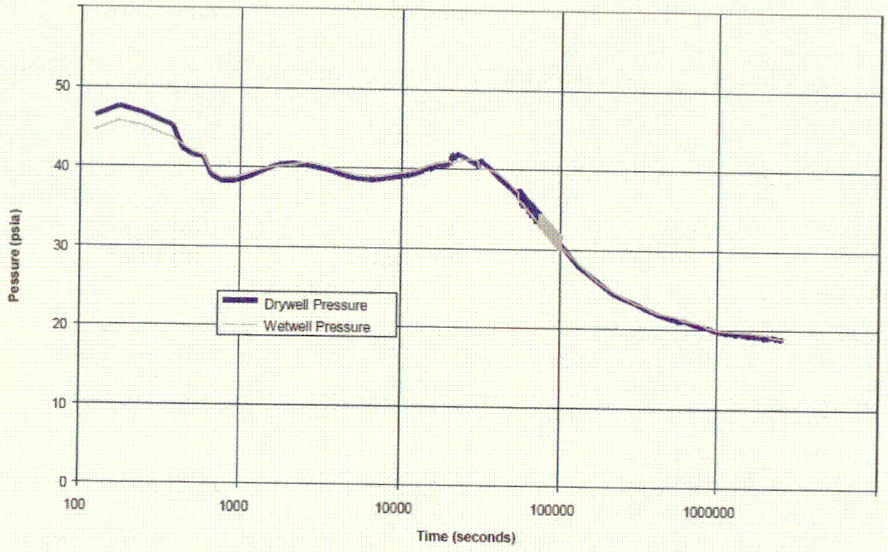
This threshold relies on operator recognition of an unexpected response for the condition and therefore a specific value is not assigned. The unexpected (UNPLANNED) response is important because it is the indicator for a containment bypass condition.

BNP Basis Reference(s):

1. BNP Updated FSAR Figure 6-11
2. BNP Updated FSAR section 6.2.1.1.1
3. NEI 99-01 Primary Containment Conditions PC Loss 1.B

CP&L A Progress Energy Company	*UPDATED FSAR* *ENGINEERED SAFETY FEATURES* *CHAPTER 6 FIGURES*	Revision: 18A Figure: 6-11 Page: 1 of 1

CONTAINMENT PRESSURE RESPONSE TO DESIGN BASIS ACCIDENT LOCA AT 120 PERCENT UPRATE



Barrier: Containment

Category: B. PC Conditions

Degradation Threat: Potential Loss

Threshold:

1. Primary Containment pressure > 62 psig

Definition(s):

None

Basis:

When the primary containment exceeds the maximum allowable value (62 psig) (ref. 1), primary containment venting may be required even if offsite radioactivity release rate limits will be exceeded (ref. 2). The drywell and suppression chamber maximum allowable value of 62 psig is based on the primary containment design pressure as identified in the BNP accident analysis (ref. 1, 3). If this threshold is exceeded, a challenge to the containment structure has occurred because assumptions used in the accident analysis are no longer valid and an unanalyzed condition exists. This constitutes a Potential Loss of the Containment barrier even if a containment breach has not occurred.

The threshold pressure is the primary containment internal design pressure. Structural acceptance testing demonstrates the capability of the primary containment to resist pressures greater than the internal design pressure. A pressure of this magnitude is greater than those expected to result from any design basis accident and, thus, represent a Potential Loss of the Containment barrier.

BNP Basis Reference(s):

1. BNP Updated FSAR section 6.2.1.1.1
2. 0EOP-02-PCCP Primary Containment Control
3. BNP Updated FSAR section 6.2.1.1
4. NEI 99-01 Primary Containment Conditions PC Potential Loss 1.A

Barrier: Containment

Category: B. PC Conditions

Degradation Threat: Potential Loss

Threshold:

2. Deflagration concentrations exist inside PC ($H_2 \geq 6\%$ **AND** $O_2 \geq 5\%$)

Definition(s):

None

Basis:

Deflagration (explosive) mixtures in the primary containment are assumed to be elevated concentrations of hydrogen and oxygen. BWR industry evaluation of hydrogen generation for development of EOPs/SAMGs indicates that any hydrogen concentration above minimum detectable is not to be expected within the short term. Post-LOCA hydrogen generation primarily caused by radiolysis is a slowly evolving, long-term condition. Hydrogen concentrations that rapidly develop are most likely caused by metal-water reaction. A metal-water reaction is indicative of an accident more severe than accidents considered in the plant design basis and would be indicative, therefore, of a potential threat to primary containment integrity. 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. 2) and readily recognizable because 6% hydrogen is well above the 0EOP-02-PCCP, Primary Containment Control, entry condition (ref. 2). The minimum global deflagration hydrogen/oxygen concentrations (6% and 5%, respectively) require intentional primary containment venting, which is defined to be a Loss of Containment (PC Loss C.4).

Monitors CAC-AT-4409 and 4410 monitor hydrogen and oxygen concentration in the drywell during normal operation. These monitors are isolated by accident isolation signals. However, monitors CAC-AT-4409 and 4410 will be realigned to the primary containment for post-accident monitoring via an operator actuated isolation signal override circuit when directed by the EOPs.

The oxygen and hydrogen concentrations from these two analyzers are recorded on two 4-channel recorders (CAC-AR-4409 and 4410) located on Panel XU-51. The indications are also displayed on the ERFIS. If concentrations exceed preset levels, recorder CAC-AR-4409 will annunciate the "Containment Atmosphere Division I 02 - H2 High" alarm in the Control Room and recorder CAC-AR-4410 will annunciate "Containment Atmosphere Division II 02 - H2 High" alarm. (ref. 3, 4).

If hydrogen concentration reaches or exceeds the lower flammability limit, as defined in plant EOPs, in an oxygen rich environment, a potentially explosive mixture exists. If the combustible mixture ignites inside the primary containment, loss of the Containment barrier could occur.

BNP Basis Reference(s):

1. BWROG EPG/SAG Revision 2, Sections PC/G
2. 0EOP-02-PCCP, Primary Containment Control
3. BNP Updated FSAR section 6.2.5.2.2
4. BNP System Description SD-04 Primary Containment
5. NEI 99-01 Primary Containment Conditions PC Potential Loss 1.B

Barrier: Containment

Category: B. PC Conditions

Degradation Threat: Potential Loss

Threshold:

3. Heat Capacity Temperature Limit (HCTL) exceeded

Definition(s):

None

Basis:

This threshold is met when the final step of section SP/T in 0EOP-02-PCCP, Primary Containment Control, is reached (ref. 1, 2).

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 maximum temperature capability of the suppression chamber and equipment within the suppression chamber which may be required to operate when the RPV is pressurized,
>
> OR
>
> Suppression chamber pressure above Primary Containment Pressure Limit A, while 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, suppression pool temperature 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.

BNP Basis Reference(s):

1. 0EOP-01-NL EOP/SAMG Numerical Limits and Values
2. 0EOP-02-PCCP Primary Containment Control
3. NEI 99-01 Primary Containment Conditions PC Potential Loss 1.C

Barrier: Primary Containment

Category: D. PC Radiation / RCS Activity

Degradation Threat: Loss

Threshold:

None

Barrier: Primary Containment

Category: D. PC Radiation / RCS Activity

Degradation Threat: Potential Loss

Threshold:

1. Drywell radiation > 20,000 R/hr

Definition(s):

None

Basis:

The Drywell High-Range Radiation Monitor (1(2)D22-RI-4195, 1(2)D22-RI-4196, 1(2)D22-RI-4197, 1(2)D22-RI-4198) reading of 20,000 R/hr indicates the release of reactor coolant into the drywell with elevated activity indicative of fuel damage.

Based on 20% clad damage, a containment radiation level of 20,000 R/hr is derived as follows:

0PEP-03.6.3 Table 3 100% Cladding Damage column 'No Spray' for 1 hour after shutdown is 100,000 R/hr. Per Step 7.2.2.1, 0.2 x 100,000 R/hr = 20,000 R/hr containment radiation corresponding to 20% clad damage.

In order to reach this Containment barrier Potential Loss threshold, a loss of the RCS barrier (RCS Loss D.5) and a loss of the Fuel Clad barrier (FC Loss D.2) have already occurred. This threshold, therefore, represents at a General Emergency classification.

The radiation monitor reading corresponds to an instantaneous release of all reactor coolant mass into the primary containment, assuming that 20% of the fuel cladding has failed. This level of fuel clad failure is well above that used to determine the analogous Fuel Clad Barrier Loss and RCS Barrier Loss thresholds.

NUREG-1228, *Source Estimations During Incident Response to Severe Nuclear Power Plant Accidents*, indicates the fuel clad failure must be greater than approximately 20% in order for there to be a major release of radioactivity requiring offsite protective actions. For this condition to exist, there must already have been a loss of the RCS Barrier and the Fuel Clad Barrier. It is therefore prudent to treat this condition as a potential loss of containment which would then escalate the emergency classification level to a General Emergency.

BNP Basis Reference(s):

1. 0PEP-03.6.3 Estimate of the Extent of Core Damage Under Accident Conditions

Barrier: Primary Containment

Category: E. PC Integrity or Bypass

Degradation Threat: Loss

Threshold:

1. UNISOLABLE direct downstream pathway to the environment exists after Primary Containment isolation signal

Definition(s):

UNISOLABLE - An open or breached system line that cannot be isolated, remotely or locally.

Basis:

This threshold addresses failure of open isolation devices which should close upon receipt of a manual or automatic containment isolation signal resulting in a significant radiological release pathway 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 "release 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, HPCI steam line or RCIC steam line breaks, unisolable RWCU system breaks, and unisloable 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, Abnormal Rad Release / Rad Effluent, EALs.

The existence of an in–line charcoal filter (SBGT) 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.

0EOP-02-PCCP, Primary Containment Control, Section PC/P 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). Under these conditions with a valid containment isolation signal, the Containment barrier should be considered lost.

The use of the modifier "direct" in defining the release path discriminates against release paths through interfacing liquid systems or minor release pathways, such as instrument lines, not protected by the Primary Containment Isolation System (PCIS).

The existence of a filter is not considered in the threshold assessment. Filters do not remove fission product noble gases. In addition, a filter could become ineffective due to iodine and/or particulate loading beyond design limits (i.e., retention ability has been exceeded) or water saturation from steam/high humidity in the release stream.

Following the leakage of RCS mass into primary containment and a rise in primary containment pressure, there may be minor radiological releases associated with allowable primary containment leakage through various penetrations or system components. Minor releases may also occur if a primary containment isolation valve(s) fails to close but the primary containment atmosphere escapes to an enclosed system. These releases do not constitute a loss or potential loss of primary containment but should be evaluated using the Recognition Category R ICs.

BNP Basis Reference(s):

1. 0EOP-02-PCCP Primary Containment Control
2. NEI 99-01 Primary Containment Isolation Failure PC Loss 3.A

Barrier: Primary Containment

Category: E. PC Integrity or Bypass

Degradation Threat: Loss

Threshold:

2. Intentional Primary Containment venting per EOPs

Definition(s):

None

Basis:

0EOP-02-PCCP, 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. 1, 2). The threshold is met when the operator begins venting the primary containment in accordance with 0EOP-01-SEP-01, not when actions are taken to bypass interlocks prior to opening the vent valves. Purge and vent actions specified in step PC/P-03 to control drywell pressure below the drywell high pressure scram setpoint or in section PC/H does not meet this threshold because such action is only permitted if offsite radioactivity release rates will remain below the ODCM limits.

EOPs may direct primary containment isolation valve logic(s) to be intentionally bypassed, even if offsite radioactivity release rate limits will be exceeded. Under these conditions with a valid primary containment isolation signal, the containment should also be considered lost if primary containment venting is actually performed.

Intentional venting of primary containment for primary containment pressure control to the secondary containment and/or the environment is a Loss of the Containment. Venting for primary containment pressure control when not in an accident situation (e.g., to control pressure below the drywell high pressure scram setpoint) does not meet the threshold condition.

BNP Basis Reference(s):

1. 0EOP-02-PCCP Primary Containment Control
2. 0EOP-01-SEP-01 Primary Containment Venting
3. NEI 99-01 Primary Containment Isolation Failure PC Loss 3.B

Barrier: Primary Containment

Category: E. PC Integrity or Bypass

Degradation Threat: Potential Loss

Threshold:

None

Barrier: Primary Containment

Category: F. SEC Judgment

Degradation Threat: Loss

Threshold:

1. **Any** condition in the opinion of the Site Emergency Coordinator that indicates loss of the Primary Containment barrier

Definition(s):

None.

Basis:

The Site Emergency Coordinator judgment threshold addresses any other factors relevant to determining if the Primary 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 Site Emergency Coordinator should be mindful of the Loss of AC power (Station Blackout) and ATWS EALs to assure timely emergency classification declarations.

This threshold addresses any other factors that may be used by the Site Emergency Coordinator in determining whether the Primary Containment Barrier is lost.

BNP Basis Reference(s):
1. NEI 99-01 Emergency Director Judgment PC Loss 6.A

Barrier: Primary Containment

Category: F. SEC Judgment

Degradation Threat: Potential Loss

Threshold:

1. **Any** condition in the opinion of the Site Emergency Coordinator that indicates potential loss of the Primary Containment barrier

Definition(s):

None

Basis:

The Site Emergency Coordinator judgment threshold addresses any other factors relevant to determining if the Primary 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 Site Emergency Coordinator should be mindful of the Loss of AC power (Station Blackout) and ATWS EALs to assure timely emergency classification declarations.

This threshold addresses any other factors that may be used by the Site Emergency Coordinator in determining whether the Primary Containment Barrier is lost.

BNP Basis Reference(s):

1. NEI 99-01 Emergency Director Judgment PC Loss 6.A

Background

NEI 99-01 Revision 6 ICs AA3 and HA5 prescribe declaration of an Alert based on impeded access to rooms or areas (due to either area radiation levels or hazardous gas concentrations) where equipment necessary for normal plant operations, cooldown or shutdown is located. These areas are intended to be plant operating mode dependent. Specifically the Developers Notes For AA3 and HA5 states:

The "site-specific list of plant rooms or areas with entry-related mode applicability identified" should specify those rooms or areas that contain equipment which require a manual/local action as specified in operating procedures used for normal plant operation, cooldown and shutdown. Do not include rooms or areas in which actions of a contingent or emergency nature would be performed (e.g., an action to address an off-normal or emergency condition such as emergency repairs, corrective measures or emergency operations). In addition, the list should specify the plant mode(s) during which entry would be required for each room or area.

The list should not include rooms or areas for which entry is required solely to perform actions of an administrative or record keeping nature (e.g., normal rounds or routine inspections).

Further, as specified in IC HA5:

The list need not include the Control Room if adequate engineered safety/design features are in place to preclude a Control Room evacuation due to the release of a hazardous gas. Such features may include, but are not limited to, capability to draw air from multiple air intakes at different and separate locations, inner and outer atmospheric boundaries, or the capability to acquire and maintain positive pressure within the Control Room envelope.

BNP Table R-2 and H-2 Bases

A review of station operating procedures identified the following mode dependent in-plant actions and associated safe shutdown areas that are required for normal plant operation, cooldown or shutdown:

Location- Safe Shutdown Area	**Modes- 1, 2**	**Modes- 3, 4, 5**
-17 North RHR Unit-1	RHR Equipment. - *No entry required*	RHR Shut Down Cooling (SDC) ***RHR Pump Discharge Isolation Valves E11-F018 A&C*** Inventory Control Equipment - *No entry required* Reactivity Control. - *No entry required*
-17 North RHR Unit-2	RHR Equipment. - *No entry required*	RHR Shut Down Cooling (SDC) ***RHR Pump Discharge Isolation Valves E11-F018 A&C*** Inventory Control Equipment. - *No entry required* Reactivity Control. - *No entry required*
-17 South RHR Unit-1	RHR Equipment. - *No entry required*	RHR Shut Down Cooling (SDC) ***RHR Pump Discharge Isolation Valves E11-F018 B&D*** Inventory Control Equipment. - *No entry required* Reactivity Control. - *No entry required*
-17 South RHR Unit-2	RHR Equipment. - *No entry required*	RHR Shut Down Cooling (SDC) ***RHR Pump Discharge Isolation Valves E11-F018 B&D*** Inventory Control Equipment. - *No entry required* Reactivity Control. - *No entry required*
-17 North Core Spray	Core Spray Equipment - *No entry required*	Inventory Control - *No entry required*
-17 South Core Spray	Core Spray Equipment - *No entry required*	Inventory Control. - *No entry required*
Service Water Building 20'	Heat Sink equipment. - *No entry required*	Heat Sink equipment. - *No entry required*

Control Building HVAC Room (Turbine Building 70')	Habitability. - *No entry required*	Habitability. - *No entry required*
Reactor Building HVAC (Reactor Building 80' West)	Habitability. - *No entry required*	Habitability. - *No entry required*
Emergency Diesel Generators (EDG Building 20')	Electrical Power, Local control parameters at EDG panel. - No entry required	Electrical Power, Local control parameters at EDG panel. - No entry required
Emergency Diesel Generators 4-Day Tank Rooms	Electrical Power. - *No entry required*	Electrical Power. - *No entry required*
EDG Building HVAC (EDG Building 70')	Habitability. - *No entry required*	Habitability. - *No entry required*
38' & 70' Turbine Building (Unit 1&2)	Electrical Power. - *No entry required*	Electrical Power. - *No entry required*
4160 VAC (EDG building 70')	- *No entry required*	- *No entry required*
480 VAC (EDG Building 20')N & S ends	- *No entry required*	- *No entry required*
120 VAC Vital (Cable Spread U-1 & U-2)	- *No entry required*	- *No entry required*
Train A & B DC (Battery Rooms U-1 & U-2)	- *No entry required*	- *No entry required*
Reactor Building 20' East & West MCC Areas	- *No entry required*	RHR SDC. - ***1 (2) E11 - F009 & F008 valve breakers (RHR SDC Suction isolation valves)*** - ***1 (2) E11 - F006 A-D valve breakers, (RHR pump suction valves)*** - ***RHR SDC Suction Fill & Vent valves (Manual Valves)*** - ***RHR suction pipe flush valve breakers (E11-F011A & B, E11-V33, E11-V32***
Reactor Building 20' Pipe Tunnel	- *No entry required*	RHR SDC - ***RHR SDC Suction Fill & Vent valves (Manual Valves)***

Table R-2 & H-2 Results

Table R-2 & H-2 Safe Operation & Shutdown Areas	
Room/Area	**Mode Applicability**
Reactor Building -17' North RHR Unit-1 & 2	3, 4, 5
Reactor Building -17' South RHR Unit-1 & 2	3, 4, 5
Reactor Building 20' East & West MCC Areas Unit-1 & 2	3, 4, 5
Reactor Building 20' Pipe Tunnel Unit-1 & 2	3, 4, 5

Plant Operating Procedures Reviewed

1. Unit 1 & 2 RHR OP-17
   - Shutdown Cooling
   - Low Pressure Coolant Injection
2. Unit-1 & 2 UAT Backfeed OP-50
3. EDG Operation OP-50.1, OP-39
4. Unit-1 & 2 Service Water OP-43
5. Unit-1 & 2 Core Spray
6. Control Building Ventilation System 2OP-37
7. Defense In Depth AP-22

REVISION SUMMARY

Revision 6 of 0PEP-02.2.1 consists of the following changes:

[xxxx]

Brunswick Steam Electric Plant, Unit Nos. 1 and 2
Renewed Facility Operating License Nos. DPR-71 and DPR-62
Docket Nos. 50-325 and 50-324
Response to Request for Additional Information Regarding Adoption of Emergency Action Level Scheme Pursuant to NEI 99-01, Revision 6, "Development of Emergency Action Levels for Non-Passive Reactors"

Revised BSEP Technical Bases Document, 0PEP-02.2.1, "Emergency Action Level Technical Bases" (Clean Version)

PLANT OPERATING MANUAL

VOLUME XIII

PLANT EMERGENCY PROCEDURE

0PEP-02.2.1

EMERGENCY ACTION LEVEL TECHNICAL BASES

REVISION 6 (RAI Response)

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1.0 PURPOSE

This document provides an explanation and rationale for each Emergency Action Level (EAL) included in the EAL Upgrade Project for Brunswick Nuclear Plant (BNP). It should be used to facilitate review of the BNP EALs and provide historical documentation for future reference. Decision-makers responsible for implementation of 0PEP-02.1 Initial Emergency Actions, may use this document as a technical reference in support of EAL interpretation. This information may assist the Site Emergency Coordinator in making classifications, particularly those involving judgment or multiple events. The basis information may also be useful in training and for explaining event classifications to offsite officials.

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 or less 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 BNP 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) Revisions 4 and 5 were 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 6 of NEI 99-01 has been issued which incorporates resolutions to numerous implementation issues including the NRC EAL Frequently Asked Questions (FAQs). Using NEI 99-01 Revision 6, "Methodology for the Development of Emergency Action Levels for Non-Passive Reactors," November 2012 (ADAMS Accession Number ML12326A805) (ref. 4.1.1), BNP conducted an EAL implementation upgrade project that produced the EALs discussed herein

2.2 Fission Product Barriers

Fission product barrier thresholds represent threats to the defense in depth design concept that precludes the release of 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.

Many of the EALs derived from the NEI methodology are fission product barrier threshold based. That is, the conditions that define the EALs are based upon thresholds that represent the 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. A "Loss" threshold means the barrier no longer assures containment of radioactive materials. A "Potential Loss" threshold implies an increased probability of barrier loss and decreased certainty of maintaining the barrier.

The primary fission product barriers are:

A. Fuel Clad (FC): The Fuel Clad Barrier consists of the cladding material that contains the fuel pellets.

B. Reactor Coolant System (RCS): The RCS Barrier is the reactor coolant system pressure boundary and includes the RPV and all reactor coolant system piping up to and including the isolation valves.

C. Primary Containment (PC): The Primary Containment Barrier includes the drywell, the wetwell, their respective interconnecting paths, and other connections up to and including the outermost containment isolation valves. Primary Containment Barrier thresholds are used as criteria for escalation of the ECL from Alert to a Site Area Emergency or a General Emergency.

2.3 Fission Product Barrier Classification Criteria

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

Alert:

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

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

2.4 EAL Organization

The BNP 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, Refueling 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 group, assignment of EALs to categories and subcategories:

Category and subcategory titles are selected to represent conditions that are operationally significant to the EAL-user. The BNP EAL categories are aligned to and represent the NEI 99-01"Recognition Categories." Subcategories are used in the BNP scheme as necessary to further divide the EALs of a category into logical sets of possible emergency classification thresholds. The BNP EAL categories and subcategories are listed below.

EAL Groups, Categories and Subcategories

EAL Group/Category	EAL Subcategory
Any Operating Mode:	
R – Abnormal **R**ad Levels / Rad Effluent	1 – Radiological Effluent 2 – Irradiated Fuel Event 3 – Area Radiation Levels
H – **H**azards and Other Conditions Affecting Plant Safety	1 – Security 2 – Seismic Event 3 – Natural or Technological Hazard 4 – Fire 5 – Hazardous Gas 6 – Control Room Evacuation 7 – SEC Judgment
E – Independent Spent Fuel Storage Installation (ISFSI)	1 – Confinement Boundary
Hot Conditions:	
S – **S**ystem Malfunction	1 – Loss of Emergency AC Power 2 – Loss of Vital DC Power 3 – Loss of Control Room Indications 4 – RCS Activity 5 – RCS Leakage 6 – RPS Failure 7 – Loss of Communications 8 – Hazardous Event Affecting Safety Systems
F – **F**ission Product Barrier Degradation	None
Cold Conditions:	
C – **C**old Shutdown / Refueling System Malfunction	1 – RPV Level 2 – Loss of Emergency AC Power 3 – RCS Temperature 4 – Loss of Vital DC Power 5 – Loss of Communications 6 – Hazardous Event Affecting Safety Systems

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 Section 3.0 and Attachments 1 & 2 of this document for such information.

2.5 Technical Bases Information

EAL technical bases are provided in Attachment 1 for each EAL according to EAL group (Any, Hot, Cold), EAL category (R, C, H, S, F and E) 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:

Category Letter & Title

Subcategory Number & Title

Initiating Condition (IC)

Site-specific description of the generic IC given in NEI 99-01 Rev. 6.

EAL Identifier (enclosed in rectangle)

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, C, H, S, F or E)
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)

Exact wording of the EAL as it appears in the EAL Classification Matrix

Mode Applicability

One or more of the following plant operating conditions comprise the mode to which each EAL is applicable: 1 - Power Operations, 2 - Startup, 3 - Hot Shutdown, 4 - Cold Shutdown, 5 - Refueling, D - Defueled, or All. (See Section 2.6 for operating mode definitions)

Definitions:

If the EAL wording contains a defined term, the definition of the term is included in this section. These definitions can also be found in Section 5.1.

Basis:

A basis section that provides BNP-relevant information concerning the EALas well as a description of the rationale for the EAL as provided in NEI 99-01 Rev. 6.

BNP Basis Reference(s):

Site-specific source documentation from which the EAL is derived

2.6 Operating Mode Applicability (ref. 4.1.7)

1 Power Operations

Reactor is critical and the mode switch is in RUN

2 Startup

The mode switch is in STARTUP/HOT STANDBY

3 Hot Shutdown

The mode switch is in SHUTDOWN, all reactor vessel head closure bolts are fully tensioned, and reactor coolant temperature is >212ºF

4 Cold Shutdown

The mode switch is in SHUTDOWN, all reactor vessel head closure bolts are fully tensioned, and reactor coolant temperature is ≤ 212ºF

5 Refuel

The mode switch is in REFUEL or SHUTDOWN and one or more reactor vessel head closure bolts are less than fully tensioned

D Defueled

RPV contains no irradiated fuel

The plant operating mode that exists at the time that the event occurs (prior to any protective system or operator action being 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.

3.0 GUIDANCE ON MAKING EMERGENCY CLASSIFICATIONS

3.1 General Considerations

When making an emergency classification, the Site Emergency Coordinator must consider all information having a bearing on the proper assessment of an Initiating Condition (IC). This includes the Emergency Action Level (EAL) plus the associated Operating Mode Applicability, Notes, and the informing basis information. In the Recognition Category F matrices, EALs are based on loss or potential loss of Fission Product Barrier Thresholds.

3.1.1 Classification Timeliness

NRC regulations require the licensee to establish and maintain the capability to assess, classify, and declare an emergency condition within 15 minutes after the availability of indications to plant operators that an emergency action level has been exceeded and to promptly declare the emergency condition as soon as possible following identification of the appropriate emergency classification level. The NRC staff has provided guidance on implementing this requirement in NSIR/DPR-ISG-01, "Interim Staff Guidance, Emergency Planning for Nuclear Power Plants" (ref. 4.1.12).

3.1.2 Valid Indications

All emergency classification assessments shall be based upon valid indications, reports or conditions. A valid indication, report, or condition, is one that has been verified through appropriate means such that there is no doubt regarding the indicator's operability, the condition's existence, or the report's accuracy. For example, verification could be accomplished through an instrument channel check, response on related or redundant indicators, or direct observation by plant personnel.

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.

3.1.3 Imminent Conditions

For ICs and EALs that have a stipulated time duration (e.g., 15 minutes, 30 minutes, etc.), the Site Emergency Coordinator 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. If an ongoing radiological release is detected and the release start time is unknown, it should be assumed that the release duration specified in the IC/EAL has been exceeded, absent data to the contrary.

3.1.4 Planned vs. Unplanned Events

A planned work activity that results in an expected event or condition which meets or exceeds an EAL does not warrant an emergency declaration provided that: 1) the activity proceeds as planned, and 2) the plant remains within the limits imposed by the operating license. Such activities include planned work to test, manipulate, repair, maintain or modify a system or component. In these cases, the controls associated with the planning, preparation and execution of the work will ensure that compliance is maintained with all aspects of the operating license provided that the activity proceeds and concludes as expected. Events or conditions of this type may be subject to the reporting requirements of 10 § CFR 50.72 (ref. 4.1.4).

3.1.5 Classification Based on Analysis

The assessment of some EALs is based on the results of analyses that are necessary to ascertain whether a specific EAL threshold has been exceeded (e.g., dose assessments, chemistry sampling, RCS leak rate calculation, etc.). For these EALs, the EAL wording or the associated basis discussion will identify the necessary analysis. In these cases, the 15-minute declaration period starts with the availability of the analysis results that show the threshold to be exceeded (i.e., this is the time that the EAL information is first available). The NRC expects licensees to establish the capability to initiate and complete EAL-related analyses within a reasonable period of time (e.g., maintain the necessary expertise on-shift).

3.1.6 Site Emergency Coordinator Judgment

While the EALs have been developed to address a full spectrum of possible events and conditions which may warrant emergency classification, a provision for classification based on operator/management experience and judgment is still necessary. The NEI 99-01 EAL scheme provides the Site Emergency Coordinator with the ability to classify events and conditions based upon judgment using EALs that are consistent with the Emergency Classification Level (ECL) definitions (refer to Category H). The Site Emergency Coordinator will need to determine if the effects or consequences of the event or condition reasonably meet or exceed a particular ECL definition. A similar provision is incorporated in the Fission Product Barrier Tables; judgment may be used to determine the status of a fission product barrier.

3.2 Classification Methodology

To make an emergency classification, the user will compare an event or condition (i.e., the relevant plant indications and reports) to an EAL(s) and determine if the EAL has been met or exceeded. The evaluation of an EAL must be consistent with the related Operating Mode Applicability and Notes. If an EAL has been met or exceeded, the associated IC is likewise met, the emergency classification process "clock" starts, and the ECL must be declared in accordance with plant procedures no later than fifteen minutes after the process "clock" started.

When assessing an EAL that specifies a time duration for the off-normal condition, the "clock" for the EAL time duration runs concurrently with the emergency classification process "clock." For a full discussion of this timing requirement, refer to NSIR/DPR-ISG-01 (ref. 4.1.14).

3.2.1 Classification of Multiple Events and Conditions

When multiple emergency events or conditions are present, the user will identify all met or exceeded EALs. The highest applicable ECL identified during this review is declared. For example:

- If an Alert EAL and a Site Area Emergency EAL are met, whether at one unit or at two different units, a Site Area Emergency should be declared.

There is no "additive" effect from multiple EALs meeting the same ECL. For example:

- If two Alert EALs are met, whether at one unit or at two different units, an Alert should be declared.

Related guidance concerning classification of rapidly escalating events or conditions is provided in Regulatory Issue Summary (RIS) 2007-02, *Clarification of NRC Guidance for Emergency Notifications During Quickly Changing Events* (ref. 4.1.2).

3.2.2 Consideration of Mode Changes During Classification

The mode in effect at the time that an event or condition occurred, and prior to any plant or operator response, is the mode that determines whether or not an IC is applicable. If an event or condition occurs, and results in a mode change before the emergency is declared, the emergency classification level is still based on the mode that existed at the time that the event or condition was initiated (and not when it was declared). Once a different mode is reached, any new event or condition, not related to the original event or condition, requiring emergency classification should be evaluated against the ICs and EALs applicable to the operating mode at the time of the new event or condition.

For events that occur in Cold Shutdown or Refueling, escalation is via EALs that are applicable in the Cold Shutdown or Refueling modes, even if Hot Shutdown (or a higher mode) is entered during the subsequent plant response. In particular, the fission product barrier EALs are applicable only to events that initiate in the Hot Shutdown mode or higher.

3.2.3 Classification of Imminent Conditions

Although EALs provide specific thresholds, the Site Emergency Coordinator must remain alert to events or conditions that could lead to meeting or exceeding an EAL within a relatively short period of time (i.e., a change in the ECL is IMMINENT). If, in the judgment of the Site Emergency Coordinator, meeting an EAL is IMMINENT, the emergency classification should be made as if the EAL has been met. While applicable to all emergency classification levels, this approach is particularly important at the higher emergency classification levels since it provides additional time for implementation of protective measures.

3.2.4 Emergency Classification Level Upgrading and Downgrading

An ECL may be downgraded when the event or condition that meets the highest IC and EAL no longer exists, and other site-specific downgrading requirements are met. If downgrading the ECL is deemed appropriate, the new ECL would then be based on a lower applicable IC(s) and EAL(s). The ECL may also simply be terminated.

As noted above, guidance concerning classification of rapidly escalating events or conditions is provided in RIS 2007-02 (ref. 4.1.2).

3.2.5 Classification of Short-Lived Events

Event-based ICs and EALs define a variety of specific occurrences that have potential or actual safety significance. By their nature, some of these events may be short-lived and, thus, over before the emergency classification assessment can be completed. If an event occurs that meets or exceeds an EAL, the associated ECL must be declared regardless of its continued presence at the time of declaration. Examples of such events include an earthquake or a failure of the reactor protection system to automatically scram the reactor followed by a successful manual scram.

3.2.6 Classification of Transient Conditions

Many of the ICs and/or EALs employ time-based criteria. These criteria will require that the IC/EAL conditions be present for a defined period of time before an emergency declaration is warranted. In cases where no time-based criterion is specified, it is recognized that some transient conditions may cause an EAL to be met for a brief period of time (e.g., a few seconds to a few minutes). The following guidance should be applied to the classification of these conditions.

<u>EAL momentarily met during expected plant response</u> - In instances where an EAL is briefly met during an expected (normal) plant response, an emergency declaration is not warranted provided that associated systems and components are operating as expected, and operator actions are performed in accordance with procedures.

<u>EAL momentarily met but the condition is corrected prior to an emergency declaration</u> – If an operator takes prompt manual action to address a condition, and the action is successful in correcting the condition prior to the emergency declaration, then the applicable EAL is not considered met and the associated emergency declaration is not required. For illustrative purposes, consider the following example:

> An ATWS occurs and the high pressure ECCS systems fail to automatically start. RPV level rapidly decreases and the plant enters an inadequate core cooling condition (a potential loss of both the fuel clad and RCS barriers). If an operator manually starts a high pressure ECCS system in accordance with an EOP step and clears the inadequate core cooling condition prior to an emergency declaration, then the classification should be based on the ATWS only.

It is important to stress that the 15-minute emergency classification assessment period (process clock) is not a "grace period" during which a classification may be delayed to allow the performance of a corrective action that would obviate the need to classify the event. Emergency classification assessments must be deliberate and timely, with no undue delays. The provision discussed above addresses only those rapidly evolving situations when an operator is able to take a successful corrective action prior to the Site Emergency Coordinator completing the review and steps necessary to make the emergency declaration. This provision is included to ensure that any public protective actions resulting from the emergency classification are truly warranted by the plant conditions.

3.2.7 After-the-Fact Discovery of an Emergency Event or Condition

In some cases, an EAL may be met but the emergency classification was not made at the time of the event or condition. This situation can occur when personnel discover that an event or condition existed which met an EAL, but no emergency was declared, and the event or condition no longer exists at the time of discovery. This may be due to the event or condition not being recognized at the time or an error that was made in the emergency classification process.

In these cases, no emergency declaration is warranted; however, the guidance contained in NUREG-1022 (ref. 4.1.3) is applicable. Specifically, the event should be reported to the NRC in accordance with 10 CFR § 50.72 (ref. 4.1.4) within one hour of the discovery of the undeclared event or condition. The licensee should also notify appropriate State and local agencies in accordance with the agreed upon arrangements.

3.2.8 Retraction of an Emergency Declaration

Guidance on the retraction of an emergency declaration reported to the NRC is discussed in NUREG-1022 (ref. 4.1.3).

4.0 REFERENCES

4.1 Developmental

4.1.1 NEI 99-01 Revision 6, Methodology for the Development of Emergency Action Levels for Non-Passive Reactors, ADAMS Accession Number ML12326A805

4.1.2 RIS 2007-02 Clarification of NRC Guidance for Emergency Notifications During Quickly Changing Events, February 2, 2007.

4.1.3 NUREG-1022 Event Reporting Guidelines: 10CFR50.72 and 50.73

4.1.4 10 § CFR 50.72 Immediate Notification Requirements for Operating Nuclear Power Reactors

4.1.5 10 § CFR 50.73 License Event Report System

4.1.6 BNP Radiological Emergency Response Plan Figure 1-1.1 Brunswick Site Plan

4.1.7 BNP Radiological Emergency Response Plan Figure 1-1.3 Brunswick Site Building and Onsite Emergency Facility Locations

4.1.8 Technical Specifications Table 1.1-1 Modes

4.1.9 Technical Specifications Section 3.6 Containment Systems

4.1.10 PRO-NGGC-0201 NGG Procedure Writers Guide

4.1.11 NSIR/DPR-ISG-01 Interim Staff Guidance, Emergency Planning for Nuclear Power Plants

4.1.12 NGGM-PM-0028 Transnuclear NUHOMS Dry Fuel Storage Program Manual

4.2 Implementing

4.2.1 0PEP-02.1 Initial Emergency Actions

4.2.2 NEI 99-01 Rev. 6 to BNP EAL Comparison Matrix

4.2.3 BNP EAL Matrix

5.0 DEFINITIONS, ACRONYMS & ABBREVIATIONS

5.1 Definitions (ref. 4.1.1 except as noted)

Selected terms used in Initiating Condition and Emergency Action Level statements are set in all capital letters (e.g., ALL CAPS). These words are defined terms that have specific meanings as used in this document. The definitions of these terms are provided below.

Alert

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 PAG exposure levels.

Can/Cannot Be Maintained Above/Below

The value of an identified parameter is/is not able to be held within the specified limit. The determination requires an evaluation of system performance and availability in relation to parameter values and trends. An instruction prescribing action when a parameter cannot be maintained above or below a specified limit neither requires nor prohibits anticipatory action-depending upon plant conditions, the action may be taken as soon as it is determined that the limit will ultimately be exceeded, or delayed until the limit is actually reached. Once the parameter does exceed the limit, however, the action must be performed; it may not be delayed while attempts are made to restore the parameter to within the desired control band.

Can/Cannot Be Restored Above/Below

The value of an identified parameter is/is not able to be brought within the specified limit. The determination requires an evaluation of system performance and availability in relation to parameter values and trends. An instruction prescribing action when a value cannot be restored and maintained above or below a specified limit does not require immediate action simply because the current values is outside the range, but does not permit extended operation beyond the limit; the action must be taken as soon as it is apparent that the specified range cannot be attained.

Confinement Boundary

The barrier(s) between spent fuel and the environment once the spent fuel is processed for dry storage. As related to the BNP ISFSI, Confinement Boundary is defined as the Dry Shielded Canister (DSC) (Ref. 4.1.12).

Containment Closure

The procedurally defined conditions or actions taken to secure Primary or Secondary Containment and its associated structures, systems, and components as a functional barrier to fission product release under shutdown conditions.

As applied to BNP, Containment Closure is established when either Primary Containment is Operable per Section 3.6.1.1 of Technical Specifications or Secondary Containment is considered functional per the requirements of 0AP-022, BNP Outage Risk Management.

Emergency Action Level (EAL)

A pre-determined, site-specific, observable threshold for an Initiating Condition that, when met or exceeded, places the plant in a given emergency classification level.

Emergency Classification Level (ECL)

One of a set of names or titles established by the US Nuclear Regulatory Commission (NRC) for grouping off-normal events or conditions according to (1) potential or actual effects or consequences, and (2) resulting onsite and offsite response actions. The emergency classification levels, in ascending order of severity, are:

- Unusual Event (UE)
- Alert
- Site Area Emergency (SAE)
- General Emergency (GE)

EPA PAGs

Environment Protection Agency Protective Action Guidelines. The EPA PAGs are expressed in terms of dose commitment: 1 Rem TEDE or 5 Rem CDE Thyroid. Actual or projected offsite exposures in excess of the EPA PAGs requires BNP to recommend protective actions for the general public to offsite planning agencies.

Explosion

A rapid, violent and catastrophic failure of a piece of equipment due to combustion, chemical reaction or overpressurization. A release of steam (from high energy lines or components) or an electrical component failure (caused by short circuits, grounding, arcing, etc.) should not automatically be considered an explosion. Such events require a post-event inspection to determine if the attributes of an explosion are present.

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.

Fission Product Barrier Threshold

A pre-determined, site-specific, observable threshold indicating the loss or potential loss of a fission product barrier.

Flooding

A condition where water is entering a room or area faster than installed equipment is capable of removal, resulting in a rise of water level within the room or area.

General Emergency

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 PAG exposure levels offsite for more than the immediate site area.

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 BNP 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 BNP. Non-terrorism-based EALs should be used to address such activities (i.e., this may include 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

The trajectory of events or conditions is such that an EAL will be met within a relatively short period of time regardless of mitigation or corrective actions.

Impede(d)

Personnel access to a room or area is hindered to an extent that extraordinary measures are necessary to facilitate entry of personnel into the affected room/area (e.g., requiring use of protective equipment, such as SCBAs, that is not routinely employed).

Initiating Condition (IC)

An event or condition that aligns with the definition of one of the four emergency classification levels by virtue of the potential or actual effects or consequences.

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.

Maintain

Take appropriate action to hold the value of an identified parameter within specified limits.

Normal Levels

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

Owner Controlled Area

Area depicted as the property boundary in BNP Radiological Emergency Response Plan Figure 1-1.1 Brunswick Site Plan (ref. 4.1.6).

Projectile

An object directed toward a Nuclear Power Plant that could cause concern for its continued operability, reliability, or personnel safety.

Protected Area

The double-fenced security area with intrusion detection devices immediately surrounding the plant structures. The Protected Area is depicted in BNP Radiological Emergency Response Plan Figure 1-1.3 Brunswick Site Building and Onsite Emergency Facility Locations (ref. 4.1.7).

RCS 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).

Refueling Pathway

The reactor refueling cavity, spent fuel pool and fuel transfer canal (cattle chute) comprise the refueling pathway.

Restore

Take the appropriate action required to return the value of an identified parameter to the applicable limits

Safety System

A system required for safe plant operation, cooling down the plant and/or placing it in the cold shutdown condition, including the ECCS. These are typically systems classified as safety-related (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 Area Emergency

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 PAG exposure levels beyond the SITE BOUNDARY.

Site Boundary

Area as depicted in BNP Radiological Emergency Response Plan Figure 1-1.1 Brunswick Site Plan (ref. 4.1.6).

Unisolable

An open or breached system line that cannot be isolated, remotely or locally.

Unplanned

A parameter change or an event that is not 1) the result of an intended evolution or 2) an expected plant response to a transient. The cause of the parameter change or event may be known or unknown.

Unusual Event

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.

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 a component or structure that is readily observable without measurements, testing, or analysis. The visual impact of the damage is sufficient to cause concern regarding the operability or reliability of the affected component or structure.

5.2 Abbreviations/Acronyms

°F .... Degrees Fahrenheit
° .... Degrees
AC .... Alternating Current
AOP .... Abnormal Operating Procedure
APRM .... Average Power Range Meter
ATWS .... Anticipated Transient Without Scram
BNP .... Brunswick Nuclear Plant
BWR .... Boiling Water Reactor
BWROG .... Boiling Water Reactor Owners Group
CDE .... Committed Dose Equivalent
CFR .... Code of Federal Regulations
CS .... Core Spray
DBA .... Design Basis Accident
DC .... Direct Current
EAL .... Emergency Action Level
ECCS .... Emergency Core Cooling System
ECL .... Emergency Classification Level
EOF .... Emergency Operations Facility
EOP .... Emergency Operating Procedure
EPA .... Environmental Protection Agency
EPG .... Emergency Procedure Guideline
EPIP .... Emergency Plan Implementing Procedure
ESF .... Engineered Safety Feature
FAA .... Federal Aviation Administration
FBI .... Federal Bureau of Investigation
FEMA .... Federal Emergency Management Agency
FSAR .... Final Safety Analysis Report
GE .... General Emergency
HCTL .... Heat Capacity Temperature Limit
HPCI .... High Pressure Coolant Injection
IC .... Initiating Condition
IPEEE .... Individual Plant Examination of External Events (Generic Letter 88-20)
ISFSI .... Independent Spent Fuel Storage Installation

K_{eff} .... Effective Neutron Multiplication Factor
LCO .... Limiting Condition of Operation
LER .... Licensee Event Report
LOCA .... Loss of Coolant Accident
LPSI .... Low Pressure Safety Injection
LWR .... Light Water Reactor
MPC .... Maximum Permissible Concentration/Multi-Purpose Canister
MPH .... Miles Per Hour
MSIV .... Main Steam Isolation Valve
MSL .... Main Steam Line
mR, mRem, mrem, mREM .... milli-Roentgen Equivalent Man
MW .... Megawatt
NEI .... Nuclear Energy Institute
NESP .... National Environmental Studies Project
NPP .... Nuclear Power Plant
NRC .... Nuclear Regulatory Commission
NSSS .... Nuclear Steam Supply System
NORAD .... North American Aerospace Defense Command
(NO)UE .... Notification of Unusual Event
OBE .... Operating Basis Earthquake
OCA .... Owner Controlled Area
ODCM/ODAM .... Offsite Dose Calculation (Assessment) Manual
ORO .... Offsite Response Organization
PA .... Protected Area
PRA/PSA .... Probabilistic Risk Assessment / Probabilistic Safety Assessment
PWR .... Pressurized Water Reactor
PSIG .... Pounds per Square Inch Gauge
R .... Roentgen
RB .... Reactor Building
RCIC .... Reactor Core Isolation Cooling
RCS .... Reactor Coolant System
Rem, rem, REM .... Roentgen Equivalent Man
RETS .... Radiological Effluent Technical Specifications
RPS .... Reactor Protection System

RPV .... Reactor Pressure Vessel
RWCU ....Reactor Water Cleanup
SAR .... Safety Analysis Report
SBGTS .... Stand-By Gas Treatment System
SBO .... Station Blackout
SCBA .... Self-Contained Breathing Apparatus
SEC .... Site Emergency Coordinator
SPDS .... Safety Parameter Display System
SRO .... Senior Reactor Operator
TEDE .... Total Effective Dose Equivalent
TAF .... Top of Active Fuel
TSC .... Technical Support Center

6.0 BNP-TO-NEI 99-01 Rev. 6 EAL CROSS-REFERENCE

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

BNP	NEI 99-01 Rev. 6	
EAL	**IC**	**Example EAL**
RU1.1	AU1	1, 2
RU1.2	AU1	3
RU2.1	AU2	1
RA1.1	AA1	1
RA1.2	AA1	2
RA1.3	AA1	3
RA1.4	AA1	4
RA2.1	AA2	1
RA2.2	AA2	2
RA2.3	AA2	3
RA3.1	AA3	1
RA3.2	AA3	2
RS1.1	AS1	1
RS1.2	AS1	2
RS1.3	AS1	3
RS2.1	AS2	1
RG1.1	AG1	1
RG1.2	AG1	2
RG1.3	AG1	3
RG2.1	AG2	1

BNP	NEI 99-01 Rev. 6	
EAL	**IC**	**Example EAL**
CU1.1	CU1	1
CU1.2	CU1	2
CU2.1	CU2	1
CU3.1	CU3	1
CU3.2	CU3	2
CU4.1	CU4	1
CU5.1	CU5	1, 2, 3
CA1.1	CA1	1
CA1.2	CA1	2
CA2.1	CA2	1
CA3.1	CA3	1, 2
CA6.1	CA6	1
CS1.1	CS1	1
CS1.2	CS1	2
CS1.3	CS1	3
CG1.1	CG1	1
CG1.2	CG1	2
FA1.1	FA1	1
FS1.1	FS1	1
FG1.1	FG1	1
HU1.1	HU1	1
HU1.2	HU1	2
HU1.3	HU1	3
HU2.1	HU2	1
HU3.1	HU3	1

BNP	NEI 99-01 Rev. 6	
EAL	IC	Example EAL
HU3.2	HU3	2
HU3.3	HU3	3
HU3.4	HU3	4
HU3.5	HU3	5
HU4.1	HU4	1
HU4.2	HU4	2
HU4.3	HU4	3
HU4.4	HU4	4
HU7.1	HU7	1
HA1.1	HA1	1
HA1.2	HA1	2
HA5.1	HA5	1
HA6.1	HA6	1
HA7.1	HA7	1
HS1.1	HS1	1
HS6.1	HS6	1
HS7.1	HS7	1
HG1.1	HG1	1
HG7.1	HG7	1
SU1.1	SU1	1
SU3.1	SU2	1
SU4.1	SU3	1
SU4.2	SU3	2
SU5.1	SU4	1, 2, 3
SU6.1	SU5	1

BNP	NEI 99-01 Rev. 6	
EAL	IC	Example EAL
SU6.2	SU5	2
SU7.1	SU6	1, 2, 3
SA1.1	SA1	1
SA3.1	SA2	1
SA6.1	SA5	1
SA8.1	SA9	1
SS1.1	SS1	1
SS2.1	SS8	1
SS6.1	SS5	1
SG1.1	SG1	1
SG1.2	SG8	1
EU1.1	E-HU1	1

7.0 ATTACHMENTS

7.1 Attachment 1, Emergency Action Level Technical Bases

7.2 Attachment 2, Fission Product Barrier 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. Radiological Effluent

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. Irradiated Fuel Event

Conditions indicative of a loss of adequate shielding or damage to irradiated fuel may preclude access to vital plant areas or result in radiological releases that warrant emergency classification.

3. Area Radiation Levels

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

Category: R – Abnormal Rad Levels / Rad Effluent

Subcategory: 1 – Radiological Effluent

Initiating Condition: Release of gaseous or liquid radioactivity > 2 times the ODCM limits for 60 minutes or longer

EAL:

RU1.1 Unusual Event

Reading on **any** Table R-1 effluent radiation monitor > column "UE" for ≥ 60 min. (Notes 1, 2, 3)

Note 1: The SEC should declare the event promptly upon determining that time limit has been exceeded, or will likely be exceeded.

Note 2: If an ongoing release is detected and the release start time is unknown, assume that the release duration has exceeded the specified time limit.

Note 3: If the effluent flow past an effluent monitor is known to have stopped, indicating that the release path is isolated, the effluent monitor reading is no longer VALID for classification purposes.

Table R-1 Effluent Monitor Classification Thresholds

	Release Point	Monitor	GE	SAE	Alert	UE
Gaseous	Main Stack Rad	D12-RM-23S	2.13E+09 μCi/sec	2.13E+08 μCi/sec	2.13E+07 μCi/sec	1.80E+06 μCi/sec
Gaseous	Reactor Bldg Vent Noble Gas	CAC-AQH-1264-3	----	----	----	6.14E+04 cpm
Gaseous	Turbine Bldg Vent	D12-RM-23	1.07E+08 μCi/sec	1.07E+07 μCi/sec	1.07E+06 μCi/sec	1.13E+04 μCi/sec
Liquid	Service Water Effluent Rad	D12-RM-K605	----	----	----	2 x hi alarm
Liquid	Radwaste Effluent Rad	D12-RM-K604	----	----	----	2 x hi-hi alarm

Mode Applicability:

All

Definition(s):

None

Basis:

Gaseous Releases

The column "UE" gaseous release values in Table R-1 represent two times the appropriate ODCM release rate limits associated with the specified monitors (ref. 2, 3).

Instrumentation that may be used to assess this EAL is listed below (ref. 1):

- Main Stack Monitoring System Noble Gas Activity Monitor 2-D12-RM-23S (1/2-D12-RR-4599-4)
- Reactor Building Ventilation Monitoring System Noble Gas Activity Monitor 1(2)-CAC-AQH-1264-3
- Turbine Building Ventilation Monitoring System Noble Gas Activity Monitor 1(2)-D12-RM-23 (1/2-DL12-RR-4548-4)

Liquid Releases

Instrumentation that may be used to assess this EAL is listed below:

- Liquid Radwaste Radioactivity Monitor 2-D12-RM-K604 (batch release)
- Main Service Water Effluent Radioactivity Monitor 1(2)-D12-RM-K605 (continuous release)

The Liquid Radwaste Radioactivity Monitor Hi-Hi alarm automatically closes Radwaste Liquid Effluent Discharge Valves D12-V27A and 27B. The Hi-Hi alarm setpoint is set in accordance with the ODCM and includes a conservative reduction factor of 20 to the ODCM release rate limit (ref. 1, 2).

The Main Service Water Effluent Radioactivity Monitor High alarm setpoint is set in accordance with the ODCM and ensures continuous liquid releases do not exceed ODCM Section 7.3.3 limits.

This IC addresses a potential decrease in the level of safety of the plant as indicated by a low-level radiological release that exceeds regulatory commitments for an extended period of time (e.g., an uncontrolled release). It includes any gaseous or liquid radiological release, monitored or un-monitored, including those for which a radioactivity discharge permit is normally prepared.

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

Radiological effluent EALs are also included to provide a basis for classifying events and conditions that cannot be readily or appropriately classified on the basis of plant conditions alone. The inclusion of both plant condition and radiological effluent EALs more fully addresses the spectrum of possible accident events and conditions.

Classification based on effluent monitor readings assumes that a release path to the environment is established. If the effluent flow past an effluent monitor is known to have stopped due to actions to isolate the release path, then the effluent monitor reading is no longer valid for classification purposes.

Releases should not be prorated or averaged. For example, a release exceeding 4 times release limits for 30 minutes does not meet the EAL.

This EAL addresses normally occurring continuous radioactivity releases from monitored gaseous or liquid effluent pathways.

Escalation of the emergency classification level would be via IC RA1.

BNP Basis Reference(s):

1. BNP ODCM Appendix E Radioactive Liquid and Gaseous Effluent Monitoring Instrumentation Numbers
2. BNP Offsite Dose Calculation Manual
3. EP-EALCALC-BNP-0801 Radiological Gaseous Effluent Values (EALs RG1, RS1, RA1 and RU1)
4. NEI 99-01 AU1

Category: R – Abnormal Rad Levels / Rad Effluent

Subcategory: 1 – Radiological Effluent

Initiating Condition: Release of gaseous or liquid radioactivity greater than 2 times the ODCM limits for 60 minutes or longer.

EAL:

RU1.2	**Unusual Event**
Sample analysis for a gaseous or liquid release indicates a concentration or release rate > 2 x ODCM limits for ≥ 60 min. (Notes 1, 2)	

Note 1: The SEC should declare the event promptly upon determining that time limit has been exceeded, or will likely be exceeded.

Note 2: If an ongoing release is detected and the release start time is unknown, assume that the release duration has exceeded the specified time limit.

Mode Applicability:

All

Definition(s):

None

Basis:

This IC addresses a potential decrease in the level of safety of the plant as indicated by a low-level radiological release that exceeds regulatory commitments for an extended period of time (e.g., an uncontrolled release). It includes any gaseous or liquid radiological release, monitored or un-monitored, including those for which a radioactivity discharge permit is normally prepared.

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

Radiological effluent EALs are also included to provide a basis for classifying events and conditions that cannot be readily or appropriately classified on the basis of plant conditions alone. The inclusion of both plant condition and radiological effluent EALs more fully addresses the spectrum of possible accident events and conditions.

Releases should not be prorated or averaged. For example, a release exceeding 4 times release limits for 30 minutes does not meet the EAL.

This EAL addresses uncontrolled gaseous or liquid releases that are detected by sample analyses or environmental surveys, particularly on unmonitored pathways (e.g., spills of radioactive liquids into storm drains, heat exchanger leakage in river water systems, etc.).

Escalation of the emergency classification level would be via IC RA1.

BNP Basis Reference(s):

1. BNP ODCM Appendix E Radioactive Liquid and Gaseous Effluent Monitoring Instrumentation Numbers
2. BNP Offsite Dose Calculation Manual
3. NEI 99-01 AU1

Category: R – Abnormal Rad Levels / Rad Effluent

Subcategory: 1 – Radiological Effluent

Initiating Condition: Release of gaseous or liquid radioactivity resulting in offsite dose greater than 10 mrem TEDE or 50 mrem thyroid CDE

EAL:

RA1.1 Alert

In the absence of real-time dose assessment, reading on **any** Table R-1 effluent radiation monitor > column "ALERT" for ≥ 15 min. (Notes 1, 2, 3, 4)

Note 1: The SEC should declare the event promptly upon determining that time limit has been exceeded, or will likely be exceeded.

Note 2: If an ongoing release is detected and the release start time is unknown, assume that the release duration has exceeded the specified time limit.

Note 3: If the effluent flow past an effluent monitor is known to have stopped, indicating that the release path is isolated, the effluent monitor reading is no longer VALID for classification purposes.

Note 4 The pre-calculated effluent monitor values presented in EALs RA1.1, RS1.1 and RG1.1 should be used for emergency classification assessments until the results from a dose assessment using actual meteorology are available.

Table R-1 Effluent Monitor Classification Thresholds

	Release Point	Monitor	GE	SAE	Alert	UE
Gaseous	Main Stack Rad	D12-RM-23S	2.13E+09 μCi/sec	2.13E+08 μCi/sec	2.13E+07 μCi/sec	1.80E+06 μCi/sec
Gaseous	Reactor Bldg Vent Noble Gas	CAC-AQH-1264-3	----	----	----	6.14E+04 cpm
Gaseous	Turbine Bldg Vent	D12-RM-23	1.07E+08 μCi/sec	1.07E+07 μCi/sec	1.07E+06 μCi/sec	1.13E+04 μCi/sec
Liquid	Service Water Effluent Rad	D12-RM-K605	----	----	----	2 x hi alarm
Liquid	Radwaste Effluent Rad	D12-RM-K604	----	----	----	2 x hi-hi alarm

Mode Applicability:

All

Definition(s):

None

Basis:

This EAL address gaseous radioactivity releases, that for whatever reason, cause effluent radiation monitor readings corresponding to site boundary doses that exceed either (ref. 2, 3):

- 10 mRem TEDE
- 50 mRem CDE Thyroid

The column "ALERT" gaseous effluent release values in Table R-1 correspond to calculated doses of 1% (10% of the SAE thresholds) of the EPA Protective Action Guidelines (TEDE or CDE Thyroid).

Instrumentation that may be used to assess this EAL is listed below (ref 1):

- Main Stack Monitoring System Noble Gas Activity Monitor 2-D12-RM-23S (1/2-D12-RR-4599-4)
- Turbine Building Ventilation Monitoring System Noble Gas Activity Monitor 1(2)-D12-RM-23 (1/2-DL12-RR-4548-4)

This IC addresses a release of gaseous or liquid radioactivity that results in projected or actual offsite doses greater than or equal to 1% of the EPA Protective Action Guides (PAGs). It includes both monitored and un-monitored releases. Releases of this magnitude represent an actual or potential substantial degradation of the level of safety of the plant as indicated by a radiological release that significantly exceeds regulatory limits (e.g., a significant uncontrolled release).

Radiological effluent EALs are also included to provide a basis for classifying events and conditions that cannot be readily or appropriately classified on the basis of plant conditions alone. The inclusion of both plant condition and radiological effluent EALs more fully addresses the spectrum of possible accident events and conditions.

The TEDE dose is set at 1% of the EPA PAG of 1,000 mrem while the 50 mrem thyroid CDE was established in consideration of the 1:5 ratio of the EPA PAG for TEDE and thyroid CDE.

Classification based on effluent monitor readings assumes that a release path to the environment is established. If the effluent flow past an effluent monitor is known to have stopped due to actions to isolate the release path, then the effluent monitor reading is no longer valid for classification purposes.

Escalation of the emergency classification level would be via IC RS1.

BNP Basis Reference(s):

1. BNP ODCM Appendix E Radioactive Liquid and Gaseous Effluent Monitoring Instrumentation Numbers
2. BNP Offsite Dose Calculation Manual
3. EP-EALCALC-BNP-0801 Radiological Gaseous Effluent Values (EALs RG1, RS1, RA1 and RU1)
4. NEI 99-01 AA1

Category: R – Abnormal Rad Levels / Rad Effluent

Subcategory: 1 – Radiological Effluent

Initiating Condition: Release of gaseous or liquid radioactivity resulting in offsite dose greater than 10 mrem TEDE or 50 mrem thyroid CDE

EAL:

RA1.2	**Alert**

Dose assessment using actual meteorology indicates doses > 10 mrem TEDE or 50 mrem thyroid CDE at or beyond the SITE BOUNDARY (Note, 4)

Note 4: The pre-calculated effluent monitor values presented in EALs RA1.1, RS1.1 and RG1.1 should be used for emergency classification assessments until the results from a dose assessment using actual meteorology are available.

Mode Applicability:

All

Definition(s):

SITE BOUNDARY - Area as depicted in BNP Radiological Emergency Response Plan Figure 1-1.1 Brunswick Site Plan

Basis:

Dose assessments are performed by computer-based methods (ref. 1, 2, 3)

This IC addresses a release of gaseous or liquid radioactivity that results in projected or actual offsite doses greater than or equal to 1% of the EPA Protective Action Guides (PAGs). It includes both monitored and un-monitored releases. Releases of this magnitude represent an actual or potential substantial degradation of the level of safety of the plant as indicated by a radiological release that significantly exceeds regulatory limits (e.g., a significant uncontrolled release).

Radiological effluent EALs are also included to provide a basis for classifying events and conditions that cannot be readily or appropriately classified on the basis of plant conditions alone. The inclusion of both plant condition and radiological effluent EALs more fully addresses the spectrum of possible accident events and conditions.

The TEDE dose is set at 1% of the EPA PAG of 1,000 mrem while the 50 mrem thyroid CDE was established in consideration of the 1:5 ratio of the EPA PAG for TEDE and thyroid CDE.

Classification based on effluent monitor readings assumes that a release path to the environment is established. If the effluent flow past an effluent monitor is known to have stopped due to actions to isolate the release path, then the effluent monitor reading is no longer valid for classification purposes.

Escalation of the emergency classification level would be via IC RS1.

BNP Basis Reference(s):

1. EMG-NGGC-0002 Off-site Dose Assessment
2. 0PEP-03.4.7 Automation of Off-Site Dose Projections
3. 0E&RC-03.4.8, Offsite Dose Projections for Monitored Releases
3. NEI 99-01 AA1

Category: R – Abnormal Rad Levels / Rad Effluent

Subcategory: 1 – Radiological Effluent

Initiating Condition: Release of gaseous or liquid radioactivity resulting in offsite dose greater than 10 mrem TEDE or 50 mrem thyroid CDE

EAL:

RA1.3	**Alert**
Analysis of a liquid effluent sample indicates a concentration or release rate that would result in doses > 10 mrem TEDE or 50 mrem thyroid CDE at or beyond the SITE BOUNDARY for 60 min. of exposure (Notes 1, 2)	

Note 1: The SEC should declare the event promptly upon determining that time limit has been exceeded, or will likely be exceeded.

Note 2: If an ongoing release is detected and the release start time is unknown, assume that the release duration has exceeded the specified time limit.

Mode Applicability:

All

Definition(s):

SITE BOUNDARY - Area as depicted in BNP Radiological Emergency Response Plan Figure 1-1.1 Brunswick Site Plan

Basis:

Dose assessments based on liquid releases are performed per Offsite Dose Calculation Manual (ref. 1).

This IC addresses a release of gaseous or liquid radioactivity that results in projected or actual offsite doses greater than or equal to 1% of the EPA Protective Action Guides (PAGs). It includes both monitored and un-monitored releases. Releases of this magnitude represent an actual or potential substantial degradation of the level of safety of the plant as indicated by a radiological release that significantly exceeds regulatory limits (e.g., a significant uncontrolled release).

Radiological effluent EALs are also included to provide a basis for classifying events and conditions that cannot be readily or appropriately classified on the basis of plant conditions alone. The inclusion of both plant condition and radiological effluent EALs more fully addresses the spectrum of possible accident events and conditions.

The TEDE dose is set at 1% of the EPA PAG of 1,000 mrem while the 50 mrem thyroid CDE was established in consideration of the 1:5 ratio of the EPA PAG for TEDE and thyroid CDE.

Classification based on effluent monitor readings assumes that a release path to the environment is established. If the effluent flow past an effluent monitor is known to have stopped due to actions to isolate the release path, then the effluent monitor reading is no longer valid for classification purposes.

Escalation of the emergency classification level would be via IC RS1.

BNP Basis Reference(s):

1. BNP Offsite Dose Calculation Manual
2. NEI 99-01 AA1

Category: R – Abnormal Rad Levels / Rad Effluent

Subcategory: 1 – Radiological Effluent

Initiating Condition: Release of gaseous or liquid radioactivity resulting in offsite dose greater than 10 mrem TEDE or 50 mrem thyroid CDE

EAL:

RA1.4 Alert

Field survey results indicate **EITHER** of the following at or beyond the SITE BOUNDARY:

- Closed window dose rates > 10 mR/hr expected to continue for ≥ 60 min.
- Analyses of field survey samples indicate thyroid CDE > 50 mrem for 60 min. of inhalation.

(Notes 1, 2)

Note 1: The SEC should declare the event promptly upon determining that time limit has been exceeded, or will likely be exceeded.

Note 2: If an ongoing release is detected and the release start time is unknown, assume that the release duration has exceeded the specified time limit.

Mode Applicability:

All

Definition(s):

SITE BOUNDARY - Area as depicted in BNP Radiological Emergency Response Plan Figure 1-1.1 Brunswick Site Plan

Basis:

0PEP-02.6.6, Environmental Monitoring Team Leader and 0PEP-03.5.5 Environmental Monitoring and Plume Tracking provide guidance for emergency or post-accident radiological environmental monitoring (ref. 1, 2).

This IC addresses a release of gaseous or liquid radioactivity that results in projected or actual offsite doses greater than or equal to 1% of the EPA Protective Action Guides (PAGs). It includes both monitored and un-monitored releases. Releases of this magnitude represent an actual or potential substantial degradation of the level of safety of the plant as indicated by a radiological release that significantly exceeds regulatory limits (e.g., a significant uncontrolled release).

Radiological effluent EALs are also included to provide a basis for classifying events and conditions that cannot be readily or appropriately classified on the basis of plant conditions alone. The inclusion of both plant condition and radiological effluent EALs more fully addresses the spectrum of possible accident events and conditions.

The TEDE dose is set at 1% of the EPA PAG of 1,000 mrem while the 50 mrem thyroid CDE was established in consideration of the 1:5 ratio of the EPA PAG for TEDE and thyroid CDE.

Classification based on effluent monitor readings assumes that a release path to the environment is established. If the effluent flow past an effluent monitor is known to have stopped due to actions to isolate the release path, then the effluent monitor reading is no longer valid for classification purposes.

Escalation of the emergency classification level would be via IC RS1.

BNP Basis Reference(s):

1. 0PEP-02.6.6, Environmental Monitoring Team Leader
2. 0PEP-03.5.5 Environmental Monitoring and Plume Tracking
3. NEI 99-01 AA1

Category: R – Abnormal Rad Levels / Rad Effluent

Subcategory: 1 – Radiological Effluent

Initiating Condition: Release of gaseous radioactivity resulting in offsite dose greater than 100 mrem TEDE or 500 mrem thyroid CDE

EAL:

RS1.1 Site Area Emergency

In the absence of real-time dose assessment, reading on **any** Table R-1 effluent radiation monitor > column "SAE" for ≥ 15 min.
(Notes 1, 2, 3, 4)

Note 1: The SEC should declare the event promptly upon determining that time limit has been exceeded, or will likely be exceeded.

Note 2: If an ongoing release is detected and the release start time is unknown, assume that the release duration has exceeded the specified time limit.

Note 3: If the effluent flow past an effluent monitor is known to have stopped, indicating that the release path is isolated, the effluent monitor reading is no longer VALID for classification purposes.

Note 4: The pre-calculated effluent monitor values presented in EALs RA1.1, RS1.1 and RG1.1 should be used for emergency classification assessments until the results from a dose assessment using actual meteorology are available.

Table R-1 Effluent Monitor Classification Thresholds

Release Point		Monitor	GE	SAE	Alert	UE
Gaseous	Main Stack Rad	D12-RM-23S	2.13E+09 µCi/sec	2.13E+08 µCi/sec	2.13E+07 µCi/sec	1.80E+06 µCi/sec
Gaseous	Reactor Bldg Vent Noble Gas	CAC-AQH-1264-3	----	----	----	6.14E+04 cpm
Gaseous	Turbine Bldg Vent	D12-RM-23	1.07E+08 µCi/sec	1.07E+07 µCi/sec	1.07E+06 µCi/sec	1.13E+04 µCi/sec
Liquid	Service Water Effluent Rad	D12-RM-K605	----	----	----	2 x hi alarm
Liquid	Radwaste Effluent Rad	D12-RM-K604	----	----	----	2 x hi-hi alarm

Mode Applicability:

All

Definition(s):

None

Basis:

This EAL address gaseous radioactivity releases, that for whatever reason, cause effluent radiation monitor readings corresponding to site boundary doses that exceed either (ref. 2):

- 100 mRem TEDE
- 500 mRem CDE Thyroid

The column "SAE" gaseous effluent release values in Table R-1 correspond to calculated doses of 10% of the EPA Protective Action Guidelines (TEDE or CDE Thyroid).

Instrumentation that may be used to assess this EAL is listed below (ref 1):

- Main Stack Monitoring System Noble Gas Activity Monitor 2-D12-RM-23S (1/2-D12-RR-4599-4)
- Turbine Building Ventilation Monitoring System Noble Gas Activity Monitor 1(2)-D12-RM-23 (1/2-DL12-RR-4548-4)

This IC addresses a release of gaseous radioactivity that results in projected or actual offsite doses greater than or equal to 10% of the EPA Protective Action Guides (PAGs). It includes both monitored and un-monitored releases. Releases of this magnitude are associated with the failure of plant systems needed for the protection of the public.

Radiological effluent EALs are also included to provide a basis for classifying events and conditions that cannot be readily or appropriately classified on the basis of plant conditions alone. The inclusion of both plant condition and radiological effluent EALs more fully addresses the spectrum of possible accident events and conditions.

The TEDE dose is set at 10% of the EPA PAG of 1,000 mrem while the 500 mrem thyroid CDE was established in consideration of the 1:5 ratio of the EPA PAG for TEDE and thyroid CDE.

Classification based on effluent monitor readings assumes that a release path to the environment is established. If the effluent flow past an effluent monitor is known to have stopped due to actions to isolate the release path, then the effluent monitor reading is no longer valid for classification purposes.

Escalation of the emergency classification level would be via IC RG1.

BNP Basis Reference(s):

1. BNP ODCM Appendix E Radioactive Liquid and Gaseous Effluent Monitoring Instrumentation Numbers
2. EP-EALCALC-BNP-0801 Radiological Gaseous Effluent Values (EALs RG1, RS1, RA1 and RU1)
3. NEI 99-01 AS1

Category: R – Abnormal Rad Levels / Rad Effluent

Subcategory: 1 – Radiological Effluent

Initiating Condition: Release of gaseous radioactivity resulting in offsite dose greater than 100 mrem TEDE or 500 mrem thyroid CDE

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 (Note 4)	

Note 4: The pre-calculated effluent monitor values presented in EALs RA1.1, RS1.1 and RG1.1 should be used for emergency classification assessments until the results from a dose assessment using actual meteorology are available.

Mode Applicability:

All

Definition(s):

SITE BOUNDARY - Area as depicted in BNP Radiological Emergency Response Plan Figure 1-1.1 Brunswick Site Plan

Basis:

Dose assessments are performed by computer-based methods (ref. 1, 2, 3)

This IC addresses a release of gaseous radioactivity that results in projected or actual offsite doses greater than or equal to 10% of the EPA Protective Action Guides (PAGs). It includes both monitored and un-monitored releases. Releases of this magnitude are associated with the failure of plant systems needed for the protection of the public.

Radiological effluent EALs are also included to provide a basis for classifying events and conditions that cannot be readily or appropriately classified on the basis of plant conditions alone. The inclusion of both plant condition and radiological effluent EALs more fully addresses the spectrum of possible accident events and conditions.

The TEDE dose is set at 10% of the EPA PAG of 1,000 mrem while the 500 mrem thyroid CDE was established in consideration of the 1:5 ratio of the EPA PAG for TEDE and thyroid CDE.

Classification based on effluent monitor readings assumes that a release path to the environment is established. If the effluent flow past an effluent monitor is known to have stopped due to actions to isolate the release path, then the effluent monitor reading is no longer valid for classification purposes.

Escalation of the emergency classification level would be via IC RG1.

BNP Basis Reference(s):

1. EMG-NGGC-0002 Off-site Dose Assessment
2. 0PEP-03.4.7 Automation of Off-Site Dose Projections
3. 0E&RC-03.4.8, Offsite Dose Projections for Monitored Releases
4. NEI 99-01 AS1

Category: R – Abnormal Rad Levels / Rad Effluent

Subcategory: 1 – Radiological Effluent

Initiating Condition: Release of gaseous radioactivity resulting in offsite dose greater than 100 mrem TEDE or 500 mrem thyroid CDE

EAL:

RS1.3 Site Area Emergency

Field survey results indicate EITHER of the following at or beyond the SITE BOUNDARY:

- Closed window dose rates > 100 mR/hr expected to continue for ≥ 60 min.
- Analyses of field survey samples indicate thyroid CDE > 500 mrem for 60 min. of inhalation.

(Notes 1, 2)

Note 1: The SEC should declare the event promptly upon determining that time limit has been exceeded, or will likely be exceeded.

Note 2: If an ongoing release is detected and the release start time is unknown, assume that the release duration has exceeded the specified time limit.

Mode Applicability:

All

Definition(s):

SITE BOUNDARY - Area as depicted in BNP Radiological Emergency Response Plan Figure 1-1.1 Brunswick Site Plan

Basis:

0PEP-02.6.6, Environmental Monitoring Team Leader and 0PEP-03.5.5 Environmental Monitoring and Plume Tracking provide guidance for emergency or post-accident radiological environmental monitoring (ref. 1, 2).

This IC addresses a release of gaseous radioactivity that results in projected or actual offsite doses greater than or equal to 10% of the EPA Protective Action Guides (PAGs). It includes both monitored and un-monitored releases. Releases of this magnitude are associated with the failure of plant systems needed for the protection of the public.

Radiological effluent EALs are also included to provide a basis for classifying events and conditions that cannot be readily or appropriately classified on the basis of plant conditions alone. The inclusion of both plant condition and radiological effluent EALs more fully addresses the spectrum of possible accident events and conditions.

The TEDE dose is set at 10% of the EPA PAG of 1,000 mrem while the 500 mrem thyroid CDE was established in consideration of the 1:5 ratio of the EPA PAG for TEDE and thyroid CDE.

Escalation of the emergency classification level would be via IC RG1.

BNP Basis Reference(s):

1. 0PEP-02.6.6, Environmental Monitoring Team Leader
2. 0PEP-03.5.5 Environmental Monitoring and Plume Tracking
3. NEI 99-01 AS1

Category: R – Abnormal Rad Levels / Rad Effluent

Subcategory: 1 – Radiological Effluent

Initiating Condition: Release of gaseous radioactivity resulting in offsite dose greater than 1,000 mrem TEDE or 5,000 mrem thyroid CDE

EAL:

RG1.1 General Emergency

In the absence of real-time dose assessment, reading on **any** Table R-1 effluent radiation monitor > column "GE" for ≥ 15 min.
(Notes 1, 2, 3, 4)

Note 1: The SEC should declare the event promptly upon determining that time limit has been exceeded, or will likely be exceeded.

Note 2: If an ongoing release is detected and the release start time is unknown, assume that the release duration has exceeded the specified time limit.

Note 3: If the effluent flow past an effluent monitor is known to have stopped, indicating that the release path is isolated, the effluent monitor reading is no longer VALID for classification purposes.

Note 4: The pre-calculated effluent monitor values presented in EALs RA1.1, RS1.1 and RG1.1 should be used for emergency classification assessments until the results from a dose assessment using actual meteorology are available.

Table R-1 Effluent Monitor Classification Thresholds

Release Point		Monitor	GE	SAE	Alert	UE
Gaseous	Main Stack Rad	D12-RM-23S	2.13E+09 µCi/sec	2.13E+08 µCi/sec	2.13E+07 µCi/sec	1.80E+06 µCi/sec
Gaseous	Reactor Bldg Vent Noble Gas	CAC-AQH-1264-3	----	----	----	6.14E+04 cpm
Gaseous	Turbine Bldg Vent	D12-RM-23	1.07E+08 µCi/sec	1.07E+07 µCi/sec	1.07E+06 µCi/sec	1.13E+04 µCi/sec
Liquid	Service Water Effluent Rad	D12-RM-K605	----	----	----	2 x hi alarm
Liquid	Radwaste Effluent Rad	D12-RM-K604	----	----	----	2 x hi-hi alarm

Mode Applicability:

All

Definition(s):

None

Basis:

This EAL address gaseous radioactivity releases, that for whatever reason, cause effluent radiation monitor readings corresponding to site boundary doses that exceed either (ref. 2):

- 1000 mRem TEDE
- 5000 mRem CDE Thyroid

The column "SAE" gaseous effluent release values in Table R-1 correspond to calculated doses of 100% of the EPA Protective Action Guidelines (TEDE or CDE Thyroid).

Instrumentation that may be used to assess this EAL is listed below (ref 1):

- Main Stack Monitoring System Noble Gas Activity Monitor 2-D12-RM-23S (1/2-D12-RR-4599-4)
- Turbine Building Ventilation Monitoring System Noble Gas Activity Monitor 1(2)-D12-RM-23 (1/2-DL12-RR-4548-4)

This IC addresses a release of gaseous radioactivity that results in projected or actual offsite doses greater than or equal to the EPA Protective Action Guides (PAGs). It includes both monitored and un-monitored releases. Releases of this magnitude will require implementation of protective actions for the public.

Radiological effluent EALs are also included to provide a basis for classifying events and conditions that cannot be readily or appropriately classified on the basis of plant conditions alone. The inclusion of both plant condition and radiological effluent EALs more fully addresses the spectrum of possible accident events and conditions.

The TEDE dose is set at the EPA PAG of 1,000 mrem 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.

Classification based on effluent monitor readings assumes that a release path to the environment is established. If the effluent flow past an effluent monitor is known to have stopped due to actions to isolate the release path, then the effluent monitor reading is no longer valid for classification purposes.

BNP Basis Reference(s):

1. BNP ODCM Appendix E Radioactive Liquid and Gaseous Effluent Monitoring Instrumentation Numbers
2. EP-EALCALC-BNP-0801 Radiological Gaseous Effluent Values (EALs RG1, RS1, RA1 and RU1)
3. NEI 99-01 AG1

Category: R – Abnormal Rad Levels / Rad Effluent

Subcategory: 1 – Radiological Effluent

Initiating Condition: Release of gaseous radioactivity resulting in offsite dose greater than 1,000 mrem TEDE or 5,000 mrem thyroid CDE

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 (Note 4)	

Note 4: The pre-calculated effluent monitor values presented in EALs RA1.1, RS1.1 and RG1.1 should be used for emergency classification assessments until the results from a dose assessment using actual meteorology are available.

Applicability:

All

Definition(s):

SITE BOUNDARY - Area as depicted in BNP Radiological Emergency Response Plan Figure 1-1.1 Brunswick Site Plan

BNP Basis:

Dose assessments are performed by computer-based methods (ref. 1, 2, 3)

This IC addresses a release of gaseous radioactivity that results in projected or actual offsite doses greater than or equal to the EPA Protective Action Guides (PAGs). It includes both monitored and un-monitored releases. Releases of this magnitude will require implementation of protective actions for the public.

Radiological effluent EALs are also included to provide a basis for classifying events and conditions that cannot be readily or appropriately classified on the basis of plant conditions alone. The inclusion of both plant condition and radiological effluent EALs more fully addresses the spectrum of possible accident events and conditions.

The TEDE dose is set at the EPA PAG of 1,000 mrem 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.

Classification based on effluent monitor readings assumes that a release path to the environment is established. If the effluent flow past an effluent monitor is known to have stopped due to actions to isolate the release path, then the effluent monitor reading is no longer valid for classification purposes.

BNP Basis Reference(s):

1. EMG-NGGC-0002 Off-site Dose Assessment
2. 0PEP-03.4.7 Automation of Off-Site Dose Projections
3. 0E&RC-03.4.8, Offsite Dose Projections for Monitored Releases
4. NEI 99-01 AG1

Category: R – Abnormal Rad Levels / Rad Effluent

Subcategory: 1 – Radiological Effluent

Initiating Condition: Release of gaseous radioactivity resulting in offsite dose greater than 1,000 mrem TEDE or 5,000 mrem thyroid CDE

EAL:

RG1.3 General Emergency

Field survey results indicate EITHER of the following at or beyond the SITE BOUNDARY:

- Closed window dose rates > 1,000 mR/hr expected to continue for ≥ 60 min.
- Analyses of field survey samples indicate thyroid CDE > 5,000 mrem for 60 min. of inhalation.

(Notes 1, 2)

Note 1: The SEC should declare the event promptly upon determining that time limit has been exceeded, or will likely be exceeded.

Note 2: If an ongoing release is detected and the release start time is unknown, assume that the release duration has exceeded the specified time limit.

Mode Applicability:

All

Definition(s):

SITE BOUNDARY - Area as depicted in BNP Radiological Emergency Response Plan Figure 1-1.1 Brunswick Site Plan

Basis:

0PEP-02.6.6, Environmental Monitoring Team Leader and 0PEP-03.5.5 Environmental Monitoring and Plume Tracking provide guidance for emergency or post-accident radiological environmental monitoring (ref. 1, 2).

This IC addresses a release of gaseous radioactivity that results in projected or actual offsite doses greater than or equal to the EPA Protective Action Guides (PAGs). It includes both monitored and un-monitored releases. Releases of this magnitude will require implementation of protective actions for the public.

Radiological effluent EALs are also included to provide a basis for classifying events and conditions that cannot be readily or appropriately classified on the basis of plant conditions alone. The inclusion of both plant condition and radiological effluent EALs more fully addresses the spectrum of possible accident events and conditions.

The TEDE dose is set at the EPA PAG of 1,000 mrem 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.

BNP Basis Reference(s):

1. 0PEP-02.6.6, Environmental Monitoring Team Leader
2. 0PEP-03.5.5 Environmental Monitoring and Plume Tracking
3. NEI 99-01 AG1

Category: R – Abnormal Rad Levels / Rad Effluent

Subcategory: 2 – Irradiated Fuel Event

Initiating Condition: Unplanned loss of water level above irradiated fuel

EAL:

RU2.1 Unusual Event

UNPLANNED water level drop in the REFUELING PATHWAY as indicated by low water level alarm (A-04 6-6) or indication

AND

UNPLANNED rise in area radiation levels as indicated by **any** of the following radiation monitors:

- ARM Channel 26 New Fuel Vault
- ARM Channel 27 North of Fuel Pool
- ARM Channel 28 Between Reactor and Fuel Pool
- ARM Channel 29 Cask Wash Area

Mode Applicability:

All

Definition(s):

UNPLANNED-. A parameter change or an event that is not 1) the result of an intended evolution or 2) an expected plant response to a transient. The cause of the parameter change or event may be known or unknown.

REFUELING PATHWAY-. The reactor refueling cavity, spent fuel pool and fuel transfer canal (cattle chute) comprise the refueling pathway.

Basis:

The spent fuel pool low water level alarm setpoint is actuated by level switch G410-LSHL-N001 at a setpoint of 37' 5". Water level restoration instructions are performed in accordance with 1(2)APP A-04 6-6 Fuel Pool Level Low (ref. 1).

When the spent fuel pool and reactor cavity are connected, there could exist the possibility of uncovering irradiated fuel. Therefore, this EAL is applicable for conditions in which irradiated fuel is being transferred to and from the RPV and spent fuel pool.

This IC addresses a decrease in water level above irradiated fuel sufficient to cause elevated radiation levels. This condition could be a precursor to a more serious event and is also indicative of a minor loss in the ability to control radiation levels within the plant. It is therefore a potential degradation in the level of safety of the plant.

A water level decrease will be primarily determined by indications from available level instrumentation. Other sources of level indications may include reports from plant personnel (e.g., from a refueling crew) or video camera observations (if available). A significant drop in the water level may also cause an increase in the radiation levels of adjacent areas that can be detected by monitors in those locations.

The effects of planned evolutions should be considered. For example, a refueling bridge area radiation monitor reading may increase due to planned evolutions such as lifting of the reactor vessel head or movement of a fuel assembly. Note that this EAL is applicable only in cases where the elevated reading is due to an unplanned loss of water level.

A drop in water level above irradiated fuel within the reactor vessel may be classified in accordance Recognition Category C during the Cold Shutdown and Refueling modes.

Escalation of the emergency classification level would be via IC RA2.

BNP Basis Reference(s):

1. 1(2)APP-A-04 6-6 Fuel Pool Level Low
2. DBD-11 Radiation Monitoring System
3. NEI 99-01 AU2

Category: R – Abnormal Rad Levels / Rad Effluent

Subcategory: 2 – Irradiated Fuel Event

Initiating Condition: Significant lowering of water level above, or damage to, irradiated fuel

EAL:

RA2.1	**Unusual Event**
Uncovery of irradiated fuel in the REFUELING PATHWAY	

Mode Applicability:

All

Definition(s):

REFUELING PATHWAY-. The reactor refueling cavity, spent fuel pool and fuel transfer canal (cattle chute) comprise the refueling pathway.

Basis:

None.

This IC addresses events that have caused imminent or actual damage to an irradiated fuel assembly, or a significant lowering of water level within the spent fuel pool. These events present radiological safety challenges to plant personnel and are precursors to a release of radioactivity to the environment. As such, they represent an actual or potential substantial degradation of the level of safety of the plant.

This EAL escalates from RU2.1 in that the loss of level, in the affected portion of the REFUELING PATHWAY, is of sufficient magnitude to have resulted in uncovery of irradiated fuel. Indications of irradiated fuel uncovery may include direct or indirect visual observation (e.g., reports from personnel or camera images), as well as significant changes in water and radiation levels, or other plant parameters. Computational aids may also be used (e.g., a boil-off curve). Classification of an event using this EAL should be based on the totality of available indications, reports and observations.

While an area radiation monitor could detect an increase in a dose rate due to a lowering of water level in some portion of the REFUELING PATHWAY, the reading may not be a reliable indication of whether or not the fuel is actually uncovered. To the degree possible, readings should be considered in combination with other available indications of inventory loss.

A drop in water level above irradiated fuel within the reactor vessel may be classified in accordance Recognition Category C during the Cold Shutdown and Refueling modes.

Escalation of the emergency classification level would be via IC RS1.

BNP Basis Reference(s):

1. 1(2)APP-A4 6-6 (Fuel pool Level Low)
2. 1(2)APP-A7 2-2 (Reactor Water Level Hi/Low)
3. NEI 99-01 AA2

Category: R – Abnormal Rad Levels / Rad Effluent

Subcategory: 2 – Irradiated Fuel Event

Initiating Condition: Significant lowering of water level above, or damage to, irradiated fuel

EAL:

RA2.2 Alert

Damage to irradiated fuel resulting in a release of radioactivity

AND

Any of the following radiation monitor indications:

- Reactor Bldg Vent Rad Monitor Channel A or B (> 3 mR/hr)
- ARM Channel 26 New Fuel Vault (> 6 mR/hr)
- ARM Channel 27 North of Fuel Pool (>10 mR/hr)
- ARM Channel 28 Between Reactor and Fuel Pool (> 1000 mR/hr)
- ARM Channel 29 Cask Wash Area (>40 mR/hr)

Mode Applicability:

All

Definition(s):

None

Basis:

The high alarm setpoints for the radiation monitors are (ref. 1, 2, 3 4):

- Reactor Building Exhaust Plenum Rad Monitor Channel A or B > 3 mR/hr
- ARM Channel 26 New Fuel Vault > 6 mR/hr
- ARM Channel 27 North of Fuel Pool > 10 mR/hr
- ARM Channel 28 Between Reactor and Fuel Pool > 1000 mR/hr
- ARM Channel 29 Cask Wash Area > 40mR/hr

This IC addresses events that have caused imminent or actual damage to an irradiated fuel assembly, or a significant lowering of water level within the spent fuel pool. These events present radiological safety challenges to plant personnel and are precursors to a release of radioactivity to the environment. As such, they represent an actual or potential substantial degradation of the level of safety of the plant.

This IC applies to irradiated fuel that is licensed for dry storage up to the point that the loaded storage cask is sealed. Once sealed, damage to a loaded cask causing loss of the CONFINEMENT BOUNDARY is classified in accordance with EAL EU1.1.

Escalation of the emergency would be based on either Recognition Category R or C ICs.

This EAL addresses a release of radioactive material caused by mechanical damage to irradiated fuel. Damaging events may include the dropping, bumping or binding of an assembly, or dropping a heavy load onto an assembly. A rise in readings on radiation monitors should be considered in conjunction with in-plant reports or observations of a potential fuel damaging event (e.g., a fuel handling accident).

Escalation of the emergency classification level would be via IC RS1.

BNP Basis Reference(s):

1. 1(2)APP-UA-03 3-7
2. 1(2)APP-UA-03 4-5
3. 1(2)APP-UA-03 4-7
4. DBD-11 Radiation Monitoring System
5. NEI 99-01 AA2

Category: R – Abnormal Rad Levels / Rad Effluent

Subcategory: 2 – Irradiated Fuel Event

Initiating Condition: Significant lowering of water level above, or damage to, irradiated fuel

EAL:

RA2.3	**Alert**
Lowering of spent fuel pool level to ≤ 105 ft. 3 in. ele.	

Mode Applicability:

All

Definition(s):

None

Basis:

Post-Fukushima order EA-12-051 required the installation of reliable SFP level indication capable of identifying normal level (Level 1 – 116 ft. 1 in. ele.), SFP level 10 ft. above the top of the fuel racks (Level 2 – 105 ft. 3 in. ele.) and SFP level at the top of the fuel racks (Level 3 – 95 ft. 3 in. ele.).

An indicated level of 105 ft. 3 in. corresponds to the Level 2 setpoint (Ref. 1).

This IC addresses events that have caused imminent or actual damage to an irradiated fuel assembly, or a significant lowering of water level within the spent fuel pool. These events present radiological safety challenges to plant personnel and are precursors to a release of radioactivity to the environment. As such, they represent an actual or potential substantial degradation of the level of safety of the plant.

Escalation of the emergency would be based on either Recognition Category R or C ICs.

Spent fuel pool water level at this value is within the lower end of the level range necessary to prevent significant dose consequences from direct gamma radiation to personnel performing operations in the vicinity of the spent fuel pool. This condition reflects a significant loss of spent fuel pool water inventory and thus it is also a precursor to a loss of the ability to adequately cool the irradiated fuel assembles stored in the pool.

Escalation of the emergency classification level would be via IC RS1.

BNP Basis Reference(s):

1. PCHG-DESG Engineering Change 0000089578R0
2. NEI 99-01 AA2

Category: R – Abnormal Rad Levels / Rad Effluent

Subcategory: 2 – Irradiated Fuel Event

Initiating Condition: Spent fuel pool level at the top of the fuel racks

EAL:

RS2.1	**Site Area Emergency**
Lowering of spent fuel pool level to ≤ 95 ft. 3 in. ele.	

Mode Applicability:

All

Definition(s):

None

Basis:

Post-Fukushima order EA-12-051 required the installation of reliable SFP level indication capable of identifying normal level (Level 1 – 116 ft. 1 in. ele.), SFP level 10 ft. above the top of the fuel racks (Level 2 – 105 ft. 3 in. ele.) and SFP level at the top of the fuel racks (Level 3 – 95 ft. 3 in. ele.).

An indicated level of 95 ft. 3 in. corresponds to the Level 3 setpoint (Ref. 1).

This EAL addresses a significant loss of spent fuel pool inventory control and makeup capability leading to IMMINENT fuel damage. This condition entails major failures of plant functions needed for protection of the public and thus warrant a Site Area Emergency declaration.

It is recognized that this IC would likely not be met until well after another Site Area Emergency IC was met; however, it is included to provide classification diversity.

Escalation of the emergency classification level would be via IC RG1 or RG2.

BNP Basis Reference(s):

1. PCHG-DESG Engineering Change 0000089578R0
2. NEI 99-01 AS2

Category: R – Abnormal Rad Levels / Rad Effluent

Subcategory: 2 – Irradiated Fuel Event

Initiating Condition: Spent fuel pool level cannot be restored to at least the top of the fuel racks for 60 minutes or longer

EAL:

RG2.1	**General Emergency**
Spent fuel pool level cannot be restored $\geq$ 95 ft. 3 in. ele. for $\geq$ 60 min. (Note 1)	

Note 1: The SEC should declare the event promptly upon determining that time limit has been exceeded, or will likely be exceeded.

Mode Applicability:

All

Definition(s):

None

Basis:

Post-Fukushima order EA-12-051 required the installation of reliable SFP level indication capable of identifying normal level (Level 1 – 116 ft. 1 in. ele.), SFP level 10 ft. above the top of the fuel racks (Level 2 – 105 ft. 3 in. ele.) and SFP level at the top of the fuel racks (Level 3 – 95 ft. 3 in. ele.).

An indicated level of 95 ft. 3 in. corresponds to the Level 3 setpoint (Ref. 1).

This EAL addresses a significant loss of spent fuel pool inventory control and makeup capability leading to a prolonged uncovery of spent fuel. This condition will lead to fuel damage and a radiological release to the environment.

It is recognized that this IC would likely not be met until well after another General Emergency IC was met; however, it is included to provide classification diversity.

BNP Basis Reference(s):

1. PCHG-DESG Engineering Change 0000089578R0
2. NEI 99-01 AG2

Category: R – Abnormal Rad Levels / Rad Effluent

Subcategory: 3 – Area Radiation Levels

Initiating Condition: Radiation levels that IMPEDE access to equipment necessary for normal plant operations, cooldown or shutdown

EAL:

RA3.1	**Alert**
Dose rates > 15 mR/hr in **EITHER** of the following areas: Control Room (ARM Channel 1-1) **OR** Central Alarm Station (by survey)	

Mode Applicability:

All

Definition(s):

None

Basis:

Areas that meet this threshold include the Control Room and the Central Alarm Station (CAS). ARM Channel 1-1 monitors the Control room for area radiation (ref. 1). The CAS is included in this EAL because of its' importance to permitting access to areas required to assure safe plant operations.

There is no permanently installed CAS area radiation monitors that may be used to assess this EAL threshold. Therefore this threshold must be assessed via local radiation survey for the CAS.

This IC addresses elevated radiation levels in certain plant rooms/areas sufficient to preclude or impede personnel from performing actions necessary to maintain normal plant operation, or to perform a normal plant cooldown and shutdown. As such, it represents an actual or potential substantial degradation of the level of safety of the plant. The Site Emergency Coordinator should consider the cause of the increased radiation levels and determine if another IC may be applicable.

Escalation of the emergency classification level would be via Recognition Category R, C or F ICs.

BNP Basis Reference(s):

1. 1(2)APP-UA-03 6-7 (Area RAD Control Room Hi)
2. NEI 99-01 AA3

Category: R – Abnormal Rad Levels / Rad Effluent

Subcategory: 3 – Area Radiation Levels

Initiating Condition: Radiation levels that IMPEDE access to equipment necessary for normal plant operations, cooldown or shutdown

EAL:

RA3.2 Alert

An UNPLANNED event results in radiation levels that prohibit or IMPEDE access to **any** Table R-2 rooms or areas (Note 5)

Note 5: If the equipment in the listed room or area was already inoperable or out-of-service before the event occurred, then no emergency classification is warranted.

Table R-2 Safe Operation & Shutdown Areas	
Room/Area	**Mode Applicability**
Reactor Building -17' North RHR Unit-1 & 2	3, 4, 5
Reactor Building -17' South RHR Unit-1 & 2	3, 4, 5
Reactor Building 20' East & West MCC Areas Unit-1 & 2	3, 4, 5
Reactor Building 20' Pipe Tunnel Unit-1 & 2	3, 4, 5

Mode Applicability:

All

Definition(s):

IMPEDE(D)- Personnel access to a room or area is hindered to an extent that extraordinary measures are necessary to facilitate entry of personnel into the affected room/area (e.g., requiring use of protective equipment, such as SCBAs, that is not routinely employed).

UNPLANNED- A parameter change or an event that is not 1) the result of an intended evolution or 2) an expected plant response to a transient. The cause of the parameter change or event may be known or unknown.

Basis:

If the equipment in the listed room or area was already inoperable, or out-of-service, before the event occurred, then no emergency should be declared since the event will have no adverse impact beyond that already allowed by Technical Specifications at the time of the event.

The list of plant rooms or areas with entry-related mode applicability identified specify those rooms or areas that contain equipment which require a manual/local action as specified in operating procedures used for normal plant operation, cooldown and shutdown. Rooms or areas in which actions of a contingent or emergency nature would be performed (e.g., an action to address an off-normal or emergency condition such as emergency repairs, corrective measures or emergency operations) are not included. In addition, the list specifies the plant mode(s) during which entry would be required for each room or area (ref. 1).

This IC addresses elevated radiation levels in certain plant rooms/areas sufficient to preclude or impede personnel from performing actions necessary to maintain normal plant operation, or to perform a normal plant cooldown and shutdown. As such, it represents an actual or potential substantial degradation of the level of safety of the plant. The Site Emergency Coordinator should consider the cause of the increased radiation levels and determine if another IC may be applicable.

For RA3.2, an Alert declaration is warranted if entry into the affected room/area is, or may be, procedurally required during the plant operating mode in effect at the time of the elevated radiation levels. The emergency classification is not contingent upon whether entry is actually necessary at the time of the increased radiation levels. Access should be considered as impeded if extraordinary measures are necessary to facilitate entry of personnel into the affected room/area (e.g., installing temporary shielding, requiring use of non-routine protective equipment, requesting an extension in dose limits beyond normal administrative limits).

An emergency declaration is not warranted if any of the following conditions apply:

- The plant is in an operating mode different than the mode specified for the affected room/area (i.e., entry is not required during the operating mode in effect at the time of the elevated radiation levels). For example, the plant is in Mode 1 when the radiation increase occurs, and the procedures used for normal operation, cooldown and shutdown do not require entry into the affected room until Mode 4.

- The increased radiation levels are a result of a planned activity that includes compensatory measures which address the temporary inaccessibility of a room or area (e.g., radiography, spent filter or resin transfer, etc.).

- The action for which room/area entry is required is of an administrative or record keeping nature (e.g., normal rounds or routine inspections).

- The access control measures are of a conservative or precautionary nature, and would not actually prevent or impede a required action.

- If the equipment in the listed room or area was already inoperable, or out-of-service, before the event occurred, then no emergency should be declared since the event will have no adverse impact beyond that already allowed by Technical Specifications at the time of the event.

Escalation of the emergency classification level would be via Recognition Category R, C or F ICs.

BNP Basis Reference(s):

1. Attachment 3 Safe Operation & Shutdown Areas Tables R-2 & H-2 Bases
2. NEI 99-01 AA3

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 - Refueling, D – Defueled).

The events of this category pertain to the following subcategories:

1. RPV Level

Reactor Pressure Vessel water level is directly related to the status of adequate core cooling and, therefore, fuel clad integrity.

2. Loss of Emergency AC Power

Loss of essential 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 4160 VAC emergency buses.

3. RCS Temperature

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

4. Loss of Vital 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 or degraded voltage on the 125 VDC vital buses.

5. Loss of Communications

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

6. Hazardous Event Affecting Safety Systems

Certain hazardous natural and technological events may result in visible damage to or degraded performance of safety systems warranting classification.

Category: C – Cold Shutdown / Refueling System Malfunction

Subcategory: 1 – RPV Level

Initiating Condition: UNPLANNED loss of RPV inventory for 15 minutes or longer

EAL:

CU1.1	**Unusual Event**
UNPLANNED loss of reactor coolant results in RPV water level less than a required lower limit for ≥ 15 min. (Note 1)	

Note 1: The SEC should declare the event promptly upon determining that time limit has been exceeded, or will likely be exceeded.

Mode Applicability:

4 - Cold Shutdown, 5 - Refueling

Definition(s):

UNPLANNED-. A parameter change or an event that is not 1) the result of an intended evolution or 2) an expected plant response to a transient. The cause of the parameter change or event may be known or unknown.

Basis:

Figure C-1 illustrates the elevations of the RPV level instrument ranges (ref. 2).

With the plant in Cold Shutdown, RPV water level is normally maintained above the RPV low level scram setpoint of 166 in. above TAF (ref. 1, 3). However, if RPV level is being controlled below the RPV low level scram setpoint, or if level is being maintained in a designated band in the reactor vessel it is the inability to maintain level above the low end of the designated control band due to a loss of inventory resulting from a leak in the RCS that is the concern.

With the plant in Refueling mode, RPV water level is normally maintained at or above the reactor vessel flange (Technical Specification LCO 3.9.6 requires at least 23 ft of water above the top of the reactor vessel flange in the refueling cavity during refueling operations). The RPV flange is at an indicated level of 355 in. as indicated on the red scale of B21-LI-R605A/B Shutdown Range Reactor Water Level Indication (ref. 4).

This IC addresses the inability to restore and maintain water level to a required minimum level (or the lower limit of a level band), or a loss of the ability to monitor RPV level concurrent with indications of coolant leakage. Either of these conditions is considered to be a potential degradation of the level of safety of the plant.

Refueling evolutions that decrease RCS water inventory are carefully planned and controlled. An UNPLANNED event that results in water level decreasing below a procedurally required limit warrants the declaration of an Unusual Event due to the reduced water inventory that is available to keep the core covered.

This EAL recognizes that the minimum required RPV level can change several times during the course of a refueling outage as different plant configurations and system lineups are implemented. This EAL is met if the minimum level, specified for the current plant conditions, cannot be maintained for 15 minutes or longer. The minimum level is typically specified in the applicable operating procedure but may be specified in another controlling document.

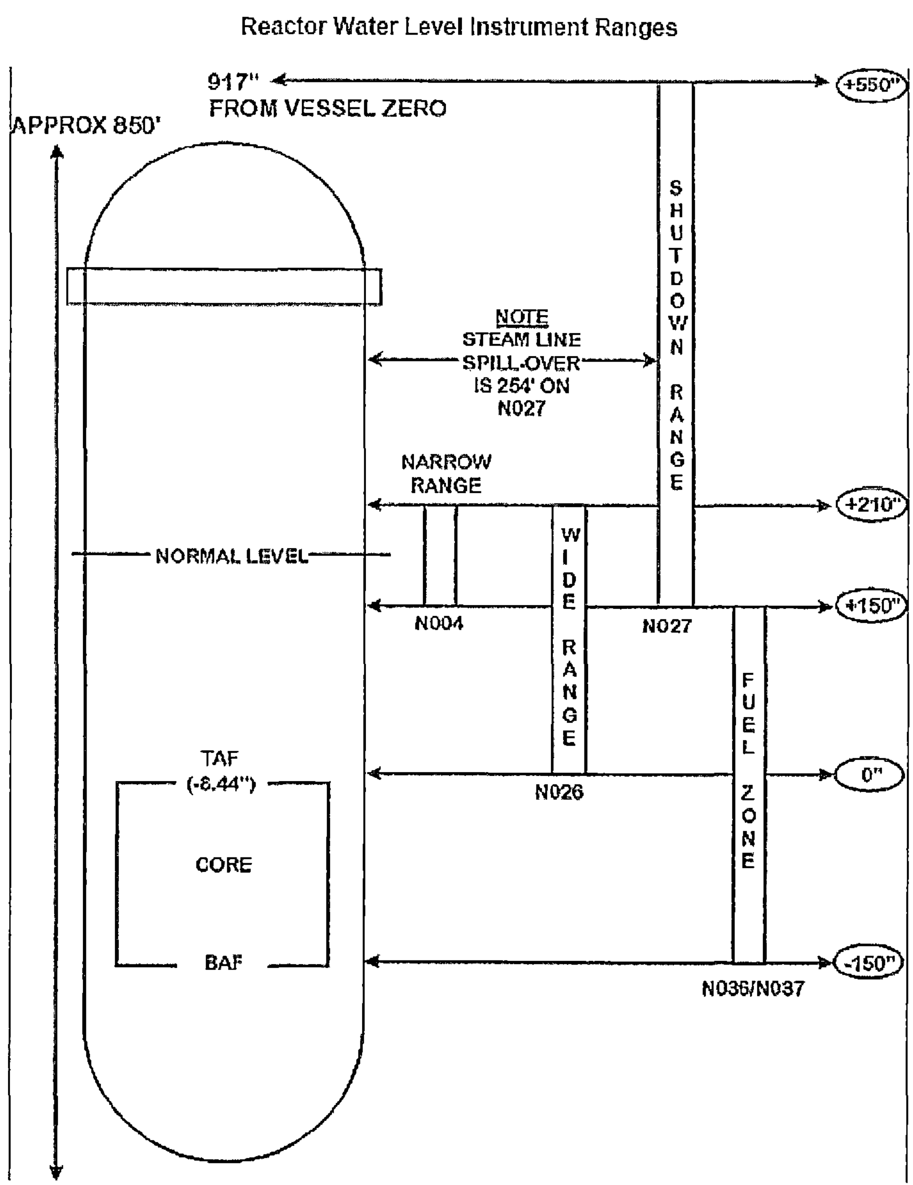
The 15-minute threshold duration allows sufficient time for prompt operator actions to restore and maintain the expected water level. This criterion excludes transient conditions causing a brief lowering of water level.

Continued loss of RCS inventory may result in escalation to the Alert emergency classification level via either IC CA1 or CA3.

BNP Basis Reference(s):

1. 0EOP-01-NL EOP/SAMG NUMERICAL LIMITS AND VALUES, Table 1E
2. SD-01.2 Reactor Vessel Instrumentation Figure 01.2-1 Reactor Water Level Instrument Ranges
3. 1(2) APP A7 2-2 (Reactor Water Level Hi/Low)
4. 0GP-06 Cold Shutdown to Refueling (Head Unbolted) step 5.1.14
5. NEI 99-01 CU1

Figure C-1 RPV Levels (ref. 2)



Category: C – Cold Shutdown / Refueling System Malfunction

Subcategory: 1 – RPV Level

Initiating Condition: UNPLANNED loss of RPV inventory

EAL:

CU1.2 Unusual Event

RPV water level cannot be monitored

AND

UNPLANNED increase in **any** Table C-1 sump or tank levels due to a loss of RPV inventory

Table C-1 Sumps & Tanks
• Drywell Floor Drain Sump
• Drywell Equipment Drain Sump
• RB Floor Drain Sump
• RB Equipment Drain Sump
• Torus
• Visual Observation

Mode Applicability:

4 - Cold Shutdown, 5 – Refueling

Definition(s):

UNPLANNED-. A parameter change or an event that is not 1) the result of an intended evolution or 2) an expected plant response to a transient. The cause of the parameter change or event may be known or unknown.

Basis:

In Cold Shutdown mode, the RCS will normally be intact and standard RPV level monitoring means are available. RPV level in the Refuel mode is normally monitored using the red scale of B21-LI-R605A/B Shutdown Range Reactor Water Level Indication.

In this EAL, all water level indication is unavailable and the RPV inventory loss must be detected by the leakage indications listed in Table C-1. 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 (ref. 1). A Reactor Building equipment or floor drain sump level rise may also be indicative of RCS inventory losses external to the Primary Containment from systems connected to the RPV. With RHR System operating in the Shutdown Cooling mode, an unexplained rise in torus level could be indicative of RHR valve misalignment or leakage (ref. 5). 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 that cannot be isolated could be indicative of a loss of RPV inventory.

This IC addresses the inability to restore and maintain water level to a required minimum level (or the lower limit of a level band), or a loss of the ability to monitor RPV level concurrent with indications of coolant leakage. Either of these conditions is considered to be a potential degradation of the level of safety of the plant.

Refueling evolutions that decrease RCS water inventory are carefully planned and controlled. An UNPLANNED event that results in water level decreasing below a procedurally required limit warrants the declaration of an Unusual Event due to the reduced water inventory that is available to keep the core covered.

This EAL addresses a condition where all means to determine RPV level have been lost. In this condition, operators may determine that an inventory loss is occurring by observing changes in sump and/or tank levels. Sump and/or tank level changes must be evaluated against other potential sources of water flow to ensure they are indicative of leakage from the RCS.

Continued loss of RCS inventory may result in escalation to the Alert emergency classification level via either IC CA1 or CA3.

BNP Basis Reference(s):

1. 0OP-47 Floor and Equipment Drain System Operating Procedure
2. 1OI-03.1 Control Room Operator Daily Surveillance Report
3. 2OI-03.2 Control Room Operator Daily Surveillance Report
4. 0AOP-14.0 Abnormal Primary Containment Conditions
5. 1(2)OP-17 Residual Heat Removal System Operating Procedure
6. NEI 99-01 CU1

Category: C – Cold Shutdown / Refueling System Malfunction

Subcategory: 1 – RPV Level

Initiating Condition: Loss of RPV inventory

EAL:

CA1.1 **Alert**
Loss of RPV inventory as indicated by RPV water level < 105 in. above TAF (Level 2)

Mode Applicability:

4 - Cold Shutdown, 5 – Refuel

Definition(s):

None

Basis:

The threshold RPV level of 105 in. is the low-low ECCS actuation setpoint (ref. 1). RPV level is normally monitored using the instruments in Figure C-1 (ref. 2).

When reactor vessel water level drops to 105 in. above TAF high pressure steam-driven injection sources HPCI (ECCS) and RCIC receive an initiation signal (ref. 1). Although these systems cannot restore RCS inventory in the cold condition, the Low-Low (Level 2) ECCS actuation setpoint is operationally significant and is indicative of a loss of RCS inventory significantly below the low level scram setpoint specified in CU1.1.

This IC addresses conditions that are precursors to a loss of the ability to adequately cool irradiated fuel (i.e., a precursor to a challenge to the fuel clad barrier). This condition represents a potential substantial reduction in the level of plant safety.

For this EAL, a lowering of water level below 105 in above TAF indicates that operator actions have not been successful in restoring and maintaining RPV water level. The heat-up rate of the coolant will increase as the available water inventory is reduced. A continuing decrease in water level will lead to core uncovery.

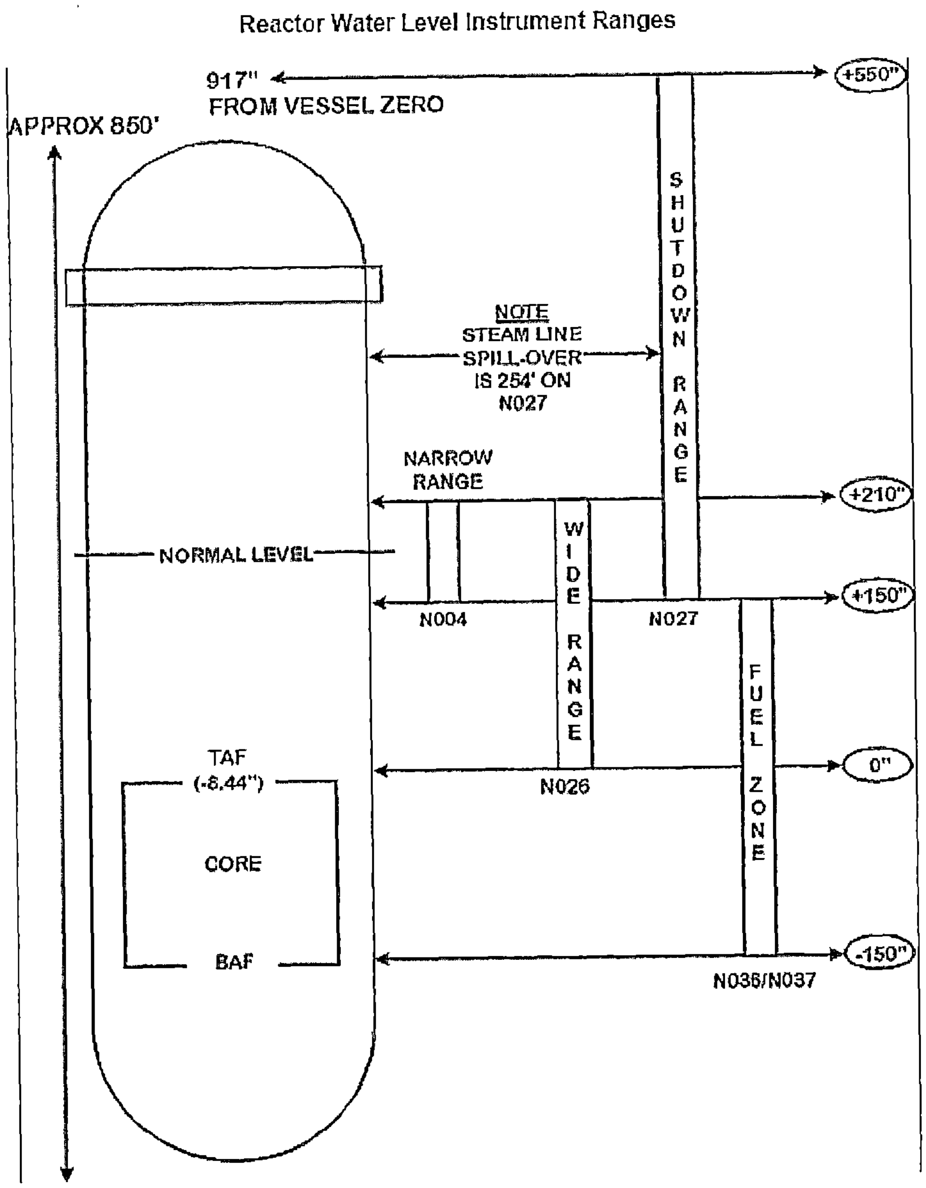
Although related, this EAL is concerned with the loss of RCS inventory and not the potential concurrent effects on systems needed for decay heat removal (e.g., loss of a Decay Heat Removal suction point). An increase in RCS temperature caused by a loss of decay heat removal capability is evaluated under IC CA3.

If RPV water level continues to lower, then escalation to Site Area Emergency would be via IC CS1.

BNP Basis Reference(s):

1. 0EOP-01-NL EOP/SAMG NUMERICAL LIMITS AND VALUES, Table 1E
2. SD-01.2 Reactor Vessel Instrumentation Figure 01.2-1 Reactor Water Level Instrument Ranges

Figure C-1 RPV Levels (ref. 2)



Category: C – Cold Shutdown / Refueling System Malfunction

Subcategory: 1 – RPV Level

Initiating Condition: Loss of RPV inventory

EAL:

CA1.2 Alert
RPV water level cannot be monitored for ≥ 15 min. (Note 1) AND UNPLANNED increase in **any** Table C-1 sump or tank levels due to a loss of RPV inventory

Table C-1 Sumps & Tanks
• Drywell Floor Drain Sump • Drywell Equipment Drain Sump • RB Floor Drain Sump • RB Equipment Drain Sump • Torus • Visual Observation

Mode Applicability:

4 - Cold Shutdown, 5 – Refuel

Definition(s):

UNPLANNED-. A parameter change or an event that is not 1) the result of an intended evolution or 2) an expected plant response to a transient. The cause of the parameter change or event may be known or unknown.

Basis:

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

In this EAL, all water level indication would be unavailable, and the RPV inventory loss must be detected by the leakage indications listed in Table C-1. Sump level increases must be evaluated against other potential sources of leakage. Drywell equipment and floor drain sump level rise is the normal method of monitoring and calculating leakage from the RPV (ref. 1 thru 4). A Reactor Building equipment or floor drain sump level rise may also be indicative of RCS inventory losses external to the Primary Containment from systems connected to the RPV. With RHR System operating in the Shutdown Cooling mode, an unexplained rise in torus level could be indicative of RHR valve misalignment or leakage (ref. 5). 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 that cannot be isolated could be indicative of a loss of RPV inventory.

This IC addresses conditions that are precursors to a loss of the ability to adequately cool irradiated fuel (i.e., a precursor to a challenge to the fuel clad barrier). This condition represents a potential substantial reduction in the level of plant safety.

For this EAL, the inability to monitor RPV level may be caused by instrumentation and/or power failures, or water level dropping below the range of available instrumentation. If water level cannot be monitored, operators may determine that an inventory loss is occurring by observing changes in sump and/or tank levels. Sump and/or tank level changes must be evaluated against other potential sources of water flow to ensure they are indicative of leakage from the RCS.

The 15-minute duration for the loss of level indication was chosen because it is half of the EAL duration specified in IC CS1.

If the RCS inventory level continues to lower, then escalation to Site Area Emergency would be via IC CS1.

BNP Basis Reference(s):

1. 0OP-47 Floor and Equipment Drain System Operating Procedure
2. 1OI-03.1 Control Room Operator Daily Surveillance Report
3. 2OI-03.2 Control Room Operator Daily Surveillance Report
4. 0AOP-14.0 Abnormal Primary Containment Conditions
5. 1(2)OP-17 Residual Heat Removal System Operating Procedure
6. NEI 99-01 CA1

Category: C – Cold Shutdown / Refueling System Malfunction

Subcategory: 1 – RPV Level

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

EAL:

CS1.1 Site Area Emergency
CONTAINMENT CLOSURE **not** established
AND
RPV level < 45 in. (Level 3)

Mode Applicability:

4 - Cold Shutdown, 5 - Refuel

Definition(s):

CONTAINMENT CLOSURE - The procedurally defined conditions or actions taken to secure Primary or Secondary Containment and its associated structures, systems, and components as a functional barrier to fission product release under shutdown conditions.

As applied to BNP, Containment Closure is established when either Primary Containment is Operable per Section 3.6.1.1 of Technical Specifications or Secondary Containment is considered functional per the requirements of 0AP-022, BNP Outage Risk Management.

Basis:

RPV level is normally monitored using the instruments in Figure C-1 (ref. 2).

When RPV level decreases to 45 in., RPV water level is below the low-low-low ECCS actuation setpoint (Level 3) (ref. 1).

The magnitude of this loss of water indicates that makeup systems have not been effective and may not be capable of preventing further RPV water level decrease and potential core uncovery. The inability to restore and maintain level after reaching this setpoint infers a failure of the RCS barrier and Potential Loss of the Fuel Clad barrier.

This IC addresses a significant and prolonged loss of RCS inventory control and makeup capability leading to IMMINENT fuel damage. The lost inventory may be due to a RCS component failure, a loss of configuration control or prolonged boiling of reactor coolant.

These conditions entail major failures of plant functions needed for protection of the public and thus warrant a Site Area Emergency declaration.

Following an extended loss of core decay heat removal and inventory makeup, decay heat will cause reactor coolant boiling and a further reduction in reactor vessel level. If RPV level cannot be restored, fuel damage is probable.

Outage/shutdown contingency plans typically provide for re-establishing or verifying CONTAINMENT CLOSURE following a loss of heat removal or RCS inventory control functions. The difference in the specified RPV levels of CS1.1 and CS1.2 reflect the fact that with CONTAINMENT CLOSURE established, there is a lower probability of a fission product release to the environment.

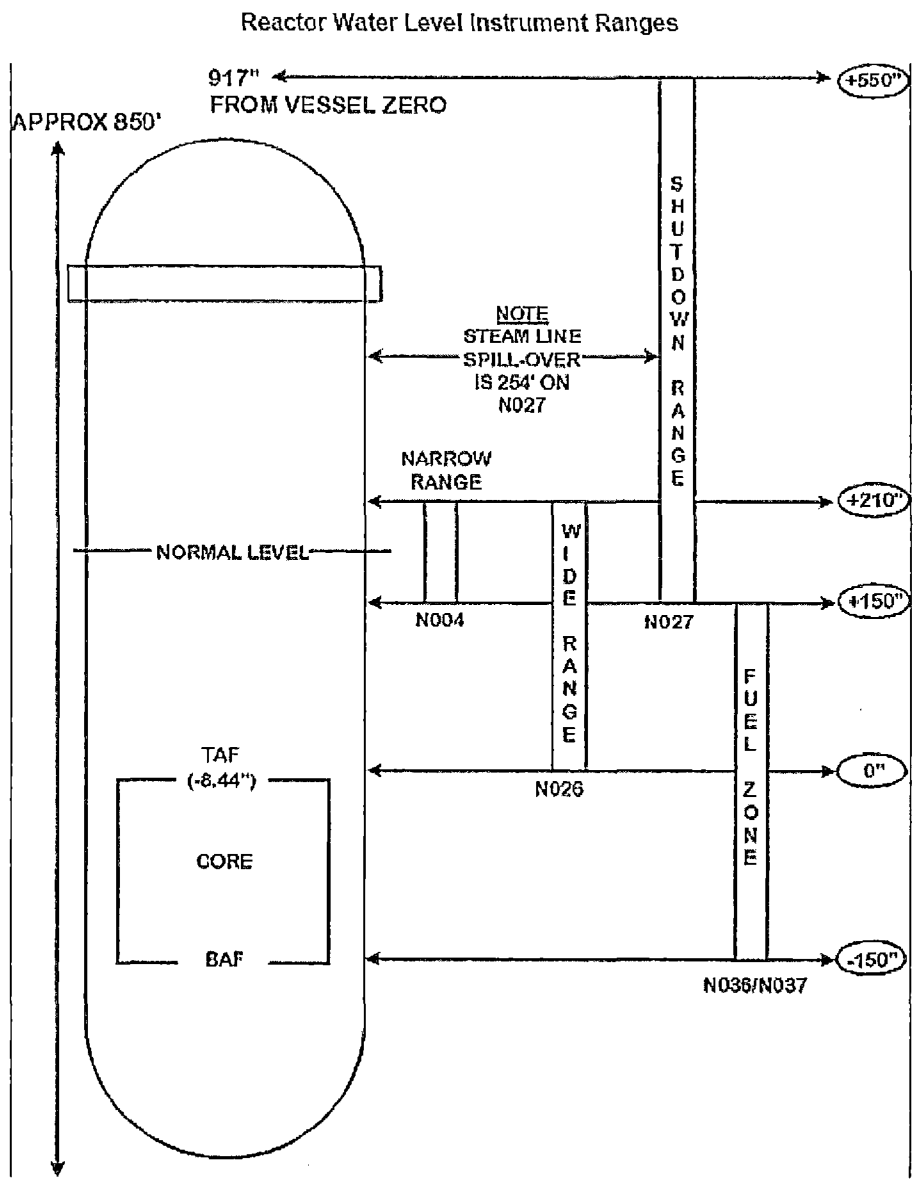
This EAL addresses concerns raised by Generic Letter 88-17, Loss of Decay Heat Removal; SECY 91-283, Evaluation of Shutdown and Low Power Risk Issues; NUREG-1449, Shutdown and Low-Power Operation at Commercial Nuclear Power Plants in the United States; and NUMARC 91-06, Guidelines for Industry Actions to Assess Shutdown Management.

Escalation of the emergency classification level would be via IC CG1 or RG1.

BNP Basis Reference(s):

1. 0EOP-01-NL EOP/SAMG NUMERICAL LIMITS AND VALUES, Table 1E
2. BNP Technical Specifications, Sections 3.6.1.1
3. 0AP-022, BNP Outage Risk Management, Section 6.5
4. SD-01.2 Reactor Vessel Instrumentation Figure 01.2-1 Reactor Water Level Instrument Ranges
5. NEI 99-01 CS1

Figure C-1 RPV Levels (ref. 4)



Category: C – Cold Shutdown / Refueling System Malfunction

Subcategory: 1 – RPV Level

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

EAL:

CS1.2 Site Area Emergency
CONTAINMENT CLOSURE established
AND
RPV level < TAF

Mode Applicability:

4 - Cold Shutdown, 5 - Refuel

Definition(s):

CONTAINMENT CLOSURE - The procedurally defined conditions or actions taken to secure Primary or Secondary Containment and its associated structures, systems, and components as a functional barrier to fission product release under shutdown conditions.

As applied to BNP, Containment Closure is established when either Primary Containment is Operable per Section 3.6.1.1 of Technical Specifications or Secondary Containment is considered functional per the requirements of 0AP-022, BNP Outage Risk Management.

Basis:

When RPV level drops to the top of active fuel (TAF) (an indicated RPV level of ~ -8 in.), core uncovery starts to occur (ref. 1).

This IC addresses a significant and prolonged loss of RPV level control and makeup capability leading to IMMINENT fuel damage. The lost inventory may be due to a RCS component failure, a loss of configuration control or prolonged boiling of reactor coolant. These conditions entail major failures of plant functions needed for protection of the public and thus warrant a Site Area Emergency declaration.

Following an extended loss of core decay heat removal and inventory makeup, decay heat will cause reactor coolant boiling and a further reduction in reactor vessel level. If RPV level cannot be restored, fuel damage is probable.

Outage/shutdown contingency plans typically provide for re-establishing or verifying CONTAINMENT CLOSURE following a loss of heat removal or RCS inventory control functions. The difference in the specified RCS/reactor vessel levels of CS1.1 and CS1.2 reflect the fact that with CONTAINMENT CLOSURE established, there is a lower probability of a fission product release to the environment.

This EAL addresses concerns raised by Generic Letter 88-17, Loss of Decay Heat Removal; SECY 91-283, Evaluation of Shutdown and Low Power Risk Issues; NUREG-1449, Shutdown and Low-Power Operation at Commercial Nuclear Power Plants in the United States; and NUMARC 91-06, Guidelines for Industry Actions to Assess Shutdown Management.

Escalation of the emergency classification level would be via IC CG1 or RG1.

BNP Basis Reference(s):

1. 0EOP-01-NL EOP/SAMG NUMERICAL LIMITS AND VALUES
2. BNP Technical Specifications, Sections 3.6.1.1 and 3.6.4.1
3. 0AP-022, BNP Outage Risk Management, Section 6.5

Category: C – Cold Shutdown / Refueling System Malfunction

Subcategory: 1 – RPV Level

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

EAL:

CS1.3 Site Area Emergency

RPV water level cannot be monitored for ≥ 30 min. (Note 1)

AND

Core uncovery is indicated by **EITHER** of the following:

- UNPLANNED increase in **any** Table C-1 sump or tank levels due to a loss of RPV inventory
- UNPLANNED increase in ARM Channel 28 Between Reactor and Fuel Pool > 1000 mR/hr

Note 1: The SEC should declare the event promptly upon determining that time limit has been exceeded, or will likely be exceeded.

Table C-1 Sumps & Tanks
• Drywell Floor Drain Sump
• Drywell Equipment Drain Sump
• RB Floor Drain Sump
• RB Equipment Drain Sump
• Torus
• Visual Observation

Mode Applicability:

4 – Cold Shutdown, 5 – Refuel

Definition(s):

UNPLANNED-. A parameter change or an event that is not 1) the result of an intended evolution or 2) an expected plant response to a transient. The cause of the parameter change or event may be known or unknown.

Basis:

If RPV level monitoring capability is unavailable, the RPV inventory loss must be detected by the leakage indications listed in Table C-1. Sump level increases must be evaluated against other potential sources of leakage. Drywell equipment and floor drain sump level rise is the normal method of monitoring and calculating leakage from the RPV (ref. 1 thru 4). A Reactor Building equipment or floor drain sump level rise may also be indicative of RCS inventory losses external to the Primary Containment from systems connected to the RPV. With RHR System operating in the Shutdown Cooling mode, an unexplained rise in torus level could be indicative of RHR valve misalignment or leakage (ref. 5). 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.

In the Refueling Mode, as water level in the reactor vessel lowers, the dose rate above the core will increase. The dose rate due to this core shine should result in indications on installed area radiation monitors. ARM Channel 28 Between Reactor and Fuel Pool is located on the Refuel Floor in the Reactor Building and is designed to provide monitoring of radiation due to a fuel handling event or loss of shielding during refueling operations. If this radiation monitor reaches and exceeds the alarm setpoint of 1,000 mr/hr, a loss of inventory with potential to uncover the core is likely to have occurred (Ref. 6, 7).

This IC addresses a significant and prolonged loss of reactor vessel/RCS inventory control and makeup capability leading to IMMINENT fuel damage. The lost inventory may be due to a RCS component failure, a loss of configuration control or prolonged boiling of reactor coolant. These conditions entail major failures of plant functions needed for protection of the public and thus warrant a Site Area Emergency declaration.

Following an extended loss of core decay heat removal and inventory makeup, decay heat will cause reactor coolant boiling and a further reduction in reactor vessel level. If RPV level cannot be restored, fuel damage is probable.

The 30-minute criterion is tied to a readily recognizable event start time (i.e., the total loss of ability to monitor level), and allows sufficient time to monitor, assess and correlate reactor and plant conditions to determine if core uncovery has actually occurred (i.e., to account for various accident progression and instrumentation uncertainties). It also allows sufficient time for performance of actions to terminate leakage, recover inventory control/makeup equipment and/or restore level monitoring.

The inability to monitor RPV level may be caused by instrumentation and/or power failures, or water level dropping below the range of available instrumentation. If water level cannot be monitored, operators may determine that an inventory loss is occurring by observing changes in sump and/or tank levels. Sump and/or tank level changes must be evaluated against other potential sources of water flow to ensure they are indicative of leakage from the RCS.

This EAL addresses concerns raised by Generic Letter 88-17, Loss of Decay Heat Removal; SECY 91-283, Evaluation of Shutdown and Low Power Risk Issues; NUREG-1449, Shutdown and Low-Power Operation at Commercial Nuclear Power Plants in the United States; and NUMARC 91-06, Guidelines for Industry Actions to Assess Shutdown Management.

Escalation of the emergency classification level would be via IC CG1 or RG1

BNP Basis Reference(s):

1. 0OP-47 Floor and Equipment Drain System Operating Procedure
2. 1OI-03.1 Control Room Operator Daily Surveillance Report
3. 2OI-03.2 Control Room Operator Daily Surveillance Report
4. 0AOP-14.0 Abnormal Primary Containment Conditions
5. 1(2)OP-17 Residual Heat Removal System Operating Procedure
6. 1(2)APP-UA-03 4-7
7. DBD-11 Radiation Monitoring System
8. NEI 99-01 CS1

Category: C – Cold Shutdown / Refueling System Malfunction

Subcategory: 1 – RPV Level

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

EAL:

CG1.1   **General Emergency**
RPV level < TAF for ≥ 30 min. (Note 1)
AND
Any Containment Challenge indication, Table C-2

Note 1: The SEC should declare the event promptly upon determining that time limit has been exceeded, or will likely be exceeded.

Note 6: If CONTAINMENT CLOSURE is re-established prior to exceeding the 30-minute time limit, declaration of a General Emergency is **not** required.

Table C-2 Containment Challenge Indications
• CONTAINMENT CLOSURE **not** established (Note 6)
• Primary Containment hydrogen concentration > 6%
• UNPLANNED rise in PC pressure
•

Mode Applicability:

4 - Cold Shutdown, 5 – Refuel

Definition(s):

CONTAINMENT CLOSURE - The procedurally defined conditions or actions taken to secure Primary or Secondary Containment and its associated structures, systems, and components as a functional barrier to fission product release under shutdown conditions.

As applied to BNP, Containment Closure is established when either Primary Containment is Operable per Section 3.6.1.1 of Technical Specifications or Secondary Containment is considered functional per the requirements of 0AP-022, BNP Outage Risk Management.

Basis:

When RPV level drops below ~ -8 in., the top of active fuel, core uncovery starts to occur (ref. 5).

Three conditions are associated with a challenge to Primary Containment (PC) integrity:

- CONTAINMENT CLOSURE is not established (Ref. 6).
- In the early stages of a core uncovery event, it is unlikely that hydrogen buildup due to a core uncovery could result in an explosive mixture of dissolved gases in the Primary Containment. However, Primary Containment monitoring and/or sampling should be performed to verify this assumption and a General Emergency declared if it is determined that an explosive mixture exists. An explosive mixture can be formed when hydrogen gas concentration in the Primary Containment atmosphere is greater than 6% by volume in the presence of oxygen (>5%) (ref. 2, 3). In Cold Shutdown and Refueling modes it is assumed that the Primary Containment is de-inerted.

  Monitors CAC-AT-4409 and 4410 monitor hydrogen and oxygen concentration in the drywell during normal operation. These monitors are isolated by accident isolation signals. However, monitors CAC-AT-4409 and 4410 will be realigned to the primary containment for post-accident monitoring via an operator actuated isolation signal override circuit when directed by the EOPs. (ref. 4).
- Any unplanned increase in PC pressure in the Cold Shutdown or Refueling mode indicates a potential loss of containment closure capability. Unplanned Primary Containment pressure increases indicates containment closure cannot be assured and the Primary Containment cannot be relied upon as a barrier to fission product release.

This IC addresses the inability to restore and maintain reactor vessel level above the top of active fuel with containment challenged. This condition represents actual or IMMINENT substantial core degradation or melting with potential for loss of containment integrity. Releases can be reasonably expected to exceed EPA PAG exposure levels offsite for more than the immediate site area.

Following an extended loss of core decay heat removal and inventory makeup, decay heat will cause reactor coolant boiling and a further reduction in reactor vessel level. If RPV level cannot be restored, fuel damage is probable.

ATTACHMENT 1
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EAL Bases

With CONTAINMENT CLOSURE not established, there is a high potential for a direct and unmonitored release of radioactivity to the environment. If CONTAINMENT CLOSURE is re-established prior to exceeding the 30-minute time limit, then declaration of a General

Emergency is not required.

The existence of an explosive mixture means, at a minimum, that the containment atmospheric hydrogen concentration is sufficient to support a hydrogen burn (i.e., at the lower deflagration limit). A hydrogen burn will raise containment pressure and could result in collateral equipment damage leading to a loss of containment integrity. It therefore represents a challenge to Containment integrity.

In the early stages of a core uncovery event, it is unlikely that hydrogen buildup due to a core uncovery could result in an explosive gas mixture in containment. If all installed hydrogen gas monitors are out-of-service during an event leading to fuel cladding damage, it may not be possible to obtain a containment hydrogen gas concentration reading as ambient conditions within the containment will preclude personnel access. During periods when installed containment hydrogen gas monitors are out-of-service, operators may use the other listed indications to assess whether or not containment is challenged.

The 30-minute criterion is tied to a readily recognizable event start time (i.e., the total loss of ability to monitor level), and allows sufficient time to monitor, assess and correlate reactor and plant conditions to determine if core uncovery has actually occurred (i.e., to account for various accident progression and instrumentation uncertainties). It also allows sufficient time for performance of actions to terminate leakage, recover inventory control/makeup equipment and/or restore level monitoring.

This EAL addresses concerns raised by Generic Letter 88-17, *Loss of Decay Heat Removal*; SECY 91-283, *Evaluation of Shutdown and Low Power Risk Issues*; NUREG-1449, *Shutdown and Low-Power Operation at Commercial Nuclear Power Plants in the United States*; and NUMARC 91-06, *Guidelines for Industry Actions to Assess Shutdown Management*.

BNP Basis Reference(s):

1. BNP Technical Specifications Sections 3.6.1.1 and 3.6.4.1
2. BWROG EPG/SAG Revision 2, Sections PC/G
3. 0EOP-02-PCCP, Primary Containment Control
4. Updated FSAR section 6.2.5.2.2
5. 0EOP-01-NL EOP/SAMG NUMERICAL LIMITS AND VALUES
6. 0AP-022 BNP Outage Risk Management, Section 6.5
7. NEI 99-01 CG1

Category: C – Cold Shutdown / Refueling System Malfunction

Subcategory: 1 – RPV Level

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

EAL:

CG1.2 General Emergency

RPV level **cannot** be monitored for ≥ 30 min. (Note 1)

AND

Core uncovery is indicated by **EITHER** of the following:

- UNPLANNED increase in **any** Table C-1 sump or tank levels due to a loss of RPV inventory
- UNPLANNED increase in ARM Channel 28 Between Reactor and Fuel Pool > 1000 mR/hr

AND

Any Containment Challenge indication, Table C-2

Note 1: The SEC should declare the event promptly upon determining that time limit has been exceeded, or will likely be exceeded.

Note 6: If CONTAINMENT CLOSURE is re-established prior to exceeding the 30-minute time limit, declaration of a General Emergency is not required.

Table C-1 Sumps & Tanks
• Drywell Floor Drain Sump • Drywell Equipment Drain Sump • RB Floor Drain Sump • RB Equipment Drain Sump • Torus • Visual Observation

Table C-2 Containment Challenge Indications
• CONTAINMENT CLOSURE **not** established (Note 6) • Primary Containment hydrogen concentration > 6% • UNPLANNED rise in PC pressure •

Mode Applicability:

4 - Cold Shutdown, 5 – Refuel

Definition(s):

CONTAINMENT CLOSURE - The procedurally defined conditions or actions taken to secure Primary or Secondary Containment and its associated structures, systems, and components as a functional barrier to fission product release under shutdown conditions.

As applied to BNP, Containment Closure is established when either Primary Containment is Operable per Section 3.6.1.1 of Technical Specifications or Secondary Containment is considered functional per the requirements of 0AP-022, BNP Outage Risk Management.

UNPLANNED-. A parameter change or an event that is not 1) the result of an intended evolution or 2) an expected plant response to a transient. The cause of the parameter change or event may be known or unknown.

Basis:

If RPV level monitoring capability is unavailable, the RPV inventory loss must be detected by the leakage indications listed in Table C-1. Sump level increases must be evaluated against other potential sources of leakage. Drywell equipment and floor drain sump level rise is the normal method of monitoring and calculating leakage from the RPV (ref. 1 thru 4). A Reactor Building equipment or floor drain sump level rise may also be indicative of RCS inventory losses external to the Primary Containment from systems connected to the RPV. With RHR System operating in the Shutdown Cooling mode, an unexplained rise in torus level could be indicative of RHR valve misalignment or leakage (ref. 5). 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 that cannot be isolated could be indicative of a loss of RPV inventory.

In the Refueling Mode, as water level in the reactor vessel lowers, the dose rate above the core will increase. The dose rate due to this core shine should result in indications on installed area radiation monitors. ARM Channel 28 Between Reactor and Fuel Pool is located on the

Refuel Floor in the Reactor Building and is designed to provide monitoring of radiation due to a fuel handling event or loss of shielding during refueling operations. If this radiation monitor reaches and exceeds the alarm setpoint of 1,000 mr/hr, a loss of inventory with potential to uncover the core is likely to have occurred (Ref. 6, 7).

Three conditions are associated with a challenge to Primary Containment (PC) integrity:

- CONTAINMENT CLOSURE is not established (Ref. 12).
- In the early stages of a core uncovery event, it is unlikely that hydrogen buildup due to a core uncovery could result in an explosive mixture of dissolved gases in the Primary Containment. However, Primary Containment monitoring and/or sampling should be performed to verify this assumption and a General Emergency declared if it is determined that an explosive mixture exists. An explosive mixture can be formed when hydrogen gas concentration in the Primary Containment atmosphere is greater than 6% by volume in the presence of oxygen (>5%) (ref. 9, 10). In Cold Shutdown and Refueling modes it is assumed that the Primary Containment is de-inerted.

  Monitors CAC-AT-4409 and 4410 monitor hydrogen and oxygen concentration in the drywell during normal operation. These monitors are isolated by accident isolation signals. However, monitors CAC-AT-4409 and 4410 will be realigned to the primary containment for post-accident monitoring via an operator actuated isolation signal override circuit when directed by the EOPs. (ref. 11).
- Any unplanned increase in PC pressure in the Cold Shutdown or Refueling mode indicates a potential loss of containment closure capability. Unplanned Primary Containment pressure increases indicates containment closure cannot be assured and the Primary Containment cannot be relied upon as a barrier to fission product release.

This IC addresses the inability to restore and maintain reactor vessel level above the top of active fuel with containment challenged. This condition represents actual or IMMINENT substantial core degradation or melting with potential for loss of containment integrity. Releases can be reasonably expected to exceed EPA PAG exposure levels offsite for more than the immediate site area.

Following an extended loss of core decay heat removal and inventory makeup, decay heat will cause reactor coolant boiling and a further reduction in reactor vessel level. If RPV level cannot be restored, fuel damage is probable.

With CONTAINMENT CLOSURE not established, there is a high potential for a direct and unmonitored release of radioactivity to the environment. If CONTAINMENT CLOSURE is re-established prior to exceeding the 30-minute time limit, then declaration of a General Emergency is not required.

The existence of an explosive mixture means, at a minimum, that the containment atmospheric hydrogen concentration is sufficient to support a hydrogen burn (i.e., at the lower deflagration limit). A hydrogen burn will raise containment pressure and could result in collateral equipment damage leading to a loss of containment integrity. It therefore represents a challenge to Containment integrity.

In the early stages of a core uncovery event, it is unlikely that hydrogen buildup due to a core uncovery could result in an explosive gas mixture in containment. If all installed hydrogen gas monitors are out-of-service during an event leading to fuel cladding damage, it may not be possible to obtain a containment hydrogen gas concentration reading as ambient conditions within the containment will preclude personnel access. During periods when installed containment hydrogen gas monitors are out-of-service, operators may use the other listed indications to assess whether or not containment is challenged.

The 30-minute criterion is tied to a readily recognizable event start time (i.e., the total loss of ability to monitor level), and allows sufficient time to monitor, assess and correlate reactor and plant conditions to determine if core uncovery has actually occurred (i.e., to account for various accident progression and instrumentation uncertainties). It also allows sufficient time for performance of actions to terminate leakage, recover inventory control/makeup equipment and/or restore level monitoring.

The inability to monitor RPV level may be caused by instrumentation and/or power failures, or water level dropping below the range of available instrumentation. If water level cannot be monitored, operators may determine that an inventory loss is occurring by observing changes in sump and/or tank levels. Sump and/or tank level changes must be evaluated against other potential sources of water flow to ensure they are indicative of leakage from theRCS.

This EAL addresses concerns raised by Generic Letter 88-17, *Loss of Decay Heat Removal*; SECY 91-283, *Evaluation of Shutdown and Low Power Risk Issues*; NUREG-1449, *Shutdown and Low-Power Operation at Commercial Nuclear Power Plants in the United States*; and NUMARC 91-06, *Guidelines for Industry Actions to Assess Shutdown Management.*

BNP Basis Reference(s):

1. 0OP-47 Floor and Equipment Drain System Operating Procedure
2. 1OI-03.1 Control Room Operator Daily Surveillance Report
3. 2OI-03.2 Control Room Operator Daily Surveillance Report
4. 0AOP-14.0 Abnormal Primary Containment Conditions
5. 1(2)OP-17 Residual Heat Removal System Operating Procedure
6. 1(2)APP-UA-03 4-7
7. DBD-11 Radiation Monitoring System
8. BNP Technical Specifications Section 3.6.1.1 and 3.6.4.1
9. BWROG EPG/SAG Revision 2, Sections PC/G
10. 0EOP-02-PCCP Primary Containment Control
11. Updated FSAR section 6.2.5.2.2
12. 0AP-022 BNP Outage Risk Management, Section 6.5
14. NEI 99-01 CG1

Category: C – Cold Shutdown / Refueling System Malfunction

Subcategory: 2 – Loss of Emergency AC Power

Initiating Condition: Loss of all but one AC power source to emergency buses for 15 minutes or longer

EAL:

CU2.1 Unusual Event

AC power capability to Emergency 4 KV Buses E1(E3) and E2(E4) reduced to a single power source, Table C-6, for ≥ 15 min. (Note 1)

AND

Any additional single power source failure will result in loss of **all** unit-specific AC power to SAFETY SYSTEMS

Note 1: The SEC should declare the event promptly upon determining that time limit has been exceeded, or will likely be exceeded.

Table C-6 AC Power Sources
Offsite: • SAT • UAT backfed through MPT (**only** if already aligned) **Onsite:** • UAT via Main Generator • EDG1(3) • EDG2(4)

Mode Applicability:

4 - Cold Shutdown, 5 – Refuel, D - Defueled

Definition(s):

SAFETY SYSTEM - A system required for safe plant operation, cooling down the plant and/or placing it in the cold shutdown condition, including the ECCS. These are typically systems classified as safety-related (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.

Basis:

The 4160 VAC System provides the power requirements for operation and safe shutdown of the plant. 4160 VAC power is divided into Balance of Plant (BOP Bus), and Emergency (E-Bus) distribution. The BOP Buses consist of the Conventional Buses (Common A/B Buses; Buses 1B/2B, 1C/2C, 1D/2D) and Caswell Beach Buses A/B. The Emergency switchgear are Buses E1/E3 (Division I) and E2/E4 (Division II) (Ref. 1, 2).

The E-Buses are normally powered through the respective BOP Buses (1D to E1, 1C to E2, 2D to E3, 2C to E4) via a master/slave breaker arrangement. Each E-Bus has a dedicated Diesel Generator to supply an emergency source of power to safe shutdown loads in the event of a loss of the normal power source or loss of offsite power. The DGs will automatically start and tie onto the E-Buses if the normal power source or offsite power is lost. The DGs can provide power to the E-Buses only. In the event the diesel generator is unavailable for an E-Bus, crosstie capability exists for each E-Bus from the same division of the opposite unit (E1 to E3, E2 to E4). Although the E-Buses within the unit also have crosstie capability, this alignment is not permitted by plant procedures, with the exception of E1 to E2 during specific Alternate Safe Shutdown (ASSD) conditions.

During periods of unit shutdown, when the Startup Auxiliary Transformer (SAT) would be the only normal source of offsite power, the Unit Auxiliary Transformer (UAT) can be made available by establishing a UAT backfeed. Backfeed from the UAT will require the use of keys for the control selector switches and opening of the respective generator's manual no-load disconnect (Ref. 3).

Because 2 RHR pumps on each unit are powered from the unaffected unit, the words "unit-specific" have been added to clarify that the cross-connected RHR pump power cannot be credited as an AC power source relative to this EAL.

This cold condition EAL is equivalent to the hot condition EAL SA1.1.

This IC describes a significant degradation of offsite and onsite AC power sources such that any additional single failure would result in a loss of all AC power to SAFETY SYSTEMS. In this condition, the sole AC power source may be powering one, or more than one, train of safety-related equipment.

When in the cold shutdown, refueling, or defueled mode, this condition is not classified as an Alert because of the increased time available to restore another power source to service. Additional time is available due to the reduced core decay heat load, and the lower temperatures and pressures in various plant systems. Thus, when in these modes, this condition is considered to be a potential degradation of the level of safety of the plant.

An "AC power source" is a source recognized in AOPs, and capable of supplying required power to an emergency bus. Some examples of this condition are presented below.

- A loss of all offsite power with a concurrent failure of one division of emergency power sources (e.g., onsite diesel generators).
- A loss of all offsite power and loss of all emergency power sources (e.g., onsite diesel generators) with a single train of emergency buses being back-fed from the unit main generator.
- A loss of emergency power sources (e.g., onsite diesel generators) with a single division of emergency buses being back-fed from an offsite power source.

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

The subsequent loss of the remaining single power source would escalate the event to an Alert in accordance with IC CA2.

BNP Basis Reference(s):

1. Drawing BN-50.0.01 Electrical Distribution
2. BNP Updated FSAR Chapter 8
3. 1(2)OP-50 Plant Electric System Operating Procedure
4. 1(2)APP-UA15 2-1 (Bus E-1 Undervoltage)
5. 1(2)APP-UA16 2-1 (Bus E-2 Undervoltage)
6. 1(2)APP-UA17 2-1 (Bus E-3 Undervoltage)
7. 1(2)APP-UA18 2-1 (Bus E-4 Undervoltage)
8. NEI 99-01 CU2

Category: C – Cold Shutdown / Refueling System Malfunction

Subcategory: 2 – Loss of Emergency AC Power

Initiating Condition: Loss of **all** offsite and **all** onsite AC power to emergency buses for 15 minutes or longer

EAL:

CA2.1	**Alert**
Loss of **all** offsite and **all** onsite AC power capability to Emergency 4 KV Buses E1(E3) and E2(E4) for ≥ 15 min. (Note 1)	

Note 1: The SEC should declare the event promptly upon determining that time limit has been exceeded, or will likely be exceeded.

Mode Applicability:

4 - Cold Shutdown, 5 - Refuel, D - Defueled

Basis:

The 4160 VAC System provides the power requirements for operation and safe shutdown of the plant. 4160 VAC power is divided into Balance of Plant (BOP Bus), and Emergency (E-Bus) distribution. The BOP Buses consist of the Conventional Buses (Common A/B Buses; Buses 1B/2B, 1C/2C, 1D/2D) and Caswell Beach Buses A/B. The Emergency switchgear are Buses E1/E3 (Division I) and E2/E4 (Division II) (Ref. 1, 2).

The E-Buses are normally powered through the respective BOP Buses (1D to E1, 1C to E2, 2D to E3, 2C to E4) via a master/slave breaker arrangement. Each E-Bus has a dedicated Diesel Generator to supply an emergency source of power to safe shutdown loads in the event of a loss of the normal power source or loss of offsite power. The DGs will automatically start and tie onto the E-Buses if the normal power source or offsite power is lost. The DGs can provide power to the E-Buses only. In the event the diesel generator is unavailable for an E-Bus, crosstie capability exists for each E-Bus from the same division of the opposite unit (E1 to E3, E2 to E4). Although the E-Buses within the unit also have crosstie capability, this alignment is not permitted by plant procedures, with the exception of E1 to E2 during specific Alternate Safe Shutdown (ASSD) conditions.

During periods of unit shutdown, when the Startup Auxiliary Transformer (SAT) would be the only normal source of offsite power, the Unit Auxiliary Transformer (UAT) can be made available by establishing a UAT backfeed. Backfeed from the UAT will require the use of keys for the control selector switches and opening of the respective generator's manual no-load disconnect (Ref. 3).

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

This IC addresses a total loss of AC power that compromises the performance of all SAFETY SYSTEMS requiring electric power including those necessary for emergency core cooling, containment heat removal/pressure control, spent fuel heat removal and the ultimate heat sink.

When in the cold shutdown, refueling, or defueled mode, this condition is not classified as a Site Area Emergency because of the increased time available to restore an emergency bus to service. Additional time is available due to the reduced core decay heat load, and the lower temperatures and pressures in various plant systems. Thus, when in these modes, this condition represents an actual or potential substantial degradation of the level of safety of the plant.

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

Escalation of the emergency classification level would be via IC CS1 or RS1.

BNP Basis Reference(s):

1. Drawing BN-50.0.01 Electrical Distribution
2. BNP Updated FSAR Chapter 8
3. 1(2)OP-50 Plant Electric System Operating Procedure
4. 1(2)APP-UA15 2-1 (Bus E-1 Undervoltage)
5. 1(2)APP-UA16 2-1 (Bus E-2 Undervoltage)
6. 1(2)APP-UA17 2-1 (Bus E-3 Undervoltage)
7. 1(2)APP-UA18 2-1 (Bus E-4 Undervoltage)
8. NEI 99-01 CA2

Category: C – Cold Shutdown / Refueling System Malfunction

Subcategory: 3 – RCS Temperature

Initiating Condition: UNPLANNED increase in RCS temperature

EAL:

CU3.1	**Unusual Event**
UNPLANNED increase in RCS temperature to > 212°F due to loss of decay heat removal capability	

Mode Applicability:

4 - Cold Shutdown, 5 - Refuel

Definition(s):

UNPLANNED-. A parameter change or an event that is not 1) the result of an intended evolution or 2) an expected plant response to a transient. The cause of the parameter change or event may be known or unknown.

Basis:

Several instruments are capable of providing indication of RCS temperature with respect to the Technical Specification cold shutdown temperature limit (212°F, ref. 1). These include (ref. 2):

- Recirculation Suction Temperatures read on B32-TR-R650 located on panel P-603 (if recirculation loop is in operation)
- RHR HX 2A(B) Inlet Temperature as read on E41-TR-R605 Point 1(2), on Panel H12-P614 (RHR HX in service)
- RHR HX 2A(B) Outlet Temperature as read on E41-TR-R605 Point 3(4), on Panel H12-P614 (RHR HX **not** in service)
- PPC Display 815, RPV HEATUP/COOLDOWN MONITOR (natural circulation)

This IC addresses an UNPLANNED increase in RCS temperature above the Technical Specification cold shutdown temperature limitand represents a potential degradation of the level of safety of the plant. If the RCS is not intact and CONTAINMENT CLOSURE is not established during this event, the Site Emergency Coordinator should also refer to IC CA3.

A momentary UNPLANNED excursion above the Technical Specification cold shutdown temperature limit when the heat removal function is available does not warrant a classification.

This EAL involves a loss of decay heat removal capability, or an addition of heat to the RCS in excess of that which can currently be removed, such that reactor coolant temperature cannot be maintained below the cold shutdown temperature limit specified in Technical Specifications. During this condition, there is no immediate threat of fuel damage because the core decay heat load has been reduced since the cessation of power operation.

During an outage, the level in the reactor vessel will normally be maintained at or above the reactor vessel flange. Refueling evolutions that lower water level below the reactor vessel flange are carefully planned and controlled. A loss of forced decay heat removal at reduced inventory may result in a rapid increase in reactor coolant temperature depending on the time after shutdown.

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

Escalation to Alert would be via IC CA1 based on an inventory loss or IC CA3 based on exceeding plant configuration-specific time criteria.

BNP Basis Reference(s):

1. BNP Technical Specifications Table 1.1-1
2. 1(2)PT-01.7 Heatup/Cooldown Monitoring
3. NEI 99-01 CU3

Category: C – Cold Shutdown / Refueling System Malfunction

Subcategory: 3 – RCS Temperature

Initiating Condition: UNPLANNED increase in RCS temperature

EAL:

CU3.2	**Unusual Event**
Loss of **all** RCS temperature and RPV level indication for ≥ 15 min. (Note 1)	

Note 1: The SEC should declare the event promptly upon determining that time limit has been exceeded, or will likely be exceeded.

Mode Applicability:

4 - Cold Shutdown, 5 - Refuel

Definition(s):

None

Basis:

RPV water level is normally monitored using the instruments in Figure C-1 (ref. 1).

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

- Recirculation Suction Temperatures read on B32-TR-R650 located on panel P-603 (if recirculation loop is in operation)
- RHR HX 2A(B) Inlet Temperature as read on E41-TR-R605 Point 1(2), on Panel H12-P614 (RHR HX in service)
- RHR HX 2A(B) Outlet Temperature as read on E41-TR-R605 Point 3(4), on Panel H12-P614 (RHR HX **not** in service)
- PPC Display 815, RPV HEATUP/COOLDOWN MONITOR (natural circulation)

This EAL addresses the inability to determine RCS temperature and RPV level, and represents a potential degradation of the level of safety of the plant. If the RCS is not intact and CONTAINMENT CLOSURE is not established during this event, the Site Emergency Coordinator should also refer to IC CA3.

This EAL reflects a condition where there has been a significant loss of instrumentation capability necessary to monitor RCS conditions and operators would be unable to monitor key parameters necessary to assure core decay heat removal. During this condition, there is no immediate threat of fuel damage because the core decay heat load has been reduced since the cessation of power operation.

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

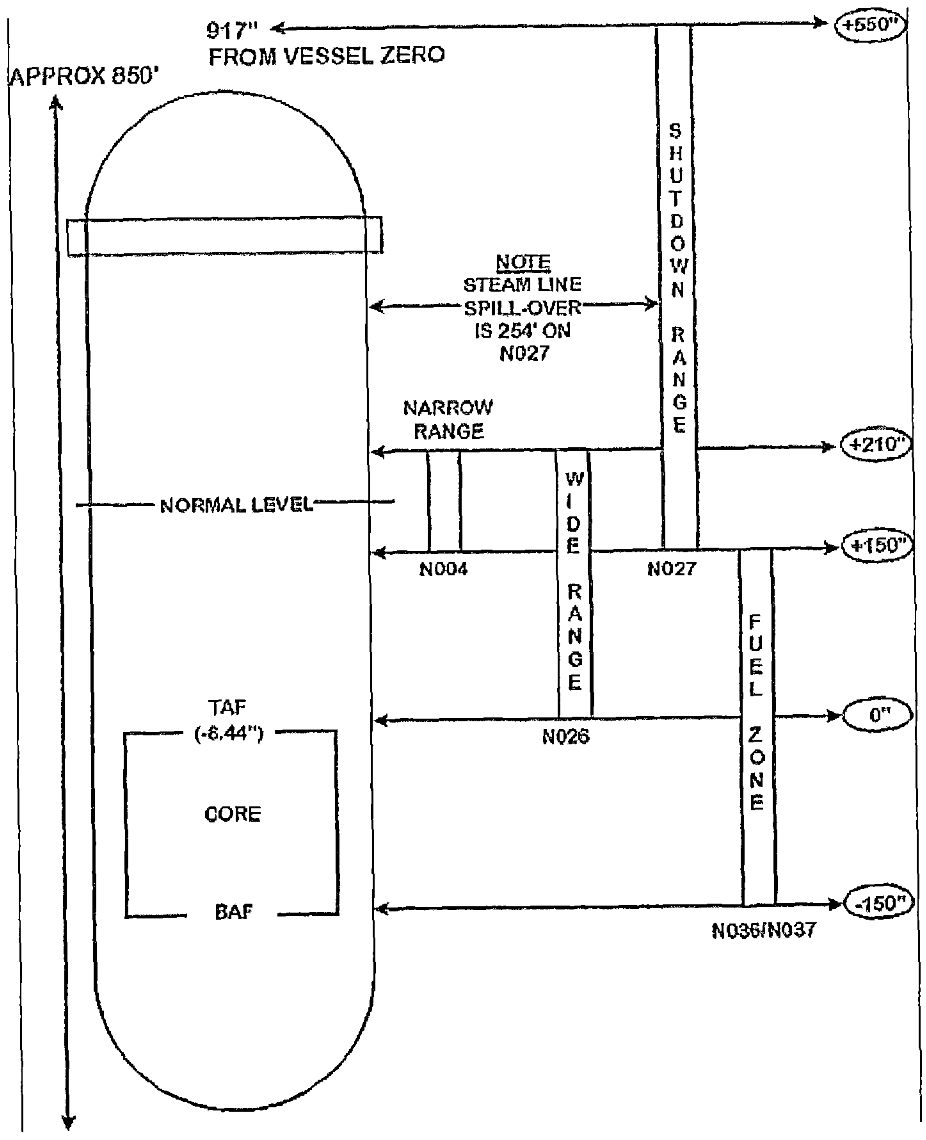
Escalation to Alert would be via IC CA1 based on an inventory loss or IC CA3 based on exceeding plant configuration-specific time criteria.

BNP Basis Reference(s):

1. SD-01.2 Reactor Vessel Instrumentation Figure 01.2-1 Reactor Water Level Instrument Ranges
2. Technical Specifications Table 1.1-1
3. 1(2)PT-01.7 Heatup/Cooldown Monitoring
4. NEI 99-01 CU3

Figure C-1 RPV Levels (ref. 1)

Reactor Water Level Instrument Ranges



Category: C – Cold Shutdown / Refueling System Malfunction

Subcategory: 3 – RCS Temperature

Initiating Condition: Inability to maintain plant in cold shutdown

EAL:

CA3.1 Alert

UNPLANNED increase in RCS temperature to > 212°F for > Table C-3 duration (Note 1)

OR

UNPLANNED RPV pressure increase > 10 psig due to a loss of RCS cooling

Note 1: The SEC should declare the event promptly upon determining that the applicable time has been exceeded, or will likely be exceeded.

Table C-3: RCS Heat-up Duration Thresholds		
RCS Status	**Containment Closure Status**	**Heat-up 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

Definition(s):

CONTAINMENT CLOSURE - The procedurally defined conditions or actions taken to secure Primary or Secondary Containment and its associated structures, systems, and components as a functional barrier to fission product release under shutdown conditions.

As applied to BNP, Containment Closure is established when either Primary Containment is Operable per Section 3.6.1.1 of Technical Specifications or Secondary Containment is considered functional per the requirements of 0AP-022, BNP Outage Risk Management.

UNPLANNED -. A parameter change or an event that is not 1) the result of an intended evolution or 2) an expected plant response to a transient. The cause of the parameter change or event may be known or unknown.

Basis:

A 10 psig RPV pressure increase can be read on (ref. 1):

- Indicator PI-R605A located on Panel P603
- Indicator PI-R605B located on Panel P601
- Recorder LPR-R608 located on P603
- Indicator C32-PI-3332 located on the Remote Shutdown Panel

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

- Recirculation Suction Temperatures read on B32-TR-R650 located on panel P-603 (if recirculation loop is in operation)
- RHR HX 2A(B) Inlet Temperature as read on E41-TR-R605 Point 1(2), on Panel H12-P614 (RHR HX in service)
- RHR HX 2A(B) Outlet Temperature as read on E41-TR-R605 Point 3(4), on Panel H12-P614 (RHR HX **not** in service)
- PPC Display 815, RPV HEATUP/COOLDOWN MONITOR (natural circulation)

This IC addresses conditions involving a loss of decay heat removal capability or an addition of heat to the RCS in excess of that which can currently be removed. Either condition represents an actual or potential substantial degradation of the level of safety of the plant.

A momentary UNPLANNED excursion above the Technical Specification cold shutdown temperature limit when the heat removal function is available does not warrant a classification.

The RCS Heat-up Duration Thresholds table addresses an increase in RCS temperature when CONTAINMENT CLOSURE is established but the RCS is not intact.. The 20-minute criterion was included to allow time for operator action to address the temperature increase.

The RCS Heat-up Duration Thresholds table also addresses an increase in RCS temperature with the RCS intact. The status of CONTAINMENT CLOSURE is not crucial in this condition since the intact RCS is providing a high pressure barrier to a fission product release. The 60-minute time frame should allow sufficient time to address the temperature increase without a substantial degradation in plant safety.

Finally, in the case where there is an increase in RCS temperature, the RCS is not intact , and CONTAINMENT CLOSURE is not established, no heat-up duration is allowed (i.e., 0 minutes). This is because 1) the evaporated reactor coolant may be released directly into the Primary Containment or Reactor Building atmosphere and subsequently to the environment, and 2) there is reduced reactor coolant inventory above the top of irradiated fuel.

The RCS pressure increase threshold provides a pressure-based indication of RCS heat-up in the absence of RCS temperature monitoring capability.

Escalation of the emergency classification level would be via IC CS1 or RS1.

BNP Basis Reference(s):

1. Reactor Vessel Instrumentation System Description SD-01.2
2. BNP Technical Specifications Table 1.1-1
3. 1(2)PT-01.7 Heatup/Cooldown Monitoring
4. Technical Specifications Sections 3.6.1.1 and 3.6.4.1
5. 0AP-022, BNP Outage Risk Management
6. NEI 99-01 CA3

Category: C – Cold Shutdown / Refueling System Malfunction

Subcategory: 4 – Loss of Vital DC Power

Initiating Condition: Loss of Vital DC power for 15 minutes or longer

EAL:

CU4.1	**Unusual Event**
< 105 VDC bus voltage indications on Technical Specification **required** 125 VDC buses for ≥ 15 min. (Note 1)	

Note 1: The SEC should declare the event promptly upon determining that time limit has been exceeded, or will likely be exceeded.

Mode Applicability:

4 - Cold Shutdown, 5 - Refuel

Definition(s):

None

Basis:

There are two independent divisions per unit, designated Division I and Division II (Batteries 1(2)A-1 and 1(2)A-2 for Division I and Batteries 1(2)B-1 and 1(2)B-2 for Division II) . Each division consists of a 250 VDC battery center tapped to form two 125 VDC batteries. Each 125 VDC battery has an associated full capacity battery charger. The chargers are supplied from the same AC load groups for which the associated DC subsystem supplies the control power. During normal operation, the DC loads are powered from the battery chargers with the batteries floating on the system. In case of loss of normal power to the battery charger, the DC loads are automatically powered from the station batteries.

105 VDC is the minimum design voltage limit (ref. 1).

Note that the Control Room DC voltage indicator only reads battery charger output voltage and not battery voltage unless the charger output breaker is closed. However ERFIS does provide DC battery voltage, otherwise battery voltage must be read locally.

In the event that DC battery voltage indication is not available via ERFIS, local voltage indication is available for each bus based on dispatching a field operator in accordance with 0AOP-39.0 Loss of DC Power. In this case the 15 minute classification clock begins upon receipt of the low voltage alarm in the Control Room. If battery voltage cannot be verified to be greater than or equal to 105 VDC within the 15 minutes, emergency classification must be made under this EAL.

This EAL is the cold condition equivalent of the hot condition loss of DC power EAL SS7.1.

This IC addresses a loss of vital DC power which compromises the ability to monitor and control operable SAFETY SYSTEMS when the plant is in the cold shutdown or refueling mode. In these modes, the core decay heat load has been significantly reduced, and coolant system temperatures and pressures are lower; these conditions increase the time available to restore a vital DC bus to service. Thus, this condition is considered to be a potential degradation of the level of safety of the plant.

As used in this EAL, "required" means the vital DC buses necessary to support operation of the in-service, or operable, train or trains of SAFETY SYSTEM equipment. For example, if Division I is out-of-service (inoperable) for scheduled outage maintenance work and Division II is in-service (operable), then a loss of Vital DC power affecting Division II would require the declaration of an Unusual Event. A loss of Vital DC power to Division I would not warrant an emergency classification.

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

Depending upon the event, escalation of the emergency classification level would be via IC CA1 or CA3, or an IC in Recognition Category R.

BNP Basis Reference(s):

1. BNP Technical Specification Bases B.3.8.4
2. 0AOP-39.0 LOSS OF DC POWER
3. NEI 99-01 CU4

Category: C – Cold Shutdown / Refueling System Malfunction

Subcategory: 5 – Loss of Communications

Initiating Condition: Loss of **all** onsite or offsite communications capabilities

EAL:

CU5.1 Unusual Event

Loss of **all** Table C-4 onsite communication methods

OR

Loss of **all** Table C-4 Offsite communication methods

OR

Loss of **all** Table C-4 NRC communication methods

Table C-4 Communication Methods

System	Onsite	Offsite	NRC
Public Address System	X		
PBX Telephone System	X	X	X
DEMNET		X	X
Commercial Telephones	X	X	X
Satellite Phones		X	X
Cellular Phones		X	X
NRC Emergency Telecommunications System			X

Mode Applicability:

4 - Cold Shutdown, 5 - Refuel, D – Defueled

Definition(s):

None

Basis:

Onsite/offsite/NRC communications include one or more of the systems listed in Table C-4 (ref. 1).

Public Address System

The Brunswick Plant public address system provides paging and party line communications between stations located throughout the plant. Inside and outside type wall and desk-mounted stations are used to communicate between roaming personnel and fixed work locations. Plant-wide instructions are issued using the paging feature. This system is powered from the plant uninterruptible power supply which employs battery reserve as well as diesel generator emergency supply.

PBX Telephone System

The Brunswick Site PBX telephone system provides communication capability between telephone stations located within the plant by dialing the four-digit telephone station code. The PBX telephone system also provides for outside communications. The PBX switch located in the TSC/EOF building is also backed up by a battery UPS capable of supplying power for a minimum of 8 hours and is augmented by a Diesel Generator capable of supplying power to the TSC/EOF building for at least 5 days.

DEMNET

DEMNET is the primary means of offsite communication. This circuit allows intercommunication among the EOF, TSC, control room, counties, and states. DEMNET operates as an internet based (VoIP) communications system with a satellite back-up. Should the internet transfer rate become slow or unavailable, the DEMNET will automatically transfer to satellite mode.

Commercial Telephones

Commercial telephone lines, which supply public telephone communications, are employed by Duke Energy in four ways: (1) tie-ins through the PBX to any other plant location, (2) lines to plant emergency facilities, (3) lines to the Joint Information Center for public information purposes, and (4) lines to the AEF. The local service provider provides primary and secondary power for their lines at the Central Office.

Satellite Phones

A total of three portable satellite telephones are available which enable communication when all other phone systems are inoperable, e.g. following a major hurricane. These portable systems can be powered by internal batteries, external DC sources as well as external AC sources. Two of these phones require outside use, while one phone may used either outside or in the EOF with a permanently mounted external antenna.

Cellular Phones

Selected plant personnel are provided with cellular telephones. These phones may be used during emergencies if other communications means are not readily available or are inoperable. These phones are not expected to be used in the Control Room or Power Block due to interference with plant equipment and loss of signal to the phone.

NRC Emergency Telecommunications System

The NRC uses a Duke Energy dedicated telephone line which allows direct telephone communications from the plant to NRC regional and national offices. The Duke Energy communications line provides a link independent of the local public telephone network. Telephones connected to this network are located in the Brunswick Control Room, Technical Support Center, and Emergency Operations Facility. These lines will not function if the PBX Telephone System fails.

This EAL is the cold condition equivalent of the hot condition EAL SU7.1.

This IC addresses a significant loss of onsite or offsite communications capabilities. While not a direct challenge to plant or personnel safety, this event warrants prompt notifications to OROs and the NRC.

This IC should be assessed only when extraordinary means are being utilized to make communications possible (e.g., use of non-plant, privately owned equipment, relaying of onsite information via individuals or multiple radio transmission points, individuals being sent to offsite locations, etc.).

The first EAL condition addresses a total loss of the communications methods used in support of routine plant operations.

The second EAL condition addresses a total loss of the communications methods used to notify all OROs of an emergency declaration. The OROs referred to here are the State, Brunswick and New Hanover County EOCs

The third EAL addresses a total loss of the communications methods used to notify the NRC of an emergency declaration.

BNP Basis Reference(s):

1. 0ERP Radiological Emergency Response Plan Appendix A
2. SD-48 Communication Systems
3. NEI 99-01 CU5

Category: C – Cold Shutdown / Refueling System Malfunction

Subcategory: 6 – Hazardous Event Affecting Safety Systems

Initiating Condition: Hazardous event affecting a SAFETY SYSTEM needed for the current operating mode

EAL:

CA6.1 Alert

The occurrence of **any** Table C-5 hazardous event

AND

EITHER of the following:

- Event damage has caused indications of degraded performance in at least one train of a SAFETY SYSTEM needed for the current operating mode
- The event has caused VISIBLE DAMAGE to a SAFETY SYSTEM component or structure needed for the current operating mode

Table C-5 Hazardous Events
• Seismic event (earthquake) • Internal or external FLOODING event • High winds or tornado strike • FIRE • EXPLOSION • Other events with similar hazard characteristics as determined by the Shift Manager

Mode Applicability:

4 - Cold Shutdown, 5 - Refuel

Definition(s):

EXPLOSION - A rapid, violent and catastrophic failure of a piece of equipment due to combustion, chemical reaction or overpressurization. A release of steam (from high energy lines or components) or an electrical component failure (caused by short circuits, grounding, arcing, etc.) should not automatically be considered an explosion. Such events require a post-event inspection to determine if the attributes of an explosion are present.

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.

FLOODING - A condition where water is entering a room or area faster than installed equipment is capable of removal, resulting in a rise of water level within the room or area.

SAFETY SYSTEM - A system required for safe plant operation, cooling down the plant and/or placing it in the cold shutdown condition, including the ECCS. These are typically systems classified as safety-related (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.

VISIBLE DAMAGE - Damage to a component or structure that is readily observable without measurements, testing, or analysis. The visual impact of the damage is sufficient to cause concern regarding the operability or reliability of the affected component or structure.

Basis:

- The significance of seismic events are discussed under EAL HU2.1 (ref. 1).
- Internal FLOODING may be caused by events such as component failures, equipment misalignment, or outage activity mishaps (ref. 2, 3).
- Seismic Category I structures are analyzed to withstand a sustained, design wind velocity of at least 135 mph. (ref. 4).

- Areas containing functions and systems required for safe shutdown of the plant are identified by Fire Zone in the fire response procedure (ref. 5, 6).
- An explosion that degrades the performance of a SAFETY SYSTEM train or visibly damages a SAFETY SYSTEM component or structure would be classified under this EAL.

This IC addresses a hazardous event that causes damage to a SAFETY SYSTEM, or a structure containing SAFETY SYSTEM components, needed for the current operating mode. This condition significantly reduces the margin to a loss or potential loss of a fission product barrier, and therefore represents an actual or potential substantial degradation of the level of safety of the plant.

The first conditional addresses damage to a SAFETY SYSTEM train that is in service/operation since indications for it will be readily available. The indications of degraded performance should be significant enough to cause concern regarding the operability or reliability of the SAFETY SYSTEM train.

The second conditional addresses damage to a SAFETY SYSTEM component that is not in service/operation or readily apparent through indications alone, or to a structure containing SAFETY SYSTEM components. Operators will make this determination based on the totality of available event and damage report information. This is intended to be a brief assessment not requiring lengthy analysis or quantification of the damage.

Escalation of the emergency classification level would be via IC CS1 or RS1.

BNP Basis Reference(s):

1. 1(2)APP-UA-28 6-4 Seismic Event
2. 0AOP-13.0 Operations During Hurricane, Flood Conditions, Tornado or Earthquake
3. Updated FSAR section 3.4.2 Protection From Internal Flooding
4. Updated FSAR Section 2.3.1.2.7
5. BNP-E-9. 010 NFPA 805 Nuclear Safety Capability Assessment (NSCA)
6. 0PFP-PBAA Power Block Auxiliary Areas Prefire Plan
7. NEI 99-01 CA6

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.

1. Security

Unauthorized entry attempts into the Protected Area, bomb threats, sabotage attempts, and actual security compromises threatening loss of physical control of the plant.

2. Seismic Event

Natural events such as earthquakes have potential to cause plant structure or equipment damage of sufficient magnitude to threaten personnel or plant safety.

3. Natural or Technology Hazard

Other natural and non-naturally occurring events that can cause damage to plant facilities include tornados, FLOODING, hazardous material releases and events restricting site access warranting classification.

4. Fire

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

5. Hazardous Gas

Toxic, corrosive, asphyxiant or flammable gas leaks can affect normal plant operations or preclude access to plant areas required to safely shutdown the plant.

6. 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.

7. SEC 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 Site Emergency Coordinator the latitude to classify emergency conditions consistent with the established classification criteria based upon Site Emergency Coordinator judgment.

Category: H – Hazards

Subcategory: 1 – Security

Initiating Condition: Confirmed SECURITY CONDITION or threat

EAL:

HU1.1	**Unusual Event**
A SECURITY CONDITION that does **not** involve a HOSTILE ACTION as reported by the Security Shift Supervision	

Mode Applicability:

All

Definition(s):

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.

HOSTILE ACTION - An act toward BNP 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 BNP. Non-terrorism-based EALs should be used to address such activities (i.e., this may include violent acts between individuals in the owner controlled area).

Basis:

The Security Shift Supervision is defined as either the Security Shift Lieutenant or the Security Shift Sergeant.

This EAL is based on the BNP Physical Security Plan (ref. 1).

This IC addresses events that pose a threat to plant personnel or SAFETY SYSTEM equipment, and thus represent a potential degradation in the level of plant safety. Security events which do not meet one of these EALs are adequately addressed by the requirements of 10 CFR § 73.71 or 10 CFR § 50.72. Security events assessed as HOSTILE ACTIONS are classifiable under ICs HA1, HS1 and HG1.

Timely and accurate communications between Security Shift Supervision and the Control Room is essential for proper classification of a security-related event. Classification of these events will initiate appropriate threat-related notifications to plant personnel and Offsite Response Organizations.

Security plans and terminology are based on the guidance provided by NEI 03-12, *Template for the Security Plan, Training and Qualification Plan, Safeguards Contingency Plan*.

This EAL references the Security Shift Supervison because these are the individuals trained to confirm that a security event is occurring or has occurred. Training on security event confirmation and classification is controlled due to the nature of Safeguards and 10 CFR § 2.39 information.

Emergency plans and implementing procedures are public documents; therefore, EALs should not incorporate Security-sensitive information. This includes information that may be advantageous to a potential adversary, such as the particulars concerning a specific threat or threat location. Security-sensitive information should be contained in non-public documents such as the BNP Physical Security Plan (ref. 1).

Escalation of the emergency classification level would be via IC HA1.

BNP Basis Reference(s):

1. BNP Physical Security Plan
2. SEC-NGGC-2170 Security Event Procedures
3. 0AOP-40.0 Security Events
4. NEI 99-01 HU1

Category: H – Hazards

Subcategory: 1 – Security

Initiating Condition: Confirmed SECURITY CONDITION or threat

EAL:

HU1.2	**Unusual Event**
Notification of a credible security threat directed at the site	

Mode Applicability:

All

Definition(s):

None

Basis:

The Security Shift Supervision is defined as either the Security Shift Lieutenant or the Security Shift Sergeant.

This EAL is based on the BNP Physical Security Plan (ref. 1).

This IC addresses events that pose a threat to plant personnel or SAFETY SYSTEM equipment, and thus represent a potential degradation in the level of plant safety. Security events which do not meet one of these EALs are adequately addressed by the requirements of 10 CFR § 73.71 or 10 CFR § 50.72. Security events assessed as HOSTILE ACTIONS are classifiable under ICs HA1, HS1 and HG1.

Timely and accurate communications between Security Shift Supervision and the Control Room is essential for proper classification of a security-related event. Classification of these events will initiate appropriate threat-related notifications to plant personnel and Offsite Response Organizations.

Security plans and terminology are based on the guidance provided by NEI 03-12, *Template for the Security Plan, Training and Qualification Plan, Safeguards Contingency Plan.*

This EAL addresses the receipt of a credible security threat. The credibility of the threat is assessed in accordance with the BNP Physical Security Plan (ref. 1).

Emergency plans and implementing procedures are public documents; therefore, EALs should not incorporate Security-sensitive information. This includes information that may be advantageous to a potential adversary, such as the particulars concerning a specific threat or threat location. Security-sensitive information should be contained in non-public documents such as the BNP Physical Security Plan (ref. 1).

Escalation of the emergency classification level would be via IC HA1.

BNP Basis Reference(s):

1. BNP Physical Security Plan
2. SEC-NGGC-2170 Security Event Procedures
3. 0AOP-40.0 Security Events
4. NEI 99-01 HU1

Category: H – Hazards

Subcategory: 1 – Security

Initiating Condition: Confirmed SECURITY CONDITION or threat

EAL:

HU1.3	**Unusual Event**
A validated notification from the NRC providing information of an aircraft threat	

Mode Applicability:

All

Definition(s):

None

Basis:

The Security Shift Supervision is defined as either the Security Shift Lieutenant or the Security Shift Sergeant.

This EAL is based on the BNP Physical Security Plan (ref. 1).

This IC addresses events that pose a threat to plant personnel or SAFETY SYSTEM equipment, and thus represent a potential degradation in the level of plant safety. Security events which do not meet one of these EALs are adequately addressed by the requirements of 10 CFR § 73.71 or 10 CFR § 50.72. Security events assessed as HOSTILE ACTIONS are classifiable under ICs HA1, HS1 and HG1.

Timely and accurate communications between Security Shift Supervision and the Control Room is essential for proper classification of a security-related event. Classification of these events will initiate appropriate threat-related notifications to plant personnel and Offsite Response Organizations.

Security plans and terminology are based on the guidance provided by NEI 03-12, *Template for the Security Plan, Training and Qualification Plan, Safeguards Contingency Plan.*

This EAL addresses the threat from the impact of an aircraft on the plant. The NRC Headquarters Operations Officer (HOO) will communicate to the licensee if the threat involves an aircraft. The status and size of the plane may also be provided by NORAD through the NRC. Validation of the threat is performed in accordance with the BNP Physical Security Plan (ref. 1).

Emergency plans and implementing procedures are public documents; therefore, EALs should not incorporate Security-sensitive information. This includes information that may be advantageous to a potential adversary, such as the particulars concerning a specific threat or threat location. Security-sensitive information should be contained in non-public documents such as the BNP Physical Security Plan (ref. 1).

Escalation of the emergency classification level would be via IC HA1.

BNP Basis Reference(s):

1. BNP Physical Security Plan
2. SEC-NGGC-2170 Security Event Procedures
3. 0AOP-40.0 Security Events
4. NEI 99-01 HU1

Category: H – Hazards

Subcategory: 1 – Security

Initiating Condition: Hostile action within the owner controlled area or airborne attack threat within 30 minutes

EAL:

HA1.1	**Alert**
A HOSTILE ACTION is occurring or has occurred within the OWNER CONTROLLED AREA as reported by the Security Shift Supervision	

Mode Applicability:

All

Definition(s):

HOSTILE ACTION - An act toward BNP 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 BNP. Non-terrorism-based EALs should be used to address such activities (i.e., this may include violent acts between individuals in the owner controlled area).

OWNER CONTROLLED AREA - Area depicted as the property boundary in BNP Radiological Emergency Response Plan Figure 1-1.1 Brunswick Site Plan.

Basis:

The Security Shift Supervision is defined as either the Security Shift Lieutenant or the Security Shift Sergeant.

This IC addresses the occurrence of a HOSTILE ACTION within the OWNER CONTROLLED AREA or notification of an aircraft attack threat. This event will require rapid response and assistance due to the possibility of the attack progressing to the PROTECTED AREA, or the need to prepare the plant and staff for a potential aircraft impact.

Timely and accurate communications between the Security Shift Supervision and the Control Room is essential for proper classification of a security-related event.

Security plans and terminology are based on the guidance provided by NEI 03-12, *Template for the Security Plan, Training and Qualification Plan, Safeguards Contingency Plan.*

As time and conditions allow, these events require a heightened state of readiness by the plant staff and implementation of onsite protective measures (e.g., evacuation, dispersal or sheltering). The Alert declaration will also heighten the awareness of Offsite Response Organizations (OROs), allowing them to be better prepared should it be necessary to consider further actions.

This IC does not apply to incidents that are accidental events, acts of civil disobedience, or otherwise are not a HOSTILE ACTION perpetrated by a HOSTILE FORCE. Examples include the crash of a small aircraft, shots from hunters, physical disputes between employees, etc. Reporting of these types of events is adequately addressed by other EALs, or the requirements of 10 CFR § 73.71 or 10 CFR § 50.72.

This EAL is applicable for any HOSTILE ACTION occurring, or that has occurred, in the OWNER CONTROLLED AREA.

Emergency plans and implementing procedures are public documents; therefore, EALs should not incorporate Security-sensitive information. This includes information that may be advantageous to a potential adversary, such as the particulars concerning a specific threat or threat location. Security-sensitive information should be contained in non-public documents such as the BNP Physical Security Plan (ref. 1).

BNP Basis Reference(s):

1. BNP Physical Security Plan
2. SEC-NGGC-2170 Security Event Procedures
3. 0AOP-40.0 Security Events
4. NEI 99-01 HA1

Category: H – Hazards

Subcategory: 1 – Security

Initiating Condition: Hostile action within the owner controlled area or airborne attack threat within 30 minutes

EAL:

HA1.2 Alert

A validated notification from NRC of an aircraft attack threat within 30 min. of the site

Mode Applicability:

All

Definition(s):

None

Basis:

The Security Shift Supervision is defined as either the Security Shift Lieutenant or the Security Shift Sergeant.

This IC addresses the occurrence of a HOSTILE ACTION within the OWNER CONTROLLED AREA or notification of an aircraft attack threat. This event will require rapid response and assistance due to the possibility of the attack progressing to the PROTECTED AREA, or the need to prepare the plant and staff for a potential aircraft impact.

Timely and accurate communications between the Security Shift Supervision and the Control Room is essential for proper classification of a security-related event.

Security plans and terminology are based on the guidance provided by NEI 03-12, *Template for the Security Plan, Training and Qualification Plan, Safeguards Contingency Plan.*

As time and conditions allow, these events require a heightened state of readiness by the plant staff and implementation of onsite protective measures (e.g., evacuation, dispersal or sheltering). The Alert declaration will also heighten the awareness of Offsite Response Organizations (OROs), allowing them to be better prepared should it be necessary to consider further actions.

This IC does not apply to incidents that are accidental events, acts of civil disobedience, or otherwise are not a HOSTILE ACTION perpetrated by a HOSTILE FORCE. Examples include the crash of a small aircraft, shots from hunters, physical disputes between employees, etc.

Reporting of these types of events is adequately addressed by other EALs, or the requirements of 10 CFR § 73.71 or 10 CFR § 50.72.

This EAL addresses the threat from the impact of an aircraft on the plant, and the anticipated arrival time is within 30 minutes. The intent of this EAL is to ensure that threat-related notifications are made in a timely manner so that plant personnel and OROs are in a heightened state of readiness. This EAL is met when the threat-related information has been validated in accordance with site-specific security procedures.

The NRC Headquarters Operations Officer (HOO) will communicate to the licensee if the threat involves an aircraft. The status and size of the plane may be provided by NORAD through the NRC.

In some cases, it may not be readily apparent if an aircraft impact within the OWNER CONTROLLED AREA was intentional (i.e., a HOSTILE ACTION). It is expected, although not certain, that notification by an appropriate Federal agency to the site would clarify this point. In this case, the appropriate federal agency is intended to be NORAD, FBI, FAA or NRC. The emergency declaration, including one based on other ICs/EALs, should not be unduly delayed while awaiting notification by a Federal agency.

Emergency plans and implementing procedures are public documents; therefore, EALs should not incorporate Security-sensitive information. This includes information that may be advantageous to a potential adversary, such as the particulars concerning a specific threat or threat location. Security-sensitive information should be contained in non-public documents such as the BNP Physical Security Plan (ref. 1).

BNP Basis Reference(s):

1. BNP Physical Security Plan
2. SEC-NGGC-2170 Security Event Procedures
3. 0AOP-40.0 Security Events
4. NEI 99-01 HA1

Category: H – Hazards

Subcategory: 1 – Security

Initiating Condition: Hostile Action within the Protected Area

EAL:

HS1.1 **Site Area Emergency**
A HOSTILE ACTION is occurring or has occurred within the PROTECTED AREA as reported by the Security Shift Supervision

Mode Applicability:

All

Definition(s):

HOSTILE ACTION - An act toward BNP 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 BNP. Non-terrorism-based EALs should be used to address such activities (i.e., this may include violent acts between individuals in the owner controlled area).

PROTECTED AREA - The double-fenced security area with intrusion detection devices immediately surrounding the plant structures. The Protected Area is depicted in BNP Radiological Emergency Response Plan Figure 1-1.3 Brunswick Site Building and Onsite Emergency Facility Locations.

Basis:

The Security Shift Supervision is defined as either the Security Shift Lieutenant or the Security Shift Sergeant.

These individuals are the designated onsite personnel 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 BNP Physical Security Plan (Safeguards) information. (ref. 1)

This IC addresses the occurrence of a HOSTILE ACTION within the PROTECTED AREA. This event will require rapid response and assistance due to the possibility for damage to plant equipment.

Timely and accurate communications between Security Shift Supervision and the Control Room is essential for proper classification of a security-related event.

Security plans and terminology are based on the guidance provided by NEI 03-12, *Template for the Security Plan, Training and Qualification Plan, Safeguards Contingency Plan.*

As time and conditions allow, these events require a heightened state of readiness by the plant staff and implementation of onsite protective measures (e.g., evacuation, dispersal or sheltering). The Site Area Emergency declaration will mobilize Offsite Response Organization (ORO) resources and have them available to develop and implement public protective actions in the unlikely event that the attack is successful in impairing multiple safety functions.

This IC does not apply to incidents that are accidental events, acts of civil disobedience, or otherwise are not a HOSTILE ACTION perpetrated by a HOSTILE FORCE. Examples include the crash of a small aircraft, shots from hunters, physical disputes between employees, etc. Reporting of these types of events is adequately addressed by other EALs, or the requirements of 10 CFR § 73.71 or 10 CFR § 50.72.

Emergency plans and implementing procedures are public documents; therefore, EALs should not incorporate Security-sensitive information. This includes information that may be advantageous to a potential adversary, such as the particulars concerning a specific threat or threat location. Security-sensitive information should be contained in non-public documents such as the BNP Physical Security Plan (ref. 1).

Escalation of the emergency classification level would be via IC HG1.

BNP Basis Reference(s):

1. BNP Physical Security Plan
2. SEC-NGGC-2170 Security Event Procedures
3. 0AOP-40.0 Security Events
4. NEI 99-01 HS1

Category: H – Hazards

Subcategory: 1 – Security

Initiating Condition: Hostile Action resulting in loss of physical control of the facility

EAL:

HG1.1 General Emergency

A HOSTILE ACTION is occurring or has occurred within the PROTECTED AREA as reported by the Security Shift Supervision

AND EITHER of the following has occurred:

Any of the following safety functions cannot be controlled or maintained

- Reactivity
- RPV water level
- RCS heat removal

OR

Damage to spent fuel has occurred or is IMMINENT

Mode Applicability:

All

Definition(s):

HOSTILE ACTION - An act toward BNP 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 BNP. Non-terrorism-based EALs should be used to address such activities (i.e., this may include violent acts between individuals in the owner controlled area).

IMMINENT - The trajectory of events or conditions is such that an EAL will be met within a relatively short period of time regardless of mitigation or corrective actions.

PROTECTED AREA - The double-fenced security area with intrusion detection devices immediately surrounding the plant structures. The Protected Area is depicted in BNP Radiological Emergency Response Plan Figure 1-1.3 Brunswick Site Building and Onsite Emergency Facility Locations.

Basis:

The Security Shift Supervision is defined as either the Security Shift Lieutenant or the Security Shift Sergeant.

This IC addresses an event in which a HOSTILE FORCE has taken physical control of the facility to the extent that the plant staff can no longer operate equipment necessary to maintain key safety functions. It also addresses a HOSTILE ACTION leading to a loss of physical control that results in actual or IMMINENT damage to spent fuel due to 1) damage to a spent fuel pool cooling system (e.g., pumps, heat exchangers, controls, etc.) or, 2) loss of spent fuel pool integrity such that sufficient water level cannot be maintained.

Timely and accurate communications between Security Shift Supervision and the Control Room is essential for proper classification of a security-related event.

Security plans and terminology are based on the guidance provided by NEI 03-12, *Template for the Security Plan, Training and Qualification Plan, Safeguards Contingency Plan.*

Emergency plans and implementing procedures are public documents; therefore, EALs should not incorporate Security-sensitive information. This includes information that may be advantageous to a potential adversary, such as the particulars concerning a specific threat or threat location. Security-sensitive information should be contained in non-public documents such as the BNP Physical Security Plan (ref.1).

BNP Basis Reference(s):

1. BNP Physical Security Plan
2. SEC-NGGC-2170 Security Event Procedures
3. 0AOP-40.0 Security Events
4. NEI 99-01 HG1

Category: H – Hazards and Other Conditions Affecting Plant Safety

Subcategory: 2 – Seismic Event

Initiating Condition: Seismic event greater than OBE levels

EAL:

HU2.1	**Unusual Event**
Seismic event > OBE per 0AOP-13.0	

Mode Applicability:

All

Definition(s):

None

Basis:

Ground motion acceleration of 0.08g is the Operating Basis Earthquake for BNP (ref. 1).

Unit 2 has an active Kinematrics Condor Seismic Monitoring System with the following components used for seismic detection for the Brunswick Site:

> The system will detect and digitally record the response to actual earthquake loading in terms of acceleration time history from the existing accelerometers mounted in the Unit 2 - 17ft. elevation (basement) of the Reactor Building and also at +89 foot elevation mounted on the Reactor Containment structure. The system will automatically evaluate the recorded acceleration time history in order to determine the response spectra of the events and compare those to the Operating Basis Earthquake (OBE) parameters graphically. It will also determine the exceedance of the OBE, and provides a hard copy of this comparison. The system will provide an immediate Event Alarm output signal at a trigger threshold value of 0.01g to alarm the existing Annunciator 1(2)UA-28 6-4 *SEISMIC EVENT* in the Control Room back to alert the Operators to a seismic event. (ref. 1, 2)

The BNP seismic instrumentation supports readily assessable OBE indications (> 0.08g acceleration) within the Control Room at panel 2-ENV-XU-823. 0AOP-13.0 provides the guidance for determining if the OBE earthquake threshold is exceeded. (ref. 3).

The Shift Manager or Site Emergency Coordinator may seek external verification if deemed appropriate (e.g., a call to the USGS, check internet news sources, etc.); however, the verification action must not preclude a timely emergency declaration.

To avoid inappropriate emergency classification resulting from spurious actuation of the seismic instrumentation or felt motion not attributable to seismic activity, an offsite agency (USGS, National Earthquake Information Center) can confirm that an earthquake has occurred in the area of the plant. As stated above, such confirmation should not, however, preclude a timely emergency declaration. The NEIC can be contacted by calling **(303) 273-8500**. Select **option #1** and inform the analyst you wish to confirm recent seismic activity in the vicinity of BNP. Provide the analyst with the following BNP coordinates: **33° 57' 30" north latitude, 78° 00' 30" west longitude** (ref. 4). Alternatively, near real-time seismic activity can be accessed via the NEIC website:

http://earthquake.usgs.gov/eqcenter/

This IC addresses a seismic event that results in accelerations at the plant site greater than those specified for an Operating Basis Earthquake (OBE). An earthquake greater than an OBE but less than a Safe Shutdown Earthquake (SSE) should have no significant impact on safety-related systems, structures and components; however, some time may be required for the plant staff to ascertain the actual post-event condition of the plant (e.g., performs walk-downs and post-event inspections). Given the time necessary to perform walk-downs and inspections, and fully understand any impacts, this event represents a potential degradation of the level of safety of the plant.

Event verification with external sources should not be necessary during or following an OBE. Earthquakes of this magnitude should be readily felt by onsite personnel and recognized as a seismic event (e.g., lateral accelerations in excess of 0.08g). The Shift Manager or Site Emergency Coordinator may seek external verification if deemed appropriate (e.g., a call to the USGS, check internet news sources, etc.); however, the verification action must not preclude a timely emergency declaration.

Depending upon the plant mode at the time of the event, escalation of the emergency classification level would be via IC CA6 or SA8.

BNP Basis Reference(s):

1. Updated FSAR section 2.5.2.6
2. 1(2)APP-UA-28 6-4 Seismic Event
3. 0AOP-13.0 Operations During Hurricane, Flood Conditions, Tornado or Earthquake
4. Updated FSAR section 2.1.1.1
5. NEI 99-01 HU2

Category: H – Hazards and Other Conditions Affecting Plant Safety

Subcategory: 3 – Natural or Technological Hazard

Initiating Condition: Hazardous event

EAL:

HU3.1	**Unusual Event**
A tornado strike within the PROTECTED AREA	

Mode Applicability:

All

Definition(s):

PROTECTED AREA - The double-fenced security area with intrusion detection devices immediately surrounding the plant structures. The Protected Area is depicted in BNP Radiological Emergency Response Plan Figure 1-1.3 Brunswick Site Building and Onsite Emergency Facility Locations.

Basis:

If damage is confirmed visually or by other in-plant indications, the event may be escalated to an Alert under EAL CA6.1 or SA8.1.

A tornado striking (touching down) within the Protected Area warrants declaration of an Unusual Event regardless of the measured wind speed at the meteorological tower. A tornado is defined as a violently rotating column of air in contact with the ground and extending from the base of a thunderstorm.

This IC addresses hazardous events that are considered to represent a potential degradation of the level of safety of the plant.

EAL HU3.1 addresses a tornado striking (touching down) within the PROTECTED AREA.

Escalation of the emergency classification level would be based on ICs in Recognition Categories R, F, S or C.

BNP Basis Reference(s):

1. NEI 99-01 HU3

Category: H – Hazards and Other Conditions Affecting Plant Safety

Subcategory: 3 – Natural or Technological Hazard

Initiating Condition: Hazardous event

EAL:

HU3.2	**Unusual Event**
Internal room or area FLOODING of a magnitude sufficient to require manual or automatic electrical isolation of a SAFETY SYSTEM component needed for the current operating mode	

Mode Applicability:

All

Definition(s):

FLOODING - A condition where water is entering a room or area faster than installed equipment is capable of removal, resulting in a rise of water level within the room or area.

SAFETY SYSTEM - A system required for safe plant operation, cooling down the plant and/or placing it in the cold shutdown condition, including the ECCS. These are typically systems classified as safety-related (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.

Basis:

This IC addresses hazardous events that are considered to represent a potential degradation of the level of safety of the plant.

This EAL addresses FLOODING of a building room or area that results in operators isolating power to a SAFETY SYSTEM component due to water level or other wetting concerns. Classification is also required if the water level or related wetting causes an automatic isolation of a SAFETY SYSTEM component from its power source (e.g., a breaker or relay trip). To warrant classification, operability of the affected component must be required by Technical Specifications for the current operating mode.

Escalation of the emergency classification level would be based on ICs in Recognition Categories R, F, S or C.

BNP Basis Reference(s):

1. Updated FSAR section 3.4.2 Protection From Internal Flooding
2. NEI 99-01 HU3

Category: H – Hazards and Other Conditions Affecting Plant Safety

Subcategory: 3 – Natural or Technological Hazard

Initiating Condition: Hazardous event

EAL:

HU3.3 **Unusual Event**
Movement of personnel within the PROTECTED AREA is IMPEDED due to an event external to the PROTECTED AREA involving hazardous materials (e.g., an offsite chemical spill or toxic gas release)

Mode Applicability:

All

Definition(s):

IMPEDE(D)- Personnel access to a room or area is hindered to an extent that extraordinary measures are necessary to facilitate entry of personnel into the affected room/area (e.g., requiring use of protective equipment, such as SCBAs, that is not routinely employed).

PROTECTED AREA - The double-fenced security area with intrusion detection devices immediately surrounding the plant structures. The Protected Area is depicted in BNP Radiological Emergency Response Plan Figure 1-1.3 Brunswick Site Building and Onsite Emergency Facility Locations.

Basis:

The following documents provide additional information on hazardous substances and spills.

- 0AOP-34.0 Chlorine Emergencies (Ref. 1)
- 0AOP-44.0 Sodium Hypochlorite or Acti-Brom Leak Response (Ref. 2)
- 0AOP-43.0 Hydrogen Emergency (Ref. 3)
- 0AOP-05.0 Radioactive Spills, High Radiation, and Airborne Activity (Ref. 4)
- Regulatory Guide 1.78, Evaluating the Habitability of a Nuclear Power Plant Control Room During a Postulated Hazardous Chemical Release, Table 1, Toxicity Limits (IDLH Limits) for Some Hazardous Chemicals (Ref. 5)

This IC addresses hazardous events that are considered to represent a potential degradation of the level of safety of the plant.

This EAL addresses a hazardous materials event originating external to the PROTECTED AREA and of sufficient magnitude to impede the movement of personnel within the PROTECTED AREA.

Escalation of the emergency classification level would be based on ICs in Recognition Categories R, F, S or C.

BNP Basis Reference(s):

1. 0AOP-34.0 Chlorine Emergencies
2. 0AOP-44.0 Sodium Hypochlorite or Acti-Brom Leak Response
3. 0AOP-43.0 Hydrogen Emergency
4. 0AOP-05.0 Radioactive Spills, High Radiation, and Airborne Activity
5. Regulatory Guide 1.78, Evaluating the Habitability of a Nuclear Power Plant Control Room During a Postulated Hazardous Chemical Release, Table 1, Toxicity Limits (IDLH Limits) for Some Hazardous Chemicals
6. NEI 99-01 HU3

Category: H – Hazards and Other Conditions Affecting Plant Safety

Subcategory: 3 – Natural or Technological Hazard

Initiating Condition: Hazardous event

EAL:

HU3.4	**Unusual Event**
A hazardous event that results in onsite conditions sufficient to prohibit the plant staff from accessing the site via personal vehicles (Note 7)	

Note 7: This EAL does not apply to routine traffic impediments such as fog, snow, ice, or vehicle breakdowns or accidents.

Mode Applicability:

All

Definition(s):

None

Basis:

This IC addresses hazardous events that are considered to represent a potential degradation of the level of safety of the plant.

This EAL addresses a hazardous event, either onsite or offsite, that causes an onsite impediment to vehicle movement and significant enough to prohibit the plant staff from accessing the site using personal vehicles. Examples of such an event include site FLOODING caused by a hurricane, heavy rains, up-river water releases, dam failure, etc., or an onsite train derailment blocking the access road.

This EAL is not intended apply to routine impediments such as fog, snow, ice, or vehicle breakdowns or accidents, but rather to more significant conditions such as the Hurricane Andrew strike on Turkey Point in 1992, the flooding around the Cooper Station during the Midwest floods of 1993, or the flooding around Ft. Calhoun Station in 2011.

Escalation of the emergency classification level would be based on ICs in Recognition Categories R, F, S or C.

BNP Basis Reference(s):

1. NEI 99-01 HU3

Category: H – Hazards and Other Conditions Affecting Plant Safety

Subcategory: 3 – Natural or Technological Hazard

Initiating Condition: Hazardous event

EAL:

HU3.5	**Unusual Event**
Intake Canal water level > +19 ft Mean Sea Level	
OR	
Intake Canal water level < -7.75 ft Mean Sea Level	

Mode Applicability:

All

BNP Basis:

The high Intake Canal level is the highest remotely measurable Intake Canal water level. Otherwise it would have been based based the plant design that Class I structures and engineered safety features systems are protected against still water flooding (elevation 22.0 feet). BNP is geographically located in close proximity to the Atlantic coastal storm track and has an approximate grade elevation of 20 feet above Mean Sea Level. Hurricanes and tropical storms are therefore, the most extreme weather phenomena that affect the site area. Potential subsequent flooding should be considered even though the plant structures were designed to compensate, via installed sump pumps, for a maximum site flooding depth of 22 feet above Mean Sea Level during the Maximum Probable Hurricane. (ref. 1).

The minimum water level predicted for the Maximum Probable Hurricane is -7.5 feet Mean Sea Level under special case circumstances. The abnormal operating procedure for a hurricane requires that each unit be shutdown prior to arrival of hurricane conditions at the site. The SW System has been analyzed in modes 4 and 5 for an intake canal water level of -7.75 feet Mean Sea Level corresponding to -8.63 feet Mean Sea Level in the pump suction bay for the maximum pressure drop, 0.88 feet, across the traveling screens. (ref. 2, 3).

This IC addresses hazardous events that are considered to represent a potential degradation of the level of safety of the plant.

This EAL addresses high and low external water levels as a result of a hurricane.

Escalation of the emergency classification level would be based on ICs in Recognition Categories R, F, S or C.

BNP Basis Reference(s):

1. Updated FSAR section 2.4.10.2
2. Updated FSAR section 9.2.1.2.3
3. 0AOP-13.0 Operations During Hurricane, Flood Conditions, Tornado or Earthquake

Category: H – Hazards and Other Conditions Affecting Plant Safety

Subcategory: 4 – Fire

Initiating Condition: FIRE potentially degrading the level of safety of the plant

EAL:

HU4.1 Unusual Event

A FIRE is not extinguished within 15 min. of **any** of the following FIRE detection indications (Note 1):

- Report from the field (i.e., visual observation)
- Receipt of multiple (more than 1) fire alarms or indications
- Field verification of a single fire alarm

AND

The FIRE is located within **any** Table H-1 area

Note 1: The SEC should declare the event promptly upon determining that time limit has been exceeded, or will likely be exceeded.

Table H-1 Fire Areas
• Reactor Building
• Diesel Generator Building
• Diesel 4-Day Tank Rooms
• Service Water Building
• Turbine Building
• Control Building
• CSTs
• Diesel Fuel Oil Storage Tank

Mode Applicability:

All

Definition(s):

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.

Basis:

The 15 minute requirement begins with a credible notification that a fire is occurring, or receipt of multiple valid fire detection system alarms or field validation of a single fire alarm. The alarm is to be validated using available Control Room indications or alarms to prove that it is not spurious, or by reports from the field. Actual field reports must be made within the 15 minute time limit or a classification must be made. If a fire is verified to be occurring by field report, the 15 minute time limit is from the original receipt of the fire detection alarm.

Table H-1 Fire Areas are based on BNP-E-9. 010 NFPA 805 Nuclear Safety Capability Assessment (NSCA) and 0PFP-PBAA Power Block Auxiliary Areas Prefire Plan. Table H-1 Fire Areas include those structures containing functions and systems required for safe shutdown of the plant (SAFETY SYSTEMS) (ref. 1, 2).

This IC addresses the magnitude and extent of FIRES that may be indicative of a potential degradation of the level of safety of the plant.

For EAL HU4.1 the intent of the 15-minute duration is to size the FIRE and to discriminate against small FIRES that are readily extinguished (e.g., smoldering waste paper basket). In addition to alarms, other indications of a FIRE could be a drop in fire main pressure, automatic activation of a suppression system, etc.

Upon receipt, operators will take prompt actions to confirm the validity of an initial fire alarm, indication, or report. For EAL assessment purposes, the emergency declaration clock starts at the time that the initial alarm, indication, or report was received, and not the time that a subsequent verification action was performed. Similarly, the fire duration clock also starts at the time of receipt of the initial alarm, indication or report.

Depending upon the plant mode at the time of the event, escalation of the emergency classification level would be via IC CA6 or SA8.

BNP Basis Reference(s):

1. BNP-E-9. 010 NFPA 805 Nuclear Safety Capability Assessment (NSCA)
2. 0PFP-PBAA Power Block Auxiliary Areas Prefire Plan
3. NEI 99-01 HU4

Category: H – Hazards and Other Conditions Affecting Plant Safety

Subcategory: 4 – Fire

Initiating Condition: FIRE potentially degrading the level of safety of the plant

EAL:

HU4.2 Unusual Event

Receipt of a single fire alarm (i.e., no other indications of a FIRE)

AND

The fire alarm is indicating a FIRE within **any** Table H-1 area

AND

The existence of a FIRE is not verified within 30 min. of alarm receipt (Note 1)

Note 1: The SEC should declare the event promptly upon determining that time limit has been exceeded, or will likely be exceeded.

Table H-1 Fire Areas
• Reactor Building • Diesel Generator Building • Diesel 4-Day Tank Rooms • Service Water Building • Turbine Building • Control Building • CSTs • Diesel Fuel Oil Storage Tank

Mode Applicability:

All

Definition(s):

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.

Basis:

The 30 minute requirement begins upon receipt of a single valid fire detection system alarm. The alarm is to be validated using available Control Room indications or alarms to prove that it is not spurious, or by reports from the field. Actual field reports must be made within the 30 minute time limit or a classification must be made. If a fire is verified to be occurring by field report, classification shall be made based on EAL HU4.1.

Table H-1 Fire Areas are based on BNP-E-9. 010 NFPA 805 Nuclear Safety Capability Assessment (NSCA) and 0PFP-PBAA Power Block Auxiliary Areas Prefire Plan. Table H-1 Fire Areas include those structures containing functions and systems required for safe shutdown of the plant (SAFETY SYSTEMS) (ref. 1, 2).

This IC addresses the magnitude and extent of FIRES that may be indicative of a potential degradation of the level of safety of the plant.

This EAL addresses receipt of a single fire alarm, and the existence of a FIRE is not verified (i.e., proved or disproved) within 30-minutes of the alarm. Upon receipt, operators will take prompt actions to confirm the validity of a single fire alarm. For EAL assessment purposes, the 30-minute clock starts at the time that the initial alarm was received, and not the time that a subsequent verification action was performed.

A single fire alarm, absent other indication(s) of a FIRE, may be indicative of equipment failure or a spurious activation, and not an actual FIRE. For this reason, additional time is allowed to verify the validity of the alarm. The 30-minute period is a reasonable amount of time to determine if an actual FIRE exists; however, after that time, and absent information to the contrary, it is assumed that an actual FIRE is in progress.

If an actual FIRE is verified by a report from the field, then HU4.1 is immediately applicable, and the emergency must be declared if the FIRE is not extinguished within 15-minutes of the report. If the alarm is verified to be due to an equipment failure or a spurious activation, and this verification occurs within 30-minutes of the receipt of the alarm, then this EAL is not applicable and no emergency declaration is warranted.

Appendix R to 10 CFR 50, states in part:

> Criterion 3 of Appendix A to this part specifies that "Structures, systems, and components important to safety shall be designed and located to minimize, consistent with other safety requirements, the probability and effect of fires and explosions."

> When considering the effects of fire, those systems associated with achieving and maintaining safe shutdown conditions assume major importance to safety because damage to them can lead to core damage resulting from loss of coolant through boil-off.
>
> Because fire may affect safe shutdown systems and because the loss of function of systems used to mitigate the consequences of design basis accidents under post-fire conditions does not per se impact public safety, the need to limit fire damage to systems required to achieve and maintain safe shutdown conditions is greater than the need to limit fire damage to those systems required to mitigate the consequences of design basis accidents.

In addition, Appendix R to 10 CFR 50, requires, among other considerations, the use of 1-hour fire barriers for the enclosure of cable and equipment and associated non-safety circuits of one redundant train (G.2.c). As used in this EAL, the 30-minutes to verify a single alarm is well within this worst-case 1-hour time period.

It should be noted however, BNP is not an Appendix R plant but rather falls under the requirements of NFPA-805 for fire protection.

Depending upon the plant mode at the time of the event, escalation of the emergency classification level would be via IC CA6 or SA8.

BNP Basis Reference(s):

1. BNP-E-9. 010 NFPA 805 Nuclear Safety Capability Assessment (NSCA)
2. 0PFP-PBAA Power Block Auxiliary Areas Prefire Plan
3. NEI 99-01 HU4

Category: H – Hazards and Other Conditions Affecting Plant Safety

Subcategory: 4 – Fire

Initiating Condition: FIRE potentially degrading the level of safety of the plant

EAL:

HU4.3	**Unusual Event**
A FIRE within the plant PROTECTED AREA not extinguished within 60 min. of the initial report, alarm or indication (Note 1)	

Note 1: The SEC should declare the event promptly upon determining that time limit has been exceeded, or will likely be exceeded.

Mode Applicability:

All

Definition(s):

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.

PROTECTED AREA - The double-fenced security area with intrusion detection devices immediately surrounding the plant structures. The Protected Area is depicted in BNP Radiological Emergency Response Plan Figure 1-1.3 Brunswick Site Building and Onsite Emergency Facility Locations.

Basis:

This IC addresses the magnitude and extent of FIRES that may be indicative of a potential degradation of the level of safety of the plant.

In addition to a FIRE addressed by EAL HU4.1 or HU4.2, a FIRE within the plant PROTECTED AREA not extinguished within 60-minutes may also potentially degrade the level of plant safety.

Depending upon the plant mode at the time of the event, escalation of the emergency classification level would be via IC CA6 or SA8.

BNP Basis Reference(s):

1. NEI 99-01 HU4

Category: H – Hazards and Other Conditions Affecting Plant Safety

Subcategory: 4 – Fire

Initiating Condition: FIRE potentially degrading the level of safety of the plant

EAL:

HU4.4	**Unusual Event**
A FIRE within the plant PROTECTED AREA that requires firefighting support by an offsite fire response agency to extinguish	

Mode Applicability:

All

Definition(s):

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.

PROTECTED AREA - The double-fenced security area with intrusion detection devices immediately surrounding the plant structures. The Protected Area is depicted in BNP Radiological Emergency Response Plan Figure 1-1.3 Brunswick Site Building and Onsite Emergency Facility Locations.

Basis:

The dispatch of an offsite firefighting agency to the site requires an emergency declaration only if it is needed to actively support firefighting efforts because the fire is beyond the capability of the Fire Brigade to extinguish.

This IC addresses the magnitude and extent of FIRES that may be indicative of a potential degradation of the level of safety of the plant.

If a FIRE within the plant PROTECTED AREA is of sufficient size to require a response by an offsite firefighting agency (e.g., a local town Fire Department), then the level of plant safety is potentially degraded. The dispatch of an offsite firefighting agency to the site requires an emergency declaration only if it is needed to actively support firefighting efforts because the fire is beyond the capability of the Fire Brigade to extinguish. Declaration is not necessary if the agency resources are placed on stand-by, or supporting post-extinguishment recovery or investigation actions.

Depending upon the plant mode at the time of the event, escalation of the emergency classification level would be via IC CA6 or SA8.

BNP Basis Reference(s):

1. NEI 99-01 HU4

Category: H – Hazards and Other Conditions Affecting Plant Safety

Subcategory: 5 – Hazardous Gases

Initiating Condition: Gaseous release IMPEDING access to equipment necessary for normal plant operations, cooldown or shutdown

EAL:

HA5.1 Alert

Release of a toxic, corrosive, asphyxiant or flammable gas into **any** Table H-2 rooms or areas

AND

Entry into the room or area is prohibited or IMPEDED (Note 5)

Note 5: If the equipment in the listed room or area was already inoperable or out-of-service before the event occurred, then no emergency classification is warranted.

Table H-2 Safe Operation & Shutdown Areas	
Room/Area	**Mode Applicability**
Reactor Building -17' North RHR Unit-1 & 2	3, 4, 5
Reactor Building -17' South RHR Unit-1 & 2	3, 4, 5
Reactor Building 20' East & West MCC Areas Unit-1 & 2	3, 4, 5
Reactor Building 20' Pipe Tunnel Unit-1 & 2	3, 4, 5

Mode Applicability:

All

Definition(s):

IMPEDE(D)- Personnel access to a room or area is hindered to an extent that extraordinary measures are necessary to facilitate entry of personnel into the affected room/area (e.g., requiring use of protective equipment, such as SCBAs, that is not routinely employed).

Basis:

If the equipment in the listed room or area was already inoperable, or out-of-service, before the event occurred, then no emergency should be declared since the event will have no adverse impact beyond that already allowed by Technical Specifications at the time of the event.

The list of plant rooms or areas with entry-related mode applicability identified specify those rooms or areas that contain equipment which require a manual/local action as specified in operating procedures used for normal plant operation, cooldown and shutdown. Rooms or areas in which actions of a contingent or emergency nature would be performed (e.g., an action to address an off-normal or emergency condition such as emergency repairs, corrective measures or emergency operations) are not included. In addition, the list specifies the plant mode(s) during which entry would be required for each room or area (ref. 1).

The following documents provide additional information on hazardous substances and spills.

- 0AOP-34.0 Chlorine Emergencies (Ref. 2)
- 0AOP-44.0 Sodium Hypochlorite or Acti-Brom Leak Response (Ref. 3)
- 0AOP-43.0 Hydrogen Emergency (Ref. 4)

This IC addresses an event involving a release of a hazardous gas that precludes or impedes access to equipment necessary to maintain normal plant operation, or required for a normal plant cooldown and shutdown. This condition represents an actual or potential substantial degradation of the level of safety of the plant.

An Alert declaration is warranted if entry into the affected room/area is, or may be, procedurally required during the plant operating mode in effect at the time of the gaseous release. The emergency classification is not contingent upon whether entry is actually necessary at the time of the release.

Evaluation of the IC and EAL do not require atmospheric sampling; it only requires the Site Emergency Coordinator's judgment that the gas concentration in the affected room/area is sufficient to preclude or significantly impede procedurally required access. This judgment may be based on a variety of factors including an existing job hazard analysis, report of ill effects on personnel, advice from a subject matter expert or operating experience with the same or similar hazards. Access should be considered as impeded if extraordinary measures are necessary to facilitate entry of personnel into the affected room/area (e.g., requiring use of protective equipment, such as SCBAs, that is not routinely employed).

An emergency declaration is not warranted if any of the following conditions apply:

- The plant is in an operating mode different than the mode specified for the affected room/area (i.e., entry is not required during the operating mode in effect at the time of the gaseous release). For example, the plant is in Mode 1 when the gaseous release occurs, and the procedures used for normal operation, cooldown and shutdown do not require entry into the affected room until Mode 4.
- The gas release is a planned activity that includes compensatory measures which address the temporary inaccessibility of a room or area (e.g., fire suppression system testing).

- The action for which room/area entry is required is of an administrative or record keeping nature (e.g., normal rounds or routine inspections).
- The access control measures are of a conservative or precautionary nature, and would not actually prevent or impede a required action.
- If the equipment in the listed room or area was already inoperable, or out-of-service, before the event occurred, then no emergency should be declared since the event will have no adverse impact 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.

This EAL does not apply to firefighting activities that automatically or manually activate a fire suppression system in an area, or to intentional inerting of containment.

Escalation of the emergency classification level would be via Recognition Category R, C or F ICs.

BNP Basis Reference(s):

1. Attachment 3 Safe Operation & Shutdown Areas Tables R-2 & H-2 Bases
2. 0AOP-34.0 Chlorine Emergencies
3. 0AOP-44.0 Sodium Hypochlorite or Acti-Brom Leak Response
4. 0AOP-43.0 Hydrogen Emergency
5. NEI 99-01 HA5

Category: H – Hazards and Other Conditions Affecting Plant Safety

Subcategory: 6 – Control Room Evacuation

Initiating Condition: Control Room evacuation resulting in transfer of plant control to alternate locations

EAL:

HA6.1 **Alert**
An event has resulted in plant control being transferred from the Control Room to the Remote Shutdown Panels

Mode Applicability:

All

Definition(s):

None

Basis:

The Shift Manager (SM) determines if the Control Room is inoperable and requires evacuation. Control Room inhabitability may be caused by fire, dense smoke, noxious fumes, bomb threat in or adjacent to the Control Room, or other life threatening conditions (Ref. 1, 2).

Inability to establish plant control from outside the Control Room escalates this event to a Site Area Emergency per EAL HS6.1.

This IC addresses an evacuation of the Control Room that results in transfer of plant control to alternate locations outside the Control Room. The loss of the ability to control the plant from the Control Room is considered to be a potential substantial degradation in the level of plant safety.

Following a Control Room evacuation, control of the plant will be transferred to alternate shutdown locations. The necessity to control a plant shutdown from outside the Control Room, in addition to responding to the event that required the evacuation of the Control Room, will present challenges to plant operators and other on-shift personnel. Activation of the ERO and emergency response facilities will assist in responding to these challenges.

Escalation of the emergency classification level would be via IC HS6.

BNP Basis Reference(s):

1. 0AOP-32.0, Plant Shutdown from Outside Control Room
2. 0PLP-01.5 Alternate Shutdown Capability Controls
3. NEI 99-01 HA6

Category: H – Hazards and Other Conditions Affecting Plant Safety

Subcategory: 6 – Control Room Evacuation

Initiating Condition: Inability to control a key safety function from outside the Control Room

EAL:

HS6.1 Site Area Emergency

An event has resulted in plant control being transferred from the Control Room to the Remote Shutdown Panels

AND

Control of **any** of the following key safety functions is not reestablished within 22.5 min. (Note 1):

- Reactivity (Modes 1 and 2 **only**)
- RPV water level
- RPV pressure (Modes 1, 2, 3 and 4 **only**)

Note 1: The SEC should declare the event promptly upon determining that time limit has been exceeded, or will likely be exceeded.

Mode Applicability:

1 - Power Operations, 2 - Startup, 3 - Hot Shutdown, 4 - Cold Shutdown, 5 - Refueling

Definition(s):

None

Basis:

The Shift Manager determines if the Control Room is inoperable and requires evacuation. Control Room inhabitability may be caused by fire, dense smoke, noxious fumes, bomb threat in or adjacent to the Control Room, or other life threatening conditions (Ref. 1, 2).

The immediate actions of 0AOP-32 direct a reactor scram prior to evacuating the Control Room. Local control of high pressure injection sources and Safety Relief Valves (SRVs) establishes control of RPV water level and pressure.

This IC addresses an evacuation of the Control Room that results in transfer of plant control to alternate locations, and the control of a key safety function cannot be reestablished in a timely manner. The failure to gain control of a key safety function following a transfer of plant control to alternate locations is a precursor to a challenge to one or more fission product barriers within a relatively short period of time.

The determination of whether or not "control" is established at the remote safe shutdown location(s) is based on Site Emergency Coordinator judgment. The Site Emergency Coordinator is expected to make a reasonable, informed judgment within 22.5 minutes whether or not the operating staff has control of key safety functions from the remote safe shutdown location(s) (ref. 3).

Escalation of the emergency classification level would be via IC FG1 or CG1

BNP Basis Reference(s):

1. 0AOP-32.0, Plant Shutdown from Outside Control Room
2. 0PLP-01.5 Alternate Shutdown Capability Controls
3. Calculation No. BNP-E-9.007 ASSD Manual Action Feasibility
4. NEI 99-01 HS6

Category: H – Hazards and Other Conditions Affecting Plant Safety

Subcategory: 7 – SEC Judgment

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

EAL:

HU7.1	**Unusual Event**
Other conditions exist which in the judgment of the Site Emergency Coordinator 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

Definition(s):

None

Basis:

The Site Emergency Coordinator is the designated onsite individual having the responsibility and authority for implementing the BNP Emergency Response Plan. The Operations Shift Manager (SM) initially acts in the capacity of the Site Emergency Coordinator and takes actions as outlined in the Emergency Plan implementing procedures. If required by the emergency classification or if deemed appropriate by the Site Emergency Coordinator, emergency response personnel are notified and instructed to report to their emergency response locations. In this manner, the individual usually in charge of activities in the Control Room is responsible for initiating the necessary emergency response, but Plant Management is expected to manage the emergency response as soon as available to do so in anticipation of the possible wide-ranging responsibilities associated with managing a major emergency (ref. 1).

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

BNP Basis Reference(s):

1. 0ERP BNP Radiological Emergency Response Plan section 3.0 Emergency Response Organization
2. NEI 99-01 HU7

Category: H – Hazards and Other Conditions Affecting Plant Safety

Subcategory: 7 – SEC Judgment

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

EAL:

HA7.1	**Alert**
Other conditions exist which, in the judgment of the Site Emergency Coordinator, 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.	

Mode Applicability:

All

Definition(s):

HOSTILE ACTION - An act toward BNP 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 BNP. Non-terrorism-based EALs should be used to address such activities (i.e., this may include violent acts between individuals in the owner controlled area).

Basis:

The Site Emergency Coordinator is the designated onsite individual having the responsibility and authority for implementing the BNP Emergency Response Plan. The Operations Shift Manager (SM) initially acts in the capacity of the Site Emergency Coordinator and takes actions as outlined in the Emergency Plan implementing procedures. If required by the emergency classification or if deemed appropriate by the Site Emergency Coordinator, emergency response personnel are notified and instructed to report to their emergency response locations. In this manner, the individual usually in charge of activities in the Control Room is responsible for initiating the necessary emergency response, but Plant Management is expected to manage the emergency response as soon as available to do so in anticipation of the possible wide-ranging responsibilities associated with managing a major emergency (ref.1).

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

BNP Basis Reference(s):

1. 0ERP BNP Radiological Emergency Response Plan section 3.0 Emergency Response Organization
2. NEI 99-01 HA7

Category: H – Hazards and Other Conditions Affecting Plant Safety

Subcategory: 7 – SEC Judgment

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

EAL:

HS7.1 **Site Area Emergency**
Other conditions exist which in the judgment of the Site Emergency Coordinator 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 beyond the site boundary

Mode Applicability:

All

Definition(s):

HOSTILE ACTION - An act toward BNP 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 BNP. Non-terrorism-based EALs should be used to address such activities (i.e., this may include violent acts between individuals in the owner controlled area)

Basis:

The Site Emergency Coordinator is the designated onsite individual having the responsibility and authority for implementing the BNP Emergency Response Plan. The Operations Shift Manager (SM) initially acts in the capacity of the Site Emergency Coordinator and takes actions as outlined in the Emergency Plan implementing procedures. If required by the emergency classification or if deemed appropriate by the Site Emergency Coordinator, emergency response personnel are notified and instructed to report to their emergency response locations. In this manner, the individual usually in charge of activities in the Control Room is responsible for initiating the necessary emergency response, but Plant Management is expected to manage the emergency response as soon as available to do so in anticipation of the possible wide-ranging responsibilities associated with managing a major emergency (ref. 1).

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

BNP Basis Reference(s):

1. 0ERP BNP Radiological Emergency Response Plan section 3.0 Emergency Response Organization

2. NEI 99-01 HS7

Category: H – Hazards and Other Conditions Affecting Plant Safety

Subcategory: 7 – SEC Judgment

Initiating Condition: Other conditions exist which in the judgment of the Site Emergency Coordinator warrant declaration of a General Emergency

EAL:

HG7.1	**General Emergency**
Other conditions exist which in the judgment of the Site Emergency Coordinator 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 offsite for more than the immediate site area	

Mode Applicability:

All

Definition(s):

HOSTILE ACTION - An act toward BNP 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 BNP. Non-terrorism-based EALs should be used to address such activities (i.e., this may include violent acts between individuals in the owner controlled area).

IMMINENT - The trajectory of events or conditions is such that an EAL will be met within a relatively short period of time regardless of mitigation or corrective actions.

Basis:

The Site Emergency Coordinator is the designated onsite individual having the responsibility and authority for implementing the BNP Emergency Response Plan. The Operations Shift Manager(SM) initially acts in the capacity of the Site Emergency Coordinator and takes actions as outlined in the Emergency Plan implementing procedures. If required by the emergency classification or if deemed appropriate by the Site Emergency Coordinator, emergency response personnel are notified and instructed to report to their emergency response locations. In this manner, the individual usually in charge of activities in the Control Room is responsible for initiating the necessary emergency response, but Plant Management is expected to manage the emergency response as soon as available to do so in anticipation of the possible wide-ranging responsibilities associated with managing a major emergency (ref. 1).

Releases can reasonably be expected to exceed EPA PAG plume exposure levels outside the Site Boundary.

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

BNP Basis Reference(s):

1. 0ERP BNP Radiological Emergency Response Plan section 3.0 Emergency Response Organization
2. NEI 99-01 HG7

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 Emergency AC Power

Loss of emergency 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 sources for 4160 V emergency buses.

2. Loss of Vital DC Power

Loss of emergency 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 vital plant 125 VDC power sources.

3. Loss of Control Room Indications

Certain events that degrade plant operator ability to effectively assess plant conditions within the plant warrant emergency classification. Losses of indicators are in this subcategory.

4. RCS Activity

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 (2% - 5% clad failures) is indicative of fuel failures and is covered under the Fission Product Barrier Degradation category. 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.

5. RCS Leakage

The reactor pressure vessel provides a volume for the coolant that covers the reactor core. The reactor pressure 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 indicates potential pipe cracks that may propagate to an extent threatening fuel clad, RCS and Primary Containment integrity.

6. RPS Failure

This subcategory includes 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 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 Primary Containment integrity.

7. Loss of Communications

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

8. Hazardous Event Affecting Safety Systems

Various natural and technological events that result in degraded plant safety system performance or significant visible damage warrant emergency classification under this subcategory.

Category: S – System Malfunction

Subcategory: 1 – Loss of Emergency AC Power

Initiating Condition: Loss of **all** offsite AC power capability to emergency buses for 15 minutes or longer

EAL:

SU1.1 Unusual Event

Loss of **all** offsite AC power capability, Table S-5, to Emergency 4 KV Buses E1(E3) and E2(E4) for ≥ 15 min. (Note 1)

Note 1: The SEC should declare the event promptly upon determining that time limit has been exceeded, or will likely be exceeded.

Table S-5 AC Power Sources
Offsite: • SAT • UAT backfed through MPT (**only** if already aligned)
Onsite: • UAT via Main Generator • EDG1(3) • EDG2(4)

Mode Applicability:

1 - Power Operations, 2 - Startup, 3 - Hot Shutdown

Definition(s):

None

Basis:

The 4160 VAC System provides the power requirements for operation and safe shutdown of the plant. 4160 VAC power is divided into Balance of Plant (BOP Bus), and Emergency (E-Bus) distribution. The BOP Buses consist of the Conventional Buses (Common A/B Buses; Buses 1B/2B, 1C/2C, 1D/2D) and Caswell Beach Buses A/B. The Emergency switchgear are Buses E1/E3 (Division I) and E2/E4 (Division II).

The E-Buses are normally powered through the respective BOP Buses (1D to E1, 1C to E2, 2D to E3, 2C to E4) via a master/slave breaker arrangement. Each E-Bus has a dedicated Diesel Generator to supply an emergency source of power to safe shutdown loads in the event of a loss of the normal power source or loss of offsite power. The DGs will automatically start and tie onto the E-Buses if the normal power source or offsite power is lost. The DGs can provide power to the E-Buses only. In the event the diesel generator is unavailable for an E-Bus, crosstie capability exists for each E-Bus from the same division of the opposite unit (E1 to E3, E2 to E4). Although the E-Buses within the unit also have crosstie capability, this alignment is not permitted by plant procedures, with the exception of E1 to E2 during specific Alternate Safe Shutdown (ASSD) conditions.

During periods of unit shutdown, when the Startup Auxiliary Transformer (SAT) would be the only normal source of offsite power, the Unit Auxiliary Transformer (UAT) can be made available by establishing a UAT backfeed. Backfeed from the UAT will require the use of keys for the control selector switches and opening of the respective generator's manual no-load disconnect. (Ref. 1, 2)

The 15-minute interval was selected as a threshold to exclude transient or momentary power losses.

This IC addresses a prolonged loss of offsite power. The loss of offsite power sources renders the plant more vulnerable to a complete loss of power to AC emergency buses. This condition represents a potential reduction in the level of safety of the plant.

For emergency classification purposes, "capability" means that an offsite AC power source(s) is available to the emergency buses, whether or not the buses are powered from it.

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

Escalation of the emergency classification level would be via IC SA1.

BNP Basis Reference(s):

1. BNP Updated FSAR Chapter 8
2. 1(2)OP-50 Plant Electric System Operating Procedure
3. NEI 99-01 SU1

Category: S – System Malfunction

Subcategory: 1 – Loss of Emergency AC Power

Initiating Condition: Loss of **all but one** AC power source to emergency buses for 15 minutes or longer

EAL:

SA1.1 Alert

AC power capability to Emergency 4 KV Buses E1(E3) and E2(E4) reduced to a single power source, Table S-5, for ≥ 15 min. (Note 1)

AND

Any additional single power source failure will result in loss of **all** unit-specific AC power to SAFETY SYSTEMS

Note 1: The SEC should declare the event promptly upon determining that time limit has been exceeded, or will likely be exceeded.

Table S-5 AC Power Sources
Offsite: • SAT • UAT backfed through MPT (**only** if already aligned) **Onsite:** • UAT via Main Generator • EDG1(3) • EDG2(4)

Mode Applicability:

1 - Power Operations, 2 - Startup, 3 - Hot Shutdown

Definition(s):

SAFETY SYSTEM - A system required for safe plant operation, cooling down the plant and/or placing it in the cold shutdown condition, including the ECCS. These are typically systems classified as safety-related (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.

Basis:

The 4160 VAC System provides the power requirements for operation and safe shutdown of the plant. 4160 VAC power is divided into Balance Of Plant (BOP Bus), and Emergency (E-Bus) distribution. The BOP Buses consist of the Conventional Buses (Common A/B Buses; Buses 1B/2B, 1C/2C, 1D/2D) and Caswell Beach Buses A/B. The Emergency switchgear are Buses E1/E3 (Division I) and E2/E4 (Division II).

The E-Buses are normally powered through the respective BOP Buses (1D to E1, 1C to E2, 2D to E3, 2C to E4) via a master/slave breaker arrangement. Each E-Bus has a dedicated Diesel Generator to supply an emergency source of power to safe shutdown loads in the event of a loss of the normal power source or loss of offsite power. The DGs will automatically start and tie onto the E-Buses if the normal power source or offsite power is lost. The DGs can provide power to the E-Buses only. In the event the diesel generator is unavailable for an E-Bus, crosstie capability exists for each E-Bus from the same division of the opposite unit (E1 to E3, E2 to E4). Although the E-Buses within the unit also have crosstie capability, this alignment is not permitted by plant procedures, with the exception of E1 to E2 during specific Alternate Safe Shutdown (ASSD) conditions.

During periods of unit shutdown, when the Startup Auxiliary Transformer (SAT) would be the only normal source of offsite power, the Unit Auxiliary Transformer (UAT) can be made available by establishing a UAT backfeed. Backfeed from the UAT will require the use of keys for the control selector switches and opening of the respective generator's manual no-load disconnect (Ref. 1, 2).

The 15-minute interval was selected as a threshold to exclude transient or momentary power losses. If the capability of a second source of emergency bus power is not restored within 15 minutes, an Alert is declared under this EAL.

Because 2 RHR pumps on each unit are powered from the unaffected unit, the words "unit-specific" have been added to clarify that the cross-connected RHR pump power cannot be credited as an AC power source relative to this EAL.

This IC describes a significant degradation of offsite and onsite AC power sources such that any additional single failure would result in a loss of all AC power to SAFETY SYSTEMS. In this condition, the sole AC power source may be powering one, or more than one, train of safety-related equipment. This IC provides an escalation path from IC SU1.

An "AC power source" is a source recognized in AOPs, and capable of supplying required power to an emergency bus. Some examples of this condition are presented below.

- A loss of all offsite power with a concurrent failure of all but one emergency power source (e.g., an onsite diesel generator).
- A loss of all offsite power and loss of all emergency power sources (e.g., onsite diesel generators) with a single train of emergency buses being back-fed from the unit main generator.
- A loss of emergency power sources (e.g., onsite diesel generators) with a single train of emergency buses being back-fed from an offsite power source.

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

Escalation of the emergency classification level would be via IC SS1.

BNP Basis Reference(s):

1. BNP Updated FSAR Chapter 8
2. 1(2)OP-50 Plant Electric System Operating Procedure
3. NEI 99-01 SA1

Category: S – System Malfunction

Subcategory: 1 – Loss of Emergency AC Power

Initiating Condition: Loss of **all** offsite and **all** onsite AC power to emergency buses for 15 minutes or longer

EAL:

SS1.1 **Site Area Emergency**
Loss of **all** offsite and **all** onsite AC power capability to Emergency 4 KV Buses E1(E3) and E2(E4) for ≥ 15 min. (Note 1)

Note 1: The SEC should declare the event promptly upon determining that time limit has been exceeded, or will likely be exceeded.

Mode Applicability:

1 - Power Operations, 2 - Startup, 3 - Hot Shutdown

Definition(s):

None

Basis:

This EAL is indicated by the loss of all offsite and onsite AC power capability to Emergency 4 KV Buses E1(E3) and E2(E4) for greater than or equal to 15 minutes.

The 4160 VAC System provides the power requirements for operation and safe shutdown of the plant. 4160 VAC power is divided into Balance Of Plant (BOP Bus), and Emergency (E-Bus) distribution. The BOP Buses consist of the Conventional Buses (Common A/B Buses; Buses 1B/2B, 1C/2C, 1D/2D) and Caswell Beach Buses A/B. The Emergency switchgear are Buses E1/E3 (Division I) and E2/E4 (Division II).

The E-Buses are normally powered through the respective BOP Buses (1D to E1, 1C to E2, 2D to E3, 2C to E4) via a master/slave breaker arrangement. Each E-Bus has a dedicated Diesel Generator to supply an emergency source of power to safe shutdown loads in the event of a loss of the normal power source or loss of offsite power. The DGs will automatically start and tie onto the E-Buses if the normal power source or offsite power is lost. The DGs can provide power to the E-Buses only. In the event the diesel generator is unavailable for an E-Bus, crosstie capability exists for each E-Bus from the same division of the opposite unit (E1 to E3, E2 to E4). Although the E-Buses within the unit also have crosstie capability, this alignment is not permitted by plant procedures, with the exception of E1 to E2 during specific Alternate Safe Shutdown (ASSD) conditions.

During periods of unit shutdown, when the Startup Auxiliary Transformer (SAT) would be the only normal source of offsite power, the Unit Auxiliary Transformer (UAT) can be made available by establishing a UAT backfeed. Backfeed from the UAT will require the use of keys for the control selector switches and opening of the respective generator's manual no-load disconnect (Ref. 1, 2)

The 15-minute interval was selected as a threshold to exclude transient or momentary power losses. The interval begins when both offsite and onsite AC power are lost.

This IC addresses a total loss of AC power that compromises the performance of all SAFETY SYSTEMS requiring electric power including those necessary for emergency core cooling, containment heat removal/pressure control, spent fuel heat removal and the ultimate heat sink. In addition, fission product barrier monitoring capabilities may be degraded under these conditions. This IC represents a condition that involves actual or likely major failures of plant functions needed for the protection of the public.

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

Escalation of the emergency classification level would be via ICs RG1, FG1 or SG1.

BNP Basis Reference(s):

1. BNP Updated FSAR Chapter 8
2. 1(2)OP-50 Plant Electric System Operating Procedure
3. 0AOP-36.2 Station Blackout
4. NEI 99-01 SS1

Category: S –System Malfunction

Subcategory: 1 – Loss of Emergency AC Power

Initiating Condition: Prolonged loss of **all** offsite and **all** onsite AC power to emergency buses **OR** loss of **all** emergency AC and vital DC power sources for 15 minutes or longer

EAL:

SG1.1 General Emergency

Loss of **all** offsite and **all** onsite AC power capability to Emergency 4 KV Buses E1(E3) and E2(E4)

AND EITHER:

- Restoration of at least one emergency bus in < 4 hours is **not** likely (Note 1)
- RPV water level **cannot** be restored and maintained > MSCRWL (LL-4)

Note 1: The SEC should declare the event promptly upon determining that time limit has been exceeded, or will likely be exceeded.

Mode Applicability:

1 - Power Operations, 2 - Startup, 3 - Hot Shutdown

Definition(s):

None

Basis:

This EAL is indicated by the extended loss of all offsite and onsite AC power capability to 4 KV emergency buses E1(E3) and E2(E4) either for greater then the BNP Station Blackout (SBO) coping analysis time (4 hrs.) (ref. 1) or that has resulted in indications of an actual loss of adequate core cooling.

Indication of continuing core cooling degradation is manifested by the inability to restore and maintain RPV water level above the Minimum Steam Cooling Reactor Water Level (LL-4) (ref. 4, 5).

The 4160 VAC System provides the power requirements for operation and safe shutdown of the plant. 4160 VAC power is divided into Balance of Plant (BOP Bus), and Emergency (E-Bus) distribution. The BOP Buses consist of the Conventional Buses (Common A/B Buses; Buses 1B/2B, 1C/2C, 1D/2D) and Caswell Beach Buses A/B. The Emergency switchgear are Buses E1/E3 (Division I) and E2/E4 (Division II).

The E-Buses are normally powered through the respective BOP Buses (1D to E1, 1C to E2, 2D to E3, 2C to E4) via a master/slave breaker arrangement. Each E-Bus has a dedicated Diesel Generator to supply an emergency source of power to safe shutdown loads in the event of a loss of the normal power source or loss of offsite power. The DGs will automatically start and tie onto the E-Buses if the normal power source or offsite power is lost. The DGs can provide power to the E-Buses only. In the event the diesel generator is unavailable for an E-Bus, crosstie capability exists for each E-Bus from the same division of the opposite unit (E1 to E3, E2 to E4). Although the E-Buses within the unit also have crosstie capability, this alignment is not permitted by plant procedures, with the exception of E1 to E2 during specific Alternate Safe Shutdown (ASSD) conditions.

During periods of unit shutdown, when the Startup Auxiliary Transformer (SAT) would be the only normal source of offsite power, the Unit Auxiliary Transformer (UAT) can be made available by establishing a UAT backfeed. Backfeed from the UAT will require the use of keys for the control selector switches and opening of the respective generator's manual no-load disconnect (Ref. 2, 3).

Four hours is the station blackout coping time (ref 1).

Indication of continuing core cooling degradation must be based on fission product barrier monitoring with particular emphasis on Site Emergency Coordinator judgment as it relates to imminent Loss or Potential Loss of fission product barriers and degraded ability to monitor fission product barriers. Indication of continuing core cooling degradation is manifested by an RPV level that cannot be restored and maintained > MSCRWL (LL-4) (ref. 4, 5). Core submergence is the most desirable means of core cooling, however when RPV level is below TAF, the uncovered portion of the core can be cooled by less reliable means (i.e., steam cooling or spray cooling).

This IC addresses a prolonged loss of all power sources to AC emergency buses. A loss of all AC power compromises the performance of all SAFETY SYSTEMS requiring electric power including those necessary for emergency core cooling, containment heat removal/pressure control, spent fuel heat removal and the ultimate heat sink. A prolonged loss of these buses will lead to a loss of one or more fission product barriers. In addition, fission product barrier monitoring capabilities may be degraded under these conditions.

The EAL should require declaration of a General Emergency prior to meeting the thresholds for IC FG1. This will allow additional time for implementation of offsite protective actions.

Escalation of the emergency classification from Site Area Emergency will occur if it is projected that power cannot be restored to at least one AC emergency bus by the end of the analyzed station blackout coping period. Beyond this time, plant responses and event trajectory are subject to greater uncertainty, and there is an increased likelihood of challenges to multiple fission product barriers.

The estimate for restoring at least one emergency bus should be based on a realistic appraisal of the situation. Mitigation actions with a low probability of success should not be used as a basis for delaying a classification upgrade. The goal is to maximize the time available to prepare for, and implement, protective actions for the public.

The EAL will also require a General Emergency declaration if the loss of AC power results in parameters that indicate an inability to adequately remove decay heat from the core.

BNP Basis Reference(s):

1. 0AOP-36.2 STATION BLACKOUT, Section 4.0
2. BNP Updated FSAR Chapter 8
3. 1(2)OP-50 Plant Electric System Operating Procedure
4. 1(2)EOP-01-RVCP, Reactor Vessel Control
5. 0EOP-01-NL, EOP/SAMG NUMERICAL LIMITS AND VALUES

Category: S –System Malfunction

Subcategory: 1 – Loss of Emergency AC Power

Initiating Condition: Prolonged loss of **all** offsite and **all** onsite AC power to emergency buses **OR** loss of **all** emergency AC and vital DC power sources for 15 minutes or longer

EAL:

SG1.2 General Emergency

Loss of **all** offsite and **all** onsite AC power capability to Emergency 4 KV Buses E1(E3) and E2(E4) for ≥ 15 min.

AND

Loss of **all** 125 VDC power based on battery bus voltage indications < 105 VDC on **all** vital DC buses 1(2)A-1, A-2, B-1 and B-2 for ≥ 15 min.

(Note 1)

Note 1: The SEC should declare the event promptly upon determining that time limit has been exceeded, or will likely be exceeded.

Mode Applicability:

1 - Power Operations, 2 - Startup, 3 - Hot Shutdown

Definition(s):

None

Basis:

This EAL is indicated by the loss of all offsite and onsite emergency AC power capability to 4 KV emergency buses E1(E3) and E2(E4) for greater than 15 minutes in combination with degraded vital DC power voltage. This EAL addresses operating experience from the March 2011 accident at Fukushima Daiichi.

The 4160 VAC System provides the power requirements for operation and safe shutdown of the plant. 4160 VAC power is divided into Balance of Plant (BOP Bus), and Emergency (E-Bus) distribution. The BOP Buses consist of the Conventional Buses (Common A/B Buses; Buses 1B/2B, 1C/2C, 1D/2D) and Caswell Beach Buses A/B. The Emergency switchgear are Buses E1/E3 (Division I) and E2/E4 (Division II).

The E-Buses are normally powered through the respective BOP Buses (1D to E1, 1C to E2, 2D to E3, 2C to E4) via a master/slave breaker arrangement. Each E-Bus has a dedicated Diesel Generator to supply an emergency source of power to safe shutdown loads in the event of a loss of the normal power source or loss of offsite power. The DGs will automatically start and tie onto the E-Buses if the normal power source or offsite power is lost. The DGs can provide power to the E-Buses only. In the event the diesel generator is unavailable for an E-Bus, crosstie capability exists for each E-Bus from the same division of the opposite unit (E1 to E3, E2 to E4). Although the E-Buses within the unit also have crosstie capability, this alignment is not permitted by plant procedures, with the exception of E1 to E2 during specific Alternate Safe Shutdown (ASSD) conditions.

During periods of unit shutdown, when the Startup Auxiliary Transformer (SAT) would be the only normal source of offsite power, the Unit Auxiliary Transformer (UAT) can be made available by establishing a UAT backfeed. Backfeed from the UAT will require the use of keys for the control selector switches and opening of the respective generator's manual no-load disconnect (Ref. 2, 3).

There are two independent vital 125 VDC divisions per unit, designated Division I and Division II (Batteries 1(2)A-1 and 1(2)A-2 for Division I and Batteries 1(2)B-1 and 1(2)B-2 for Division II). Each division consists of a 250 VDC battery center tapped to form two 125 VDC batteries. Each 125 VDC battery has an associated full capacity battery charger. The chargers are supplied from the same AC load groups for which the associated DC subsystem supplies the control power. During normal operation, the DC loads are powered from the battery chargers with the batteries floating on the system. In case of loss of normal power to the battery charger, the DC loads are automatically powered from the station batteries.

105 VDC is the minimum design voltage limit (ref. 4, 5).

In the event that DC battery voltage indication is not available via ERFIS, local voltage indication is available for each bus based on dispatching a field operator in accordance with 0AOP-39.0 Loss of DC Power. In this case the 15 minute classification clock begins upon receipt of the low voltage alarm in the Control Room. If battery voltage cannot be verified to be greater than or equal to 105 VDC within the 15 minutes, emergency classification must be made under this EAL.

This IC addresses a concurrent and prolonged loss of both emergency AC and Vital DC power. A loss of all emergency AC power compromises the performance of all SAFETY SYSTEMS requiring electric power including those necessary for emergency core cooling, containment heat removal/pressure control, spent fuel heat removal and the ultimate heat sink. A loss of vital DC power compromises the ability to monitor and control SAFETY SYSTEMS. A sustained loss of both emergency AC and vital DC power will lead to multiple challenges to fission product barriers.

Fifteen minutes was selected as a threshold to exclude transient or momentary power losses. The 15-minute emergency declaration clock begins at the point when both EAL thresholds are met.

BNP Basis Reference(s):

1. 0AOP-36.2 STATION BLACKOUT, Section 4.0
2. BNP Updated FSAR Chapter 8
3. 1(2)OP-50 Plant Electric System Operating Procedure
4. BNP Technical Specification Bases B.3.8.4
5. 0AOP-39.0 LOSS OF DC POWER

Category: S – System Malfunction

Subcategory: 2 – Loss of Vital DC Power

Initiating Condition: Loss of all vital DC power for 15 minutes or longer

EAL:

SS2.1	**Site Area Emergency**
Loss of **all** 125 VDC power based on battery bus voltage indications < 105 VDC on **all** vital DC buses 1(2)A-1, A-2, B-1 and B-2 for ≥ 15 min. (Note 1)	

Note 1: The SEC should declare the event promptly upon determining that time limit has been exceeded, or will likely be exceeded.

Mode Applicability:

1 - Power Operations, 2 - Startup, 3 - Hot Shutdown

Definition(s):

None

Basis:

There are two independent vital 125 VDC divisions per unit, designated Division I and Division II (Batteries 1(2)A-1 and 1(2)A-2 for Division I and Batteries 1(2)B-1 and 1(2)B-2 for Division II). Each division consists of a 250 VDC battery center tapped to form two 125 VDC batteries. Each 125 VDC battery has an associated full capacity battery charger. The chargers are supplied from the same AC load groups for which the associated DC subsystem supplies the control power. During normal operation, the DC loads are powered from the battery chargers with the batteries floating on the system. In case of loss of normal power to the battery charger, the DC loads are automatically powered from the station batteries.

Note that the Control Room DC voltage indicator only reads battery charger output voltage and not battery voltage unless the charger output breaker is closed. However ERFIS does provide DC battery voltage, otherwise battery voltage must be read locally.

In the event that DC battery voltage indication is not available via ERFIS, local voltage indication is available for each bus based on dispatching a field operator in accordance with 0AOP-39.0 Loss of DC Power. In this case the 15 minute classification clock begins upon receipt of the low voltage alarm in the Control Room. If battery voltage cannot be verified to be greater than or equal to 105 VDC within the 15 minutes, emergency classification must be made under this EAL.

105 VDC is the minimum design voltage limit (ref. 1).

This IC addresses a loss of vital DC power which compromises the ability to monitor and control SAFETY SYSTEMS. In modes above Cold Shutdown, this condition involves a major failure of plant functions needed for the protection of the public.

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

Escalation of the emergency classification level would be via ICs RG1, FG1 or SG1.

BNP Basis Reference(s):

1. BNP Technical Specification Bases B.3.8.4
2. 0AOP-39.0 LOSS OF DC POWER
3. NEI 99-01 SS8

Category: S – System Malfunction

Subcategory: 3 – Loss of Control Room Indications

Initiating Condition: UNPLANNED loss of Control Room indications for 15 minutes or longer

EAL:

SU3.1	**Unusual Event**
An UNPLANNED event results in the inability to monitor one or more Table S-1 parameters from within the Control Room for ≥ 15 min. (Note 1)	

Note 1: The SEC should declare the event promptly upon determining that time limit has been exceeded, or will likely be exceeded.

Table S-1 Safety System Parameters
• Reactor power
• RPV water level
• RPV pressure
• Primary containment pressure
• Torus water level
• Torus temperature

Mode Applicability:

1 - Power Operations, 2 - Startup, 3 - Hot Shutdown

Definition(s):

UNPLANNED - A parameter change or an event that is not 1) the result of an intended evolution or 2) an expected plant response to a transient. The cause of the parameter change or event may be known or unknown.

Basis:

SAFETY SYSTEM parameters listed in Table S-1 are monitored in the Control Room through a combination of hard control panel indicators as well as computer based information systems. The ERFIS, which displays SPDS required information, serves as a redundant compensatory indicator which may be utilized in lieu of normal Control Room indicators (ref. 1, 2).

As used in this EAL "within the Control Room" means any available indicator available within the Control Room boundary, including back panels.

This IC addresses the difficulty associated with monitoring normal plant conditions without the ability to obtain SAFETY SYSTEM parameters from within the Control Room. This condition is a precursor to a more significant event and represents a potential degradation in the level of safety of the plant.

As used in this EAL, an “inability to monitor” means that values for one or more of the listed parameters cannot be determined from within the Control Room. This situation would require a loss of all of the Control Room sources for the given parameter(s). For example, the reactor power level cannot be determined from any analog, digital or recorder source within the Control Room.

An event involving a loss of plant indications, annunciators and/or display systems is evaluated in accordance with 10 CFR 50.72 (and associated guidance in NUREG-1022) to determine if an NRC event report is required. The event would be reported if it significantly impaired the capability to perform emergency assessments. In particular, emergency assessments necessary to implement abnormal operating procedures, emergency operating procedures, and emergency plan implementing procedures addressing emergency classification, accident assessment, or protective action decision-making.

This EAL is focused on a selected subset of plant parameters associated with the key safety functions of reactivity control, RPV level and RCS heat removal. The loss of the ability to determine one or more of these parameters from within the Control Room is considered to be more significant than simply a reportable condition. In addition, if all indication sources for one or more of the listed parameters are lost, then the ability to determine the values of other SAFETY SYSTEM parameters may be impacted as well. For example, if the value for RPV water level cannot be determined from the indications and recorders on a main control board, the SPDS or the plant computer, the availability of other parameter values may be compromised as well.

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

Escalation of the emergency classification level would be via IC SA3.

BNP Basis Reference(s):

1. Updated FSAR Update Section 7.7.1.9
2. 0OI-01.08 Control of Equipment and System Status
3. NEI 99-01 SU2

Category: S – System Malfunction

Subcategory: 3 – Loss of Control Room Indications

Initiating Condition: UNPLANNED loss of Control Room indications for 15 minutes or longer with a significant transient in progress

EAL:

SA3.1 Alert

An UNPLANNED event results in the inability to monitor one or more Table S-1 parameters from within the Control Room for ≥ 15 min. (Note 1)

AND

Any significant transient is in progress, Table S-2

Note 1: The SEC should declare the event promptly upon determining that time limit has been exceeded, or will likely be exceeded.

Table S-1 Safety System Parameters
• Reactor power
• RPV water level
• RPV pressure
• Primary containment pressure
• Torus water level
• Torus temperature

Table S-2 Significant Transients
• Reactor scram
• Runback > 25% rated thermal power
• Electrical load rejection > 25% electrical load
• ECCS injection
• Thermal power oscillations > 10% (peak to peak)

Mode Applicability:

1 - Power Operations, 2 - Startup, 3 - Hot Shutdown

Definition(s):

UNPLANNED - A parameter change or an event that is not 1) the result of an intended evolution or 2) an expected plant response to a transient. The cause of the parameter change or event may be known or unknown.

Basis:

SAFETY SYSTEM parameters listed in Table S-1 are monitored in the Control Room through a combination of hard control panel indicators as well as computer based information systems. The ERFIS, which displays SPDS required information, serves as a redundant compensatory indicator which may be utilized in lieu of normal Control Room indicators (ref. 1, 2).

Significant transients are listed in Table S-2 and include response to automatic or manually initiated functions such as scrams, runbacks (Recirculation) involving greater than 25% thermal power change, electrical load rejections of greater than 25% full electrical load, ECCS injections, or thermal power oscillations of 10% (peak to peak) or greater.

As used in this EAL "within the Control Room" means any available indicator available within the Control Room boundary, including back panels.

This IC addresses the difficulty associated with monitoring rapidly changing plant conditions during a transient without the ability to obtain SAFETY SYSTEM parameters from within the Control Room. During this condition, the margin to a potential fission product barrier challenge is reduced. It thus represents a potential substantial degradation in the level of safety of the plant.

As used in this EAL, an "inability to monitor" means that values for one or more of the listed parameters cannot be determined from within the Control Room. This situation would require a loss of all of the Control Room sources for the given parameter(s). For example, the reactor power level cannot be determined from any analog, digital or recorder source within the Control Room.

An event involving a loss of plant indications, annunciators and/or display systems is evaluated in accordance with 10 CFR 50.72 (and associated guidance in NUREG-1022) to determine if an NRC event report is required. The event would be reported if it significantly impaired the capability to perform emergency assessments. In particular, emergency assessments necessary to implement abnormal operating procedures, emergency operating procedures, and emergency plan implementing procedures addressing emergency classification, accident assessment, or protective action decision-making.

This EAL is focused on a selected subset of plant parameters associated with the key safety functions of reactivity control, RPV level and RCS heat removal. The loss of the ability to determine one or more of these parameters from within the Control Room is considered to be more significant than simply a reportable condition. In addition, if all indication sources for one or more of the listed parameters are lost, then the ability to determine the values of other SAFETY SYSTEM parameters may be impacted as well. For example, if the value for RPV water level cannot be determined from the indications and recorders on a main control board, the SPDS or the plant computer, the availability of other parameter values may be compromised as well.

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

Escalation of the emergency classification level would be via ICs FS1 or IC RS1

BNP Basis Reference(s):

1. Updated FSAR Update Section 7.7.1.9
2. 0OI-01.08 Control of Equipment and System Status
3. NEI 99-01 SA2

Category: S – System Malfunction

Subcategory: 4 – RCS Activity

Initiating Condition: Reactor coolant activity greater than Technical Specification allowable limits

EAL:

SU4.1	**Unusual Event**

Steam Jet Air Ejector Radiation Monitor 1(2)D12-RM-K601A /B Hi-Hi alarm (Process Off-Gas Rad Hi-Hi alarm 1(2)UA-03 4-2

Mode Applicability:

1 - Power Operations, 2 - Startup, 3 - Hot Shutdown

Definition(s):

None

Basis:

The Steam Jet Air Ejector radiation monitor setpoint provides reasonable assurance that the total body exposure to an individual at the exclusion area boundary will not exceed a small fraction of the limits of 10CFR100 in the event of an inadvertent release via the condenser air ejector (ref. 2, 3).

At the Hi-Hi alarm setpoint, the process Off-Gas timer is started. After the process Off-Gas timer has timed out (15 minutes), the Off-Gas system will isolate (ref. 1).

This IC addresses a reactor coolant activity value that exceeds an allowable limit specified in Technical Specifications. This condition is a precursor to a more significant event and represents a potential degradation of the level of safety of the plant.

Escalation of the emergency classification level would be via ICs FA1 or the Recognition Category R ICs.

BNP Basis Reference(s):

1. ARP 1(2)APP-UA-03 4-2 Process Off-Gas Rad Hi-Hi
2. BNP Offsite Dose Calculation Manual section 3.1.3
3. BNP Technical Specifications section 3.7.5
4. NEI 99-01 SU3

Category: S – System Malfunction

Subcategory: 4 – RCS Activity

Initiating Condition: Reactor coolant activity greater than Technical Specification allowable limits

EAL:

SU4.2	**Unusual Event**
Coolant activity > 0.2 μCi/gm I-131 dose equivalent for > 48 hours (Note 1)	
OR	
Coolant activity > 4.0 μCi/gm I-131 dose equivalent instantaneous	

Note 1: The SEC should declare the event promptly upon determining that time limit has been exceeded, or will likely be exceeded.

Mode Applicability:

1 - Power Operations, 2 - Startup, 3 - Hot Shutdown

Definition(s):

None

Basis:

The specific iodine activity is limited to ≤ 0.2 μCi/gm Dose Equivalent I-131. This limit ensures the source term assumed in the safety analysis for the Main Steam Line Break (MSLB) is not exceeded, so any release of radioactivity to the environment during an MSLB is less than a small fraction of the 10 CFR 50.67 limits (ref. 1).

The upper limit of 4.0 μCi/gm Dose Equivalent I-131 ensures that the thyroid dose from an MSLB will not exceed the dose guidelines of 10 CFR 50.67 or Control Room operator dose limits specified in GDC 19 of 10 CFR 50, Appendix A (ref. 1).

This IC addresses a reactor coolant activity value that exceeds an allowable limit specified in Technical Specifications. This condition is a precursor to a more significant event and represents a potential degradation of the level of safety of the plant.

Escalation of the emergency classification level would be via ICs FA1 or the Recognition Category R ICs.

BNP Basis Reference(s):

1. BNP Technical Specifications section 3.4.6
2. NEI 99-01 SU3

Category: S – System Malfunction

Subcategory: 5 – RCS Leakage

Initiating Condition: RCS leakage for 15 minutes or longer

EAL:

SU5.1 **Unusual Event**
RCS unidentified or pressure boundary leakage > 10 gpm for ≥ 15 min. **OR** RCS identified leakage > 25 gpm for ≥ 15 min. **OR** Leakage from the RCS to a location outside Primary Containment > 25 gpm for ≥ 15 min. (Note 1)

Note 1: The SEC should declare the event promptly upon determining that time limit has been exceeded, or will likely be exceeded.

Mode Applicability:

1 - Power Operations, 2 - Startup, 3 - Hot Shutdown

Definition(s):

None

Basis:

Leakage is monitored by utilizing the following techniques:

- Sensing excess flow in piping systems
- Sensing pressure and temperature changes in the primary containment
- Monitoring for high flow and temperature through selected drains,
- Sampling airborne particulate and gaseous radioactivity.
- Drywell floor and equipment drain sump leak rate system

Identified leakage is leakage into the drywell, such as that from pump seals or valve packing, that is captured and conducted to a sump or collecting tank; or leakage into the drywell atmosphere from sources that are both specifically located and known either not to interfere with the operation of leakage detection systems or not to be pressure boundary leakage. (ref. 1, 2)

Unidentified leakage is all leakage into the drywell that is not identified leakage. (ref. 1, 2)

Pressure boundary leakage is leakage through a nonisolable fault in a Reactor Coolant System (RCS) component body, pipe wall, or vessel wall. (ref. 1, 2)

The drywell floor drain sump flow monitoring system monitors the leakage collected in the floor drain sump. This unidentified leakage consists of leakage from control rod drives, valve flanges, floor drains, the Reactor Building Closed Cooling Water System, and drywell cooler drains, and any leakage not collected in the drywell equipment drain sump. The drywell floor drain sump is provided with two sump pumps. A flow transmitter in the common discharge line of the drywell floor drain sump pumps inputs to a flow integrator. In addition to the required instrumentation, the starting frequency and run duration of a sump pump motor are monitored by timer circuitry to provide a signal (alarm) in the Control Room indicating that leakage has reached a specified limit. (ref. 2)

RCS leakage outside of the Primary Containment that is not considered identified or unidentified leakage per Technical Specifications includes leakage via interfacing systems such as RCS to the Reactor Building Closed Cooling Water (RBCCW system), or systems that directly see RCS pressure outside primary containment such as Reactor Water Cleanup (RWCU), reactor water sampling system and Residual Heat Removal (RHR) system (when in the shutdown cooling mode) (ref. 3)

Escalation of this EAL to the Alert level is via Category F, Fission Product Barrier Degradation, EAL FA1.1. The note has been added to remind the EAL-user to review Table F-1 for possible escalation to higher emergency classifications.

This IC addresses RCS leakage which may be a precursor to a more significant event. In this case, RCS leakage has been detected and operators, following applicable procedures, have been unable to promptly isolate the leak. This condition is considered to be a potential degradation of the level of safety of the plant.

The first and second EAL conditions are focused on a loss of mass from the RCS due to "unidentified leakage", "pressure boundary leakage" or "identified leakage" (as these leakage types are defined in the plant Technical Specifications). The third condition addresses an RCS mass loss caused by an UNISOLABLE leak through an interfacing system. These conditions thus apply to leakage into the Primary Containment, or a location outside of Primary Containment.

The leak rate values for each condition were selected because they are usually observable with normal Control Room indications. Lesser values typically require time-consuming calculations to determine (e.g., a mass balance calculation). The first condition uses a lower value that reflects the greater significance of unidentified or pressure boundary leakage.

A stuck-open Safety Relief Valve (SRV) or SRV leakage is not considered either identified or unidentified leakage by Technical Specifications and, therefore, is not applicable to this EAL.

The 15-minute threshold duration allows sufficient time for prompt operator actions to isolate the leakage, if possible.

Escalation of the emergency classification level would be via ICs of Recognition Category R or F.

BNP Basis Reference(s):

1. BNP Technical Specifications Definitions section 1.1
2. BNP Technical Specifications Bases 3.4.5
3. BNP UFSAR section 5.1 Reactor Coolant System and Connected Systems
4. NEI 99-01 SU4

Category: S – System Malfunction

Subcategory: 6 – RPS Failure

Initiating Condition: Automatic or manual scram fails to shut down the reactor

EAL:

SU6.1 Unusual Event
An automatic scram did **not** reduce reactor power to < 2% (APRM downscale) after **any** RPS setpoint is exceeded **AND** A subsequent automatic scram or manual scram action taken at the reactor control console (Manual PBs, Mode Switch, ARI) is successful in shutting down the reactor as indicated by reactor power < 2% (APRM downscale) (Note 8)

Note 8: A manual scram action is any operator action, or set of actions, which causes the control rods to be rapidly inserted into the core, and **does not** include manually driving in control rods or implementation of boron injection strategies.

Mode Applicability:

1 - Power Operations, 2 - Startup

Definition(s):

None

Basis:

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 (ref. 1).

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 below the APRM downscale setpoint of 2%.

For the purposes of emergency classification, successful manual scram actions are those which can be quickly performed from the reactor control console (i.e., manual scram pushbuttons, mode switch, or ARI actuation). Reactor shutdown achieved by use of the 0EOP-01-LEP-02 actions does not constitute a successful manual scram (ref. 4).

Following any automatic RPS scram signal, 1(2)EOP-01-RSP (ref. 2) and 1(2)EOP-01-ATWS (ref. 3) 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 Unusual Event.

Taking the mode switch to shutdown is a manual scram action. When the Mode Switch is taken out of the Run position, however, the nuclear instrumentation scram setpoint is lowered. If reactor power remains above the lowered setpoint, an automatic scram is initiated.

For the purposes of this EAL, a successful <u>automatic</u> initiation of ARI that reduces reactor power below 2% is <u>not</u> considered 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. However, a successful automatic or manual initiation of ARI is an acceptable means of establishing reactor shutdown conditions relative to the EAL threshold in the absence of any required subsequent manual scram actions.

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 below 2% (ref. 2, 3), the event escalates to the Alert under EAL SA6.1.

If by procedure, operator actions include the initiation of an immediate manual scram following receipt of an automatic scram signal and there are no clear indications that the automatic scram failed (such as a time delay following indications that a scram setpoint was exceeded), it may be difficult to determine if the reactor was shut down because of automatic scram or manual actions. If a subsequent review of the scram actuation indications reveals that the automatic scram did not cause the reactor to be shut down, then consideration should be given to evaluating the fuel for potential damage, and the reporting requirements of 50.72 should be considered for the transient event.

This IC addresses a failure of the RPS to initiate or complete an automatic or manual reactor scram that results in a reactor shutdown, and either a subsequent operator manual action taken at the reactor control consoles or an automatic scram is successful in shutting down the reactor. This event is a precursor to a more significant condition and thus represents a potential degradation of the level of safety of the plant.

Following the failure on an automatic reactor scram, operators will promptly initiate manual actions at the reactor control consoles to shutdown the reactor (e.g., initiate a manual reactor scram). If these manual actions are successful in shutting down the reactor, core heat generation will quickly fall to a level within the capabilities of the plant's decay heat removal systems.

If an initial manual reactor scram is unsuccessful, operators will promptly take manual action at another location(s) on the reactor control consoles to shutdown the reactor (e.g., initiate a manual reactor scram) using a different switch). Depending upon several factors, the initial or subsequent effort to manually scram the reactor, or a concurrent plant condition, may lead to the generation of an automatic reactor scram signal. If a subsequent manual or automatic scram is successful in shutting down the reactor, core heat generation will quickly fall to a level within the capabilities of the plant's decay heat removal systems.

A manual action at the reactor control consoles is any operator action, or set of actions, which causes the control rods to be rapidly inserted into the core (e.g., initiating a manual reactor scram). This action does not include manually driving in control rods or implementation of boron injection strategies. Actions taken at back-panels or other locations within the Control Room, or any location outside the Control Room, are not considered to be "at the reactor control consoles".

Taking the Reactor Mode Switch to SHUTDOWN is considered to be a manual scram action.

The plant response to the failure of an automatic or manual reactor scram will vary based upon several factors including the reactor power level prior to the event, availability of the condenser, performance of mitigation equipment and actions, other concurrent plant conditions, etc. If subsequent operator manual actions taken at the reactor control consoles are also unsuccessful in shutting down the reactor, then the emergency classification level will escalate to an Alert via IC SA6. Depending upon the plant response, escalation is also possible via IC FA1. Absent the plant conditions needed to meet either IC SA6 or FA1, an Unusual Event declaration is appropriate for this event.

Should a reactor scram signal be generated as a result of plant work (e.g., RPS setpoint testing), the following classification guidance should be applied.

- If the signal causes a plant transient that should have included an automatic reactor scram and the RPS fails to automatically shutdown the reactor, then this IC and the EALs are applicable, and should be evaluated.
- If the signal does not cause a plant transient and the scram failure is determined through other means (e.g., assessment of test results), then this IC and the EALs are not applicable and no classification is warranted.

BNP Basis Reference(s):

1. BNP Technical Specifications section 3.3.1.1 RPS Instrumentation
2. 1(2) EOP-01-RSP, Reactor Scram
3. 1(2) EOP-01-ATWS, ATWS
4. 0EOP-01-LEP-02 Alternate Control Rod Insertion
5. NEI 99-01 SU5

Category: S – System Malfunction

Subcategory: 6 – RPS Failure

Initiating Condition: Automatic or manual scram fails to shut down the reactor

EAL:

SU6.2 **Unusual Event**
A manual scram did **not** reduce reactor power to < 2% (APRM downscale) after **any** manual scram action was initiated **AND** A subsequent automatic scram or manual scram action taken at the reactor control console (Manual PBs, Mode Switch, ARI) is successful in shutting down the reactor as indicated by reactor power < 2% (APRM downscale) (Note 8)

Note 8: A manual scram action is any operator action, or set of actions, which causes the control rods to be rapidly inserted into the core, and **does not** include manually driving in control rods or implementation of boron injection strategies.

Mode Applicability:

1 - Power Operations, 2 - Startup

Definition(s):

None

Basis:

This EAL addresses a failure of a manually initiated scram in the absence of having exceeded an automatic RPS trip setpoint and a subsequent automatic or manual scram is successful in shutting down the reactor (reactor power < 2%). (ref. 1).

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 a manual 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 below the APRM downscale setpoint of 2%.

For the purposes of emergency classification, successful manual scram actions are those which can be quickly performed from the reactor control console (i.e., manual scram pushbuttons, mode switch, or ARI actuation). Reactor shutdown achieved by use of the 0EOP-01-LEP-02 actions does not constitute a successful manual scram (ref. 2, 3).

Taking the mode switch to shutdown is a manual scram action. When the Mode Switch is taken out of the Run position, however, the nuclear instrumentation scram setpoint is lowered. If reactor power remains above the lowered setpoint, an automatic scram is initiated.

Successful automatic or manual initiation of ARI is an acceptable means of establishing reactor shutdown conditions relative to the EAL threshold in the absence of any required subsequent manual scram actions.

If both subsequent automatic and subsequent manual reactor scram actions in the Control Room fail to reduce reactor power below the power associated with the safety system design (< 2%) following a failure of an initial manual scram, the event escalates to an Alert under EAL SA6.1

This IC addresses a failure of the RPS to initiate or complete an automatic or manual reactor scram that results in a reactor shutdown, and either a subsequent operator manual action taken at the reactor control consoles or an automatic scram is successful in shutting down the reactor. This event is a precursor to a more significant condition and thus represents a potential degradation of the level of safety of the plant.

Following the failure on an automatic reactor scram, operators will promptly initiate manual actions at the reactor control consoles to shutdown the reactor (e.g., initiate a manual reactor scram). If these manual actions are successful in shutting down the reactor, core heat generation will quickly fall to a level within the capabilities of the plant's decay heat removal systems.

If an initial manual reactor scram is unsuccessful, operators will promptly take manual action at another location(s) on the reactor control consoles to shutdown the reactor (e.g., initiate a manual reactor scram) using a different switch). Depending upon several factors, the initial or subsequent effort to manually scram the reactor, or a concurrent plant condition, may lead to the generation of an automatic reactor scram signal. If a subsequent manual or automatic scram is successful in shutting down the reactor, core heat generation will quickly fall to a level within the capabilities of the plant's decay heat removal systems.

A manual action at the reactor control consoles is any operator action, or set of actions, which causes the control rods to be rapidly inserted into the core (e.g., initiating a manual reactor scram). This action does not include manually driving in control rods or implementation of boron injection strategies. Actions taken at back-panels or other locations within the Control Room, or any location outside the Control Room, are not considered to be "at the reactor control consoles".

Taking the Reactor Mode Switch to SHUTDOWN is considered to be a manual scram action.

The plant response to the failure of an automatic or manual reactor scram will vary based upon several factors including the reactor power level prior to the event, availability of the condenser, performance of mitigation equipment and actions, other concurrent plant conditions, etc. If subsequent operator manual actions taken at the reactor control consoles are also unsuccessful in shutting down the reactor, then the emergency classification level will escalate to an Alert via IC SA6. Depending upon the plant response, escalation is also possible via IC FA1. Absent the plant conditions needed to meet either IC SA6 or FA1, an Unusual Event declaration is appropriate for this event.

Should a reactor scram signal be generated as a result of plant work (e.g., RPS setpoint testing), the following classification guidance should be applied.

- If the signal causes a plant transient that should have included an automatic reactor scram and the RPS fails to automatically shutdown the reactor, then this IC and the EALs are applicable, and should be evaluated.
- If the signal does not cause a plant transient and the scram failure is determined through other means (e.g., assessment of test results), then this IC and the EALs are not applicable and no classification is warranted.

BNP Basis Reference(s):

1. BNP Technical Specifications section 3.3.1.1 RPS Instrumentation
2. 1(2) EOP-01-RSP, Reactor Scram
3. 1(2) EOP-01-ATWS, ATWS
4. NEI 99-01 SU5

Category: S – System Malfunction

Subcategory: 2 – RPS Failure

Initiating Condition: Automatic or manual scram fails to shut down the reactor and subsequent manual actions taken at the reactor control consoles are not successful in shutting down the reactor

EAL:

SA6.1 Alert

An automatic or manual scram fails to reduce reactor power to < 2% (APRM downscale)

AND

Manual scram actions taken at the reactor control console (Manual PBs, Mode Switch, ARI) are **not** successful in shutting down the reactor as indicated by reactor power ≥ 2% (Note 8)

Note 8: A manual scram action is any operator action, or set of actions, which causes the control rods to be rapidly inserted into the core, and **does not** include manually driving in control rods or implementation of boron injection strategies.

Mode Applicability:

1 - Power Operations, 2 - Startup

Definition(s):

None

Basis:

This EAL addresses any automatic or manual reactor scram signal that fails to shut down the reactor followed by a subsequent 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.

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., manual scram pushbuttons, mode switch, or ARI actuation). Reactor shutdown achieved by use of the 0EOP-01-LEP-02 actions does not constitute a successful manual scram (ref. 1).

For the purposes of this EAL, a successful automatic initiation of ARI that reduces reactor power below 2% is not considered 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. However, a successful automatic initiation of ARI is an acceptable means of establishing reactor shutdown conditions relative to the EAL threshold in the absence of any required subsequent manual scram actions.

The APRM downscale trip setpoint (2%) is a minimum reading on the power range scale that indicates power production. 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. Below the APRM downscale trip setpoint, plant response will be similar to that observed during a normal shutdown. Nuclear instrumentation (APRM) indications or other reactor parameters (steam flow, RPV pressure, torus temperature trend) can be used to determine if reactor power is greater than 2 % power (ref. 2, 3).

Escalation of this event to a Site Area Emergency would be under EAL SS6.

This IC addresses a failure of the RPS to initiate or complete an automatic or manual reactor scram that results in a reactor shutdown, and subsequent operator manual actions taken at the reactor control consoles to shutdown the reactor are also unsuccessful. This condition represents an actual or potential substantial degradation of the level of safety of the plant. An emergency declaration is required even if the reactor is subsequently shutdown by an action taken away from the reactor control consoles since this event entails a significant failure of the RPS.

A manual action at the reactor control console is any operator action, or set of actions, which causes the control rods to be rapidly inserted into the core (e.g., initiating a manual reactor scram). This action does not include manually driving in control rods or implementation of boron injection strategies. If this action(s) is unsuccessful, operators would immediately pursue additional manual actions at locations away from the reactor control console (e.g., locally opening breakers). Actions taken at backpanels or other locations within the Control Room, or any location outside the Control Room, are not considered to be "at the reactor control console".

Taking the Reactor Mode Switch to SHUTDOWN is considered to be a manual scram action.

The plant response to the failure of an automatic or manual reactor scram will vary based upon several factors including the reactor power level prior to the event, availability of the condenser, performance of mitigation equipment and actions, other concurrent plant conditions, etc. If the failure to shut down the reactor is prolonged enough to cause a challenge to RPV water level or RCS heat removal safety functions, the emergency classification level will escalate to a Site Area Emergency via IC SS6. Depending upon plant responses and symptoms, escalation is also possible via IC FS1. Absent the plant conditions needed to meet either IC SS6 or FS1, an Alert declaration is appropriate for this event.

It is recognized that plant responses or symptoms may also require an Alert declaration in accordance with the Recognition Category F ICs; however, this IC and EAL are included to ensure a timely emergency declaration.

BNP Basis Reference(s):

1. 0EOP-01-LEP-02, Alternate Control Rod Insertion
2. 1(2) EOP-01-RSP, Reactor Scram
3. 1(2) EOP-01-ATWS, ATWS
4. NEI 99-01 SA5

Category: S – System Malfunction

Subcategory: 2 – RPS Failure

Initiating Condition: Inability to shut down the reactor causing a challenge to RPV water level or RCS heat removal

EAL:

SS6.1 Site Area Emergency

An automatic or manual scram fails to reduce reactor power to < 2% (APRM downscale)

AND

All actions to shut down the reactor are **not** successful as indicated by reactor power ≥ 2%

AND EITHER:

- RPV level **cannot** be restored and maintained > LL-4 or **cannot** be determined
- Suppression pool water temperature and RPV pressure **cannot** be maintained below the HCTL

Mode Applicability:

1 - Power Operations, 2 - Startup

Definition(s):

None

Basis:

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 SA6.1), and
- Indications that either core cooling is extremely challenged or heat removal is extremely challenged.

Reactor shutdown achieved by use of 0EOP-01-LEP-02 Alternate Control Rod Insertion 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.

The APRM downscale trip setpoint (2%) is a minimum reading on the power range scale that indicates power production. 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. Below the APRM downscale trip setpoint, plant response will be similar to that observed during a normal shutdown. Nuclear instrumentation (APRM) indications or other reactor parameters (steam flow, RPV pressure, torus temperature trend) can be used to determine if reactor power is greater than 2% power (ref. 1, 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.

Indication that core cooling is extremely challenged is manifested by inability to restore and maintain RPV water level above LL-4. LL-4 is the Minimum Steam Cooling RPV Water Level (MSCRWL) (ref. 3). The MSCRWL is the lowest RPV level at which the covered portion of the reactor core will generate sufficient steam to prevent any clad temperature in the uncovered part of the core from exceeding 1500°F. This water level is utilized in the EOPs to preclude fuel damage when RPV level is below the top of active fuel. RPV level below the MSCRWL for an extended period of time without satisfactory core spray cooling could be a precursor of a core melt sequence. When RPV level cannot be determined, EOPs require entry to 0EOP-01-RXFP, Reactor Flooding. 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 0EOP-01-RXFP 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 Alternate Flooding Pressure (ref. 4).

The Heat Capacity Temperature Limit (HCTL) is the highest suppression pool water temperature from which Emergency RPV Depressurization will not raise suppression chamber pressure above Primary Containment Pressure Limit A (PCPL-A), while 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. This threshold is met when the final step SP/T-13 of section SP/T in 0EOP-02-PCCP, Primary Containment Control, is reached (ref. 5). This condition addresses loss of functions required for hot shutdown with the reactor at pressure and temperature.

This IC addresses a failure of the RPS to initiate or complete an automatic or manual reactor scram that results in a reactor shutdown, all subsequent operator actions to manually shutdown the reactor are unsuccessful, and continued power generation is challenging the capability to adequately remove heat from the core and/or the RCS. This condition will lead to fuel damage if additional mitigation actions are unsuccessful and thus warrants the declaration of a Site Area Emergency.

In some instances, the emergency classification resulting from this IC/EAL may be higher than that resulting from an assessment of the plant responses and symptoms against the Recognition Category F ICs/EALs. This is appropriate in that the Recognition Category F ICs/EALs do not address the additional threat posed by a failure to shut down the reactor. The inclusion of this IC and EAL ensures the timely declaration of a Site Area Emergency in response to prolonged failure to shutdown the reactor.

Escalation of the emergency classification level would be via IC RG1 or FG1.

BNP Basis Reference(s):

1. 1(2) EOP-01-RSP, Reactor Scram
2. 1(2) EOP-01-ATWS, ATWS
3. 0EOP-01-NL EOP/SAMG Numerical Limits and Values, Attachment 1, pg 37-40, Figures 1-10 and 1-11
4. 0EOP-01-RXFP, Reactor Flooding
5. 0EOP-02-PCCP, Primary Containment Control
6. NEI 99-01 SS5

Category: S – System Malfunction

Subcategory: 7 – Loss of Communications

Initiating Condition: Loss of **all** onsite or offsite communications capabilities

EAL:

SU7.1 Unusual Event

Loss of **all** Table S-3 onsite communication methods

OR

Loss of **all** Table S-3 offsite communication methods

OR

Loss of **all** Table S-3 NRC communication methods

Table S-3 Communication Methods

System	Onsite	Offsite	NRC
Public Address System	X		
PBX Telephone System	X	X	X
DEMNET		X	X
Commercial Telephones	X	X	X
Satellite Phones		X	X
Cellular Phones		X	X
NRC Emergency Telecommunications System			X

Mode Applicability:

1 - Power Operations, 2 - Startup, 3 - Hot Shutdown

Definition(s):

None

Basis:

Onsite/offsite/NRC communications include one or more of the systems listed in Table S-3 (ref. 1).

Public Address System

The Brunswick Plant public address system provides paging and party line communications between stations located throughout the plant. Inside and outside type wall and desk-mounted stations are used to communicate between roaming personnel and fixed work locations. Plant-wide instructions are issued using the paging feature. This system is powered from the plant uninterruptible power supply which employs battery reserve as well as diesel generator emergency supply.

PBX Telephone System

The Brunswick Site PBX telephone system provides communication capability between telephone stations located within the plant by dialing the four-digit telephone station code. The PBX telephone system also provides for outside communications. The PBX switch located in the TSC/EOF building is also backed up by a battery UPS capable of supplying power for a minimum of 8 hours and is augmented by a Diesel Generator capable of supplying power to the TSC/EOF building for at least 5 days.

DEMNET

DEMNET is the primary means of offsite communication. This circuit allows intercommunication among the EOF, TSC, control room, counties, and states. DEMNET operates as an internet based (VoIP) communications system with a satellite back-up. Should the internet transfer rate become slow or unavailable, the DEMNET will automatically transfer to satellite mode.

Commercial Telephones

Commercial telephone lines, which supply public telephone communications, are employed by Duke Energy in four ways: (1) tie-ins through the PBX to any other plant location, (2) lines to plant emergency facilities, (3) lines to the Joint Information Center for public information purposes, and (4) lines to the AEF. The local service provider provides primary and secondary power for their lines at the Central Office.

Satellite Phones

A total of three portable satellite telephones are available which enable communication when all other phone systems are inoperable, e.g. following a major hurricane. These portable systems can be powered by internal batteries, external DC sources as well as external AC sources. Two of these phones require outside use, while one phone may used either outside or in the EOF with a permanently mounted external antenna.

Cellular Phones

Selected plant personnel are provided with cellular telephones. These phones may be used during emergencies if other communications means are not readily available or are inoperable. These phones are not expected to be used in the Control Room or Power Block due to interference with plant equipment and loss of signal to the phone.

NRC Emergency Telecommunications System

The NRC uses a Duke Energy dedicated telephone line which allows direct telephone communications from the plant to NRC regional and national offices. The Duke Energy communications line provides a link independent of the local public telephone network. Telephones connected to this network are located in the Brunswick Control Room, Technical Support Center, and Emergency Operations Facility. These lines will not function if the PBX Telephone System fails.

This EAL is the hot condition equivalent of the cold condition EAL CU5.1.

This IC addresses a significant loss of onsite or offsite communications capabilities. While not a direct challenge to plant or personnel safety, this event warrants prompt notifications to OROs and the NRC.

This IC should be assessed only when extraordinary means are being utilized to make communications possible (e.g., use of non-plant, privately owned equipment, relaying of onsite information via individuals or multiple radio transmission points, individuals being sent to offsite locations, etc.).

The first EAL condition addresses a total loss of the communications methods used in support of routine plant operations.

The second EAL condition addresses a total loss of the communications methods used to notify all OROs of an emergency declaration. The OROs referred to here are the State, Brunswick and New Hanover County EOCs

The third EAL addresses a total loss of the communications methods used to notify the NRC of an emergency declaration.

BNP Basis Reference(s):

1. 0ERP Radiological Emergency Response Plan Appendix A
2. SD-48 Communication Systems
3. NEI 99-01 SU6

Category: S – System Malfunction

Subcategory: 8 – Hazardous Event Affecting Safety Systems

Initiating Condition: Hazardous event affecting a SAFETY SYSTEM needed for the current operating mode

EAL:

SA8.1 Alert

The occurrence of **any** Table S-4 hazardous event

AND EITHER:

- Event damage has caused indications of degraded performance in at least one train of a SAFETY SYSTEM needed for the current operating mode
- The event has caused VISIBLE DAMAGE to a SAFETY SYSTEM component or structure needed for the current operating mode

Table S-4 Hazardous Events
• Seismic event (earthquake) • Internal or external FLOODING event • High winds or tornado strike • FIRE • EXPLOSION • Other events with similar hazard characteristics as determined by the Shift Manager

Mode Applicability:

1 - Power Operations, 2 - Startup, 3 - Hot Shutdown

Definition(s):

EXPLOSION - A rapid, violent and catastrophic failure of a piece of equipment due to combustion, chemical reaction or overpressurization. A release of steam (from high energy lines or components) or an electrical component failure (caused by short circuits, grounding, arcing, etc.) should not automatically be considered an explosion. Such events require a post-event inspection to determine if the attributes of an explosion are present.

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.

FLOODING - A condition where water is entering a room or area faster than installed equipment is capable of removal, resulting in a rise of water level within the room or area.

SAFETY SYSTEM - A system required for safe plant operation, cooling down the plant and/or placing it in the cold shutdown condition, including the ECCS. These are typically systems classified as safety-related (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.

VISIBLE DAMAGE - Damage to a component or structure that is readily observable without measurements, testing, or analysis. The visual impact of the damage is sufficient to cause concern regarding the operability or reliability of the affected component or structure.

Basis:

- The significance of seismic events are discussed under EAL HU2.1 (ref. 1).
- Internal FLOODING may be caused by events such as component failures, equipment misalignment, or outage activity mishaps (ref. 2, 3).
- Seismic Category I structures are analyzed to withstand a sustained, design wind velocity of at least 135 mph. (ref. 4).

- Areas containing functions and systems required for safe shutdown of the plant are identified by Fire Zone in the fire response procedure (ref. 5, 6).

- An explosion that degrades the performance of a SAFETY SYSTEM train or visibly damages a SAFETY SYSTEM component or structure would be classified under this EAL.

This IC addresses a hazardous event that causes damage to a SAFETY SYSTEM, or a structure containing SAFETY SYSTEM components, needed for the current operating mode. This condition significantly reduces the margin to a loss or potential loss of a fission product barrier, and therefore represents an actual or potential substantial degradation of the level of safety of the plant.

The first condition addresses damage to a SAFETY SYSTEM train that is in service/operation since indications for it will be readily available. The indications of degraded performance should be significant enough to cause concern regarding the operability or reliability of the SAFETY SYSTEM train.

The second condition addresses damage to a SAFETY SYSTEM component that is not in service/operation or readily apparent through indications alone, or to a structure containing SAFETY SYSTEM components. Operators will make this determination based on the totality of available event and damage report information. This is intended to be a brief assessment not requiring lengthy analysis or quantification of the damage.

Escalation of the emergency classification level would be via IC FS1 or RS1.

BNP Basis Reference(s):

1. 1(2)APP-UA-28 6-4 Seismic Event
2. 0AOP-13.0 Operations During Hurricane, Flood Conditions, Tornado or Earthquake
3. Updated FSAR section 3.4.2 Protection From Internal Flooding
4. Updated FSAR Section 2.3.1.2.7
5. BNP-E-9. 010 NFPA 805 Nuclear Safety Capability Assessment (NSCA)
6. 0PFP-PBAA Power Block Auxiliary Areas Prefire Plan
7. NEI 99-01 SA9

Category E – <u>Independent Spent Fuel Storage Installation (ISFSI)</u>

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

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

An 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.

The BNP ISFSI is contained wholly within the plant Protected Area. Therefore a security event related to the ISFSI would be applicable to EALs HU1.1, HA1.1 and HS1.1..

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

Category: E - ISFSI

Sub-category: None

Initiating Condition: Damage to a loaded cask CONFINEMENT BOUNDARY

EAL:

EU1.1 Notification of Unusual Event

Damage to a loaded canister confinement boundary as indicated by an on-contact radiation reading on the surface of a loaded spent fuel cask > any of the following:

- 1,400 mrem/hr on the HSM-H front surface
- 10 mrem/hr on the HSM-H door centerline
- 20 mrem/hr on the end shield wall exterior

Mode Applicability:

All

Definition(s):

CONFINEMENT BOUNDARY-. The barrier(s) between spent fuel and the environment once the spent fuel is processed for dry storage. As related to the BNP ISFSI, Confinement Boundary is defined as the Dry Shielded Canister (DSC).

Basis:

The BNP ISFSI utilizes the NUHOMS® Type 2 -61BTH dry spent fuel storage (ref. 1, 2).

The NUHOMS® Type 2 61BTH spent fuel storage system is a modular canister based spent fuel storage and transfer system and consists of the following components:

- A 61BTH Dry Shielded Canister (DSC) provides confinement, an inert environment, structural support, and criticality control for 61 BWR fuel assemblies.
- A horizontal storage module (HSM-H) is provided for environmental protection, shielding, and heat rejection during storage.
- An OS197FC-B transfer cask that supports onsite transfer of the 61BTH DSC.

The NUHOMS® System confinement vessel is the DSC. The DSC is welded and designed to provide confinement of all radionuclides under normal, off-normal, and accident conditions.

Confinement boundary is defined as the barrier(s) between areas containing radioactive substances and the environment. Therefore, damage to a confinement boundary must be a confirmed physical breach between the spent fuel and the environment for the Dry Shield Canister (DSC).

The values shown represent 2 times the limits specified in the ISFSI Certificate of Compliance (COC) Technical Specification for radiation external to a loaded (NUHOMS® Type 2 -61BTH) MPC (HSM-H) overpack (ref. 1, 2). The survey method(s) used to assess this EAL threshold shall be consistent with those used to ensure compliance with the COC Technical Specification limits (ref. 2).

This IC addresses an event that results in damage to the CONFINEMENT BOUNDARY of a storage cask containing spent fuel. It applies to irradiated fuel that is licensed for dry storage beginning at the point that the loaded storage cask is sealed. The issues of concern are the creation of a potential or actual release path to the environment, degradation of one or more fuel assemblies due to environmental factors, and configuration changes which could cause challenges in removing the cask or fuel from storage.

The existence of "damage" is determined by radiological survey. The technical specification multiple of "2 times", which is also used in Recognition Category R IC RU1, is used here to distinguish between non-emergency and emergency conditions. The emphasis for this classification is the degradation in the level of safety of the spent fuel cask and not the magnitude of the associated dose or dose rate. It is recognized that in the case of extreme damage to a loaded cask, the fact that the "on-contact" dose rate limit is exceeded may be determined based on measurement of a dose rate at some distance from the cask.

Security-related events for ISFSIs are covered under ICs HU1 and HA1.

BNP Basis Reference(s):

1. 0PLP-36 BNP 10CFR50.72.212 Report
2. Transnuclear, Inc. Standardized NUHOMS Horizontal Modular Storage System Certificate of Compliance No. 1004, Ammendment 10 Enclosure 1
3. NGGM-PM-0028 Transnuclear NUHOMS Dry Fuel Storage Program Manual
4. NEI 99-01 E-HU1

Category F – Fission Product Barrier Degradation

EAL Group: Hot Conditions (RCS temperature > 212ºF); 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): The Fuel Clad Barrier consists of the cladding material that contains the fuel pellets.
B. Reactor Coolant System (RCS): The RCS Barrier is the reactor coolant system pressure boundary and includes the RPV and all reactor coolant system piping up to and including the isolation valves.
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 PC barrier. Primary Containment Barrier thresholds are used as criteria for escalation of the ECL from Alert to a Site Area Emergency or a General Emergency.

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:

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

The logic used for emergency classification based on fission product barrier monitoring should reflect the following considerations:

- The Fuel Clad Barrier and the RCS Barrier are weighted more heavily than the Primary Containment Barrier.

- Unusual Event ICs associated with RCS and Fuel Clad Barriers are addressed under System Malfunction ICs.

- For accident conditions involving a radiological release, evaluation of the fission product barrier thresholds will need to be performed in conjunction with dose assessments to ensure correct and timely escalation of the emergency classification. For example, an evaluation of the fission product barrier thresholds may result in a Site Area Emergency classification while a dose assessment may indicate that an EAL for General Emergency IC RG1 has been exceeded.

- The fission product barrier thresholds specified within a scheme reflect plant-specific BNP design and operating characteristics.

- As used in this category, the term RCS leakage encompasses not just those types defined in Technical Specifications but also includes the loss of RCS mass to any location– inside the primary containment, an interfacing system, or outside of the primary containment. The release of liquid or steam mass from the RCS due to the as-designed/expected operation of a relief valve is not considered to be RCS leakage.

- At the Site Area Emergency level, EAL users should maintain cognizance of how far present conditions are from meeting a threshold that would require a General Emergency declaration. For example, if the Fuel Clad and RCS fission product barriers were both lost, then there should be frequent assessments of containment radioactive inventory and integrity. Alternatively, if both the Fuel Clad and RCS fission product barriers were potentially lost, the SEC would have more assurance that there was no immediate need to escalate to a General Emergency.

Category: Fission Product Barrier Degradation

Subcategory: N/A

Initiating Condition: **Any** loss or **any** potential loss of either Fuel Clad or RCS barrier

EAL:

FA1.1	**Alert**
Any loss or **any** potential loss of either Fuel Clad or RCS (Table F-1)	

Mode Applicability:

1 - Power Operations, 2 - Startup, 3 - Hot Shutdown

Definition(s):

None

Basis:

Fuel Clad, RCS and Primary 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 FS1.1

BNP Basis Reference(s):

1. NEI 99-01 FA1

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 barriers (Table F-1)	

Mode Applicability:

1 - Power Operations, 2 - Startup, 3 - Hot Shutdown

Definition(s):

None

Basis:

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 Site Emergency Coordinator would have greater assurance that escalation to a General Emergency is less imminent.

BNP Basis Reference(s):

1. NEI 99-01 FS1

Category: Fission Product Barrier Degradation

Subcategory: N/A

Initiating Condition: Loss of **any** two barriers and loss or potential loss of third barrier

EAL:

FG1.1	**General Emergency**
Loss of **any** two barriers	
AND	
Loss or potential loss of third barrier (Table F-1)	

Mode Applicability:

1 - Power Operations, 2 - Startup, 3 - Hot Shutdown

Definition(s):

None

Basis:

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

BNP Basis Reference(s):

1. NEI 99-01 FG1

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. RCS Leak Rate

B. Primary Containment Conditions

C. Primary Containment Radiation/RCS Activity

D. Primary Containment Integrity or Bypass

E. SEC Jugement

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 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, and FA1.1 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,..., F.

Table F-1 Fission Product Barrier Threshold Matrix						
	Fuel Clad Barrier		Reactor Coolant System Barrier		Primary Containment Barrier	
Category	Loss	Potential Loss	Loss	Potential Loss	Loss	Potential Loss
A RPV Water Level	1. Entry to SAMG-01 required	1. RPV level **cannot** be restored and maintained > TAF or **cannot** be determined	1. RPV level **cannot** be restored and maintained > TAF or **cannot** be determined	None	None	1. Entry to SAMG-01 required
B RCS Leak Rate	None	None	1. UNISOLABLE break in **any** of the following: • Main steam • HPCI steam Line • RCIC steam Line • RWCU • Feedwater 2. Emergency Depressurization is required	1. UNISOLABLE primary system leakage that results in exceeding **EITHER** of the following: • One or more Secondary Containment area radiation Maximum Normal Operating Limits (0EOP-03-SCCP Table 3) • One or more Secondary Containment area temperature Maximum Normal Operating Limits (0EOP-03-SCCP Table 1)	1. UNISOLABLE primary system leakage that results in exceeding **EITHER** of the following: • One or more Secondary Containment area radiation Maximum Safe Operating Limits (0EOP-03-SCCP Table 3) • One or more Secondary Containment area temperature Maximum Safe Operating Limits (0EOP-03-SCCP Table 1)	None
C PC Conditions	None	None	1. Primary Containment pressure > 1.7 psig due to RCS leakage	None	1. UNPLANNED rapid drop in Primary Containment pressure following Primary Containment pressure rise 2. Primary Containment pressure response **not** consistent with LOCA conditions	1. Primary Containment pressure > 62 psig 2. Deflagration concentrations exist inside PC ($H_2 \geq 6\%$ **AND** $O_2 \geq 5\%$) 3. Heat Capacity Temperature Limit (HCTL) exceeded
D PC Rad / RCS Activity	1. Drywell radiation > 2,000 R/hr 2. Primary coolant activity > 300 μCi/gm I-131 dose equivalent	None	1. Drywell radiation > 27 R/hr with reactor shutdown	None	None	1. Drywell radiation > 20,000 R/hr
E PC Integrity or Bypass	None	None	None	None	1. UNISOLABLE direct downstream pathway to the environment exists after Primary Containment isolation signal 2. Intentional Primary Containment venting per EOPs	None
F ED Judgment	1. **Any** condition in the opinion of the SEC that indicates loss of the fuel clad barrier	1. **Any** condition in the opinion of the SEC that indicates potential loss of the fuel clad barrier	1. **Any** condition in the opinion of the SEC that indicates loss of the RCS barrier	1. **Any** condition in the opinion of the SEC that indicates potential loss of the RCS barrier	1. **Any** condition in the opinion of the SEC that indicates loss of the Primary Containment barrier	1. **Any** condition in the opinion of the SEC that indicates potential loss of the Primary Containment barrier

Barrier: Fuel Clad

Category: A. RPV Level

Degradation Threat: Loss

Threshold:

1. Entry to SAMG-01 required

Definition(s):

N/A

Basis:

1(2)EOP-01-RVCP, 0EOP-01-ATWS and 0EOP-01-RXFP specify the requirement for entry to SAMG-01 when core cooling is severely challenged. 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. SAMG-01 is required when (ref. 1):

- Reactor water level cannot be restored and maintained above -57.5 inches (Jet Pump Suction) with at least one core spray pump injecting into the reactor vessel
- Reactor vessel water level cannot be restored and maintained above LL-4 (MSCRWL)
- The reactor vessel flooding conditions cannot be restored and maintained (5 SRVs open and reactor vessel pressure more than 50 psig above suppression chamber pressure)
- When at least 1 SRV cannot be opened and reactor vessel pressure cannot be restored and maintained above the minimum alternate reactor vessel flooding pressure (Table 1 values that are dependent on number of open SRVs)

The above EOP conditions represent a challenge to core cooling and are the minimum values to assure core cooling without further degradation of the clad.

This threshold is also a Potential Loss of the Containment barrier (PC P-Loss A.1). Since entry to SAMG-01 occurs after core uncovery has occurred a Loss of the RCS barrier exists (RCS Loss A.1). Entry to SAMG-01, therefore, represents a Loss of two barriers and a Potential Loss of a third, which requires a General Emergency classification.

The Loss threshold represents the EOP requirement for entry to SAMG-01. This is identified in the BWROG EPGs/SAGs when the phrase, "enter all Severe Accident Guidelines" appears. Since a site-specific RPV water level is not specified here, the Loss threshold phrase, "Entry to SAMG-01 required," also accommodates the EOP need to enter SAMG-01 when RPV water level cannot be determined and core damage due to inadequate core cooling is believed to be occurring.

BNP Basis Reference(s):

1. 0SAMG-06.0 SAMG Primary Containment Flooding Basis Document
2. NEI 99-01 RPV Water Level Fuel Clad Loss 2.A

Barrier: Fuel Clad

Category: A. RPV Level

Degradation Threat: Potential Loss

Threshold:

1. RPV level **cannot** be restored and maintained > TAF or **cannot** be determined

Definition(s):

N/A

Basis:

An RPV level instrument reading of -7.5 in. indicates RPV level is at the top of active fuel (TAF) (ref. 1). When RPV level is at or above the TAF, the core is completely submerged. Core submergence is the most desirable means of core cooling. When RPV level is below TAF, the uncovered portion of the core must be cooled by less reliable means (i.e., steam cooling or spray cooling). If core uncovery is threatened, the EOPs specify alternate, more extreme, RPV level control measures in order to restore and maintain adequate core cooling. Since core uncovery begins if RPV level drops to TAF, the level is indicative of a challenge to core cooling and the Fuel Clad barrier.

When RPV level cannot be determined, EOPs require entry to 0EOP-01-RXFP, Reactor Flooding. RPV water level indication provides the primary means of knowing if adequate core cooling is being maintained (ref. 2, 3). 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 0EOP-01-RXFP 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 Alternate Reactor Vessel Flooding Pressure (in scram-failure events) (ref. 4). 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 1(2)EOP-01-ATWS, ATWS, may require intentionally lowering RPV water level to TAF and control level between the LL-4, the Minimum Steam Cooling RPV Water Level (MSCRWL) and TAF (ref. 3). Under these conditions, a high-power ATWS event exists and requires at least an Alert classification in accordance with the System Malfunction - RPS Failure EALs, however under these conditions a potential loss of the fuel clad does not exist.

This water level corresponds to the top of the active fuel and is used in the EOPs to indicate a challenge to core cooling.

The RPV water level threshold is the same as RCS barrier Loss threshold 1.A. Thus, this threshold indicates a Potential Loss of the Fuel Clad barrier and a Loss of the 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, 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) or when procedural guidance or a lack of low pressure RPV injection sources preclude Emergency RPV depressurization. EOPs allow the operator a wide choice of RPV injection sources to consider when restoring RPV water level to within prescribed limits. EOPs also specify depressurization of the RPV in order to facilitate RPV water level control with low-pressure injection sources. In some events, elevated RPV pressure may prevent restoration of RPV water level until pressure drops below the shutoff heads of available injection sources. Therefore, this Fuel Clad barrier Potential Loss is met only after either: 1) the RPV has been depressurized, or required emergency RPV depressurization has been attempted, giving the operator an opportunity to assess the capability of low-pressure injection sources to restore RPV water level or 2) no low pressure RPV injection systems are available, precluding RPV depressurization in an attempt to minimize loss of RPV inventory.

The term "cannot be restored and maintained above" means the value of RPV water level is not able to be brought above the specified limit (top of active fuel). The determination requires an evaluation of system performance and availability in relation to the RPV water level value and trend. A threshold prescribing declaration when a threshold value *cannot* be restored and maintained above a specified limit does not require immediate action simply because the current value is below the top of active fuel, but does not permit extended operation below the limit; the threshold must be considered reached as soon as it is apparent that the top of active fuel cannot be attained.

In high-power ATWS/failure to scram events, EOPs may direct the operator to deliberately lower RPV water level to the top of active fuel in order to reduce reactor power. RPV water level is then controlled between the top of active fuel and the Minimum Steam Cooling RPV Water Level (MSCRWL). Although such action is a challenge to core cooling and the Fuel Clad barrier, the immediate need to reduce reactor power is the higher priority. For such events, ICs SA6 or SS6 will dictate the need for emergency classification.

Since the loss of ability to determine if adequate core cooling is being provided presents a significant challenge to the fuel clad barrier, a potential loss of the fuel clad barrier is specified.

BNP Basis Reference(s):

1. 0EOP-01-NL EOP/SAMG Numerical Limits and Values, Attachment 1
2. 1(2)EOP-01-RVCP Reactor Vessel Control
3. 1(2)EOP-01-ATWS, ATWS
4. 0EOP-01-RXFP, Reactor Flooding
5. NEI 99-01 RPV Water Level Fuel Clad Potential Loss 2.A

Barrier: Fuel Clad

Category: B. RCS Leak Rate

Degradation Threat: Loss

Threshold:

None

Barrier: Fuel Clad

Category: B. RCS Leak Rate

Degradation Threat: Potential Loss

Threshold:

None

Barrier: Fuel Clad

Category: C. PC Conditions

Degradation Threat: Loss

Threshold:

None

Barrier: Fuel Clad

Category: C. PC Conditions

Degradation Threat: Potential Loss

Threshold:

None

Barrier: Fuel Clad

Category: D. PC Radiation / RCS Activity

Degradation Threat: Loss

Threshold:

1. Drywell radiation > 2,000 R/hr

Definition(s):

None

Basis:

The Drywell High-Range Radiation Monitor (1(2)D22-RI-4195, 1(2)D22-RI-4196, 1(2)D22-RI-4197, 1(2)D22-RI-4198) reading of 2,000 R/hr indicates the release of reactor coolant into the drywell with elevated activity indicative of fuel damage.

Based on 2% clad damage, a containment radiation level of 2000 R/hr is derived as follows:

Per 0PEP-03.6.3 Table 3, 100% Cladding Damage column 'No Spray' for 1 hour after shutdown is 100,000 R/hr. Per Step 7.2.2.1, 0.02 x 100,000 R/hr = 2000 R/hr containment radiation (ref. 1).

The radiation monitor reading corresponds to an instantaneous release of all reactor coolant mass into the primary containment, assuming that reactor coolant activity equals 300 μCi/gm dose equivalent I-131. Reactor coolant activity above this level is greater than that expected for iodine spikes and corresponds to an approximate range of 2% to 5% fuel clad damage. Since this condition indicates that a significant amount of fuel clad damage has occurred, it represents a loss of the Fuel Clad Barrier.

The radiation monitor reading in this threshold is higher than that specified for RCS Barrier Loss threshold D.1 since it indicates a loss of both the Fuel Clad Barrier and the RCS Barrier. Note that a combination of the two monitor readings appropriately escalates the emergency classification level to a Site Area Emergency.

There is no Potential Loss threshold associated with Primary Containment Radiation.

BNP Basis Reference(s):

1. 0PEP-03.6.3 Estimate of the Extent of Core Damage Under Accident Conditions
2. NEI 99-01 Primary Containment Radiation Fuel Clad Loss 4.A

Barrier: Fuel Clad

Category: D. PC Radiation / RCS Activity

Degradation Threat: Loss

Threshold:

2. Primary coolant activity > 300 μCi/gm I-131 dose equivalent

Basis:

This threshold indicates that RCS radioactivity concentration is greater than 300 μCi/gm dose equivalent I-131. Reactor coolant activity above this level is greater than that expected for iodine spikes and corresponds to an approximate range of 2% to 5% fuel clad damage. Since this condition indicates that a significant amount of fuel clad damage has occurred, it represents a loss of the Fuel Clad Barrier.

There is no Potential Loss threshold associated with RCS Activity.

There is no Potential Loss threshold associated with Primary Containment Radiation.

BNP Basis Reference(s):

1. NEI 99-01 RCS Activity Fuel Clad Loss 1.A

Barrier: Fuel Clad

Category: D. PC Radiation / RCS Activity

Degradation Threat: Potential Loss

Threshold:

None

Barrier: Fuel Clad

Category: E. PC Integrity or Bypass

Degradation Threat: Loss

Threshold:

None

Barrier: Fuel Clad

Category: E. PC Integrity or Bypass

Degradation Threat: Potential Loss

Threshold:

None

Barrier: Fuel Clad

Category: F. SEC Judgment

Degradation Threat: Loss

Threshold:

1. **Any** condition in the opinion of the Site Emergency Coordinator that indicates loss of the Fuel Clad barrier

Definition(s):

None

Basis:

The Site Emergency Coordinator 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 Site Emergency Coordinator should be mindful of the Loss of AC power (Station Blackout) and ATWS EALs to assure timely emergency classification declarations.

This threshold addresses any other factors that are to be used by the Site Emergency Coordinator in determining whether the Fuel Clad barrier is lost

BNP Basis Reference(s):

1. NEI 99-01 Emergency Director Judgment Fuel Clad Loss 6.A

Barrier: Fuel Clad

Category: F. SEC Judgment

Degradation Threat: Potential Loss

Threshold:

1. **Any** condition in the opinion of the Site Emergency Coordinator that indicates potential loss of the Fuel Clad barrier

Basis:

The Emergency Coordinator 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 Site Emergency Coordinator should be mindful of the Loss of AC power (Station Blackout) and ATWS EALs to assure timely emergency classification declarations.

This threshold addresses any other factors that are to be used by the Site Emergency Coordinator in determining whether the Fuel Clad barrier is potentially lost. The Site Emergency Coordinator should also consider whether or not to declare the barrier potentially lost in the event that barrier status cannot be monitored.

BNP Basis Reference(s):

1. NEI 99-01 Emergency Director Judgment Potential Fuel Clad Loss 6.A

Barrier: Reactor Coolant System

Category: A. RPV Water Level

Degradation Threat: Loss

Threshold:

1. RPV level **cannot** be restored and maintained > TAF or **cannot** be determined

Definition(s):

None

Basis:

An RPV level instrument reading of -7.5 in. indicates RPV level is at the top of active fuel (TAF) (ref. 1). TAF is significantly lower than the normal operating RPV level control band. To reach this level, RPV inventory loss would have previously required isolation of the RCS and Primary Containment barriers, and initiation of all ECCS. If RPV level cannot be maintained above TAF, ECCS and other sources of RPV injection have been ineffective or incapable of reversing the decreasing level trend. The cause of the loss of RPV inventory is therefore assumed to be a LOCA. By definition, a LOCA event is a Loss of the RCS barrier.

When RPV level cannot be determined, EOPs require entry to 0EOP-01-RXFP, Reactor Flooding (ref. 2). The instructions in 0EOP-01-RXFP specify emergency depressurization of the RPV, which is defined to be a Loss of the RCS barrier (RCS Loss C.4).

The conditions of this threshold are also a Potential Loss of the Fuel Clad barrier (FC P-Loss A.1). A Loss of the RCS barrier and Potential Loss of the Fuel Clad barrier requires a Site Area Emergency classification.

Note that 1(2)EOP-01-ATWS, ATWS, may require intentionally lowering RPV water level to TAF and control level between LL-4, the Minimum Steam Cooling RPV Water Level (MSCRWL), and TAF (ref. 3). Under these conditions, a high-power ATWS event exists and requires at least an Alert classification in accordance with the System Malfunction - RPS Failure EALs, however under these conditions a loss of the RCS does not exist.

This water level corresponds to the top of active fuel and is used in the EOPs to indicate challenge to core cooling.

The RPV water level threshold is the same as Fuel Clad barrier Potential Loss threshold 1.A. Thus, this threshold indicates a Loss of the RCS barrier and Potential Loss of the Fuel Clad barrier and 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, 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) or when procedural guidance or a lack of low pressure RPV injection sources preclude Emergency RPV depressurization. EOPs allow the operator a wide choice of RPV injection sources to consider when restoring RPV water level to within prescribed limits. EOPs also specify depressurization of the RPV in order to facilitate RPV water level control with low-pressure injection sources. In some events, elevated RPV pressure may prevent restoration of RPV water level until pressure drops below the shutoff heads of available injection sources. Therefore, this RCS barrier Loss is met only after either: 1) the RPV has been depressurized, or required emergency RPV depressurization has been attempted, giving the operator an opportunity to assess the capability of low-pressure injection sources to restore RPV water level or 2) no low pressure RPV injection systems are available, precluding RPV depressurization in an attempt to minimize loss of RPV inventory.

BNP Basis Reference(s):

1. 0EOP-01-NL EOP/SAMG Numerical Limits and Values, Attachment 1
2. 1(2)EOP-01-RVCP Reactor Vessel Control
3. 1(2)EOP-01ATWS, ATWS
4. 0EOP-01-RXFP, Reactor Flooding
5. NEI 99-01 RPV Water Level RCS Loss 2.A

Barrier: Reactor Coolant System

Category: A. RPV Water Level

Degradation Threat: Potential Loss

Threshold:

None

Barrier: Reactor Coolant System

Category: B. RCS Leak Rate

Degradation Threat: Loss

Threshold:

1. UNISOLABLE break outside Primary Containment in any of the following:
   - Main steam line
   - HPCI steam line
   - RCIC steam line
   - RWCU
   - Feedwater

Definition(s):

UNISOLABLE - An open or breached system line that cannot be isolated, remotely or locally.

Basis:

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. In the case of a failure of both isolation valves to close but in which no downstream flowpath exists, emergency declaration under this threshold would not be required. 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 Primary Containment (see PC Loss E.1) barriers and justifies declaration of a Site Area Emergency (i.e., Loss or Potential Loss of any two barriers).

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.

Large high-energy lines that rupture outside primary containment can discharge significant amounts of inventory and jeopardize the pressure-retaining capability of the RCS until they are isolated. If it is determined that the ruptured line cannot be promptly isolated from the Control Room, the RCS barrier Loss threshold is met.

BNP Basis Reference(s):

1. 1(2)OP-01 Nuclear Boiler System
2. 1(2)OP-25 Main Steam System Operating Procedure
3. 1(2)OP-19 High Pressure Coolant Injection System Operating Procedure
4. 1(2)OP-16 Reactor Core Isolation Cooling System Operating Procedure
5. 1(2)OP-14 Reactor Water Cleanup System Operating Procedure
6. 1(2)OP-32 Condensate and Feedwater System Operating Procedure
7. NEI 99-01 RCS Leak Rate RCS Loss 3.A

Barrier: Reactor Coolant System

Category: B. RCS Leak Rate

Degradation Threat: Loss

Threshold:

2. Emergency Depressurization is required

Definition(s):

N/A

Basis:

Plant symptoms requiring Emergency Depressurization per the EOPs are indicative of a loss of the RCS barrier. If Emergency depressurization is required, the plant operators are directed to open safety relief valves (SRVs) and keep them open regardless of any subsequent radiological release rate (ref. 1 – 6). 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.

Emergency RPV Depressurization in accordance with the EOPs is indicative of a loss of the RCS barrier. If Emergency RPV Depressurization is performed, 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 barrier exists due to the diminished effectiveness of the RCS to retain fission products within its boundary.

BNP Basis Reference(s):

1. 0EOP-01-UG User's Guide
2. 1(2)EOP-01-RVCP Reactor Vessel Control
3. 1(2)EOP-01-ATWS ATWS
4. 0EOP-02-PCCP Primary Containment Control
5. 0EOP-03-SCCP Secondary Containment Control
6. 0EOP-04-RRCP Radioactivity Release Control
7. 0EOP-01-RXFP Reactor Flooding
8. NEI 99-01 RCS Leak Rate RCS Loss 3.B

Barrier: Reactor Coolant System

Category: B. RCS Leak Rate

Degradation Threat: Potential Loss

Threshold:

1. UNISOLABLE primary system leakage that results in exceeding **EITHER** of the following:

   - One or more Secondary Containment area radiation Maximum Normal Operating Limits (0EOP-03-SCCP Table SC-3)

   - One or more Secondary Containment area temperature Maximum Normal Operating Limits (0EOP-03-SCCP Table SC-1)

Definition(s):

UNISOLABLE - An open or breached system line that cannot be isolated, remotely or locally.

Basis:

The presence of elevated general area temperatures or radiation levels in the secondary containment may be indicative of unisolable primary system leakage outside the primary containment. The Maximum Normal Operating Limit values define this RCS threshold because they are the maximum normal operating values and signify the onset of abnormal system operation. When parameters reach this 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. The locations into which the primary system discharge is of concern correspond to the areas addressed in 0EOP-03-SCCP, Secondary Containment Control Tables (ref. 1).

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.

Potential loss of RCS based on primary system leakage outside the primary containment is determined from EOP temperature or radiation Max Normal Operating values in areas such as main steam line tunnel, RCIC, HPCI, etc., which indicate a direct path from the RCS to areas outside primary containment.

A Max Normal Operating value is the highest value of the identified parameter expected to occur during normal plant operating conditions with all directly associated support and control systems functioning properly.

The indicators reaching the threshold barriers and confirmed to be caused by RCS leakage from a primary system warrant an Alert classification. A primary system is defined to be the pipes, valves, and other equipment which connect directly to the RPV such that a reduction in RPV pressure will effect a decrease in the steam or water being discharged through an unisolated break in the system.

An UNISOLABLE leak which is indicated by Max Normal Operating values escalates to a Site Area Emergency when combined with Containment Barrier Loss threshold 3.A (after a containment isolation) and a General Emergency when the Fuel Clad Barrier criteria is also exceeded.

BNP Basis Reference(s):

1. 0EOP-03-SCCP, Secondary Containment Control
2. NEI 99-01 RCS Leak Rate RCS Potential Loss 3.A

Barrier: Reactor Coolant System

Category: C. PC Conditions

Degradation Threat: Loss

Threshold:

1. Primary Containment pressure > 1.7 psig due to RCS leakage

Definition(s):

None

Basis:

The drywell high pressure scram setpoint is an entry condition to 1(2)EOP-01-RVCP Reactor Vessel Control, and 0EOP-02-PCCP, Primary Containment Control (ref. 1, 2, 3). Normal primary containment pressure control functions (e.g., operation of drywell coolers, vent through SBGT, etc.) are specified in 0EOP-02-PCCP in advance of less desirable but more effective functions (e.g., operation of drywell or suppression pool sprays, etc.).

In the BNP 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. 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. PC pressure greater than 1.7 psig with corollary indications (e.g., 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 1.7 psig should not be considered an RCS barrier Loss.

1.7 psig is the drywell high pressure setpoint which indicates a LOCA by automatically initiating the ECCS or equivalent makeup system.

There is no Potential Loss threshold associated with Primary Containment Pressure.

BNP Basis Reference(s):

1. 0EOP-01-NL EOP/SAMG Numerical Limits and Values, Attachment 3
2. 1(2)EOP-01-RVCP Reactor Vessel Control
3. 0EOP-02-PCCP Primary Containment Control
4. BNP Updated FSAR Chapter 6 Emergency Core Cooling Systems
5. NEI 99-01 Primary Containment Pressure RCS Loss 1.A

Barrier: Reactor Coolant System

Category: C. PC Conditions

Degradation Threat: Potential Loss

Threshold:

None

Barrier: Reactor Coolant System

Category: D. PC Radiation / RCS Activity

Degradation Threat: Loss

Threshold:

1. Drywell radiation > 27 R/hr with reactor shutdown

Definition(s):

N/A

Basis:

The Drywell High-Range Radiation Monitor (1(2)D22-RI-4195, 1(2)D22-RI-4196, 1(2)D22-RI-4197, 1(2)D22-RI-4198) reading of 27 R/hr is based on coolant activity at the Technical Specification limit of 4 μCi/gm I-131).

The containment radiation level of 27 R/hr is derived as follows:

0PEP-03.6.3 Table 3 100% Cladding Damage column 'No Spray' for 1 hour after shutdown is 100,000 R/hr. Assuming that 300 μCi/gm I-131 is approximately 2% cladding failure, a coolant activity of 4 μCi/gm I-131 is ratioed to approximately 0.027% (0.00027) clad failure. Per Step 7.2.2.1, 0.00027 x 100,000 R/hr = 27 R/hr containment radiation corresponding to Technical Specification coolant activity. (ref. 1)

The threshold value is only applicable with the reactor shutdown as the high range detectors normally read as high as 100 R/hr during power operations due to shine from the reactor.

The radiation monitor reading corresponds to an instantaneous release of all reactor coolant mass into the primary containment, assuming that reactor coolant activity equals Technical Specification allowable limits. This value is lower than that specified for Fuel Clad Barrier Loss threshold D.1 since it indicates a loss of the RCS Barrier only.

There is no Potential Loss threshold associated with Primary Containment Radiation.

BNP Basis Reference(s):

1. 0PEP-03.6.3 Estimate of the Extent of Core Damage Under Accident Conditions
2. NEI 99-01 Primary Containment Radiation RCS Loss 4.A

Barrier: Reactor Coolant System

Category: D. PC Radiation / RCS Activity

Degradation Threat: Potential Loss

Threshold:

None

Barrier: Reactor Coolant System

Category: E. PC Integrity or Bypass

Degradation Threat: Loss

Threshold:

None

Barrier: Reactor Coolant System

Category: E. PC Integrity or Bypass

Degradation Threat: Potential Loss

Threshold:

None

Barrier: Reactor Coolant System

Category: F. SEC Judgment

Degradation Threat: Loss

Threshold:

1. **Any** condition in the opinion of the Site Emergency Coordinator that indicates loss of the RCS barrier

Definition(s):

None

Basis:

The Site Emergency Coordinator 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 Site Emergency Coordinator should be mindful of the Loss of AC power (Station Blackout) and ATWS EALs to assure timely emergency classification declarations.

This threshold addresses any other factors that may be used by the Site Emergency Coordinator in determining whether the RCS Barrier is lost.

BNP Basis Reference(s):

1. NEI 99-01 Emergency Director Judgment RCS Loss 6.A

Barrier: Reactor Coolant System

Category: F. SEC Judgment

Degradation Threat: Potential Loss

Threshold:

1. **Any** condition in the opinion of the Site Emergency Coordinator that indicates potential loss of the RCS barrier

Definition(s):

None

Basis:

The Site Emergency Coordinator 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 Site Emergency Coordinator should be mindful of the Loss of AC power (Station Blackout) and ATWS EALs to assure timely emergency classification declarations.

This threshold addresses any other factors that may be used by the Site Emergency Coordinator in determining whether the RCS Barrier is potentially lost. The Emergency Director should also consider whether or not to declare the barrier potentially lost in the event that barrier status cannot be monitored.

BNP Basis Reference(s):

1. NEI 99-01 Emergency Director Judgment RCS Potential Loss 6.A

Fission Product Barrier Loss/Potential Loss Matrix and Bases

Barrier: Primary Containment

Category: A. RPV Water Level

Degradation Threat: Loss

Threshold:

None

Barrier: Primary Containment

Category: A. RPV Water Level

Degradation Threat: Potential Loss

Threshold:

1. Entry to SAMG-01 required

Definition(s):

None

Basis:

1(2)EOP-01-RVCP, 1(2)EOP-01-ATWS and 0EOP-01-RXFP specify the requirement for entry to SAMG-01 when core cooling is severely challenged. 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. Entry to SAMG-01 is required when (ref. 1):

- Reactor water level cannot be restored and maintained above -57.5 inches (Jet Pump Suction) with at least one core spray pump injecting into the reactor vessel
- Reactor vessel water level cannot be restored and maintained above LL-4 (MSCRWL)
- The reactor vessel flooding conditions cannot be restored and maintained (5 SRVs open and reactor vessel pressure more than 50 psig above suppression chamber pressure)
- When at least 1 SRV cannot be opened and reactor vessel pressure cannot be restored and maintained above the minimum alternate reactor vessel flooding pressure (0EOP-01-RXFP Table 1 values that are dependent on number of open SRVs)

The above EOP conditions represent a challenge to core cooling and are the minimum values to assure core cooling without further degradation of the clad.

This threshold is also a Loss of the Fuel Clad barrier (FC Loss A.1). Since entry to SAMG-01 occurs after core uncovery has occurred a Loss of the RCS barrier exists (RCS Loss A.1). Entry to SAMG-01, therefore, represents a Loss of two barriers and a Potential Loss of a third, which requires a General Emergency classification.

The Potential Loss threshold is identical to the Fuel Clad Loss RPV Water Level threshold A.1. The Potential Loss requirement for entry to SAMG-01 indicates adequate core cooling cannot be restored and maintained and that core damage is possible. BWR EPGs/SAGs specify the conditions that require entry to SAMG-01. When entry to SAMG-01 is required, the EPGs are exited and SAGs are entered. Entry into SAGs is a logical escalation in response to the inability to restore and maintain adequate core cooling.

PRA studies indicate that the condition of this Potential Loss threshold could be a core melt sequence which, if not corrected, could lead to RPV failure and increased potential for primary containment failure. In conjunction with the RPV water level Loss thresholds in the Fuel Clad and RCS barrier columns, this threshold results in the declaration of a General Emergency.

BNP Basis Reference(s):

1. 0SAMG-06.0 SAMG Primary Containment Flooding Basis Document
2. NEI 99-01 RPV Water Level PC Potential Loss 2.A

Barrier: Primary Containment

Category: B. RCS Leak Rate

Degradation Threat: Loss

Threshold:

1. UNISOLABLE primary system leakage that results in exceeding one or more Secondary Containment area temperature Maximum Safe Operating Limits (0EOP-03-SCCP Table SC-1)

Definition(s):

UNISOLABLE - An open or breached system line that cannot be isolated, remotely or locally.

Basis:

The presence of elevated general area temperatures in the secondary containment may be indicative of unisolable primary system leakage outside the primary containment. The Maximum Safe Operating Limit area temperature 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 0EOP-03-SCCP, Secondary Containment Control Table 1(ref. 1) (see below).

In general, multiple indications should be used to determine if a primary system is discharging outside Primary Containment. A high area temperature condition in conjunction with other indications (e.g. room flooding, 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.

The Max Safe Operating Temperatures are each the highest value of these parameters at which neither: (1) equipment necessary for the safe shutdown of the plant will fail, nor (2) personnel access necessary for the safe shutdown of the plant will be precluded. EOPs utilize these temperatures to establish conditions under which RPV depressurization is required.

The temperatures should be confirmed to be caused by RCS leakage from a primary system. A primary system is defined to be the pipes, valves, and other equipment which connect directly to the RPV such that a reduction in RPV pressure will effect a decrease in the steam or water being discharged through an unisolated break in the system.

In combination with RCS Potential Loss 2.A this threshold would result in a Site Area Emergency.

There is no Potential Loss threshold associated with Primary Containment Isolation Failure.

BNP Basis Reference(s):

1. 0EOP-03-SCCP Secondary Containment Control
2. NEI 99-01 RCS Leak Rate PC Loss 3.C

Barrier: Primary Containment

Category: B. RCS Leak Rate

Degradation Threat: Potential Loss

Threshold:

None

Barrier: Primary Containment

Category: C. PC Conditions

Degradation Threat: Loss

Threshold:

1. UNPLANNED rapid drop in Primary Containment pressure following Primary Containment pressure rise

Definition(s):

UNPLANNED - A parameter change or an event that is not 1) the result of an intended evolution or 2) an expected plant response to a transient. The cause of the parameter change or event may be known or unknown.

Basis:

Rapid UNPLANNED loss of primary containment pressure (i.e., not attributable to drywell spray or condensation effects) following an initial pressure increase indicates a loss of primary containment integrity.

This threshold relies on operator recognition of an unexpected response for the condition and therefore a specific value is not assigned. The unexpected (UNPLANNED) response is important because it is the indicator for a containment bypass condition.

BNP Basis Reference(s):

1. NEI 99-01 Primary Containment Conditions PC Loss 1.A

Barrier: Primary Containment

Category: C. PC Conditions

Degradation Threat: Loss

Threshold:

2. Primary Containment pressure response **not** consistent with LOCA conditions

Definition(s):

None

Basis:

The calculated pressure response of the containment is shown in Figure 6-11. Figure 6-11 shows that the maximum calculated drywell pressure is 48 psia (33 psig), which is well below the design allowable pressure of 62 psig (ref. 2). The primary containment pressure stabilizes at about 40 psia (25 psig), as shown on Figure 6-1.

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.

Primary containment pressure should increase as a result of mass and energy release into the primary containment from a LOCA. Thus, primary containment pressure not increasing under these conditions indicates a loss of primary containment integrity.

This threshold relies on operator recognition of an unexpected response for the condition and therefore a specific value is not assigned. The unexpected (UNPLANNED) response is important because it is the indicator for a containment bypass condition.

BNP Basis Reference(s):

1. BNP Updated FSAR Figure 6-11
2. BNP Updated FSAR section 6.2.1.1.1
3. NEI 99-01 Primary Containment Conditions PC Loss 1.B

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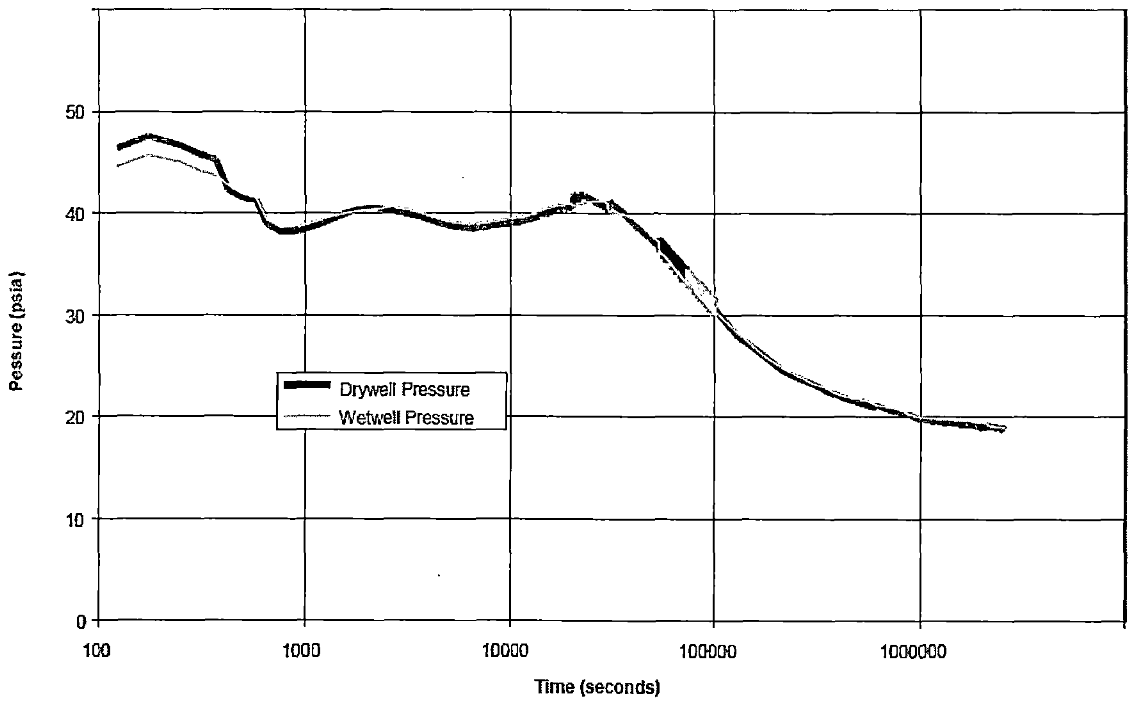
CONTAINMENT PRESSURE RESPONSE TO DESIGN BASIS ACCIDENT LOCA AT 120 PERCENT UPRATE



Barrier: Containment

Category: B. PC Conditions

Degradation Threat: Potential Loss

Threshold:

1. Primary Containment pressure > 62 psig

Definition(s):

None

Basis:

When the primary containment exceeds the maximum allowable value (62 psig) (ref. 1), primary containment venting may be required even if offsite radioactivity release rate limits will be exceeded (ref. 2). The drywell and suppression chamber maximum allowable value of 62 psig is based on the primary containment design pressure as identified in the BNP accident analysis (ref. 1, 3). If this threshold is exceeded, a challenge to the containment structure has occurred because assumptions used in the accident analysis are no longer valid and an unanalyzed condition exists. This constitutes a Potential Loss of the Containment barrier even if a containment breach has not occurred.

The threshold pressure is the primary containment internal design pressure. Structural acceptance testing demonstrates the capability of the primary containment to resist pressures greater than the internal design pressure. A pressure of this magnitude is greater than those expected to result from any design basis accident and, thus, represent a Potential Loss of the Containment barrier.

BNP Basis Reference(s):

1. BNP Updated FSAR section 6.2.1.1.1
2. 0EOP-02-PCCP Primary Containment Control
3. BNP Updated FSAR section 6.2.1.1
4. NEI 99-01 Primary Containment Conditions PC Potential Loss 1.A

Barrier: Containment

Category: B. PC Conditions

Degradation Threat: Potential Loss

Threshold:

2. Deflagration concentrations exist inside PC ($H_2 \geq 6\%$ **AND** $O_2 \geq 5\%$)

Definition(s):

None

Basis:

Deflagration (explosive) mixtures in the primary containment are assumed to be elevated concentrations of hydrogen and oxygen. BWR industry evaluation of hydrogen generation for development of EOPs/SAMGs indicates that any hydrogen concentration above minimum detectable is not to be expected within the short term. Post-LOCA hydrogen generation primarily caused by radiolysis is a slowly evolving, long-term condition. Hydrogen concentrations that rapidly develop are most likely caused by metal-water reaction. A metal-water reaction is indicative of an accident more severe than accidents considered in the plant design basis and would be indicative, therefore, of a potential threat to primary containment integrity. 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. 2) and readily recognizable because 6% hydrogen is well above the 0EOP-02-PCCP, Primary Containment Control, entry condition (ref. 2). The minimum global deflagration hydrogen/oxygen concentrations (6% and 5%, respectively) require intentional primary containment venting, which is defined to be a Loss of Containment (PC Loss C.4).

Monitors CAC-AT-4409 and 4410 monitor hydrogen and oxygen concentration in the drywell during normal operation. These monitors are isolated by accident isolation signals. However, monitors CAC-AT-4409 and 4410 will be realigned to the primary containment for post-accident monitoring via an operator actuated isolation signal override circuit when directed by the EOPs.

The oxygen and hydrogen concentrations from these two analyzers are recorded on two 4-channel recorders (CAC-AR-4409 and 4410) located on Panel XU-51. The indications are also displayed on the ERFIS. If concentrations exceed preset levels, recorder CAC-AR-4409 will annunciate the "Containment Atmosphere Division I 02 - H2 High" alarm in the Control Room and recorder CAC-AR-4410 will annunciate "Containment Atmosphere Division II 02 - H2 High" alarm. (ref. 3, 4).

If hydrogen concentration reaches or exceeds the lower flammability limit, as defined in plant EOPs, in an oxygen rich environment, a potentially explosive mixture exists. If the combustible mixture ignites inside the primary containment, loss of the Containment barrier could occur.

BNP Basis Reference(s):

1. BWROG EPG/SAG Revision 2, Sections PC/G
2. 0EOP-02-PCCP, Primary Containment Control
3. BNP Updated FSAR section 6.2.5.2.2
4. BNP System Description SD-04 Primary Containment
5. NEI 99-01 Primary Containment Conditions PC Potential Loss 1.B

Barrier: Containment

Category: B. PC Conditions

Degradation Threat: Potential Loss

Threshold:

3. Heat Capacity Temperature Limit (HCTL) exceeded

Definition(s):

None

Basis:

This threshold is met when the final step of section SP/T in 0EOP-02-PCCP, Primary Containment Control, is reached (ref. 1, 2).

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 maximum temperature capability of the suppression chamber and equipment within the suppression chamber which may be required to operate when the RPV is pressurized,
>
> OR
>
> Suppression chamber pressure above Primary Containment Pressure Limit A, while 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, suppression pool temperature 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.

BNP Basis Reference(s):

1. 0EOP-01-NL EOP/SAMG Numerical Limits and Values
2. 0EOP-02-PCCP Primary Containment Control
3. NEI 99-01 Primary Containment Conditions PC Potential Loss 1.C

Barrier: Primary Containment

Category: D. PC Radiation / RCS Activity

Degradation Threat: Loss

Threshold:

None

Barrier: Primary Containment

Category: D. PC Radiation / RCS Activity

Degradation Threat: Potential Loss

Threshold:

1. Drywell radiation > 20,000 R/hr

Definition(s):

None

Basis:

The Drywell High-Range Radiation Monitor (1(2)D22-RI-4195, 1(2)D22-RI-4196, 1(2)D22-RI-4197, 1(2)D22-RI-4198) reading of 20,000 R/hr indicates the release of reactor coolant into the drywell with elevated activity indicative of fuel damage.

Based on 20% clad damage, a containment radiation level of 20,000 R/hr is derived as follows:

0PEP-03.6.3 Table 3 100% Cladding Damage column 'No Spray' for 1 hour after shutdown is 100,000 R/hr. Per Step 7.2.2.1, 0.2 x 100,000 R/hr = 20,000 R/hr containment radiation corresponding to 20% clad damage.

In order to reach this Containment barrier Potential Loss threshold, a loss of the RCS barrier (RCS Loss D.5) and a loss of the Fuel Clad barrier (FC Loss D.2) have already occurred. This threshold, therefore, represents at a General Emergency classification.

The radiation monitor reading corresponds to an instantaneous release of all reactor coolant mass into the primary containment, assuming that 20% of the fuel cladding has failed. This level of fuel clad failure is well above that used to determine the analogous Fuel Clad Barrier Loss and RCS Barrier Loss thresholds.

NUREG-1228, *Source Estimations During Incident Response to Severe Nuclear Power Plant Accidents*, indicates the fuel clad failure must be greater than approximately 20% in order for there to be a major release of radioactivity requiring offsite protective actions. For this condition to exist, there must already have been a loss of the RCS Barrier and the Fuel Clad Barrier. It is therefore prudent to treat this condition as a potential loss of containment which would then escalate the emergency classification level to a General Emergency.

BNP Basis Reference(s):

1. 0PEP-03.6.3 Estimate of the Extent of Core Damage Under Accident Conditions

Barrier: Primary Containment

Category: E. PC Integrity or Bypass

Degradation Threat: Loss

Threshold:

1. UNISOLABLE direct downstream pathway to the environment exists after Primary Containment isolation signal

Definition(s):

UNISOLABLE - An open or breached system line that cannot be isolated, remotely or locally.

Basis:

This threshold addresses failure of open isolation devices which should close upon receipt of a manual or automatic containment isolation signal resulting in a significant radiological release pathway 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 "release 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, HPCI steam line or RCIC steam line breaks, unisolable RWCU system breaks, and unisloable 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, Abnormal Rad Release / Rad Effluent, EALs.

The existence of an in–line charcoal filter (SBGT) 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.

0EOP-02-PCCP, Primary Containment Control, Section PC/P 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). Under these conditions with a valid containment isolation signal, the Containment barrier should be considered lost.

The use of the modifier "direct" in defining the release path discriminates against release paths through interfacing liquid systems or minor release pathways, such as instrument lines, not protected by the Primary Containment Isolation System (PCIS).

The existence of a filter is not considered in the threshold assessment. Filters do not remove fission product noble gases. In addition, a filter could become ineffective due to iodine and/or particulate loading beyond design limits (i.e., retention ability has been exceeded) or water saturation from steam/high humidity in the release stream.

Following the leakage of RCS mass into primary containment and a rise in primary containment pressure, there may be minor radiological releases associated with allowable primary containment leakage through various penetrations or system components. Minor releases may also occur if a primary containment isolation valve(s) fails to close but the primary containment atmosphere escapes to an enclosed system. These releases do not constitute a loss or potential loss of primary containment but should be evaluated using the Recognition Category R ICs.

BNP Basis Reference(s):

1. 0EOP-02-PCCP Primary Containment Control
2. NEI 99-01 Primary Containment Isolation Failure PC Loss 3.A

Barrier: Primary Containment

Category: E. PC Integrity or Bypass

Degradation Threat: Loss

Threshold:

2. Intentional Primary Containment venting per EOPs

Definition(s):

None

Basis:

0EOP-02-PCCP, 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. 1, 2). The threshold is met when the operator begins venting the primary containment in accordance with 0EOP-01-SEP-01, not when actions are taken to bypass interlocks prior to opening the vent valves. Purge and vent actions specified in step PC/P-03 to control drywell pressure below the drywell high pressure scram setpoint or in section PC/H does not meet this threshold because such action is only permitted if offsite radioactivity release rates will remain below the ODCM limits.

EOPs may direct primary containment isolation valve logic(s) to be intentionally bypassed, even if offsite radioactivity release rate limits will be exceeded. Under these conditions with a valid primary containment isolation signal, the containment should also be considered lost if primary containment venting is actually performed.

Intentional venting of primary containment for primary containment pressure control to the secondary containment and/or the environment is a Loss of the Containment. Venting for primary containment pressure control when not in an accident situation (e.g., to control pressure below the drywell high pressure scram setpoint) does not meet the threshold condition.

BNP Basis Reference(s):

1. 0EOP-02-PCCP Primary Containment Control
2. 0EOP-01-SEP-01 Primary Containment Venting
3. NEI 99-01 Primary Containment Isolation Failure PC Loss 3.B

Barrier: Primary Containment

Category: E. PC Integrity or Bypass

Degradation Threat: Potential Loss

Threshold:

None

Barrier: Primary Containment

Category: F. SEC Judgment

Degradation Threat: Loss

Threshold:

1. **Any** condition in the opinion of the Site Emergency Coordinator that indicates loss of the Primary Containment barrier

Definition(s):

None

Basis:

The Site Emergency Coordinator judgment threshold addresses any other factors relevant to determining if the Primary 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 Site Emergency Coordinator should be mindful of the Loss of AC power (Station Blackout) and ATWS EALs to assure timely emergency classification declarations.

This threshold addresses any other factors that may be used by the Site Emergency Coordinator in determining whether the Primary Containment Barrier is lost.

BNP Basis Reference(s):

1. NEI 99-01 Emergency Director Judgment PC Loss 6.A

Barrier: Primary Containment

Category: F. SEC Judgment

Degradation Threat: Potential Loss

Threshold:

1. **Any** condition in the opinion of the Site Emergency Coordinator that indicates potential loss of the Primary Containment barrier

Definition(s):

None

Basis:

The Site Emergency Coordinator judgment threshold addresses any other factors relevant to determining if the Primary 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 Site Emergency Coordinator should be mindful of the Loss of AC power (Station Blackout) and ATWS EALs to assure timely emergency classification declarations.

This threshold addresses any other factors that may be used by the Site Emergency Coordinator in determining whether the Primary Containment Barrier is lost.

BNP Basis Reference(s):

1. NEI 99-01 Emergency Director Judgment PC Loss 6.A

Safe Operation & Shutdown Areas Tables R-2 & H-2 Bases

Background

NEI 99-01 Revision 6 ICs AA3 and HA5 prescribe declaration of an Alert based on impeded access to rooms or areas (due to either area radiation levels or hazardous gas concentrations) where equipment necessary for normal plant operations, cooldown or shutdown is located. These areas are intended to be plant operating mode dependent. Specifically the Developers Notes For AA3 and HA5 states:

The "site-specific list of plant rooms or areas with entry-related mode applicability identified" should specify those rooms or areas that contain equipment which require a manual/local action as specified in operating procedures used for normal plant operation, cooldown and shutdown. Do not include rooms or areas in which actions of a contingent or emergency nature would be performed (e.g., an action to address an off-normal or emergency condition such as emergency repairs, corrective measures or emergency operations). In addition, the list should specify the plant mode(s) during which entry would be required for each room or area.

The list should not include rooms or areas for which entry is required solely to perform actions of an administrative or record keeping nature (e.g., normal rounds or routine inspections).

Further, as specified in IC HA5:

The list need not include the Control Room if adequate engineered safety/design features are in place to preclude a Control Room evacuation due to the release of a hazardous gas. Such features may include, but are not limited to, capability to draw air from multiple air intakes at different and separate locations, inner and outer atmospheric boundaries, or the capability to acquire and maintain positive pressure within the Control Room envelope.

BNP Table R-2 and H-2 Bases

A review of station operating procedures identified the following mode dependent in-plant actions and associated safe shutdown areas that are required for normal plant operation, cooldown or shutdown:

Location- Safe Shutdown Area	**Modes- 1, 2**	**Modes- 3, 4, 5**
-17 North RHR Unit-1	RHR Equipment. - *No entry required*	RHR Shut Down Cooling (SDC) ***RHR Pump Discharge Isolation Valves E11-F018 A&C*** Inventory Control Equipment - *No entry required* Reactivity Control. - *No entry required*
-17 North RHR Unit-2	RHR Equipment. - *No entry required*	RHR Shut Down Cooling (SDC) ***RHR Pump Discharge Isolation Valves E11-F018 A&C*** Inventory Control Equipment. - *No entry required* Reactivity Control. - *No entry required*
-17 South RHR Unit-1	RHR Equipment. - *No entry required*	RHR Shut Down Cooling (SDC) ***RHR Pump Discharge Isolation Valves E11-F018 B&D*** Inventory Control Equipment. - *No entry required* Reactivity Control. - *No entry required*
-17 South RHR Unit-2	RHR Equipment. - *No entry required*	RHR Shut Down Cooling (SDC) ***RHR Pump Discharge Isolation Valves E11-F018 B&D*** Inventory Control Equipment. - *No entry required* Reactivity Control. - *No entry required*
-17 North Core Spray	Core Spray Equipment - *No entry required*	Inventory Control - *No entry required*
-17 South Core Spray	Core Spray Equipment - *No entry required*	Inventory Control. - *No entry required*
Service Water Building 20'	Heat Sink equipment. - *No entry required*	Heat Sink equipment. - *No entry required*

Safe Operation & Shutdown Areas Tables R-2 & H-2 Bases

Control Building HVAC Room (Turbine Building 70')	Habitability. - *No entry required*	Habitability. - *No entry required*
Reactor Building HVAC (Reactor Building 80' West)	Habitability. - *No entry required*	Habitability. - *No entry required*
Emergency Diesel Generators (EDG Building 20')	Electrical Power, Local control parameters at EDG panel. - No entry required	Electrical Power, Local control parameters at EDG panel. - No entry required
Emergency Diesel Generators 4-Day Tank Rooms	Electrical Power. - *No entry required*	Electrical Power. - *No entry required*
EDG Building HVAC (EDG Building 70')	Habitability. - *No entry required*	Habitability. - *No entry required*
38' & 70' Turbine Building (Unit 1&2)	Electrical Power. - *No entry required*	Electrical Power. - *No entry required*
4160 VAC (EDG building 70')	- *No entry required*	- *No entry required*
480 VAC (EDG Building 20')N & S ends	- *No entry required*	- *No entry required*
120 VAC Vital (Cable Spread U-1 & U-2)	- *No entry required*	- *No entry required*
Train A & B DC (Battery Rooms U-1 & U-2)	- *No entry required*	- *No entry required*
Reactor Building 20' East & West MCC Areas	- *No entry required*	RHR SDC. - ***1 (2) E11 - F009 & F008 valve breakers (RHR SDC Suction isolation valves)*** - ***1 (2) E11 - F006 A-D valve breakers, (RHR pump suction valves)*** - ***RHR SDC Suction Fill & Vent valves (Manual Valves)*** - ***RHR suction pipe flush valve breakers (E11-F011A & B, E11-V33, E11-V32***
Reactor Building 20' Pipe Tunnel	- *No entry required*	RHR SDC - ***RHR SDC Suction Fill & Vent valves (Manual Valves)***

Table R-2 & H-2 Results

Table R-2 & H-2 Safe Operation & Shutdown Areas	
Room/Area	**Mode Applicability**
Reactor Building -17' North RHR Unit-1 & 2	3, 4, 5
Reactor Building -17' South RHR Unit-1 & 2	3, 4, 5
Reactor Building 20' East & West MCC Areas Unit-1 & 2	3, 4, 5
Reactor Building 20' Pipe Tunnel Unit-1 & 2	3, 4, 5

Plant Operating Procedures Reviewed

1. Unit 1 & 2 RHR OP-17
   - Shutdown Cooling
   - Low Pressure Coolant Injection
2. Unit-1 & 2 UAT Backfeed OP-50
3. EDG Operation OP-50.1, OP-39
4. Unit-1 & 2 Service Water OP-43
5. Unit-1 & 2 Core Spray
6. Control Building Ventilation System 2OP-37
7. Defense In Depth AP-22

REVISION SUMMARY

Revision 6 of 0PEP-02.2.1 consists of the following changes:

[xxxx]

Brunswick Steam Electric Plant, Unit Nos. 1 and 2
Renewed Facility Operating License Nos. DPR-71 and DPR-62
Docket Nos. 50-325 and 50-324
Response to Request for Additional Information Regarding Adoption of Emergency Action Level Scheme Pursuant to NEI 99-01, Revision 6, "Development of Emergency Action Levels for Non-Passive Reactors"

Revised BSEP EAL Wall Charts, 0PEP-02.1,
"Initial Emergency Actions"

The 2 drawings specifically referenced in the letter have been processed into ADAMS.

These drawings can be accessed within the ADAMS package or by performing a search on the Document/Report Number.

D01 – D02X