Constellation Energy
Calvert Cliffs Nuclear Power Plant

EAL Comparison Matrix

Revision 0 [Draft D3 1/6/11]

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Introduction

This document provides a line-by-line comparison of the Initiating Conditions (ICs), Mode Applicability and Emergency Action Levels (EALs) in NEI 99-01 Rev. 5 Final, Methodology for Development of Emergency Action Levels, February 2008 (ADAMS Accession Number ML080450149), and the Calvert Cliffs Nuclear Power Plant (CCNPP) ICs, Mode Applicability and EALs. This document provides a means of assessing CCNPP differences and deviations from the NRC endorsed guidance given in NEI 99-01. Discussion of CCNPP EAL bases and lists of source document references are given in the EAL Technical Bases Document. It is, therefore, advisable to reference the EAL Technical Bases Document for background information while using this document.

Comparison Matrix Format

The ICs and EALs discussed in this document are grouped according to NEI 99-01 Recognition Categories. Within each Recognition Category, the ICs and EALs are listed in tabular format according to the order in which they are given in NEI 99-01. Generally, each row of the comparison matrix provides the following information:

- NEI EAL/IC identifier
- NEI EAL/IC wording
- CCNPP EAL/IC identifier
- CCNPP EAL/IC wording
- Description of any differences or deviations

EAL Wording

In Section 4.2, NEI recommends the following: "The method of [EAL] presentation should be one with which the operations and health physics staff are comfortable. As is the case for emergency procedures, bases for steps should be in a separate (or separable) document suitable for training and for reference by emergency response personnel and offsite agencies. Each nuclear plant should already have presentation and human factors standards as part of its procedure writing guidance. EALs that are consistent with those procedure writing standards (in particular, emergency operating procedures which most closely correspond to the conditions under which EALs must be used) should be the norm for each utility."

To assist the Emergency Director (ED), the CCNPP EALs have been written in a clear and concise style (to the extent that the differences from the NEI EAL wording could be reasonably documented and justified). As a result, unnecessary words have been removed from the CCNPP EALs to reduce EAL-user reading burden to the extent practicable.

The wording reduction gained from elimination of a few characters in a given EAL may not appear to be advantageous within the context of one EAL. When applied to the composite set of EALs, however, significant gains are realized and reading efficiency is improved. This supports timely and accurate classification in the tense atmosphere of an emergency event. The EAL differences introduced to reduce reading burden comprise almost all of the differences justified in this document.

EAL Emphasis Techniques

Due to the width of the table columns and table formatting constraints in this document, line breaks and indentation may differ slightly from the appearance of comparable wording in the source documents. NEI 99-01 is the source document for the NEI EALs; the CCNPP EAL Technical Bases Document for the CCNPP EALs.

Development of the CCNPP IC/EAL wording has attempted to minimize inconsistencies and apply sound human factors principles. As a result, differences occur between NEI and CCNPP ICs/EALs for these reasons alone. When such difference may infer a technical difference in the associated NEI IC/EAL, the difference is identified and a justification provided.

The print and paragraph formatting conventions summarized below guide presentation of the CCNPP EALs in accordance with the EAL writing criteria. Space restrictions in the EAL table of this document sometimes override this criteria in cases when following the criteria would introduce undesirable complications in the EAL layout.

- Upper case print is reserved for system abbreviations, logic terms (and, or, etc. when not used as a conjunction), annunciator window engravings.
- Bold font is used for logic terms, negative terms (**not**, **cannot**, etc.), **ANY**, **all**.

- Underscore is avoided as it can interfere with text in narrow line spacing.
- Three or more items in a list are normally introduced with "**ANY** of the following" or "**all** of the following." Items of the list begin with bullets when a priority or sequence is not inferred.
- The use of **AND/OR** logic within the same EAL has been avoided when possible. When such logic cannot be avoided, indentation and separation of subordinate contingent phrases is employed.

Global Differences

The differences listed below generally apply throughout the set of EALs and are not repeated in the Justification sections of this document. The global differences do not decrease the effectiveness of the intent of NEI 99-01.

1. The NEI phrase "Notification of Unusual Event" has been changed to "Unusual Event" or abbreviated "UE" to reduce EAL-user reading burden.
2. NEI 99-01 IC Example EALs are implemented in separate plant EALs to improve clarity and readability. For example, NEI lists all IC HU1 Example EALs under one IC. The corresponding CCNPP EALs appear as unique EALs (e.g., HU1.1 through HU1.5).
3. Mode applicability identifiers (numbers/letter) modify the NEI 99-01 mode applicability names as follows: 1 - Power Operation, 2 - Startup, 3 - Hot Standby, 4 - Hot Shutdown, 5 - Cold Shutdown, 6 - Refuel, D – Defueled. NEI 99-01defines Defueled as follows: "Reactor Vessel contains no irradiated fuel (full core off-load during refueling or extended outage)."
4. NEI 99-01 uses the terms greater than, less than, greater than or equal to, etc. in the wording of some ICs and example EALs. For consistency and reduce EAL-user reading burden, CCNPP has adopted use of boolean symbols in place of the NEI 99-01 text modifiers.
5. "min." is the standard abbreviation for "minutes" and is used to reduce EAL user reading burden.
6. IC/EAL identification:

- NEI Recognition Category A "Abnormal Radiation Levels/ Radiological Effluents" has been changed to Category R "Abnormal Rad Levels / Rad Effluents." The designator "R" is more intuitively associated with radiation (rad) or radiological events. NEI IC designators beginning with "A" have likewise been changed to "R."
- NEI 99-01 defines the thresholds requiring emergency classification (example EALs) and assigns them to ICs which, in turn, are grouped in "Recognition Categories." The Recognition Categories, however, are so broad and the IC descriptions are so varied that an EAL is difficult to locate in a timely manner when the EAL-user must refer to a set of EALs with the NEI organization and identification scheme. The NEI document clearly states that the EAL/IC/Recognition Category scheme is **not** intended to be the plant-specific EAL scheme for any plant, and appropriate human factors principles should be applied to development of an EAL scheme that helps the EAL-user make timely and accurate classifications. CCNPP endeavors to improve upon the NEI EAL organization and identification scheme to enhance usability of the plant-specific EAL set. To this end, the CCNPP IC/EAL scheme includes the following features:
    a. Division of the NEI EAL set into three 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, Hot Standby, 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, Refuel or Defueled mode.

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

b. Within each of the above three groups, assignment of EALs to categories/subcategories – Category and subcategory titles are selected to represent conditions that are operationally significant to the EAL-user. Subcategories are used as necessary to further divide the EALs of a category into logical sets of possible emergency classification thresholds. The CCNPP EAL categories/subcategories and their relationship to NEI Recognition Categories are listed in Table 1.

c. Unique identification of each EAL – Four characters comprise the EAL identifier as illustrated in Figure 1.

Figure 1 – EAL Identifier

EAL Identifier

XXX.X

Category (R, H, E, S, F, C)

Emergency classification (G, S, A, U)

Sequential number within subcategory/classification

Subcategory number (1 if no subcategory)

The first character is a letter associated with the category in which the EAL is located. The second character is a letter associated with the emergency classification level (G for General Emergency, S for Site Area Emergency, A for Alert, and U for Notification of Unusual Event). The third character is a number associated with one or more subcategories within a given category. Subcategories are sequentially numbered beginning with the number "1". If a category does not have a subcategory, this character is assigned the number "1". The fourth character is a number preceded by a period for each EAL within a subcategory. EALs are sequentially numbered within the emergency classification level of a subcategory beginning with the number "1".

The EAL identifier is designed to fulfill the following objectives:

- Uniqueness – The EAL identifier ensures that there can be no confusion over which EAL is driving the need for emergency classification.
- Speed in locating the EAL of concern – When the EALs are displayed in a matrix format, knowledge of the EAL identifier alone can lead the EAL-user to the location of the EAL within the classification matrix. The identifier conveys the category, subcategory and classification level. This assists ERO responders (who may not be in the same facility as the ED) to find the EAL of concern in a timely manner without the need for a word description of the classification threshold.
- Possible classification upgrade – The category/subcategory/identifier scheme helps the EAL-user find higher emergency classification EALs that may become active if plant conditions worsen.

Note that the NEI 99-01 identifier only identifies the IC, not the specific example EAL threshold. The NEI scheme, therefore, does not fulfill the above objectives which are desirable in facilitating timely and accurate emergency classification.

Table 2 lists the CCNPP ICs and EALs that correspond to the NEI ICs/Example EALs when the above EAL/IC organization and identification scheme is implemented.

Differences and Deviations

In accordance NRC Regulatory Issue Summary (RIS) 2003-18 "Use of Nuclear Energy Institute (NEI) 99-01, Methodology for Development of Emergency Action Levels" Supplements 1 and 2, a difference is an EAL change in which the basis scheme guidance differs in wording but agrees in meaning and intent, such that classification of an event would be the same, whether using the basis scheme guidance or the CCNPP EAL. A deviation is an EAL change in which the basis scheme guidance differs in wording and is altered in meaning or intent, such that classification of the event could be different between the basis scheme guidance and the CCNPP proposed EAL.

Administrative changes that do not actually change the textual content are neither differences nor deviations. Likewise, any format change that does not alter the wording of the IC or EAL is considered neither a difference nor a deviation.

The following are examples of differences:

- Choosing the applicable EAL based upon plant type (i.e., BWR vs. PWR).
- Using a numbering scheme other than that provided in NEI 99-01 that does not change the intent of the overall scheme.
- Where the NEI 99-01 guidance specifically provides an option to not include an EAL if equipment for the EAL does not exist at CCNPP (e.g., automatic real-time dose assessment capability).
- Pulling information from the bases section up to the actual EAL that does not change the intent of the EAL.
- Choosing to state ALL Operating Modes are applicable instead of stating N/A, or listing each mode individually under the Abnormal Rad Level/Radiological Effluent and Hazard and Other Conditions Affecting Plant Safety sections.
- Using synonymous wording (e.g., greater than or equal to vs. at or above, less than or equal vs. at or below, greater than or less than vs. above or below, etc.)
- Adding CCNPP equipment/instrument identification and/or noun names to EALs.
- Combining like ICs that are exactly the same but have different operating modes as long as the intent of each IC is maintained and the overall progression of the EAL scheme is not affected.
- Any change to the IC and/or EAL, and/or basis wording, as stated in NEI 99-01, that does not alter the intent of the IC and/or EAL, i.e., the IC and/or EAL continues to:
    - Classify at the correct classification level.
    - Logically integrate with other EALs in the EAL scheme.
    - Ensure that the resulting EAL scheme is complete (i.e., classifies all potential emergency conditions).

The following are examples of deviations:

- Use of altered mode applicability.
- Altering key words or time limits.
- Changing words of physical reference (protected area, safety-related equipment, etc.).
- Eliminating an IC. This includes the removal of an IC from the Fission Product Barrier Degradation category as this impacts the logic of Fission Product Barrier ICs.
- Changing a Fission Product Barrier from a Loss to a Potential Loss or vice-versa.
- Not using NEI 99-01definitions as the intent is for all NEI 99-01 users to have a standard set of defined terms as defined in NEI 99-01. Differences due to plant types are permissible (BWR or PWR). Verbatim compliance to the wording in NEI 99-01 is not necessary as long as the intent of the defined word is maintained. Use of the wording provided in NEI 99-01 is encouraged since the intent is for all users to have a standard set of defined terms as defined in NEI 99-01.
- Any change to the IC and/or EAL, and/or basis wording as stated in NEI 99-01 that does alter the intent of the IC and/or EAL, i.e., the IC and/or EAL:
    - Does not classify at the classification level consistent with NEI 99-01.
    - Is not logically integrated with other EALs in the EAL scheme.
    - Results in an incomplete EAL scheme (i.e., does not classify all potential emergency conditions).

The "Difference/Deviation Justification" columns in the remaining sections of this document identify each difference between the NEI 99-01 IC/EAL wording and the CCNPP IC/EAL wording. An explanation that justifies the reason for each difference is then provided. If the difference is determined to be a deviation, a statement is made to that affect and explanation is given that states why classification may be different from the NEI 99-01 IC/EAL and the reason for its acceptability. In all cases, however, the differences and deviations do not decrease the effectiveness of the intent of NEI 99-01. A summary list of CCNPP EAL deviations from NEI 99-01 is given in Table 3.

Table 1 – CCNPP EAL Categories/Subcategories

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

Table 2 – NEI / CCNPP EAL Identification Cross-Reference

NEI		CCNPP	
IC	Example EAL	Category and Subcategory	EAL
AU1	1	R – Abnormal Rad Release / Rad Effluent, 1 – Offsite Rad Conditions	RU1.1
AU1	2	R – Abnormal Rad Release / Rad Effluent, 1 – Offsite Rad Conditions	RU1.2
AU1	3	R – Abnormal Rad Release / Rad Effluent, 1 – Offsite Rad Conditions	RU1.3
AU1	4	N/A	N/A
AU1	5	N/A	N/A
AU2	1	R – Abnormal Rad Release / Rad Effluent, 2 – Onsite Rad Conditions & Spent Fuel Events	RU2.1
AU2	2	R – Abnormal Rad Release / Rad Effluent, 2 – Onsite Rad Conditions & Spent Fuel Events	RU2.2
AA1	1	R – Abnormal Rad Release / Rad Effluent, 1 – Offsite Rad Conditions	RA1.1
AA1	2	R – Abnormal Rad Release / Rad Effluent, 1 – Offsite Rad Conditions	RA1.2
AA1	3	R – Abnormal Rad Release / Rad Effluent, 1 – Offsite Rad Conditions	RA1.3
AA1	4	N/A	N/A
AA1	5	N/A	N/A
AA2	1	R – Abnormal Rad Release / Rad Effluent, 2 – Onsite Rad Conditions & Spent Fuel Events	RA2.2
AA2	2	R – Abnormal Rad Release / Rad Effluent, 2 – Onsite Rad Conditions & Spent Fuel Events	RA2.1

NEI		CCNPP	
IC	Example EAL	Category and Subcategory	EAL
AA3	1	R – Abnormal Rad Release / Rad Effluent, 2 – Onsite Rad Conditions & Spent Fuel Events	RA2.3
AS1	1	R – Abnormal Rad Release / Rad Effluent, 1 – Offsite Rad Conditions	RS1.1
AS1	2	R – Abnormal Rad Release / Rad Effluent, 1 – Offsite Rad Conditions	RS1.2
AS1	3	N/A	N/A
AS1	4	R – Abnormal Rad Release / Rad Effluent, 1 – Offsite Rad Conditions	RS1.3
AG1	1	R – Abnormal Rad Release / Rad Effluent, 1 – Offsite Rad Conditions	RG1.1
AG1	2	R – Abnormal Rad Release / Rad Effluent, 1 – Offsite Rad Conditions	RG1.2
AG1	3	N/A	N/A
AG1	4	R – Abnormal Rad Release / Rad Effluent, 1 – Offsite Rad Conditions	RG1.3
CU1	1	C – Cold SD/ Refuel System Malfunction, 3 – PCS Level	CU3.1
CU2	1	C – Cold SD/ Refuel System Malfunction, 3 – PCS Level	CU3.2
CU2	2	C – Cold SD/ Refuel System Malfunction, 3 – PCS Level	CU3.3
CU3	1	C – Cold SD/ Refuel System Malfunction, 1 – Loss of AC Power	CU1.1
CU4	1	C – Cold SD/ Refuel System Malfunction, 4 – PCS Temperature	CU4.1
CU4	2	C – Cold SD/ Refuel System Malfunction, 4 – PCS Temperature	CU4.2
CU6	1, 2	C – Cold SD/ Refuel System Malfunction, 5 – Communications	CU5.1
CU7	1	C – Cold SD/ Refuel System Malfunction, 2 – Loss of DC Power	CU2.1
CU8	1	N/A	N/A

NEI		CCNPP	
IC	**Example EAL**	**Category and Subcategory**	**EAL**
CU8	2	C – Cold SD/ Refuel System Malfunction, 6 – Inadvertent Criticality	CU6.1
CA1	1, 2	C – Cold SD/ Refuel System Malfunction, 3 – PCS Level	CA3.1
CA3	1	C – Cold SD/ Refuel System Malfunction, 1 – Loss of Power	CA1.1
CA4	1, 2	C – Cold SD/ Refuel System Malfunction, 4 – PCS Temperature	CA4.1
CS1	1	C – Cold SD/ Refuel System Malfunction, 4 – PCS Temperature	CA4.2
CS1	2	C – Cold SD/ Refuel System Malfunction, 3 – PCS Level	CS3.1
CS1	3	C – Cold SD/ Refuel System Malfunction, 3 – PCS Level	CS3.2
CG1	1	C – Cold SD/ Refuel System Malfunction, 3 – PCS Level	CS3.3
CG1	2	C – Cold SD/ Refuel System Malfunction, 3 – PCS Level	CG3.1
D-AU1 D-AU2 D-SU1 D-HU1 D-HU2 D-HU3 D-AA1 D-AA2 D-HA1 D-HA2		N/A	N/A
E-HU1	1	E - ISFSI	EU1.1
FU1	1	F – Fission Product Barriers	FU1.1

NEI		CCNPP	
IC	Example EAL	Category and Subcategory	EAL
FA1	1	F – Fission Product Barriers	FA1.1
FS1	1	F – Fission Product Barriers	FS1.1
FG1	1	F – Fission Product Barriers	FG1.1
HU1	1	H – Hazards, 1 – Natural or Destructive Phenomena	HU1.1
HU1	2	H – Hazards, 1 – Natural or Destructive Phenomena	HU1.2
HU1	3	H – Hazards, 1 – Natural or Destructive Phenomena	HU1.3
HU1	4	H – Hazards, 1 – Natural or Destructive Phenomena	HU1.4
HU1	5	H – Hazards, 1 – Natural or Destructive Phenomena	HU1.5
HU2	1	H – Hazards, 2 – Fire or Explosion	HU2.1
HU2	2	H – Hazards, 2 – Fire or Explosion	HU2.2
HU3	1	H – Hazards, 3 – Hazardous Gas	HU3.1
HU3	2	H – Hazards, 3 – Hazardous Gas	HU3.2
HU4	1, 2, 3	H – Hazards, 4 – Security	HU4.1
HU5	1	H – Hazards, 6 – Judgment	HU6.1
HA1	1	H – Hazards, 1 – Natural or Destructive Phenomena	HA1.1
HA1	2	H – Hazards, 1 – Natural or Destructive Phenomena	HA1.2
HA1	3	H – Hazards, 1 – Natural or Destructive Phenomena	HA1.3
HA1	4	H – Hazards, 1 – Natural or Destructive Phenomena	HA1.4

NEI		CCNPP	
IC	Example EAL	Category and Subcategory	EAL
HA1	5	H – Hazards, 1 – Natural or Destructive Phenomena	HA1.6
HA1	6	H – Hazards, 1 – Natural or Destructive Phenomena	HA1.5
HA2	1	H – Hazards, 2 – Fire or Explosion	HA2.1
HA3	1	H – Hazards, 3 – Hazardous Gas	HA3.1
HA4	1, 2	H – Hazards, 4 – Security	HA4.1
HA5	1	H – Hazards, 5 – Control Room Evacuation	HA5.1
HA6	1	H – Hazards, 6 –Judgment	HA6.1
HS2	1	H – Hazards, 5 – Control Room Evacuation	HS5.1
HS3	1	H – Hazards, 6 – Judgment	HS6.1
HS4	1	H – Hazards, 4 – Security	HS4.1
HG1	1	H – Hazards, 4 – Security	HG4.1
HG1	2	H – Hazards, 4 – Security	HG4.2
HG2	1	H – Hazards, 6 – Judgment	HG6.1
SU1	1	S – System Malfunction, 1 – Loss of AC Power	SU1.1
SU2	1	S – System Malfunction, 4 – Inability to Reach or Maintain Shutdown Conditions	SU4.1
SU3	1	S – System Malfunction, 5 – Instrumentation	SU5.1
SU4	1	S – System Malfunction, 7 – Fuel Clad Degradation	SU7.2
SU4	2	S – System Malfunction, 7 – Fuel Clad Degradation	SU7.1

NEI		CCNPP	
IC	Example EAL	Category and Subcategory	EAL
SU5	1, 2	S – System Malfunction, 8 – RCS Leakage	SU8.1
SU6	1, 2	S – System Malfunction, 6 –Communications	SU6.1
SU8	1 (BWR)	N/A	N/A
SU8	1 (PWR)	S – System Malfunction, 3 – Criticality & RPS Failure	SU3.1
SA2	1	S – System Malfunction, 3 – Criticality & RPS Failure	SA3.1
SA4	1	S – System Malfunction, 5 – Instrumentation	SA5.1
SA5	1	S – System Malfunction, 1 – Loss of AC Power	SA1.1
SS1	1	S – System Malfunction, 1 – Loss of AC Power	SS1.1
SS2	1	S – System Malfunction, 3 – Criticality & RPS Failure	SS3.1
SS3	1	S – System Malfunction, 1 – Loss of DC Power	SS2.1
SS6	1	S – System Malfunction, 5 – Instrumentation	SS5.1
SG1	1	S – System Malfunction, 1 – Loss of AC Power	SG1.1
SG2	1	S – System Malfunction, 3 – Criticality & RPS Failure	SG3.1

Table 3 – Summary of Deviations

NEI		CCNPP EAL	Description
IC	Example EAL		
SU5	1, 2	SU8.1	The phrase "for ≥ 15 min. (Note 4)" has been added to the CCNPP EAL to allow mitigation by operating procedures prior to declaration. This is a deviation from NEI 99-01 Revision 5.
HU2	1	HU2.1	The third paragraph of the NEI basis has been edited to clarify the significance of the 15-minute duration. If the alarm cannot be verified by redundant Control Room or nearby Fire Panel indications, notification from the field that a fire exists starts the concurrent 15-minute classification and fire suppression clocks. This change is consistent with the manner in which the Control Room and Fire Brigade leaders verify fires. This change is necessary to avoid declaring Unusual Event emergencies for spurious alarms that, due to the sensor location, cannot be verified within 15 minutes of receipt of the alarm. This is a deviation from NEI 99-01 Revision 5.

Category R

Abnormal Rad Levels / Radiological Effluents

NEI IC#	NEI IC Wording and Mode Applicability	CCNPP IC#(s)	CCNPP IC Wording and Mode Applicability	Difference/Deviation Justification
AU1	Any release of gaseous or liquid radioactivity to the environment greater than 2 times the Radiological Effluent Technical Specifications/ODCM for 60 minutes or longer. MODE: All	RU1	**ANY** release of gaseous or liquid radioactivity to the environment greater than 2 times the ODCM for 60 minutes or longer MODE: All	The CCNPP ODCM limits provide the site-specific Radiological Effluent Technical Specifications.

NEI Ex. EAL #	NEI Example EAL Wording	CCNPP EAL #	CCNPP EAL Wording	Difference/Deviation Justification
1	VALID reading on **ANY** of the following radiation monitors greater than the reading shown for 60 minutes or longer: (site specific monitor list and threshold values) **Note:** The Emergency Director should not wait until the applicable time has elapsed, but should declare the event as soon as it is determined that the release duration has exceeded, or will likely exceed, the applicable time. In the absence of data to the contrary, assume that the release duration has exceeded the applicable time if an ongoing release is detected and the release start time is unknown.	RU1.1	**ANY** gaseous monitor reading > Table R-1 column "UE" for ≥ 60 min. (Note 2) Note 2: The ED should **not** wait until the applicable time has elapsed, but should declare the event as soon as it is determined that the release duration has exceeded, or will likely exceed, the applicable time. In the absence of data to the contrary, assume that the release duration has exceeded the applicable time if an ongoing release is detected and the release start time is unknown.	The NEI phrase "VALID reading on **ANY**..." has been changed to "**ANY**...reading." All EAL thresholds assume valid readings for emergency classification. This change implements EAL FAQ #4. The NEI phrase "...of the following radiation monitors greater than the reading shown ...(site specific monitor list and threshold values)" has been replaced with "...Gaseous monitors > Table R-1 column "UE"..." UE, Alert, SAE and GE thresholds for all CCNPP continuously monitored gaseous release pathways are listed in Table R-1 to consolidate the information in a single location and, thereby, simplify identification of the thresholds by the EAL user. The values shown in Table R-1 column "UE", consistent with the NEI bases, represent two times the ODCM release limits for both liquid and gaseous release. Gaseous release is emphasized in this EAL to be consistent with the NEI basis, which states: "Some sites may find it advantageous to address gaseous and liquid releases with separate initiating conditions and EALs." The NEI phrase "...for 60 minutes or longer" has been replaced with "...≥ 60 min." to reduce EAL user reading

NEI Ex. EAL #	NEI Example EAL Wording	CCNPP EAL #	CCNPP EAL Wording	Difference/Deviation Justification
				burden. The symbol "≥" means "greater than or equal to" and thus implements the intent of the NEI phrase. Reference to the NEI note is included in the EAL wording "(Note 2)." Numbering the note facilitates referencing in the EAL matrix.
2	VALID reading on any effluent monitor reading greater than 2 times the alarm setpoint established by a current radioactivity discharge permit for 60 minutes or longer. **Note:** The Emergency Director should not wait until the applicable time has elapsed, but should declare the event as soon as it is determined that the release duration has exceeded, or will likely exceed, the applicable time. In the absence of data to the contrary, assume that the release duration has exceeded the applicable time if an ongoing release is detected and the release start time is unknown.	RU1.2	Liquid monitor reading > Table R-1 column "UE" for ≥ 60 min. (Note 2) Note 2: The ED should **not** wait until the applicable time has elapsed, but should declare the event as soon as it is determined that the release duration has exceeded, or will likely exceed, the applicable time. In the absence of data to the contrary, assume that the release duration has exceeded the applicable time if an ongoing release is detected and the release start time is unknown.	The NEI phrase "VALID reading on any…" has been changed to "Liquid monitor reading." All EAL thresholds assume valid readings for emergency classification. This change implements EAL FAQ #4. The NEI phrase "…effluent monitor reading greater than 2 times the alarm setpoint established by a current radioactivity discharge permit …" has been replaced with "…liquid monitor > Table R-1 column "UE"…" UE and Alert thresholds for all CCNPP effluent pathways governed by a radioactivity discharge permit are listed in Table R-1 to consolidate the information in a single location and, thereby, simplify identification of the thresholds by the EAL user. The values shown in Table R-1 column "UE", consistent with the NEI bases, represent two times the ODCM release limits for both liquid and gaseous release. Liquid release is emphasized in this EAL to be consistent with the NEI basis, which states "This alarm setpoint may be associated with a planned batch release, or a continuous release path." Liquid releases at CCNPP are the only planned batch releases subject to the discharge permit process. This change is also consistent with the NEI basis, which states "Some sites may find it advantageous to address gaseous and liquid releases with separate initiating conditions and EALs." The NEI phrase "…for 60 minutes or longer" has been replaced with "…≥ 60 min." to reduce EAL user reading burden. The symbol "≥" means "greater than or equal to" and thus implements the intent of the NEI phrase. The phrase "*with Waste Water effluent discharge **not** isolated"

NEI Ex. EAL #	NEI Example EAL Wording	CCNPP EAL #	CCNPP EAL Wording	Difference/Deviation Justification
				has been associated with the liquid release path in Table R-1. At low classification levels, NEI states in the AU1/AA1 bases that the concern for classification is the continuing, uncontrolled release of radioactivity and not the magnitude of the release. When the liquid release is isolated, the release is no longer continuing nor is it uncontrolled. Therefore, the classification is not appropriate when the liquid release is isolated. Reference to the NEI note is included in the EAL wording "(Note 2)." Numbering the note facilitates referencing in the EAL matrix.
3	Confirmed sample analyses for gaseous or liquid releases indicates concentrations or release rates greater than 2 times (site specific RETS values) for 60 minutes or longer. **Note**: The Emergency Director should not wait until the applicable time has elapsed, but should declare the event as soon as it is determined that the release duration has exceeded, or will likely exceed, the applicable time. In the absence of data to the contrary, assume that the release duration has exceeded the applicable time if an ongoing release is detected and the release start time is unknown.	RU1.3	Confirmed sample analyses for gaseous or liquid releases indicate concentrations or release rates > 2 x ODCM limits for ≥ 60 min. (Note 2) Note 2: The ED should **not** wait until the applicable time has elapsed, but should declare the event as soon as it is determined that the release duration has exceeded, or will likely exceed, the applicable time. In the absence of data to the contrary, assume that the release duration has exceeded the applicable time if an ongoing release is detected and the release start time is unknown.	The CCNPP ODCM is the site-specific radiological effluent Technical Specifications. The NEI phrase "2 times" has been replaced with phrase "2 x" to reduce EAL user reading burden. The phrases have the same meaning. The NEI phrase "The NEI phrase "...for 60 minutes or longer" has been replaced with "...≥ 60 min." to reduce EAL-user reading burden. The symbol "≥" means "greater than or equal to" and thus implements the intent of the NEI phrase. Reference to the NEI note is included in the EAL wording "(Note 2)." Numbering the note facilitates referencing in the EAL matrix.
4	VALID reading on perimeter radiation monitoring system reading greater than 0.10 mR/hr above normal* background for 60	N/A	N/A	Deleted NEI Example EAL #4 because the plant is not equipped with a perimeter radiation monitoring system. This threshold is properly addressed by the radiation monitors

NEI Ex. EAL #	NEI Example EAL Wording	CCNPP EAL #	CCNPP EAL Wording	Difference/Deviation Justification
	minutes or longer. [for sites having telemetered perimeter monitors]			listed in Table R-1 and dose assessment capabilities.
5	VALID indication on automatic real-time dose assessment capability indicating greater than (site specific value) for 60 minutes or longer. [for sites having such capability]	N/A	N/A	Deleted NEI Example EAL #5 because the plant is not equipped with real-time dose assessment. This threshold is properly addressed by the radiation monitors listed in Table R-1 and dose assessment capabilities.

Table R-1 Effluent Monitor Classification Thresholds

Monitor	GE	SAE	Alert	UE
Gaseous				
WRNGM (RIC-5415)	3.2E+09 µCi/sec	3.2E+08 µCi/sec	3.2E+07 µCi/sec	3.2E+05 µCi/sec
Main Steam Effluent (RI-5421, RI-5422)	40.0 rem/hr	4.0 rem/hr	0.40 rem/hr	N/A
Main Vent (RI-5415)	N/A	N/A	N/A	2.0E+05 cpm
Waste Processing (RI-5410)	N/A	N/A	N/A	4.0E+05 cpm
Fuel Handling Area Vent (RI-5420)	N/A	N/A	N/A	3.4E+05 cpm
Liquid				
Liquid Waste Disch* (RE-2201)	N/A	N/A	off-scale hi	8.4E+05 cpm

* with effluent discharge **not** isolated

NEI IC#	NEI IC Wording and Mode Applicability	CCNPP IC#(s)	CCNPP IC Wording and Mode Applicability	Difference/Deviation Justification
AU2	UNPLANNED rise in plant radiation levels MODE: All	RU2	Unplanned rise in plant radiation levels MODE: All	None

NEI Ex. EAL #	NEI Example EAL Wording	CCNPP EAL #	CCNPP EAL Wording	Difference/Deviation Justification
1	a. UNPLANNED water level drop in a reactor refueling pathway as indicated by (site specific level or indication). **AND** b. VALID Area Radiation Monitor reading rise on (site specific list).	RU2.1	UNPLANNED water level drop in a reactor refueling pathway as indicated by **ANY** of the following (Note 3): • Inability to restore and maintain SFP level > Technical Specification limit (65 ft 7 in) • Inability to restore and maintain RFP level > Technical Specification limit (56 ft 8.5 in) • Report of visual observation of an uncontrolled drop in water level in the RFP or SFP **AND** Area radiation monitor reading rise on **ANY** of the following: • SFP Area RM-320 EL-69 (RI-7024) • Spent Fuel Handling Machine (RI-7025) • Unit 1/2 CNTMT EL-69 (RI-5316A/B/C/D) Note 3: If loss of water level in the refueling pathway occurs while in Mode 5, 6 or D, consider classification	The site specific level or indication of an unplanned water level drop that may result in increased area radiation are either: the inability to restore and maintain SFP water level > Technical Specifications (65 ft 8.5 in.), the inability to restore and maintain RFP level > Technical Specification limit (56 ft 8.5 in) or report of visual observation of an uncontrolled drop in water level in the RFP or SFP. The NEI term "VALID" has been deleted. All EAL thresholds assume valid readings for emergency classification. This change implements EAL FAQ #4. The site specific area radiation monitors are listed. Note 3 has been added to the plant EAL wording to ensure subcategory C.3 EALs are reviewed when loss of water shielding above spent fuel adversely affects area radiation levels.

			under EALs CU3.1, CU3.2 or CU3.3	
2	UNPLANNED VALID Area Radiation Monitor readings or survey results indicate a rise by a factor of 1000 over normal* levels. *Normal can be considered as the highest reading in the past twenty-four hours excluding the current peak value.	RU2.2	UNPLANNED area radiation readings increases by a factor of 1,000 over NORMAL LEVELS	The NEI term "monitor" has been deleted to clarify that radiation readings obtained by portable survey instruments are an acceptable source for assessing this EAL. The NEI term "VALID" has been deleted. All EAL thresholds assume valid readings for emergency classification. This change implements EAL FAQ #4. Deleted the asterisk phrase and added the defined phrase to the EAL Technical Bases: "Normal Levels As applied to radiological IC/EALs, the highest reading in the past twenty-four hours excluding the current peak value." This change implements EAL FAQ #5.

NEI IC#	NEI IC Wording	CCNPP IC#(s)	CCNPP IC Wording	Difference/Deviation Justification
AA1	Any release of gaseous or liquid radioactivity to the environment greater than 200 times the Radiological Effluent Technical Specifications/ODCM for 15 minutes or longer. MODE: All	RA1	**ANY** release of gaseous or liquid radioactivity to the environment greater than 200 times the ODCM for 15 minutes or longer MODE: All	The CCNPP ODCM limits provide the site-specific Radiological Effluent Technical Specifications.

NEI Ex. EAL #	NEI Example EAL Wording	CCNPP EAL #	CCNPP EAL Wording	Difference/Deviation Justification
1	VALID reading on **ANY** of the following radiation monitors greater than the reading shown for 15 minutes or longer: (site specific monitor list and threshold values) **Note:** The Emergency Director should not wait until the applicable time has elapsed, but should declare the event as soon as it is determined that the release duration has exceeded, or will likely exceed, the applicable time. In the absence of data to the contrary, assume that the release duration has exceeded the applicable time if an ongoing release is detected and the release start time is unknown.	RA1.1	**ANY** gaseous monitor reading > Table R-1 column "Alert" for ≥ 15 min. (Note 2) Note 2: The ED should **not** wait until the applicable time has elapsed, but should declare the event as soon as it is determined that the release duration has exceeded, or will likely exceed, the applicable time. In the absence of data to the contrary, assume that the release duration has exceeded the applicable time if an ongoing release is detected and the release start time is unknown.	The NEI phrase "VALID reading on **ANY**…" has been changed to "**ANY**…reading." All EAL thresholds assume valid readings for emergency classification. This change implements EAL FAQ #4. The NEI phrase "…of the following radiation monitors greater than the reading shown …" has been replaced with "…gaseous monitors > Table R-1 column "Alert"…" The CCNPP radiation monitors that detect radioactivity effluent release to the environment are listed in Table R-1. UE, Alert, SAE and GE thresholds for all CCNPP continuously monitored gaseous release pathways are listed in Table R-1 to consolidate the information in a single location and, thereby, simplify identification of the thresholds by the EAL-user. The values shown in Table R-1 column "Alert", consistent with the NEI bases, represent two hundred times the ODCM release limits for both liquid and gaseous release. The NEI phrase "…for 15 minutes or longer" has been replaced with "…≥ 15 min." to reduce EAL-user reading burden. The symbol "≥" means "greater than or equal to" and thus implements the intent of the NEI phrase.

				Reference to the NEI note is included in the EAL wording "(Note 2)." Numbering the note facilitates referencing in the EAL matrix.
2	VALID reading on any effluent monitor reading greater than 200 times the alarm setpoint established by a current radioactivity discharge permit for 15 minutes or longer. **Note:** The Emergency Director should not wait until the applicable time has elapsed, but should declare the event as soon as it is determined that the release duration has exceeded, or will likely exceed, the applicable time. In the absence of data to the contrary, assume that the release duration has exceeded the applicable time if an ongoing release is detected and the release start time is unknown.	RA1.2	Liquid monitor reading > Table R-1 column "Alert" for ≥ 15 min. (Note 2) Note 2: The ED should **not** wait until the applicable time has elapsed, but should declare the event as soon as it is determined that the release duration has exceeded, or will likely exceed, the applicable time. In the absence of data to the contrary, assume that the release duration has exceeded the applicable time if an ongoing release is detected and the release start time is unknown.	The NEI phrase "VALID reading on any…" has been changed to "Liquid monitor reading." All EAL thresholds assume valid readings for emergency classification. This change implements EAL FAQ #4. The NEI phrase "…effluent monitor reading greater than 200 times the alarm setpoint established by a current radioactivity discharge permit …" has been replaced with "…liquid monitors > Table R-1 column "ALERT" for ≥ 15 min. …" UE, Alert, SAE and GE thresholds for all CCNPP continuously monitored release pathways are listed in Table R-1 to consolidate the information in a single location and, thereby, simplify identification of the thresholds by the EAL user. Liquid release is emphasized in this EAL to be consistent with the NEI basis, which states "This alarm setpoint may be associated with a planned batch release, or a continuous release path." Liquid releases at CCNPP are the only planned batch releases subject to the discharge permit process. This change is also consistent with the NEI basis, which states "Some sites may find it advantageous to address gaseous and liquid releases with separate initiating conditions and EALs." The phrase "with Waste Water effluent **not** isolated" has been associated with the liquid release path. At low classification levels, NEI states in the RU1/RA1 bases that the concern for classification is the continuing, uncontrolled release of radioactivity and not the magnitude of the release. When the liquid release is isolated, the release is no longer continuing nor is it uncontrolled. Therefore, the classification is not appropriate when the liquid release is isolated. The NEI phrase "…for 15 minutes or longer" has been replaced with "…≥ 15 min." to reduce EAL-user reading burden. The symbol "≥" means "greater than or equal to" and thus implements the intent of the NEI phrase. Reference to the NEI note is included in the EAL wording

				"(Note 2)." Numbering the note facilitates referencing in the EAL matrix.
3	Confirmed sample analyses for gaseous or liquid releases indicates concentrations or release rates greater than 200 times (site specific RETS values) for 15 minutes or longer. **Note:** The Emergency Director should not wait until the applicable time has elapsed, but should declare the event as soon as it is determined that the release duration has exceeded, or will likely exceed, the applicable time. In the absence of data to the contrary, assume that the release duration has exceeded the applicable time if an ongoing release is detected and the release start time is unknown.	RA1.3	Confirmed sample analyses for gaseous or liquid releases indicate concentrations or release rates > 200 x ODCM limits for ≥ 15 min. Note 2: The ED should **not** wait until the applicable time has elapsed, but should declare the event as soon as it is determined that the release duration has exceeded, or will likely exceed, the applicable time. In the absence of data to the contrary, assume that the release duration has exceeded the applicable time if an ongoing release is detected and the release start time is unknown.	The NEI phrase "200 times" has been replaced with phrase "200 x" to reduce EAL-user reading burden. The phrases have the same meaning. The CCNPP ODCM is the site-specific effluent Technical Specifications. The NEI phrase "...for 15 minutes or longer" has been replaced with "...≥ 15 min." to reduce EAL-user reading burden. The symbol "≥" means "greater than or equal to" and thus implements the intent of the NEI phrase. Reference to the NEI note is included in the EAL wording "(Note 2)." Numbering the note facilitates referencing in the EAL matrix.
4	VALID reading on perimeter radiation monitoring system reading greater than 10.0 mR/hr above normal* background for 15 minutes or longer. [for sites having telemetered perimeter monitors]	N/A	N/A	Deleted NEI Example EAL #4 because the plant is not equipped with a perimeter radiation monitoring system. This threshold is properly addressed by the radiation monitors listed in Table R-1 and dose assessment capabilities.
5	VALID indication on automatic real-time dose assessment capability indicating greater than (site specific value) for 15 minutes or longer. [for sites having such capability]	N/A	N/A	Deleted NEI Example EALs #5 because the plant is not equipped with and real-time dose assessment. This threshold is properly addressed by the radiation monitors listed in Table R-1 and dose assessment capabilities.

NEI IC#	NEI IC Wording	CCNPP IC#(s)	CCNPP IC Wording	Difference/Deviation Justification
AA2	Damage to irradiated fuel or loss of water level that has resulted or will result in the uncovering of irradiated fuel outside the reactor vessel. MODE: All	RA2	Damage to irradiated fuel or loss of water level that has resulted or will result in the uncovering of irradiated fuel outside the Reactor Vessel. MODE: All	None

NEI Ex. EAL #	NEI Example EAL Wording	CCNPP EAL #	CCNPP EAL Wording	Difference/Deviation Justification
1	A water level drop in the reactor refueling cavity, spent fuel pool or fuel transfer canal that will result in irradiated fuel becoming uncovered.	RA2.2	A water level drop in a reactor refueling pathway that will result in irradiated fuel becoming uncovered	The terms "reactor refueling cavity, spent fuel pool or fuel transfer canal" have been replaced with the term "reactor refueling pathway" to encompass all three volumes where irradiated fuel may be located. This change implements EAL FAQ #6.
2	A VALID alarm or (site specific elevated reading) on **ANY** of the following due to damage to irradiated fuel or loss of water level. (site specific radiation monitors)	RA2.1	Alarm on **ANY** of the following radiation monitors due to damage to irradiated fuel or loss of water level: • Fuel Handling Area Vent (RI-5420) • SFP Area RM-320 EL-69 (RI-7024) • Spent Fuel Handling Machine (RI-7025) • Unit 1/2 CNTMT EL-69 (RI-5316A/B/C/D)	The NEI term "VALID" has been deleted. All EAL thresholds assume valid readings for emergency classification. This change implements EAL FAQ #4. The EAL provides a site-specific list of radiation monitors applicable to this threshold.

NEI IC#	NEI IC Wording	CCNPP IC#(s)	CCNPP IC Wording	Difference/Deviation Justification
AA3	Rise in radiation levels within the facility that impedes operation of systems required to maintain plant safety functions MODE: All	RA3	Rise in radiation levels within the facility that impedes operation of systems required to maintain plant safety functions MODE: All	None

NEI Ex. EAL #	NEI Example EAL Wording	CCNPP EAL #	CCNPP EAL Wording	Difference/Deviation Justification
1	VALID (site-specific) radiation monitor readings GREATER THAN 15 mR/hr in areas requiring continuous occupancy to maintain plant safety functions: (Site-specific) list	RA3.1	Dose rates > 15 mRem/hr in **ANY** of the following areas requiring continuous occupancy to maintain plant safety functions: • Control Room • CAS • SAS	The NEI term "VALID" has been deleted. All EAL thresholds assume valid readings for emergency classification. This change implements EAL FAQ #4. The words "VALID (site-specific) radiation monitor readings GREATER THAN" was replaced with "Dose rates >…" It doesn't matter if the 15 mRem/hr was measured with an ARM or survey instrument, therefore, the term radiation monitor was deleted to not confuse those who may think that only implies a fixed ARM. The symbol ">" means "greater than." The Control Room, CAS and SAS are the CCNPP areas requiring continuous occupancy to maintain plant safety functions. Since all three areas require continuous occupancy, elevated dose rates in any one area could preclude occupancy and, therefore, satisfy the intent of the IC.

NEI IC#	NEI IC Wording	CCNPP IC#(s)	CCNPP IC Wording	Difference/Deviation Justification
AS1	Off-site dose resulting from an actual or IMMINENT release of gaseous radioactivity greater than 100 mrem TEDE or 500 mrem Thyroid CDE for the actual or projected duration of the release. MODE: All MODE: All	RS1	Offsite dose resulting from an actual or imminent release of gaseous radioactivity exceeds 100 mRem TEDE or 500 mRem thyroid CDE for the actual or projected duration of the release using actual meteorology MODE: All	The NEI abbreviation "mrem" has been replaced with the plant abbreviation "mRem" to agree with units of measure given in the EPA PAGs. This change implements EAL FAQ #8. The phrase "using actual meteorology" has been added to the CCNPP IC for consistency with RG1.1 IC wording. This change implements EAL FAQ #9.

NEI Ex. EAL #	NEI Example EAL Wording	CCNPP EAL #	CCNPP EAL Wording	Difference/Deviation Justification
1	VALID reading on ANY of the following radiation monitors greater than the reading shown for 15 minutes or longer: (site specific monitor list and threshold values) **Note:** The Emergency Director should not wait until the applicable time has elapsed, but should declare the event as soon as it is determined that the condition will likely exceed the applicable time. If dose assessment results are available, declaration should be based on dose assessment instead of radiation monitor values. Do not delay declaration awaiting dose assessment results.	RS1.1	**ANY** radiation monitor reading > Table R-1 column "SAE" for ≥ 15 min. (Note 1) • Do **not** delay declaration awaiting dose assessment results • If dose assessment results are available, declaration should be based on dose assessment instead of radiation monitor values (see EAL RS1.2) Note 1: The ED should **not** wait until the applicable time has elapsed, but should declare the event as soon as it is determined that the condition will likely exceed the applicable time	The NEI phrase "VALID reading on **ANY**..." has been changed to "**ANY**...reading." All EAL thresholds assume valid readings for emergency classification. This change implements EAL FAQ #4. The NEI phrase "...of the following... the reading shown..." has been replaced with "...Table R-1 column "SAE"..." The site-specific list is provided in Table R-1. The NEI phrases "greater than" and "...15 minutes or longer" have been replaced with ">" and "...≥ 15 min.", respectively, to reduce EAL-user reading burden. The symbols implement the intent of the NEI phrase. Reference to the NEI note is included in the EAL wording "(Note 1)." Numbering the note facilitates referencing in the EAL matrix. The second and third sentences of the NEI note have been incorporated in the wording of the CCNPP EAL for clarification. EAL validation exercises demonstrated the need to emphasis this information in a form other than a note.

2	Dose assessment using actual meteorology indicates doses greater than 100 mrem TEDE or 500 mrem thyroid CDE at or beyond the site boundary.	RS1.2	Dose assessment using actual meteorology indicates doses > 100 mRem TEDE or 500 mRem thyroid CDE at or beyond the site boundary	The NEI abbreviation "mrem" has been replaced with the plant abbreviation "mRem" to agree with units of measure given in the EPA PAGs. This change implements EAL FAQ #8.
3	VALID perimeter radiation monitoring system reading greater than 100 mR/hr for 15 minutes or longer. [for sites having telemetered perimeter monitors]	N/A	N/A	Deleted NEI Example EAL #3 because the plant is not equipped with a perimeter radiation monitoring system. This threshold is properly addressed by the radiation monitors listed in Table R-1 and dose assessment capabilities.
4	Field survey results indicate closed window dose rates greater than 100 mR/hr expected to continue for 60 minutes or longer; or analyses of field survey samples indicate thyroid CDE greater than 500 mrem for one hour of inhalation, at or beyond the site boundary.	RS1.3	Field survey results indicate closed window dose rates > 100 mRem/hr expected to continue for ≥ 60 min. at or beyond the site boundary **OR** Analyses of field survey samples indicate thyroid CDE > 500 mRem for 1 hr of inhalation at or beyond the site boundary (Note 1)	Split the example into two logical conditions separated by the "OR" logical connector for usability. The NEI abbreviation "R" has been replaced with the plant abbreviation "Rem" to agree with units of measure given in the EPA PAGs. This change implements EAL FAQ #8. The NEI phrase "one hour" has been abbreviated "1 hr" to reduce EAL-user reading burden. Reference to the NEI note is included in the EAL wording "(Note 1)." Numbering the note facilitates referencing in the EAL matrix.

NEI IC#	NEI IC Wording	CCNPP IC#(s)	CCNPP IC Wording	Difference/Deviation Justification
AG1	Off-site dose resulting from an actual or IMMINENT release of gaseous radioactivity greater than 1000 mrem TEDE or 5000 mrem Thyroid CDE for the actual or projected duration of the release using actual meteorology. MODE: All	RG1	Offsite dose resulting from an actual or imminent release of gaseous radioactivity greater than 1,000 mRem TEDE or 5,000 mRem thyroid CDE for the actual or projected duration of the release using actual meteorology MODE: All	The NEI abbreviation "mrem" has been replaced with the plant abbreviation "mRem" to agree with units of measure given in the EPA PAGs.

NEI Ex. EAL #	NEI Example EAL Wording	CCNPP EAL #	CCNPP EAL Wording	Difference/Deviation Justification
1	VALID reading on **ANY** of the following radiation monitors greater than the reading shown for 15 minutes or longer: (site specific monitor list and threshold values) **Note:** The Emergency Director should not wait until the applicable time has elapsed, but should declare the event as soon as it is determined that the condition will likely exceed the applicable time. If dose assessment results are available, declaration should be based on dose assessment instead of radiation monitor values. Do not delay declaration awaiting dose assessment results.	RG1.1	**ANY** radiation monitor reading > Table R-1 column "GE" for ≥ 15 min. (Note 1) • Do **not** delay declaration awaiting dose assessment results • If dose assessment results are available, declaration should be based on dose assessment instead of radiation monitor values (see EAL RG1.2) Note 1: The ED should **not** wait until the applicable time has elapsed, but should declare the event as soon as it is determined that the condition will likely exceed the applicable time	The NEI phrase "VALID reading on **ANY**..." has been changed to "**ANY**...reading." All EAL thresholds assume valid readings for emergency classification. This change implements EAL FAQ #4. The NEI phrase "...of the following... the reading shown..." has been replaced with "...Table R-1 column "GE"..." The site-specific list is provided in Table R-1. The symbol ">" means "greater than." The NEI phrases "greater than" and "...15 minutes or longer" have been replaced with ">" and "...≥ 15 min.", respectively, to reduce EAL-user reading burden. The symbols implement the intent of the NEI phrase. Reference to the NEI note is included in the EAL wording "(Note 1)." Numbering the note facilitates referencing in the EAL matrix. The second and third sentences of the NEI note have been incorporated in the wording of the CCNPP EAL for clarification. EAL validation exercises demonstrated the need to emphasis this information in a form other than a note.

2	Dose assessment using actual meteorology indicates doses greater than 1000 mrem TEDE or 5000 mrem thyroid CDE at or beyond the site boundary.	RG1.2	Dose assessment using actual meteorology indicates doses > 1,000 mRem TEDE or 5,000 mRem thyroid CDE at or beyond the site boundary	The NEI phrase "greater than" has been replaced with the symbol ">" to reduce EAL-user reading burden. The symbol ">" means "greater than" and thus implements the intent of the NEI phrase.
3	VALID perimeter radiation monitoring system reading greater than 1000 mR/hr for 15 minutes or longer. [for sites having telemetered perimeter monitors]	N/A	N/A	Deleted NEI Example EAL #3 because the plant is not equipped with a perimeter radiation monitoring system. This threshold is properly addressed by the radiation monitors listed in Table R-1 and dose assessment capabilities.
4	Field survey results indicate closed window dose rates greater than 1000 mR/hr expected to continue for 60 minutes or longer; or analyses of field survey samples indicate thyroid CDE greater than 5000 mrem for one hour of inhalation, at or beyond site boundary.	RG1.3	Field survey results indicate closed window dose rates > 1,000 mRem/hr expected to continue for ≥ 60 min. at or beyond the site boundary **OR** Analyses of field survey samples indicate thyroid CDE > 5,000 mRem for 1 hr of inhalation at or beyond the site boundary (Note 1)	Split the example into two logical conditions separated by the "OR" logical connector for usability. The NEI abbreviation "R" has been replaced with the plant abbreviation "Rem" to agree with units of measure given in the EPA PAGs. This change implements EAL FAQ #8. The NEI phrase "one hour" has been abbreviated "1 hr" to reduce EAL-user reading burden. Reference to the NEI note is included in the EAL wording "(Note 1)." Numbering the note facilitates referencing in the EAL matrix.

Category C

Cold Shutdown / Refueling System Malfunction

NEI IC#	NEI IC Wording	CCNPP IC#(s)	CCNPP IC Wording	Difference/Deviation Justification
CU1	RCS Leakage MODE: Cold Shutdown	CU3	RCS Leakage MODE: 5 - Cold Shutdown	None

NEI Ex. EAL #	NEI Example EAL Wording	CCNPP EAL #	CCNPP EAL Wording	Difference/Deviation Justification
1	**Note:** The Emergency Director should not wait until the applicable time has elapsed, but should declare the event as soon as it is determined that the condition will likely exceed the applicable time. 1. RCS leakage results in the inability to maintain or restore RPV level greater than (site specific low level RPS actuation setpoint) for 15 minutes or longer. [*BWR*] 1. RCS leakage results in the inability to maintain or restore level within (site specific pressurizer or RCS/RPV level target band) for 15 minutes or longer. [*PWR*]	CU3.1	RCS leakage results in the inability to maintain or restore **EITHER** of the following for ≥ 15 min. (Note 4): Pressurizer level > 101 in. **OR** RCS level within the target band established by procedure (when the level band was established below 101 in.) Note 4: The ED should **not** wait until the applicable time has elapsed, but should declare the event as soon as it is determined that the condition has exceeded, or will likely exceed, the applicable time	The BWR portion of the NEI EAL has not been implemented because CCNPP is a PWR. The site-specific pressurizer low level heater trip actuation setpoint is specified. The NEI phrase "15 minutes or longer" has been replaced with "≥ 15 min." to reduce EAL-user reading burden. The symbol "≥" means "greater than or equal to" and thus implements the intent of the NEI phrase. Reference to the NEI note is included in the EAL wording "(Note 4)." Numbering the note facilitates referencing in the EAL matrix.

NEI IC#	NEI IC Wording	CCNPP IC#(s)	CCNPP IC Wording	Difference/Deviation Justification
CU2	UNPLANNED loss of RCS/RPV inventory. MODE: Refueling	CU3	RCS leakage MODE: 6 - Refuel	IC wording aligned with NEI IC CU1 to support grouping NEI IC CU1 and CU2 EALs under the same subcategory. There is no fundamental difference between an unplanned loss of RCS inventory and RCS leakage. This change implements EAL FAQ #41.

NEI Ex. EAL #	NEI Example EAL Wording	CCNPP EAL #	CCNPP EAL Wording	Difference/Deviation Justification
1	**Note:** The Emergency Director should not wait until the applicable time has elapsed, but should declare the event as soon as it is determined that the condition will likely exceed the applicable time. UNPLANNED RCS/RPV level drop as indicated by either of the following: • RCS/RPV water level drop below the RPV flange for 15 minutes or longer when the RCS/RPV level band is established above the RPV flange. • RCS/RPV water level drop below the RCS level band for 15 minutes or longer when the RCS/RPV level band is established below the RPV flange.	CU3.2	UNPLANNED RCS level drop below **EITHER** of the following for ≥ 15 min. (Note 4): Reactor Vessel flange (44 ft) (when the level band was established above the flange) **OR** Target band (when the level band was established below the flange) Note 4: The ED should **not** wait until the applicable time has elapsed, but should declare the event as soon as it is determined that the condition has exceeded, or will likely exceed, the applicable time.	The NEI abbreviation "RCS/RPV" has been changed to "RCS" to use terminology commonly accepted at PWRs. The NEI phrase "15 minutes or longer" has been replaced with "≥ 15 min." to reduce EAL-user reading burden. The symbol "≥" means "greater than or equal to" and thus implements the intent of the NEI phrase. Reference to the NEI note is included in the EAL wording "(Note 4)." Numbering the note facilitates referencing in the EAL matrix. The NEI phrase "RPV flange" has been replaced with "Reactor Vessel flange" to use terminology commonly accepted at PWRs. In the second bullet, the NEI phrase "RCS level band" has been replaced with "Target band" for consistency with terminology used in EAL CU3.1. The NEI introductory clause and the two NEI bulleted conditions have been reworded for clarification.
2	RCS/RPV level cannot be monitored with a loss of	CU3.3	RCS level **cannot** be monitored with a loss of RCS inventory as	The NEI abbreviation "RCS/RPV" has been changed to "RCS" to

	RCS/RPV inventory as indicated by an unexplained level rise in (site specific sump or tank).		indicated by an unexplained level rise in **ANY** Table C-2 sump / tank attributable to RCS leakage	use terminology commonly accepted at PWRs. The NEI phrase "(site-specific sump or tank)" has been replaced with "**ANY** Table C-2 sump / tank attributable to RCS leakage" for clarification. The list of sumps and tanks is too large to include within the wording of the EAL and maintain readability. Table C-2 contains the site-specific list of sumps and tanks.

Table C-2 RCS Leakage Indications
• Containment sump • Auxiliary Building sumps • Miscellaneous Waste System Tanks • RWT • RC Waste System Tank

NEI IC#	NEI IC Wording	CCNPP IC#(s)	CCNPP IC Wording	Difference/Deviation Justification
CU3	AC power capability to emergency busses reduced to a single power source for 15 minutes or longer such that any additional single failure would result in station blackout. MODE: Cold Shutdown, Refueling	CU1	AC power capability to 4kV vital buses reduced to a single power source for ≥ 15 min. such that **ANY** additional single failure would result in a complete loss of all 4kV vital bus power MODE: 5 - Cold Shutdown, 6 - Refuel, D - Defueled	"...emergency busses..." replaced with "...4kV vital buses..." as the site specific terminology for emergency buses. "...station blackout." replaced with "...a complete loss of all 4kV vital bus power" as this describes the intended condition for CCNPP. The NEI phrase "15 minutes or longer" has been replaced with "≥ 15 min." to reduce EAL-user reading burden. The symbol "≥" means "greater than or equal to" and thus implements the intent of the NEI phrase. Added "D - Defueled" to the mode applicability to correct omission in NEI 99-01.

NEI Ex. EAL #	NEI Example EAL Wording	CCNPP EAL #	CCNPP EAL Wording	Difference/Deviation Justification
1	**Note:** The Emergency Director should not wait until the applicable time has elapsed, but should declare the event as soon as it is determined that the condition will likely exceed the applicable time. a. AC power capability to (site specific emergency busses) reduced to a single power source for 15 minutes or longer. **AND** b. Any additional single power source failure will result in station blackout.	CU1.1	AC power capability to 4kV vital buses 11(21) and 14(24) reduced to a single power source, Table C-1, for ≥ 15 min. (Note 4) **AND** **ANY** additional single power source failure will result in a complete loss of all 4kV vital bus power Note 4: The ED should **not** wait until the applicable time has elapsed, but should declare the event as soon as it is determined that the condition has exceeded, or will likely exceed, the applicable time.	The NEI phrase "15 minutes or longer" has been replaced with "≥ 15 min." to reduce EAL-user reading burden. The symbol "≥" means "greater than or equal to" and thus implements the intent of the NEI phrase. 4kV vital buses 11(21) and 14(24) are the CCNPP emergency buses. Table C-1 provides a list of CCNPP onsite and offsite AC power sources. "..station blackout." replaced with "...a complete loss of all 4kV vital bus power" as this describes the intended condition for CCNPP. This change implements EAL FAQ #36. Reference to the NEI note is included in the EAL wording "(Note 4)." Numbering the note facilitates referencing in the EAL matrix.

Table C-1 AC Power Sources	
Onsite	• 1(2)A DG • 1(2)B DG • 0C DG , if aligned
Offsite	• 500kV transmission line 5051* • 500kV transmission line 5052* • 500kV transmission line 5072* • SMECO line , if aligned * A credited 500kV line must have an independent 13kV service transformer

NEI IC#	NEI IC Wording	CCNPP IC#(s)	CCNPP IC Wording	Difference/Deviation Justification
CU4	UNPLANNED loss of decay heat removal capability with irradiated fuel in the RPV. MODE: Cold Shutdown, Refueling	CU4	Unplanned loss of decay heat removal capability MODE: 5 - Cold Shutdown, 6 - Refuel	The NEI acronym "RPV" has been replaced with the phrase "Reactor Vessel" to use terminology commonly accepted at PWRs. The NEI phrase "with irradiated fuel in the Reactor Vessel" has been deleted to implement EAL FAQ #11.

NEI Ex. EAL #	NEI Example EAL Wording	CCNPP EAL #	CCNPP EAL Wording	Difference/Deviation Justification
1	UNPLANNED event results in RCS temperature exceeding the Technical Specification cold shutdown temperature limit.	CU4.1	UNPLANNED event results in RCS temperature > 200°F	The NEI phrase "...exceeding the Technical Specification cold shutdown temperature limit" has been replaced with "> 200°F." >200°F is the Technical Specification cold shutdown temperature limit and is specified in the EAL instead of the NEI wording to reduce EAL-user reading burden.
2	Note: The Emergency Director should not wait until the applicable time has elapsed, but should declare the event as soon as it is determined that the condition will likely exceed the applicable time. Loss of all RCS temperature and RCS/RPV level indication for 15 minutes or longer.	CU4.2	Loss of **all** RCS temperature and RCS level indication for ≥ 15 min. (Note 4) Note 4: The ED should **not** wait until the applicable time has elapsed, but should declare the event as soon as it is determined that the condition has exceeded, or will likely exceed, the applicable time.	The NEI abbreviation "RCS/RPV" has been changed to "RCS" to use terminology commonly accepted at PWRs. The NEI phrase "15 minutes or longer" has been replaced with "≥ 15 min." to reduce EAL-user reading burden. The symbol "≥" means "greater than or equal to" and thus implements the intent of the NEI phrase. Reference to the NEI note is included in the EAL wording "(Note 4)." Numbering the note facilitates referencing in the EAL matrix.

NEI IC#	NEI IC Wording	CCNPP IC#(s)	CCNPP IC Wording	Difference/Deviation Justification
CU6	Loss of all On-site or Off-site communications capabilities. MODE: Cold Shutdown, Refueling, Defueled	CU5	Loss of **all** onsite or offsite communications capabilities MODE: 5 - Cold Shutdown, 6 - Refuel, D - Defueled	None

NEI Ex. EAL #	NEI Example EAL Wording	CCNPP EAL #	CCNPP EAL Wording	Difference/Deviation Justification
1	Loss of all of the following on-site communication methods affecting the ability to perform routine operations: (site specific list of communications methods)	CU5.1	Loss of **all** Table C-5 onsite (internal) communication methods affecting the ability to perform routine operations **OR** Loss of **all** Table C-5 offsite (external) communication methods affecting the ability to perform offsite notifications to any agency	CU5.1 implements Example EALs #1 and #2. These were combined for improved usability. The NEI example EALs specify site-specific lists of onsite and offsite communications methods. The CCNPP EAL lists these methods in Table C-5 because the number of communications methods is too long to include within the text of the EAL. The adjectives "(internal)" and "(external)" have been added to the CCNPP EAL for clarification. The terms "onsite/offsite" could be interpreted as the location in which the communication originates instead of the location to which communication is directed. Added the words "...to any agency" to clarify the intent of the bases statement: "The availability of one method of ordinary off-site communications is sufficient to inform federal, state, and local authorities of plant issues."
2	Loss of all of the following off-site communication methods affecting the ability to perform offsite notifications: (site specific list of communications methods)			

Table C-5 Communications Systems		
System	**Onsite** (internal)	**Offsite** (external)
Commercial phone system	X	X
Plant page system	X	
Microwave telephone (Hot-Lines) (EOB)	X	X
Dedicated offsite agency telephone system		X
FTS 2001 telephone system		X
CCNPP Radio System	X	X
Satellite Phone System		X
Cellular Phone System	X	X

NEI IC#	NEI IC Wording	CCNPP IC#(s)	CCNPP IC Wording	Difference/Deviation Justification
CU7	Loss of required DC power for 15 minutes or longer. MODE: Cold Shutdown, Refueling	CU2	Loss of **required** DC power for ≥ 15 min. MODE: 5 - Cold Shutdown, 6 - Refuel	The NEI phrase "15 minutes or longer" has been replaced with "≥ 15 min." to reduce EAL-user reading burden. The symbol "≥" means "greater than or equal to" and thus implements the intent of the NEI phrase.

NEI Ex. EAL #	NEI Example EAL Wording	CCNPP EAL #	CCNPP EAL Wording	Difference/Deviation Justification
1	**Note:** The Emergency Director should not wait until the applicable time has elapsed, but should declare the event as soon as it is determined that the condition will likely exceed the applicable time. Less than (site specific bus voltage indication) on required (site specific Vital DC busses) for 15 minutes or longer.	CU2.1	< 105 VDC for ≥ 15 min. on the 125 VDC buses (11, 12, 21 or 22) that are required to monitor and control the removal of decay heat (Note 4) Note 4: The ED should **not** wait until the applicable time has elapsed, but should declare the event as soon as it is determined that the condition has exceeded, or will likely exceed, the applicable time.	The NEI IC phrase "less than" has been replaced with "<" to reduce EAL-user reading burden. The symbol "<" means "less than" and thus implements the intent of the NEI phrase. "105 VDC" is the site-specific bus voltage indication. Emergency DC Buses (125V) 11, 12, 21 and 22 are the CCNPP vital DC buses. The NEI phrase "15 minutes or longer" has been replaced with "≥ 15 min." to reduce EAL-user reading burden. The symbol "≥" means "greater than or equal to" and thus implements the intent of the NEI phrase. The phrase "that are required to monitor and control the removal of decay heat" has been added to the EAL wording for clarification. It was not clear during EAL validation exercises which vital DC buses may be "required." The added phrase implements the first sentence of the NEI basis. Reference to the NEI note is included in the EAL wording "(Note 4)." Numbering the note facilitates referencing in the EAL matrix.

NEI IC#	NEI IC Wording	CCNPP IC#(s)	CCNPP IC Wording	Difference/Deviation Justification
CU8	Inadvertent Criticality MODE: Cold Shutdown, Refueling	CU6	Inadvertent criticality MODE: 5 - Cold Shutdown, 6 - Refueling	None

NEI Ex. EAL #	NEI Example EAL Wording	CCNPP EAL #	CCNPP EAL Wording	Difference/Deviation Justification
1	UNPLANNED sustained positive period observed on nuclear instrumentation. (BWR)	N/A	N/A	NEI Example EAL #1 has not been implemented because it applies only to BWR plants. CCNPP is a PWR. PWRs are not equipped with period meters.
2	UNPLANNED sustained positive startup rate observed on nuclear instrumentation. (PWR)	CU6.1	An UNPLANNED sustained positive startup rate observed on nuclear instrumentation	None

NEI IC#	NEI IC Wording	CCNPP IC#(s)	CCNPP IC Wording	Difference/Deviation Justification
CA1	Loss of RCS/RPV inventory. MODE: Cold Shutdown, Refueling	CA3	Loss of RCS inventory MODE: 5 - Cold Shutdown, 6 - Refuel	The NEI abbreviation "RCS/RPV" has been changed to "RCS" to use terminology commonly accepted at PWRs.

NEI Ex. EAL #	NEI Example EAL Wording	CCNPP EAL #	CCNPP EAL Wording	Difference/Deviation Justification
1	Loss of RCS/RPV inventory as indicated by level less than (site specific level). [Low-Low ECCS actuation setpoint / Level 2 (BWR)] [Bottom ID of the RCS loop (PWR)]	CA3.1	Loss of inventory as indicated by RCS water level < 35.6 ft (29 in. 6th alarm on RVLMS) **OR** RCS level **cannot** be monitored for ≥ 15 min. with a loss of RCS inventory as indicated by an unexplained level rise in **ANY** Table C-2 sump / tank attributable to RCS leakage (Note 4) Note 4: The ED should **not** wait until the applicable time has elapsed, but should declare the event as soon as it is determined that the condition has exceeded, or will likely exceed, the applicable time.	CA3.1 implements Example EALs #1 and #2. These were combined for improved usability. The NEI abbreviation "RCS/RPV" has been changed to "RCS" to use terminology commonly accepted at PWRs. The bottom ID of the RCS hot leg is indicated by RFP level at 35.6 ft or 29 in. (6th alarm) on RVLMS. CCNPP is a PWR and is not equipped with the BWR low-low ECCS actuation setpoint. The NEI phrase "15 minutes or longer" has been replaced with "≥ 15 min." to reduce EAL-user reading burden. The symbol "≥" means "greater than or equal to" and thus implements the intent of the NEI phrase. The NEI phrase "(site-specific sump or tank)" has been replaced with "**ANY** Table C-2 sump / tank attributable to RCS leakage" for clarification. The list of sumps and tanks is too large to include within the wording of the EAL and maintain readability. Table C-2 contains the site-specific list of sumps and tanks as well as observation of unisolable RCS leakage. Reference to the NEI note is included in the EAL wording "(Note 4)." Numbering the note facilitates referencing in the EAL matrix.
2	Note: The Emergency Director should not wait until the applicable time has elapsed, but should declare the event as soon as it is determined that the condition will likely exceed the applicable time. RCS/RPV level cannot be monitored for 15 minutes or longer with a loss of RCS/RPV inventory as indicated by an unexplained level rise in (site specific sump or tank).			

NEI IC#	NEI IC Wording	CCNPP IC#(s)	CCNPP IC Wording	Difference/Deviation Justification
CA3	Loss of all Off-site and all On-Site AC power to emergency busses for 15 minutes or longer. MODE: Cold Shutdown, Refueling, Defueled	CA1	Loss of **all** offsite and **all** onsite AC power to 4kV vital buses for ≥ 15 min. MODE: 5 - Cold Shutdown, 6 - Refuel, D - Defueled	"...emergency busses..." replaced with "...4kV vital buses..." as the site specific terminology for emergency buses. The NEI phrase "15 minutes or longer" has been replaced with "≥ 15 min." to reduce EAL-user reading burden. The symbol "≥" means "greater than or equal to" and thus implements the intent of the NEI phrase.

NEI Ex. EAL #	NEI Example EAL Wording	CCNPP EAL #	CCNPP EAL Wording	Difference/Deviation Justification
1	**Note:** The Emergency Director should not wait until the applicable time has elapsed, but should declare the event as soon as it is determined that the condition will likely exceed the applicable time. Loss of all Off-Site and all On-Site AC Power to (site specific emergency busses) for 15 minutes or longer.	CA1.1	Loss of **all** offsite and **all** onsite AC power, Table C-1, to 4kV vital buses 11(21) and 14(24) for ≥ 15 min. (Note 4) Note 4: The ED should **not** wait until the applicable time has elapsed, but should declare the event as soon as it is determined that the condition has exceeded, or will likely exceed, the applicable time.	The NEI phrase "15 minutes or longer" has been replaced with "≥ 15 min." to reduce EAL-user reading burden. The symbol "≥" means "greater than or equal to" and thus implements the intent of the NEI phrase. 4kV vital buses 11(21) and 14(24) are the CCNPP emergency buses. Table C-1 provides a list of CCNPP onsite and offsite AC power sources. Reference to the NEI note is included in the EAL wording "(Note 4)." Numbering the note facilitates referencing in the EAL matrix.

NEI IC#	NEI IC Wording	CCNPP IC#(s)	CCNPP IC Wording	Difference/Deviation Justification
CA4	Inability to maintain plant in cold shutdown. MODE: Cold Shutdown, Refueling	CA4	Inability to maintain plant in cold shutdown MODE: 5 - Cold Shutdown, 6 - Refuel	None

NEI Ex. EAL #	NEI Example EAL Wording	CCNPP EAL #	CCNPP EAL Wording	Difference/Deviation Justification
1	An UNPLANNED event results in RCS temperature greater than (site specific Technical Specification cold shutdown temperature limit) for greater than the specified duration on table.	CA4.1	An UNPLANNED event results in **EITHER**: RCS temperature > 200°F for > Table C-4 duration **OR** RCS pressure increase > 10 psi due to an unplanned loss of decay heat removal capability (this condition is **not** applicable in solid plant conditions)	CA4.1 implements NEI EALs #1 and #2. The NEI example EALs have been combined for simplification. The NEI phrase "greater than" has been replaced with ">" to reduce EAL-user reading burden. The symbol ">" means "greater than" and thus implements the intent of the NEI phrase. The NEI phrase "...exceeding the Technical Specification cold shutdown temperature limit" has been replaced with "> 200ºF." 200°F is the Technical Specification cold shutdown temperature limit. NEI criteria associated with RCS temperature exceeding the Technical Specification cold shutdown temperature limit are given in Table C-4. The NEI phrase "An UNPLANNED event results in RCS pressure increase greater than 10 psi due to a loss of RCS cooling" has been changed to "RCS pressure increase > 10 psi due to an unplanned loss of decay heat removal capability" for clarification. This change implements EAL FAQ #13. The CCNPP pressure of 10 psig is the site-specific RCS pressure based on the accuracy of RCS pressure instruments PI-103, PI-103-1 and PI-105.
2	An UNPLANNED event results in RCS pressure increase greater than 10 psi due to a loss of RCS cooling. (PWR-This EAL does not apply in Solid Plant conditions.)			

Table: RCS Reheat Duration Thresholds		
RCS	**Containment Closure**	**Duration**
Intact (but not RCS Reduced Inventory [PWR])	N/A	60 minutes*
Not intact or RCS Reduced Inventory *(PWR)*	Established	20 minutes*
	Not Established	0 minutes
* If an RCS heat removal system is in operation within this time frame and RCS temperature is being reduced, the EAL is not applicable.		

Table C-4 RCS Reheat Duration Thresholds		
RCS Status	Containment Closure Status	Duration
Intact **AND not** reduced inventory	N/A	60 min.*
Not intact **OR** reduced inventory	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.

NEI IC#	NEI IC Wording	CCNPP IC#(s)	CCNPP IC Wording	Difference/Deviation Justification
CS1	Loss of RCS/RPV inventory affecting core decay heat removal capability MODE: Cold Shutdown, Refueling	CS2	Loss of RCS inventory affecting core decay heat removal capability MODE: 5 - Cold Shutdown, 6 - Refuel	The NEI abbreviation "RCS/RPV" has been changed to "RCS" to use terminology commonly accepted at PWRs.

NEI Ex. EAL #	NEI Example EAL Wording	CCNPP EAL #	CCNPP EAL Wording	Difference/Deviation Justification
1	With CONTAINMENT CLOSURE not established, RCS/RPV level less than (site specific level). [6" below the bottom ID of the RCS loop (PWR)] [6" below the low-low ECCS actuation setpoint (BWR)]	CS3.1	With CONTAINMENT CLOSURE **not** established, RCS level < 34.7 ft (19 in. 7th alarm on RVLMS)	The NEI abbreviation "RCS/RPV" has been changed to "RCS" to use terminology commonly accepted at PWRs. 10 inches below the bottom ID of the RCS hot leg loop is indicated by the 7th alarm (19 in. above TOAF) on RVLMS. This value was selected instead of 6 in. below the hotleg as it is operationally significant and readily recognized on RVLMS. CCNPP is a PWR and is not equipped with the BWR low-low ECCS actuation setpoint.
2	With CONTAINMENT CLOSURE established, RCS/RPV level less than (site specific level for TOAF).	CS3.2	With CONTAINMENT CLOSURE established, RCS level < 32.9 ft (10 in. last alarm light on RVLMS (Note 6)) Note 6: The lowest RVLMS indication is the 10 in. alarm, which is 10 in. above top of active fuel. Therefore, this indicator should only be used when a valid RFP/RCS level indication is **not** available.	The NEI abbreviation "RCS/RPV" has been changed to "RCS" to use terminology commonly accepted at PWRs. The site-specific level for TOAF is 32.9 ft RFP. The lowest RVLMS indicating light is 10 in. light which indicates at 10 in. above TOAF. This information is provided in Note 6.
3	Note: The Emergency Director should not wait until the	CS3.3	RCS level **cannot** be monitored for ≥ 30 min. with a loss of RCS	The NEI abbreviation "RCS/RPV" has been changed to "RCS" to use terminology commonly accepted at PWRs.

	applicable time has elapsed, but should declare the event as soon as it is determined that the condition will likely exceed the applicable time. RCS/RPV level cannot be monitored for 30 minutes or longer with a loss of RCS/RPV inventory as indicated by ANY of the following: • (Site specific radiation monitor) reading greater than (site specific value). • Erratic Source Range Monitor Indication. • Unexplained level rise in (site specific sump or tank).		inventory as indicated by **ANY** of the following (Note 4): • Containment radiation > 6 R/hr • Erratic WRNI indication • Unexplained level rise in **ANY** Table C-2 sump / tank attributable to RCS leakage Note 4: The ED should **not** wait until the applicable time has elapsed, but should declare the event as soon as it is determined that the condition has exceeded, or will likely exceed, the applicable time.	The NEI phrase "30 minutes or longer" has been replaced with "≥ 30 min." to reduce EAL-user reading burden. The symbol "≥" means "greater than or equal to" and thus implements the intent of the NEI phrase. Reference to the NEI note is included in the EAL wording "(Note 4)." Numbering the note facilitates referencing in the EAL matrix. Containment radiation is indicated on 1(2)-RI-5317 A&B. Typical Containment radiation readings at full power are 1 to 1.2 R/hr. The Containment radiation monitors alarm at 6 R/hr. The 6 R/hr setpoint has been selected to be operationally significant and above that expected under normal plant conditions while in the Refuel mode. The NEI phrase "(site-specific sump or tank)" has been replaced with "**ANY** Table C-2 sump / tank attributable to RCS leakage" for clarification. The list of sumps and tanks is too large to include within the wording of the EAL and maintain readability. Table C-2 contains the site-specific list of sumps and tanks as well as observation of unisolable RCS leakage.

NEI IC#	NEI IC Wording	CCNPP IC#(s)	CCNPP IC Wording	Difference/Deviation Justification
CG1	Loss of RCS/RPV inventory affecting fuel clad integrity with containment challenged. MODE: Cold Shutdown, Refueling	CG3	Loss of RCS inventory affecting fuel clad integrity with Containment challenged MODE: 5 - Cold Shutdown, 6 - Refueling	The NEI abbreviation "RCS/RPV" has been changed to "RCS" to use terminology commonly accepted at PWRs.

NEI Ex. EAL #	NEI Example EAL Wording	CCNPP EAL #	CCNPP EAL Wording	Difference/Deviation Justification
1	**Note:** The Emergency Director should not wait until the applicable time has elapsed, but should declare the event as soon as it is determined that the condition will likely exceed the applicable time. a. RCS/RPV level less than (site specific level for TOAF) for 30 minutes or longer. **AND** b. **ANY** containment challenge indication (see Table):	CG3.1	RCS level < 32.9 ft (10 in. alarm on RVLMS, Note 6) for ≥ 30 min. (Note 4) **AND** **ANY** Containment Challenge Indication, Table C-3 Note 4: The ED should **not** wait until the applicable time has elapsed, but should declare the event as soon as it is determined that the condition has exceeded, or will likely exceed, the applicable time. Note 6: The lowest RVLMS indication is the 10 in. alarm, which is 10 in. above top of active fuel. Therefore, this indicator should only be used when a valid RFP/RCS level indication is **not** available.	The NEI abbreviation "RCS/RPV" has been changed to "RCS" to use terminology commonly accepted at PWRs. The site-specific level for TOAF is 32.9 ft RFP. The lowest RVLMS indicating light is the 10 in. light which indicates at 10 in. above TOAF. This information is provided in Note 6. The NEI phrase "30 minutes or longer" has been replaced with "≥ 30 min." to reduce EAL-user reading burden. The symbol "≥" means "greater than or equal to" and thus implements the intent of the NEI phrase. Reference to the NEI note is included in the EAL wording "(Note 4)." Numbering the note facilitates referencing in the EAL matrix. Table C-3 lists the Containment Challenge indications. "Secondary Containment radiation monitor reading above" has not been incorporated in Table C-3 because CCNPP is a PWR and not equipped with a secondary Containment. The site-specific level for TOAF is 32.9 ft RFP. The lowest RVLMS indicating light is 10 in. light which indicates at 10 in. above TOAF. This information is provided in Note 6.
2	a. RCS/RPV level cannot be monitored with core uncovery	CG3.2	RCS level **cannot** be monitored	The NEI abbreviation "RCS/RPV" has been changed to "RCS" to

	indicated by **ANY** of the following for 30 minutes or longer. • (Site specific radiation monitor) reading greater than (site specific setpoint). • Erratic source range monitor indication • UNPLANNED level rise in (site specific sump or tank). • *[Other site specific indications]* **AND** b. **ANY** containment challenge indication (see Table):		with core uncovery indicated by **ANY** of the following for ≥ 30 min. (Note 4): • Containment radiation > 6 R/hr • Erratic WRNI indication • Unexplained level rise in **ANY** Table C-2 sump / tank attributable to RCS leakage **AND** **ANY** Containment Challenge Indication, Table C-3 Note 4: The ED should **not** wait until the applicable time has elapsed, but should declare the event as soon as it is determined that the condition has exceeded, or will likely exceed, the applicable time	use terminology commonly accepted at PWRs. The NEI phrase "30 minutes or longer" has been replaced with "≥ 30 min." to reduce EAL-user reading burden. The symbol "≥" means "greater than or equal to" and thus implements the intent of the NEI phrase. Reference to the NEI note is included in the EAL wording "(Note 4)." Numbering the note facilitates referencing in the EAL matrix. Containment radiation is indicated on 1(2)-RI-5317 A&B. Typical Containment radiation readings at full power are 1 to 1.2 R/hr. The Containment radiation monitors alarm at 6 R/hr. The 6 R/hr setpoint has been selected to be operationally significant and above that expected under normal plant conditions while in the Refuel mode. The NEI term "UNPLANNED" has been changed to "Unexplained" for consistency with NEI IC CS1 Example EAL #3f. The NEI phrase "(site-specific sump or tank)" has been replaced with "**ANY** Table C-2 sump / tank attributable to RCS leakage" for clarification. The list of sumps and tanks is too large to include within the wording of the EAL and maintain readability. Table C-2 contains the site-specific list of sumps and tanks as well as observation of unisolable RCS leakage. Other site-specific indications of core uncovery could not be identified. Table C-3 lists the Containment Challenge indications. "Secondary Containment radiation monitor reading above" has not been incorporated in Table C-3 because CCNPP is a PWR and not equipped with a secondary containment.

Table: Containment Challenge Indications
• CONTAINMENT CLOSURE not established. • (Site specific explosive mixture) inside containment. • UNPLANNED rise in containment pressure. • Secondary containment radiation monitor reading above (site specific value). [*BWR only*]

Table C-3 Containment Challenge Indications
• Containment closure **not** established • Hydrogen concentration in Containment ≥ 4% • Unplanned rise in Containment pressure

Category D

Permanently Defueled Station Malfunction

NEI IC#	NEI IC Wording	CCNPP IC#(s)	CCNPP IC Wording	Difference/Deviation Justification
D-AU1 D-AU2 D-SU1 D-HU1 D-HU2 D-HU3 D-AA1 D-AA2 D-HA1 D-HA2	Recognition Category D Permanently Defueled Station Malfunction	N/A	N/A	NEI Recognition Category D ICs and EALs are applicable only to permanently defueled stations. CCNPP is not a defueled station.

Category E

Events Related to Independent Spent Fuel Storage Installations

NEI IC#	NEI IC Wording	CCNPP IC#(s)	CCNPP IC Wording	Difference/Deviation Justification
E-HU1	Damage to a loaded cask CONFINEMENT BOUNDARY MODE: Not applicable	EU1	Damage to a loaded cask confinement boundary	None

NEI Ex. EAL #	NEI Example EAL Wording	CCNPP EAL #	CCNPP EAL Wording	Difference/Deviation Justification
1	Damage to a loaded cask CONFINEMENT BOUNDARY	EU1.1	Damage to a loaded cask CONFINEMENT BOUNDARY	None

Category F

Fission Product Barrier Degradation

NEI IC#	NEI IC Wording	CCNPP IC#(s)	CCNPP IC Wording	Difference/Deviation Justification
FU1	ANY Loss or ANY Potential Loss of Containment MODE: Power Operation, Hot Standby, Startup, Hot Shutdown	FU1	**ANY** loss or **ANY** potential loss of Containment MODE: 1 - Power Operation, 2 - Startup, 3 - Hot Standby, 4 - Hot Shutdown	None

NEI Ex. EAL #	NEI Example EAL Wording	CCNPP EAL #	CCNPP EAL Wording	Difference/Deviation Justification
1	ANY Loss or ANY Potential Loss of Containment	FU1.1	**ANY** loss or **ANY** potential loss of Containment (Table F-1)	Table F-1 contains the loss and potential loss thresholds for the three fission product barriers and is the plant representation of NEI Table 5-F-3.

NEI IC#	NEI IC Wording	CCNPP IC#(s)	CCNPP IC Wording	Difference/Deviation Justification
FA1	ANY Loss or ANY Potential Loss of EITHER Fuel Clad OR RCS MODE: Power Operation, Hot Standby, Startup, Hot Shutdown	FA1	**ANY** loss or **ANY** potential loss of either Fuel Clad or RCS MODE: 1 - Power Operation, 2 - Startup, 3 - Hot Standby, 4 - Hot Shutdown	None

NEI Ex. EAL #	NEI Example EAL Wording	CCNPP EAL #	CCNPP EAL Wording	Difference/Deviation Justification
1	ANY Loss or ANY Potential Loss of EITHER Fuel Clad OR RCS	FA1.1	**ANY** loss or **ANY** potential loss of either Fuel Clad or RCS (Table F-1)	Table F-1 contains the loss and potential loss thresholds for the three fission product barriers and is the plant representation of NEI Table 5-F-3.

NEI IC#	NEI IC Wording	CCNPP IC#(s)	CCNPP IC Wording	Difference/Deviation Justification
FS1	Loss or Potential Loss of ANY Two Barriers MODE: Power Operation, Hot Standby, Startup, Hot Shutdown	FS1	Loss or potential loss of **ANY** two barriers MODE: 1 - Power Operation, 2 - Startup, 3 - Hot Standby, 4 - Hot Shutdown	None

NEI Ex. EAL #	NEI Example EAL Wording	CCNPP EAL #	CCNPP EAL Wording	Difference/Deviation Justification
1	Loss or Potential Loss of ANY Two Barriers	FS1.1	Loss or potential loss of **ANY** two barriers (Table F-1)	Table F-1 contains the loss and potential loss thresholds for the three fission product barriers and is the plant representation of NEI Table 5-F-3.

NEI IC#	NEI IC Wording	CCNPP IC#(s)	CCNPP IC Wording	Difference/Deviation Justification
FG1	Loss of ANY Two Barriers AND Loss or Potential Loss of Third Barrier MODE: Power Operation, Hot Standby, Startup, Hot Shutdown	FG1	Loss of **ANY** two barriers and loss or potential loss of the third barrier MODE: 1 - Power Operation, 2 - Startup, 3 - Hot Standby, 4 - Hot Shutdown	None

NEI Ex. EAL #	NEI Example EAL Wording	CCNPP EAL #	CCNPP EAL Wording	Difference/Deviation Justification
1	Loss of ANY Two Barriers AND Loss or Potential Loss of Third Barrier	FG1.1	Loss of **ANY** two barriers **AND** Loss or potential loss of the third barrier (Table F-1)	Table F-1 contains the loss and potential loss thresholds for the three fission product barriers and is the plant representation of NEI Table 5-F-3.

NEI Ex. EAL #	NEI Table 5-F-1 Notes	CCNPP EAL #	CCNPP EAL Notes	Difference/Deviation Justification
N/A	**NOTES** The logic used for these initiating conditions reflects the following considerations: • The Fuel Clad Barrier and the RCS Barrier are weighted more heavily than the Containment Barrier (See Sections 3.4 and 3.8). NOUE ICs associated with RCS and Fuel Clad Barriers are addressed under System Malfunction ICs. • At the Site Area Emergency level, there must be some ability to dynamically assess how far present conditions are from the threshold for a General Emergency. For example, if Fuel Clad and RCS Barrier "Loss" EALs existed, that, in addition to off-site dose assessments, would require continual assessments of radioactive inventory and containment integrity. Alternatively, if both Fuel Clad and RCS Barrier "Potential Loss" EALs existed, the Emergency Director would have more assurance that there was no immediate need	FU1.1 FA1.1 FS1.1 FG1.1	The logic used for Category F EALs reflects the following considerations: • The Fuel Clad Barrier and the RCS Barrier are weighted more heavily than the Containment Barrier. UE EALs associated with RCS and Fuel Clad Barriers are addressed under Category S. • At the Site Area Emergency level, there must be some ability to dynamically assess how far present conditions are from the threshold for a General Emergency. For example, if Fuel Clad and RCS Barrier "Loss" thresholds existed, that, in addition to off-site dose assessments, would require continual assessments of radioactive inventory and Containment integrity. Alternatively, if both Fuel Clad and RCS Barrier "Potential Loss" thresholds existed, the ED would have more assurance that there was no immediate need to escalate to a General	First bullet: The NEI parenthetical phrase "See Sections 3.4 and 3.8" has been deleted because it refers to NEI EAL developmental information. First bullet: The NEI acronym "NOUE" has been implemented as "UE" for simplification. The NEI abbreviation "ICs" has been changed to "EALs" for clarification. Second bullet: The NEI abbreviation "EALs" has been changed to "thresholds" for clarification. The second sentence in the fourth bullet of the NEI notes "When no event is in progress (Loss or Potential Loss of either Fuel Clad and/or RCS) the Containment Barrier status is addressed by Technical Specifications" has been deleted to implement EAL FAQ #14.

	to escalate to a General Emergency. • The ability to escalate to higher emergency classification levels as an event deteriorates must be maintained. For example, RCS leakage steadily increasing would represent an increasing risk to public health and safety. • The Containment Barrier should not be declared lost or potentially lost based on exceeding Technical Specification action statement criteria, unless there is an event in progress requiring mitigation by the Containment barrier. When no event is in progress (Loss or Potential Loss of either Fuel Clad and/or RCS) the Containment Barrier status is addressed by Technical Specifications.		Emergency. • The ability to escalate to higher emergency classification levels as an event deteriorates must be maintained. For example, RCS leakage steadily increasing would represent an increasing risk to public health and safety. • The Containment Barrier should **not** be declared lost or potentially lost based on exceeding Technical Specification action statement criteria, unless there is an event in progress requiring mitigation by the Containment barrier.	

Table F-1 Fission Product Barrier Matrix

	Fuel Clad Barrier		Reactor Coolant System Barrier		Containment Barrier	
Category	**Loss**	**Potential Loss**	**Loss**	**Potential Loss**	**Loss**	**Potential Loss**
A **Core Cooling / Heat Removal**	1. CET readings > 1,200°F	1. CET readings > 700ºF 2. RCS heat removal **cannot** be established **AND EITHER**: RCS pressure > PORV setpoint **OR** RCS subcooling < 25°F	None	1. OTCC flow established 2. RCS heat removal **cannot** be established **AND EITHER**: RCS pressure > PORV setpoint **OR** RCS subcooling < 25°F 3. Uncontrolled RCS cooldown and to left of Max Operating Pressure Curve (EOP Attachment 1, RCS Pressure Temperature Limits)	None	1. CET readings **cannot** be restored < 1,200°F within 15 min. 2. CET readings > 700ºF **AND** Reactor vessel water level **cannot** be restored > RVLMS 10 in. alarm within 15 min.
B **Inventory**	None	3. RVLMS level ≤ 10 in. alarm	1. RCS leak rate > available makeup capacity as indicated by a loss of RCS subcooling (< 25°F) 2. RUPTURED S/G results in an ECCS (SIAS) actuation	4. RCS leak rate > 50 gpm with letdown isolated	1. A Containment pressure rise followed by a rapid unexplained drop in Containment pressure 2. Containment pressure or sump level response **not** consistent with LOCA conditions 3. RUPTURED S/G is also FAULTED outside of Containment 4. Primary-to-secondary leakrate > 10 gpm **AND** Unisolable prolonged steam release from affected S/G to the environment	3. Containment pressure ≥ 50 psig and rising 4. Containment hydrogen concentration ≥ 4% 5. Containment pressure > 4.25 psig **AND cannot** meet **ANY** of the following conditions: • 2 Containment Spray Pumps Operating • 3 CACs Operating • 1 Containment Spray Pump and 2 CACs Operating
C **Radiation / Coolant Activity**	2. Containment radiation monitor (5317A/B) reading > 3,500 R/hr 3. Post-accident sample dose rate ≥ 40 mRem/hr (1 ft from sample) 4. Coolant activity >300 µCi/cc DEQ I-131	None	3. Containment radiation monitor (5317A/B) reading > 6 R/hr (Note 8)	None	None	6. Containment radiation monitor (5317A/B) reading > 14,000 R/hr
D **Isolation Status**	None	None	None	None	5. Failure of **all** valves in **ANY one** line to close **AND** Direct downstream pathway to the environment exists after Containment isolation signal	None
E **Judgment**	5. **ANY** condition in the opinion of the Emergency Director that indicates loss of the fuel clad barrier	4. **ANY** condition in the opinion of the Emergency Director that indicates potential loss of the fuel clad barrier	4. **ANY** condition in the opinion of the Emergency Director that indicates loss of the RCS barrier	5. **ANY** condition in the opinion of the Emergency Director that indicates potential loss of the RCS barrier	6. **ANY** condition in the opinion of the Emergency Director that indicates loss of the Containment barrier	7. **ANY** condition in the opinion of the Emergency Director that indicates potential loss of the Containment barrier

Fuel Clad Fission Product Barrier Degradation Thresholds

NEI FPB#	NEI Threshold Wording	CCNPP FPB #(s)	CCNPP FPB Wording	Difference/Deviation Justification
FC Loss 1	**Critical Safety Function Status** A. Core-Cooling Red Entry Conditions Met.	N/A	N/A	CCNPP is a Combustion Engineering (CE) designed PWR. CCNPP does not implement Westinghouse Owners Group (WOG) Critical Safety Function Status Trees and therefore this threshold is not applicable to CCNPP.
FC Loss 2	**Primary Coolant Activity Level** A. Coolant activity greater than (site specific value).	FC Loss C.4	Coolant activity > 300 µCi/cc DEQ I-131	300 µCi/gm dose equivalent I-131 is the site-specific value for coolant activity. The NEI phrase "greater than" has been replaced with ">" to reduce EAL-user reading burden. The symbol ">" means "greater than" and thus implements the intent of the NEI phrase.
FC Loss 3	**Core Exit Thermocouple Readings** A. Core exit thermocouples reading greater than (site specific degree F).	FC Loss A.1	CET readings > 1,200°F	CETs is the CCNPP equivalent of NEI "thermocouple readings." The NEI phrase "greater than" has been replaced with ">" to reduce EAL-user reading burden. The symbol ">" means "greater than" and thus implements the intent of the NEI phrase. The NEI word "degree" has been replaced with "°" to reduce EAL-user reading burden. The symbol "°" is commonly understood to mean "degree." 1,200°F is the CCNPP specific temperature corresponding to significant core exit superheating and core uncovery.
FC Loss 4	**Reactor Vessel Water Level** Not Applicable	N/A	N/A	N/A
FC Loss 5	**Not Applicable** Not Applicable	N/A	N/A	N/A

NEI FPB#	NEI Threshold Wording	CCNPP FPB #(s)	CCNPP FPB Wording	Difference/Deviation Justification
FC Loss 6	**Containment Radiation Monitoring** A. Containment radiation monitor reading greater than (site specific value).	FC Loss C.2	Containment radiation monitor (5317A/B) reading > 3,500 R/hr	The NEI phrase "greater than" has been replaced with ">" to reduce EAL-user reading burden. The symbol ">" means "greater than" and thus implements the intent of the NEI phrase. 3,500 R/hr is the site-specific Containment rad monitor reading.
FC Loss 7	**Other (Site-Specific) Indications** A. (Site-specific) as applicable	FC Loss C.3	Post-accident sample dose rate ≥ 40 mRem/hr (1 ft from sample)	A shielded 12.5 ml pressurized bomb sample would read 40 mRem/hr at one foot from the sample (168 mRem/hr unshielded) for 5% fuel cladding damage. When reactor coolant activity reaches this level, significant clad heating has occurred and thus the Fuel Cladding barrier is considered lost per BG&E Fuel Degradation EALs Calculation Worksheet, JSB Associates, February 18, 1993.
FC Loss 8	**Emergency Director Judgment** A. Any condition in the opinion of the Emergency Director that indicates Loss of the Fuel Clad Barrier.	FC Loss E.5	**ANY** condition in the opinion of the Emergency Director that indicates loss of the Fuel Clad barrier	None
FC P-Loss 1	**Critical Safety Function Status** A. Core Cooling-Orange Entry Conditions Met. **OR** B. Heat Sink-Red Entry Conditions Met.	N/A	N/A	CCNPP is a Combustion Engineering (CE) designed PWR. CCNPP does not implement Westinghouse Owners Group (WOG) Critical Safety Function Status Trees and therefore this threshold is not applicable to CCNPP.
FC P-Loss 2	**Primary Coolant Activity Level** Not Applicable	N/A	N/A	N/A

NEI FPB#	NEI Threshold Wording	CCNPP FPB #(s)	CCNPP FPB Wording	Difference/Deviation Justification
FC P-Loss 3	**Core Exit Thermocouple Readings** A.. Core exit thermocouples reading greater than (site specific degree F).	FC P-Loss A.1	CET readings > 700ºF	CET is the CCNPP equivalent of NEI "thermocouple readings." The NEI phrase "greater than" has been replaced with ">" to reduce EAL-user reading burden. The symbol ">" means "greater than" and thus implements the intent of the NEI phrase. The NEI word "degree" has been replaced with "º" to reduce EAL-user reading burden. The symbol "º" is commonly understood to mean "degree." Core Exit Thermocouples (CETs) are a component of the Inadequate Core Cooling Instrumentation and provide an indirect indication of fuel cladding temperature by measuring the temperature of the reactor coolant that leaves the core region. The RCS Pressure Safety Limit is 2750 psia per CCNPP Technical Specifications. The saturation temperature for this pressure is 682.2°F. Per Action Value Bases Document EOP-24.33, the uncertainty on CET Temperature is +/- 39.8°F. If one or more CETs indicate 722°F (682.2 + 39.8), subcooling has been lost for at least some locations in the core. CET indications at or above 722°F are a clear sign that core heat removal capability is lost or greatly reduced and one fission product barrier, the fuel clad, is threatened due to elevated fuel temperatures. 700°F qualifies as a condition representing a potential loss of the fuel clad barrier.

NEI FPB#	NEI Threshold Wording	CCNPP FPB #(s)	CCNPP FPB Wording	Difference/Deviation Justification
FC P-Loss 4	**Reactor Vessel Water Level** A. RCS/RPV level less than (site specific level for TOAF).	FC P-Loss B.3	RVLMS < 10 in. alarm	The NEI phrase "RCS/RPV" has been replaced with "RVLMS" to use terminology consistent with the CCNPP EOPs. The NEI phrase "less than" has been replaced with "<" to reduce EAL-user reading burden. The symbol "<" means "less than" and thus implements the intent of the NEI phrase. The Reactor Vessel Level Monitoring System (RVLMS) is based on the CE Heated Junction Thermocouple (HJTC) system. The HJTC system measures reactor coolant liquid inventory with discrete HJTC sensors located at different levels within a separator tube ranging from the fuel alignment plate (i.e., near top of active fuel) to the top of the Reactor Vessel head. The basic principle of system operation is detection of a temperature difference between heated and unheated thermocouples. When Reactor Vessel/RCS water level drops to 32.9 ft el., core uncovery is about to occur. The closest RVLMS indication is the 10 in. alarm. This signals inadequate coolant inventory, loss of subcooling and the occurrence of possible fuel cladding damage.
FC P-Loss 5	**Not Applicable** Not Applicable	N/A	N/A	N/A
FC P-Loss 6	**Containment Radiation Monitoring** Not Applicable	N/A	N/A	N/A

NEI FPB#	NEI Threshold Wording	CCNPP FPB #(s)	CCNPP FPB Wording	Difference/Deviation Justification
FC P-Loss 7	**Other (Site-Specific) Indications** A. (Site-specific) as applicable	FC P-Loss A.2	RCS heat removal **cannot** be established **AND EITHER**: RCS pressure > PORV setpoint **OR** RCS subcooling < 25°F	If RCS pressure approaches the PORV setpoint (2400 psia), heat input to the RCS is likely raising pressure instead of reaching the ultimate heat sink. If RCS subcooling approaches 25°F, the margin to superheated conditions is being reduced. Following an uncomplicated reactor trip, subcooling margin should be in excess of 50°F. Subcooling margin greater than 25°F ensures the fluid surrounding the core is sufficiently cooled and provides margin for reestablishing flow should subcooling deteriorate when SI flow is secured. Voids may exist in some parts of the RCS (e.g., Reactor Vessel head) but are permissible as long as core heat removal is maintained. The combination of these conditions indicates the ultimate heat sink function is under extreme challenge. This threshold addresses loss of functions required for hot shutdown with the reactor at pressure and temperature and thus a potential loss of the Fuel Cladding barrier.
FC P-Loss 8	**Emergency Director Judgment** A. Any condition in the opinion of the Emergency Director that indicates Potential Loss of the Fuel Clad Barrier.	FC P-Loss E.4	**ANY** condition in the opinion of the Emergency Director that indicates potential loss of the Fuel Clad barrier	None

RCS Fission Product Barrier Degradation Thresholds

NEI FPB#	NEI IC Wording	CCNPP FPB #(s)	CCNPP FPB Wording	Difference/Deviation Justification
RCS Loss 1	**Critical Safety Function Status** Not Applicable	N/A	N/A	None
RCS Loss 2	**RCS Leak Rate** A. RCS leak rate greater than available makeup capacity as indicated by a loss of RCS subcooling.	RCS Loss B.1	RCS leak rate > available makeup capacity as indicated by a loss of RCS subcooling (< 25°F)	The NEI phrase "greater than" has been replaced with ">" to reduce EAL-user reading burden. The symbol ">" means "greater than" and thus implements the intent of the NEI phrase. AOP-2A, Excessive Reactor Coolant Leakage, provides a list of conditions that may be observed when excessive RCS leakage occurs and provides appropriate actions to prevent and mitigate the consequences of RCS leakage. Following an uncomplicated reactor trip, subcooling margin should be in the range of 50°F to 75°F. Subcooling margin greater than or equal to 25°F ensures the fluid surrounding the core is sufficiently cooled and provides margin for reestablishing flow should subcooling deteriorate when SIS flow is secured. Voids may exist in some parts of the RCS (e.g., Reactor Vessel head) but are permissible as long as core heat removal is maintained. The loss of subcooling is therefore the fundamental indication that the inventory control systems are incapable of counteracting the mass loss through the leak in the RCS.
RCS Loss 3	**Not Applicable** Not Applicable	N/A	N/A	None
RCS Loss 4	**SG Tube Rupture** A. RUPTURED SG results in an ECCS (SI) actuation.	RCS Loss B.2	RUPTURED S/G results in an ECCS (SIAS) actuation	SIAS is the site specific name for an SI actuation signal.
RCS Loss 5	**Not Applicable** Not Applicable	N/A	N/A	None

NEI FPB#	NEI IC Wording	CCNPP FPB #(s)	CCNPP FPB Wording	Difference/Deviation Justification
RCS Loss 6	**Containment Radiation Monitoring** A. Containment radiation monitor reading greater than (site specific value).	RCS Loss C.3	Containment radiation monitor (5317A/B) reading > 6 R/hr (Note 8) Note 8: High temperature in Containment may induce a current error in the Mineral Insulated (MI) cable running through Containment to the meter. The CHRRM 1(2)-RI-5317 A&B may **not** detect this value (6 R/hr) under these conditions. When Containment temperature reaches 300°F, the meter will indicate approximately 40 R/hr for a few minutes then drop to approximately 10 R/hr after three hours. This information is to provide guidance on determining the validity of the readings under the specified high temperature conditions.	The specified reading is based assuming the instantaneous release and dispersal of the reactor coolant noble gas and iodine inventory associated with normal operating concentrations (i.e., within Technical Specifications) into the Containment atmosphere. Because of the very high fuel cladding integrity, only small amounts of noble gases would be dissolved in the reactor coolant. Only leakage from the RCS is assumed for this barrier loss threshold. The Containment radiation monitors alarm at 6 R/hr and so is operationally significant. The NEI phrase "greater than" has been replaced with ">" to reduce EAL-user reading burden. The symbol ">" means "greater than" and thus implements the intent of the NEI phrase. Note 8 has been added to provide guidance on potential effects on Containment radiation indications due to high Containment temperatures.
RCS Loss 7	**Other (Site-Specific) Indications** A. (Site-specific) as applicable	N/A	N/A	Other site-specific indications of RCS loss have not been identified.
RCS Loss 8	**Emergency Director Judgment** A. Any condition in the opinion of the Emergency Director that indicates Loss of the RCS Barrier.	RCS Loss E.4	**ANY** condition in the opinion of the Emergency Director that indicates loss of the RCS barrier	None

NEI FPB#	NEI IC Wording	CCNPP FPB #(s)	CCNPP FPB Wording	Difference/Deviation Justification
RCS P-Loss 1	**Critical Safety Function Status** A. RCS Integrity-Red Entry Conditions Met. **OR** B. Heat Sink-Red Entry Conditions Met.	N/A	N/A	CCNPP is a Combustion Engineering (CE) designed PWR. CCNPP does not implement Westinghouse Owners Group (WOG) Critical Safety Function Status Trees and therefore this threshold is not applicable to CCNPP.
RCS P-Loss 2	**RCS Leak Rate** A. RCS leak rate indicated greater than (site specific capacity of one charging pump in the normal charging mode) with Letdown isolated.	RCS P-Loss B.4	RCS leak rate > 50 gpm with letdown isolated	The CVCS includes three positive displacement horizontal pumps with a capacity of 44 gpm each. The single charging pump capacity is rounded up to 50 gpm for this threshold and clearly signals that operation of more than one charging pump is needed.
RCS P-Loss 3	**Not Applicable** Not Applicable	N/A	N/A	N/A
RCS P-Loss 4	**SG Tube Rupture** Not Applicable	N/A	N/A	N/A
RCS P-Loss 5	**Not Applicable** Not Applicable	N/A	N/A	N/A
RCS P-Loss 6	**Containment Radiation Monitoring** Not Applicable	N/A	N/A	N/A
RCS P-Loss 7	**Other (Site-Specific) Indications** A. (Site-specific) as applicable	RCS P-Loss A.1	OTCC flow established	CCNPP is a CE plant with Once Through Core Cooling (OTCC) capability and has a procedure that intentionally opens the RCS barrier to cool the core when normal means fail. This procedure is employed when the heat removal function is extremely challenged. Establishment of OTCC flow represents a potential loss of the RCS barrier due to PORVs being intentionally maintained open to establish adequate core heat removal capability.

NEI FPB#	NEI IC Wording	CCNPP FPB #(s)	CCNPP FPB Wording	Difference/Deviation Justification
		RCS P-Loss A.2	RCS heat removal **cannot** be established **AND EITHER**: RCS pressure > PORV setpoint **OR** RCS subcooling < 25°F	If RCS pressure approaches the PORV setpoint (2400 psia), heat input to the RCS is likely raising pressure instead of reaching the ultimate heat sink. If RCS subcooling approaches 25°F, the margin to superheated conditions is being reduced. Following an uncomplicated reactor trip, subcooling margin should be in excess of 50°F. Subcooling margin greater than 25°F ensures the fluid surrounding the core is sufficiently cooled and provides margin for reestablishing flow should subcooling deteriorate when SI flow is secured. Voids may exist in some parts of the RCS (e.g., Reactor Vessel head) but are permissible as long as core heat removal is maintained. The combination of these conditions indicates the ultimate heat sink function is under extreme challenge. This threshold addresses loss of functions required for hot shutdown with the reactor at pressure and temperature and thus a potential loss of the RCS barrier.
		RCS P-Loss A.3	Uncontrolled RCS cooldown and to left of Max Operating Pressure Curve (EOP Attachment 1, RCS Pressure Temperature Limits)	Among the EOP safety functions to be maintained is RCS Pressure Control. Per EOP-4, Excess Steam Demand Event, the potential exists for pressurized thermal shock from an excessive cooldown rate followed by a repressurization. The Max Operating Pressure Curve and RCS cooldown rate limits are established to prevent the effects of pressurized thermal shock. The region to the left of the curve is labeled the "Non-Operating Area." Several curves are included in EOP Attachment 1 based on the combinations of Reactor Coolant Pumps (RCPs) in operation. The combination of the conditions of this potential loss threshold indicates the RCS barrier is under significant challenge.
RCS P-Loss 8	**Emergency Director Judgment** A. Any condition in the opinion of the Emergency Director that indicates Potential Loss of the RCS Barrier.	RCS P-Loss E.5	**ANY** condition in the opinion of the Emergency Director that indicates potential loss of the RCS barrier	None

Containment Fission Product Barrier Degradation Thresholds

NEI FPB#	NEI IC Wording	CCNPP FPB #(s)	CCNPP FPB Wording	Difference/Deviation Justification
CNMT Loss 1	**Critical Safety Function Status** Not Applicable	N/A	N/A	None
CNMT Loss 2	**Containment Pressure** A. A containment pressure rise followed by a rapid unexplained drop in containment pressure. **OR** B. Containment pressure or sump level response not consistent with LOCA conditions.	CNMT Loss B.1	A Containment pressure rise followed by a rapid unexplained drop in Containment pressure	The NEI threshold has been divided into two CCNPP thresholds to improve clarity.
		CNMT Loss B.2	Containment pressure or sump level response **not** consistent with LOCA conditions	The NEI threshold has been divided into two CCNPP thresholds to improve clarity.
CNMT Loss 3	**Core Exit Thermocouple Readings** Not applicable	N/A	N/A	N/A
CNMT Loss 4	**SG Secondary Side Release with P-to-S Leakage** A. RUPTURED SG is also FAULTED	CNMT Loss B.3	RUPTURED S/G is also FAULTED outside of Containment	None

NEI FPB#	NEI IC Wording	CCNPP FPB #(s)	CCNPP FPB Wording	Difference/Deviation Justification
	outside of containment. **OR** B. a. Primary-to-Secondary leakrate greater than 10 gpm. **AND** b. UNISOLABLE steam release from affected SG to the environment.	CNMT Loss B.4	Primary-to-secondary leakrate > 10 gpm **AND** UNISOLABLE or prolonged steam release from affected S/G to the environment	The NEI threshold has been divided into two CCNPP thresholds to improve clarity. The NEI phrase "greater than" has been replaced with ">" to reduce EAL-user reading burden. The symbol ">" means "greater than" and thus implements the intent of the NEI phrase. "Prolonged" as used here is in the context meaning that the release from the affected S/G within the time frame expected when implementing EOP-6 Steam Generator Tube Rupture or EOP-8 Functional Recovery Procedure. Cooldowns conducted to allow controlled isolation of the affected S/G per emergency procedures are not considered prolonged releases. The criterion for prolonged release is met if the objective of EOP-6 or EOP-8 to isolate the affected S/G cannot be met.
CNMT Loss 5	**Containment Isolation Failure or Bypass** A. a. Failure of all valves in any one line to close. **AND** b. Direct downstream pathway to the environment exists after containment isolation signal.	CNMT Loss D.5	Failure of **all** valves in **ANY one** line to close **AND** Direct downstream pathway to the environment exists after Containment isolation signal	None
CNMT Loss 6	**Containment Radiation Monitoring** Not Applicable	N/A	N/A	N/A
CNMT Loss 7	**Other (Site-Specific) Indications** A. (Site-specific) as applicable	N/A	N/A	Other site-specific indications of Containment loss have not been identified.

NEI FPB#	NEI IC Wording	CCNPP FPB #(s)	CCNPP FPB Wording	Difference/Deviation Justification
CNMT Loss 8	**Emergency Director Judgment** A. Any condition in the opinion of the Emergency Director that indicates Loss of the Containment Barrier.	CNMT Loss E.6	**ANY** condition in the opinion of the Emergency Director that indicates loss of the Containment barrier	None
CNMT P-Loss 1	**Critical Safety Function Status** A. Containment-Red Entry Conditions Met.	N/A	N/A	CCNPP is a Combustion Engineering (CE) designed PWR. CCNPP does not implement Westinghouse Owners Group (WOG) Critical Safety Function Status Trees and therefore this threshold is not applicable to CCNPP.
CNMT P-Loss 2	**Containment Pressure** A. Containment pressure greater than (site specific value) and rising. **OR** B. Explosive mixture exists inside containment. **OR** C. a. Pressure greater than containment depressurization actuation setpoint. **AND** b. Less than one full train of	CNMT P-Loss B.3	Containment pressure ≥ 50 psig and rising	The NEI threshold has been divided into three plant thresholds to improve clarity. This threshold is the Containment design pressure and is in excess of that expected from the design basis loss of coolant accident (LOCA). The pressure-time responses for the spectrum of LOCAs considered in the plant design basis are described in Section 14 of the UFSAR.
		CNMT P-Loss B.4	Containment hydrogen concentration ≥ 4%	The NEI threshold has been divided into three CCNPP thresholds to improve clarity. Containment hydrogen concentration of 4% is the minimum concentration associated with an explosive mixture.

NEI FPB#	NEI IC Wording	CCNPP FPB #(s)	CCNPP FPB Wording	Difference/Deviation Justification
	depressurization equipment operating.	CNMT P-Loss B.5	Containment pressure > 4.25 psig **AND** **cannot** meet **ANY** of the following conditions: • 2 Containment Spray Pumps Operating • 3 CACs Operating • 1 Containment Spray Pump and 2 CACs Operating	The NEI threshold has been divided into three CCNPP thresholds to improve clarity. The word "Containment" has been added to the plant threshold for clarification. The Containment pressure setpoint (4.25 psig) is the Containment depressurization actuation setpoint. The NEI phrase "greater than" has been replaced with ">" to reduce EAL-user reading burden. The symbol ">" means "greater than" and thus implements the intent of the NEI phrase. The phrase "Less than one full train of depressurization equipment operating" has been replaced with the site specific minimum required depressurization equipment per FSAR accident analysis.
CNMT P-Loss 3	**Core Exit Thermocouple Readings** A. a. Core exit thermocouples in excess of (site specific) ° F. AND b. Restoration procedures not effective within 15 minutes. OR B. a Core exit thermocouples in excess of (site-specific) F. AND b. Reactor vessel level below (site specific level).	CNMT P-Loss A.1	CET readings **cannot** be restored < 1,200°F within 15 min.	The NEI threshold has been divided into two CCNPP thresholds to improve clarity. "CET" is the CCNPP equivalent of NEI "Core exit thermocouples." The NEI phrase "in excess of (site specific) ° F AND Restoration procedures not effective..." has been replaced with " **cannot** be restored < 1,200°F...". The phrase "**cannot** be restored <" infers CET readings have exceeded the threshold temperature and procedural guidance used to restore RCS inventory has been attempted but is thus far unsuccessful. Whether or not guidance is effective should be apparent within fifteen minutes.

NEI FPB#	NEI IC Wording	CCNPP FPB #(s)	CCNPP FPB Wording	Difference/Deviation Justification
	AND c. Restoration procedures not effective within 15 minutes.	CNMT P-Loss A.2	CET readings > 700°F **AND** Reactor vessel water level **cannot** be restored > RVLMS 10 in. alarm within 15 min.	The NEI threshold has been divided into two CCNPP thresholds to improve clarity. "CET" is the CCNPP equivalent of NEI "Core exit thermocouples." The NEI phrase "in excess of (site specific) ° F AND reactor vessel level below...AND Restoration procedures not effective..." has been replaced with "CET readings indicate superheat **AND** Reactor vessel water level **cannot** be restored > RVLMS 10 in. alarm..." Core Exit Thermocouples (CETs) are a component of the Inadequate Core Cooling Instrumentation and provide an indirect indication of fuel cladding temperature by measuring the temperature of the reactor coolant that leaves the core region. The RCS Pressure Safety Limit is 2750 psia per CCNPP Technical Specifications. The saturation temperature for this pressure is 682.2°F. Per Action Value Bases Document EOP-24.33, the uncertainty on CET Temperature is +/- 39.8°F. If one or more CETs indicate 722°F (682.2 + 39.8), subcooling has been lost for at least some locations in the core. CET indications at or above 722°F are a clear sign that core heat removal capability is lost or greatly reduced and one fission product barrier, the fuel clad, is threatened due to elevated fuel temperatures. 700°F qualifies as a condition representing a potential loss of the fuel clad barrier. The phrase "**cannot** be restored >" infers RVLMS level readings have exceeded the threshold level and procedural guidance used to restore RCS inventory has been attempted but is thus far unsuccessful. Whether or not guidance is effective should be apparent within fifteen minutes.

NEI FPB#	NEI IC Wording	CCNPP FPB #(s)	CCNPP FPB Wording	Difference/Deviation Justification
CNMT P-Loss 4	**SG Secondary Side Release with P-to-S Leakage** Not applicable	N/A	N/A	N/A
CNMT P-Loss 5	**Containment Isolation Failure or Bypass** Not Applicable	N/A	N/A	N/A
CNMT P-Loss 6	**Containment Radiation Monitoring** A. Containment radiation monitor reading greater than (site specific value).	CNMT P-Loss C.6	Containment radiation monitor (5317A/B) reading > 14,000 R/hr	The NEI phrase "greater than" has been replaced with ">" to reduce EAL-user reading burden. The symbol ">" means "greater than" and thus implements the intent of the NEI phrase. 14,000 R/hr is the site-specific Containment rad monitor reading corresponding to ~20% clad damage per ERPIP-801 Core Damage Assessment Using Containment Radiation Dose Rates.
CNMT P-Loss 7	**Other (Site-Specific) Indications** A. (Site-specific) as applicable	N/A	N/A	Other site-specific indications of Containment potential loss have not been identified.
CNMT P-Loss 8	**Emergency Director Judgment** A. Any condition in the opinion of the Emergency Director that indicates Potential Loss of the Containment Barrier.	CNMT P-Loss E.7	**ANY** condition in the opinion of the Emergency Director that indicates potential loss of the Containment barrier	None

Category H

Hazards and Other Conditions Affecting Plant Safety

NEI IC#	NEI IC Wording	CCNPP IC#(s)	CCNPP IC Wording	Difference/Deviation Justification
HU1	Natural or destructive phenomena affecting the PROTECTED AREA. MODE: All	HU1	Natural or destructive phenomena affecting the Protected Area MODE: All	None

NEI Ex. EAL #	NEI Example EAL Wording	CCNPP EAL #	CCNPP EAL Wording	Difference/Deviation Justification
1	Seismic event identified by **ANY** 2 of the following: • Seismic event confirmed by (site specific indication or method) • Earthquake felt in plant • National Earthquake Center	HU1.1	Seismic event identified by **ANY two** of the following: • Seismic Acceleration Recorder (0-YRC-001) Event Indicator alarms indicate seismic event detected • Earthquake felt in plant • National Earthquake Information Center (Note 7) Note 7: 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 Calvert Cliffs Nuclear Power Plant. Provide the analyst with the following CCNPP coordinates: **38° 25' 39.7" north latitude, 76° 26' 45" west longitude**.	Actuation of the CCNPP Seismic Acceleration Recorder (0-YRC-001) Event Indicator provides the site specific indication or method of detecting a seismic event. Note 7 provides guidance for contacting the NEIC and verifying seismic activity near the CCNPP site.
2	Tornado striking within PROTECTED AREA boundary or high winds greater than (site specific mph).	HU1.2	Tornado striking within PROTECTED AREA boundary **OR** Sustained high winds > 45 m/sec	The NEI phrase "greater than" has been replaced with the symbol ">" to reduce EAL-user reading burden. The symbol means "greater than" and thus implements the intent of the NEI phrase. The wind speed of 45 m/sec (100 mph) is the sustained design wind speed for Class 1 safe shutdown structures 30 ft above

			(100 mph)	ground and incorporates a gust factor of 1.1.
3	Internal flooding that has the potential to affect safety related equipment required by Technical Specifications for the current operating mode in **ANY** of the following areas: (site specific area list)	HU1.3	Internal flooding that has the potential to affect **ANY** SAFETY-RELATED STRUCTURE, SYSTEM, OR COMPONENT required by Technical Specifications for the current operating mode in **ANY** Table H-1 area	The NEI phrase "safety related equipment" has been changed to "**ANY** SAFETY-RELATED STRUCTURE, SYSTEM, OR COMPONENT ..." for clarification CCNPP areas containing safety related equipment are specified in Table H-1. This change implements EAL FAQ #44.
4	Turbine failure resulting in casing penetration or damage to turbine or generator seals.	HU1.4	Turbine failure resulting in casing penetration or damage to turbine or generator seals	None
5	(Site specific occurrences affecting the PROTECTED AREA).	HU1.5	Bay water level ≥ bottom of the traveling screen cover housing (+ 120 in. Mean Sea Level) **OR** Bay water level < 13.6 ft below intake concrete level (- 43.2 in. Mean Sea Level)	+120 in. (12 ft) MSL (approximately bottom of the travelling screen cover) is the still water level used for the Intake Structural Analysis. This value was selected to be anticipatory to the design level of 18 ft MSL (top of the travelling screen cover). The predicted extreme low tide elevation is -43.2 in. (-3.6 ft) MSL. However, the plant has been designed for -4.0 ft MSL and can continue to operate with an extreme low water Elevation of -6.0 ft MSL. The top of the saltwater pump intakes is at -9.5 ft MSL.

NEI IC#	NEI IC Wording	CCNPP IC#(s)	CCNPP IC Wording	Difference/Deviation Justification
HU2	FIRE within the PROTECTED AREA not extinguished within 15 minutes of detection or EXPLOSION within the PROTECTED AREA. MODE: All	HU2	Fire within the Protected Area **not** extinguished within 15 min. of detection or explosion within the Protected Area MODE: All	

NEI Ex. EAL #	NEI Example EAL Wording	CCNPP EAL #	CCNPP EAL Wording	Difference/Deviation Justification
1	FIRE not extinguished within 15 minutes of control room notification or verification of a control room FIRE alarm in **ANY** of the following areas: (site specific area list)	HU2.1	FIRE **not** extinguished within 15 min. of Control Room notification or verification of a Control Room fire alarm in the North Service Building, Turbine Building, Butler Building* or **ANY** Table H-1 area (Note 4) * Butler Building is only considered adjacent in Modes 5, 6 and D. Note 4: The ED should **not** wait until the applicable time has elapsed, but should declare the event as soon as it is determined that the condition has exceeded, or will likely exceed, the applicable time.	The NEI phrase "of the following areas...(site specific area list)" has been changed to "in the North Service Building, Turbine Building or **ANY** Table H-1 area." The areas listed in Table H-1 are areas containing functions and systems required for safe shutdown. This change implements EAL FAQ #44. The North Service Building, Turbine Building and Butler Building (only when in modes 4, 5 or D) are adjacent structures. Note 4 has been added consistent with other NEI based EALs that include the 15 min. transitory condition exclusion. The third paragraph of the NEI basis has been edited to clarify the significance of the 15-minute duration. If the alarm cannot be verified by redundant Control Room or nearby Fire Panel indications, notification from the field that a fire exists starts the concurrent 15-minute classification and fire suppression clocks. This change is consistent with the manner in which the Control Room and Fire Brigade leaders verify fires. This change is necessary to avoid declaring Unusual Event emergencies for spurious alarms that, due to the sensor location, cannot be verified within 15 minutes of receipt of the alarm. This is a deviation from NEI 99-01 Revision 5.
2	EXPLOSION within the	HU2.2	EXPLOSION within the	None

	PROTECTED AREA.		PROTECTED AREA	

Table H-1 Safe Shutdown Areas
• Control Room • Containment • Auxiliary Building • Diesel Generator Rooms • Intake Structure • 1A/0C DG Buildings • RWT • RWT Rooms • CST No. 12 • FOST No. 21 • Auxiliary Feed Pump Rooms

NEI IC#	NEI IC Wording	CCNPP IC#(s)	CCNPP IC Wording	Difference/Deviation Justification
HU3	Release of toxic, corrosive, asphyxiant, or flammable gases deemed detrimental to NORMAL PLANT OPERATIONS. MODE: All	HU3	Release of toxic, corrosive, asphyxiant or flammable gases deemed detrimental to normal plant operations MODE: All	None

NEI Ex. EAL #	NEI Example EAL Wording	CCNPP EAL #	CCNPP EAL Wording	Difference/Deviation Justification
1	Toxic, corrosive, asphyxiant or flammable gases in amounts that have or could adversely affect NORMAL PLANT OPERATIONS.	HU3.1	Toxic, corrosive, asphyxiant or flammable gases in amounts that have or could adversely affect NORMAL PLANT OPERATIONS	None
2	Report by local, county or state officials for evacuation or sheltering of site personnel based on an off-site event.	HU3.2	Recommendation by local, county or state officials to evacuate or shelter site personnel based on offsite event	Reworded EAL for readability.

NEI IC#	NEI IC Wording	CCNPP IC#(s)	CCNPP IC Wording	Difference/Deviation Justification
HU4	Confirmed SECURITY CONDITION or threat which indicates a potential degradation in the level of safety of the plant. MODE: All	HU4	Confirmed security condition or threat which indicates a potential degradation in the level of safety of the plant MODE: All	None

NEI Ex. EAL #	NEI Example EAL Wording	CCNPP EAL #	CCNPP EAL Wording	Difference/Deviation Justification
1	A SECURITY CONDITION that does NOT involve a HOSTILE ACTION as reported by the (site specific security shift supervision).	HU4.1	A SECURITY CONDITION that does **not** involve a HOSTILE ACTION as reported by Security Shift Supervisor **OR** A credible site-specific security threat notification **OR** A validated notification from NRC providing information of an aircraft threat	The NEI Example EALs have been combined in one plant EAL for simplification. "Security Shift Supervisor" is the site-specific security supervision that are qualified and trained to confirm that a security event is occurring or has occurred.
2	A credible site specific security threat notification.			
3	A validated notification from NRC providing information of an aircraft threat.			

NEI IC#	NEI IC Wording	CCNPP IC#(s)	CCNPP IC Wording	Difference/Deviation Justification
HU5	Other conditions exist which in the judgment of the Emergency Director warrant declaration of a NOUE. MODE: All	HU6	Other conditions existing that in the judgment of the Emergency Director warrant declaration of a UE MODE: All	The NEI acronym "NOUE" has been implemented as "UE" for simplification. Replaced the word "which" with "that" for proper grammar.

NEI Ex. EAL #	NEI Example EAL Wording	CCNPP EAL #	CCNPP EAL Wording	Difference/Deviation Justification
1	Other conditions exist which in the judgment of the Emergency Director indicate that events are in progress or have occurred which indicate a potential degradation of the level of safety of the plant or indicate a security threat to facility protection has been initiated. No releases of radioactive material requiring off-site response or monitoring are expected unless further degradation of safety systems occurs.	HU6.1	Other conditions exist which in the judgment of the Emergency Director indicate that events are in progress or have occurred which: Indicate a potential degradation of the level of safety of the plant **OR** Indicate a security threat to facility protection has been initiated **No** releases of radioactive material requiring offsite response or monitoring are expected unless further degradation of safety systems occurs	Reformatted for readability.

NEI IC#	NEI IC Wording	CCNPP IC#(s)	CCNPP IC Wording	Difference/Deviation Justification
HA1	Natural or destructive phenomena affecting VITAL AREAS MODE: All	HA1	Natural or destructive phenomena affecting Vital Areas MODE: All	None

NEI Ex. EAL #	NEI Example EAL Wording	CCNPP EAL #	CCNPP EAL Wording	Difference/Deviation Justification
1	a. Seismic event greater than Operating Basis Earthquake (OBE) as indicated by (site specific seismic instrumentation) reading (site specific OBE limit). **AND** b. Earthquake confirmed by **ANY** of the following: • Earthquake felt in plant • National Earthquake Center • Control Room indication of degraded performance of systems required for the safe shutdown of the plant.	HA1.1	**EITHER**: Seismic Acceleration Recorder (0-YRC-001) Event Indicator indicates seismic event > OBE (0.08 g horizontal, 0.053g vertical) **OR** Control Room indication of degraded performance of **ANY** SAFETY-RELATED STRUCTURE, SYSTEM, OR COMPONENT **AND** Earthquake confirmed by **EITHER**: Earthquake felt in plant **OR** National Earthquake Information Center (Note 7) Note 7: 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 Calvert Cliffs	The site-specific instrumentation used to indicate a seismic event > OBE cannot be analyzed in a timely manner. To allow for timely classification under this threshold, actual indication of degraded performance of safety-related structures systems or component has been included as a primary indicator of exceeding the OBE threshold. The site-specific instrumentation used to indicate a seismic event > OBE is the Seismic Acceleration Recorder (0-YRC-001) Event Indicator indicates seismic event > OBE (0.08 g horizontal, 0.053g vertical). The NEI phrase "greater than" has been replaced with the symbol ">" to reduce EAL-user reading burden. The symbol means "greater than" and thus implements the intent of the NEI phrase. Note 7 provides guidance for contacting the NEIC for confirmation of seismic activity in the vicinity of CCNPP. The NEI phrase "systems required for the safe shutdown of the plant" has been changed to "**ANY** SAFETY-RELATED STRUCTURE, SYSTEM, OR COMPONENT" to be consistent with the definition of visible damage and related HA1 EAL thresholds.

			Nuclear Power Plant. Provide the analyst with the following CCNPP coordinates: **38° 25' 39.7" north latitude, 76° 26' 45" west longitude**	
2	Tornado striking or high winds greater than (site specific mph) resulting in VISIBLE DAMAGE to **ANY** of the following structures containing safety systems or components **OR** control room indication of degraded performance of those safety systems: (site specific structure list)	HA1.2	Tornado striking or sustained high winds > 45 m/sec (100 mph) resulting in **EITHER**: VISIBLE DAMAGE to **ANY** SAFETY-RELATED STRUCTURE, SYSTEM, OR COMPONENT within **ANY** Table H-1 area **OR** Control Room indication of degraded performance of **ANY** SAFETY-RELATED STRUCTURE, SYSTEM, OR COMPONENT within **ANY** Table H-1 area	The NEI phrase "greater than" has been replaced with the symbol ">" to reduce EAL-user reading burden. The symbol means "greater than" and thus implements the intent of the NEI phrase. The wind speed of 45 m/sec (100 mph) is the sustained design wind speed for Class 1 safe shutdown structures 30 ft above ground and incorporates a gust factor of 1.1. The logic term "**EITHER**" has been added to the threshold so that the two indicated results of the tornado/high wind could be presented in list format. The NEI phrase "**ANY** of the following structures containing safety systems or components" has been changed to "**ANY** SAFETY-RELATED STRUCTURE, SYSTEM, OR COMPONENT within **ANY** Table H-1 area" to be consistent with the definition of visible damage and related HA1 EAL thresholds. This also permits presentation of the site specific list in a table. Table H-1 provides the list of structures containing safety systems or components. This change implements EAL FAQ #44. The NEI phrase "those safety systems" has been changed to "**ANY** SAFETY-RELATED STRUCTURE, SYSTEM, OR COMPONENT within **ANY** Table H-1 area" for clarification.
3	Internal flooding in **ANY** of the following areas resulting in an electrical shock hazard that precludes access to operate or monitor safety equipment OR control room indication of degraded performance of those safety systems: (site specific area list)	HA1.3	Internal flooding in **ANY** Table H-1 area resulting in **EITHER**: An electrical shock hazard that precludes access to operate or monitor **ANY** SAFETY-RELATED STRUCTURE, SYSTEM, OR COMPONENT within **ANY** Table H-1 area **OR** Control Room indication of	CCNPP areas containing safety related equipment are specified in Table H-1. This change implements EAL FAQ #44. The logic term "**EITHER**" has been added to the threshold so that the two indicated results of the flooding could be presented in list format. The NEI phrase "those safety systems" has been changed to "**ANY** SAFETY-RELATED STRUCTURE, SYSTEM, OR COMPONENT within **ANY** Table H-1 area" for clarification.

			degraded performance of **ANY** SAFETY-RELATED STRUCTURE, SYSTEM, OR COMPONENT within **ANY** Table H-1 area	
4	Turbine failure-generated PROJECTILES resulting in VISIBLE DAMAGE to or penetration of **ANY** of the following structures containing safety systems or components **OR** control room indication of degraded performance of those safety systems: (site specific structure list)	HA1.4	Turbine failure-generated PROJECTILES resulting in **EITHER**: VISIBLE DAMAGE to or penetration of **ANY** SAFETY-RELATED STRUCTURE, SYSTEM, OR COMPONENT within **ANY** Table H-1 area **OR** Control Room indication of degraded performance of **ANY** SAFETY-RELATED STRUCTURE, SYSTEM, OR COMPONENT within **ANY** Table H-1 area	The logic term "**EITHER**" has been added to the threshold so that the two indicated results of the flooding could be presented in list format. The NEI phrase "**ANY** of the following structures containing safety systems or components" has been changed to "**ANY** SAFETY-RELATED STRUCTURE, SYSTEM, OR COMPONENT within **ANY** Table H-1 area" to be consistent with the definition of visible damage and related HA1 EAL thresholds. This also permits presentation of the site specific list in a table. Table H-1 provides the list of areas/structures containing safety systems or components. This change implements EAL FAQ #44. The NEI phrase "those safety systems" has been changed to "**ANY** SAFETY-RELATED STRUCTURE, SYSTEM, OR COMPONENT within **ANY** Table H-1 area" for clarification.
5	Vehicle crash resulting in VISIBLE DAMAGE to **ANY** of the following structures containing safety systems or components **OR** control room indication of degraded performance of those safety systems: (site specific structure list)	HA1.6	Vehicle crash resulting in **EITHER**: VISIBLE DAMAGE to **ANY** SAFETY-RELATED STRUCTURE, SYSTEM, OR COMPONENT within **ANY** Table H-1 area **OR** Control Room indication of degraded performance of **ANY** SAFETY-RELATED STRUCTURE, SYSTEM, OR COMPONENT within **ANY**	The logic term "**EITHER**" has been added to the threshold so that the two indicated results of the flooding could be presented in list format. The NEI phrase "**ANY** of the following structures containing safety systems or components" has been changed to "**ANY** safety-related structure, system, or component within **ANY** Table H-1 area" to be consistent with the definition of visible damage and related HA1 EAL thresholds. This also permits presentation of the site specific list in a table. Table H-1 provides the list of areas/structures containing safety systems or components. This change implements EAL FAQ #44. The NEI phrase "those safety systems" has been changed to "**ANY** safety-related structure, system, or component within **ANY** Table

			Table H-1 area	H-1 area" for clarification.
6	(Site specific occurrences) resulting in VISIBLE DAMAGE to **ANY** of the following structures containing safety systems or components **OR** control room indication of degraded performance of those safety systems:	HA1.5	Bay water level ≥ top of the traveling screen cover housing **OR** Bay water level or inside traveling screen water level < 16.0 ft below intake concrete level (-72.0 in. Mean Sea Level)	18 ft (top of the traveling screen cover) is the design flood level. The predicted extreme low tide elevation is (-) 3.6 ft Mean Sea Level (MSL). However, the plant has been designed for (-) 4.0 ft MSL and can continue to operate with an extreme low water Elevation of -6.0 ft MSL. This EAL criterion is met if the water is 16 ft below the intake (-72 in. MSL) concrete level by observation. This measurement requires judgment because the Bay surface is not normally still. The top of the saltwater pump intakes is at -9.5 ft MSL.

Table H-1 Safe Shutdown Areas
• Control Room • Containment • Auxiliary Building • Diesel Generator Rooms • Intake Structure • 1A/0C DG Buildings • RWT • RWT Rooms • CST No. 12 • FOST No. 21 • Auxiliary Feed Pump Rooms

NEI IC#	NEI IC Wording	CCNPP IC#(s)	CCNPP IC Wording	Difference/Deviation Justification
HA2	FIRE or EXPLOSION affecting the operability of plant safety systems required to establish or maintain safe shutdown MODE: All	HA2	Fire or explosion affecting the operability of plant safety systems required to establish or maintain safe shutdown MODE: All	None

NEI Ex. EAL #	NEI Example EAL Wording	CCNPP EAL #	CCNPP EAL Wording	Difference/Deviation Justification
1	FIRE or EXPLOSION resulting in VISIBLE DAMAGE to **ANY** of the following structures containing safety systems or components **OR** control room indication of degraded performance of those safety systems: (site specific structure list)	HA2.1	FIRE or EXPLOSION resulting in **EITHER**: VISIBLE DAMAGE to **ANY** SAFETY-RELATED STRUCTURE, SYSTEM, OR COMPONENT within **ANY** Table H-1 area **OR** Control Room indication of degraded performance of **ANY** SAFETY-RELATED STRUCTURE, SYSTEM, OR COMPONENT within **ANY** Table H-1 area	The logic term "**EITHER**" has been added to the threshold so that the two indicated results of the fire/explosion could be presented in list format. The NEI phrase "**ANY** of the following structures containing safety systems or components" has been changed to "**ANY** safety-related structure, system, or component within **ANY** Table H-1 area" so that the site specific list could be presented in a table. Table H-1 provides the list of areas/structures containing safety systems or components. This change implements EAL FAQ #44. The NEI phrase "those safety systems" has been changed to "**ANY** safety-related structure, system, or component within **ANY** Table H-1 area" for clarification.

Table H-1 Safe Shutdown Areas
• Control Room • Containment • Auxiliary Building • Diesel Generator Rooms • Intake Structure • 1A/0C DG Buildings • RWT • RWT Rooms • CST No. 12 • FOST No. 21 • Auxiliary Feed Pump Rooms

NEI IC#	NEI IC Wording	CCNPP IC#(s)	CCNPP IC Wording	Difference/Deviation Justification
HA3	Access to a VITAL AREA is prohibited due to toxic, corrosive, asphyxiant or flammable gases which jeopardize operation of operable equipment required to maintain safe operations or safely shutdown the reactor. MODE: All	HA3	Access to a Vital Area is prohibited due to toxic, corrosive, asphyxiant or flammable gases which jeopardize operation of operable equipment required to maintain safe operations or safely shutdown the reactor MODE: All	None

NEI Ex. EAL #	NEI Example EAL Wording	CCNPP EAL #	CCNPP EAL Wording	Difference/Deviation Justification
1	**Note:** If the equipment in the stated area was already inoperable, or out of service, before the event occurred, then this EAL should not be declared as it will have no adverse impact on the ability of the plant to safely operate or safely shutdown beyond that already allowed by Technical Specifications at the time of the event. 1. Access to a VITAL AREA is prohibited due to toxic, corrosive, asphyxiant or flammable gases which jeopardize operation of systems required to maintain safe operations or safely shutdown the reactor.	HA3.1	Access to **ANY** Table H-1 area is prohibited due to toxic, corrosive, asphyxiant or flammable gases which jeopardize operation of **ANY** SAFETY-RELATED STRUCTURE, SYSTEM, OR COMPONENT (Note 5) Note 5: If the equipment in the stated area was already inoperable, or out of service, before the event occurred, then EAL HA3.1 should **not** be declared as it will have no adverse impact on the ability of the plant to safely operate or safely shutdown beyond that already allowed by Technical Specifications at the time of the event.	CCNPP vital areas are specified in Table H-1. This change implements EAL FAQ #44. The NEI phrase "systems required to maintain safe operations or safely shutdown the reactor" has been changed to "**ANY** safety-related structure, system, or component" for consistency with subcategory H1 and H2 Alert EALs. Safety-related structures, systems, and components are systems that are required to maintain safe operations or safely shutdown the reactor. Reference to the NEI note is included in the EAL wording "(Note 5)." Numbering the note facilitates referencing in the EAL matrix. The NEI phrase "this EAL" has been changed to "EAL HA3.1" for clarification.

NEI IC#	NEI IC Wording	CCNPP IC#(s)	CCNPP IC Wording	Difference/Deviation Justification
HA4	HOSTILE ACTION within the OWNER CONTROLLED AREA or airborne attack threat. MODE: All	HA4	Hostile action within the Owner Controlled Area or airborne attack threat. MODE: All	None

NEI Ex. EAL #	NEI Example EAL Wording	CCNPP EAL #	CCNPP EAL Wording	Difference/Deviation Justification
1	A HOSTILE ACTION is occurring or has occurred within the OWNER CONTROLLED AREA as reported by the (site specific security shift supervision).	HA4.1	A HOSTILE ACTION is occurring or has occurred within the Owner Controlled Area as reported by Security Shift Supervisor **OR** A validated notification from NRC of an AIRLINER attack threat within 30 min. of the site	The NEI Example EALs have been combined in one plant EAL for simplification. "Security Shift Supervisor" is the site-specific security supervision that are qualified and trained to confirm that a security event is occurring or has occurred.
2	A validated notification from NRC of an airliner attack threat within 30 minutes of the site.			

NEI IC#	NEI IC Wording	CCNPP IC#(s)	CCNPP IC Wording	Difference/Deviation Justification
HA5	Control Room Evacuation Has Been Initiated MODE: All	HA5	Control Room evacuation has been initiated MODE: All	None

NEI Ex. EAL #	NEI Example EAL Wording	CCNPP EAL #	CCNPP EAL Wording	Difference/Deviation Justification
1	(Site-specific procedure) requires control room evacuation.	HA5.1	Control Room evacuation has been initiated	AOP-9A, Control Room Evacuation and Safe Shutdown Due to a Severe Control Room Fire and AOP-11 Control Room Evacuation and Safe Shutdown - Non-Fire Conditions provide specific instructions for evacuating the Control Room if it becomes uninhabitable. The IC wording has been utilized since the intent is to classify the Alert based on Control Room evacuation, regardless whether the associated procedure has been entered or executed. This change implements EAL FAQ #28.

NEI IC#	NEI IC Wording	CCNPP IC#(s)	CCNPP IC Wording	Difference/Deviation Justification
HA6	Other conditions exist which in the judgment of the Emergency Director warrant declaration of an Alert. MODE: All	HA6	Other conditions exist that in the judgment of the Emergency Director warrant declaration of an Alert MODE: All	Replaced the word "which" with "that" for proper grammar.

NEI Ex. EAL #	NEI Example EAL Wording	CCNPP EAL #	CCNPP EAL Wording	Difference/Deviation Justification
1	Other conditions exist which in the judgment of the Emergency Director indicate that events are in progress or have occurred which involve an actual or potential substantial degradation of the level of safety of the plant or a security event that involves probable life threatening risk to site personnel or damage to site equipment because of HOSTILE ACTION. Any releases are expected to be limited to small fractions of the EPA Protective Action Guideline exposure levels.	HA6.1	Other conditions exist which in the judgment of the Emergency Director indicate that events are in progress or have occurred which involve: An actual or potential substantial degradation of the level of safety of the plant **OR** A security event that involves probable life threatening risk to site personnel or damage to site equipment because of HOSTILE ACTION **ANY** releases are expected to be limited to small fractions of the EPA Protective Action Guideline exposure levels (1,000 mRem TEDE and 5,000 mRem thyroid CDE)	Reformatted for readability. EPA PAG values have been added for clarification.

NEI IC#	NEI IC Wording	CCNPP IC#(s)	CCNPP IC Wording	Difference/Deviation Justification
HS4	A HOSTILE ACTION within the Protected Area MODE: All	HS4	Hostile action within the Protected Area MODE: All	Deleted unnecessary preposition "A".

NEI Ex. EAL #	NEI Example EAL Wording	CCNPP EAL #	CCNPP EAL Wording	Difference/Deviation Justification
1	A HOSITLE ACTION is occurring or has occurred within the PROTECTED AREA as reported by the (site-security shift supervision).	HS4.1	A HOSTILE ACTION is occurring or has occurred within the PROTECTED AREA as reported by Security Shift Supervisor	"Security Shift Supervisor" is the site-specific security supervision that are qualified and trained to confirm that a security event is occurring or has occurred.

NEI IC#	NEI IC Wording	CCNPP IC#(s)	CCNPP IC Wording	Difference/Deviation Justification
HS2	Control room evacuation has been initiated and plant control cannot be established. MODE: All	HS5	Control Room evacuation has been initiated and plant control **cannot** be established MODE: All	None

NEI Ex. EAL #	NEI Example EAL Wording	CCNPP EAL #	CCNPP EAL Wording	Difference/Deviation Justification
1	a. Control room evacuation has been initiated. **AND** b. Control of the plant cannot be established within (site specific minutes).	HS5.1	Control Room evacuation has been initiated **AND EITHER**: Inability to establish Auxiliary Feedwater to at least one steam generator within 30 min. (Note 4) **OR** Inability to establish reactor coolant make-up (charging pump flow) within 60 min. (Note 4) Note 4: The ED should **not** wait until the applicable time has elapsed, but should declare the event as soon as it is determined that the condition has exceeded, or will likely exceed, the applicable time.	An analysis was performed to determine how quickly control must be re-established at CCNPP without core uncovery or damage. A RETRAN simulation shows that the steam generators go dry at about 47 minutes for the AOP-9 (station fire) scenario. RCS pressure reaches the lowest pressurizer safety valve setpoint soon thereafter. Restoring feedwater within 45 minutes assures that RCS pressure remains below the safety valve setpoint thus avoiding inventory loss. The maximum time allowable to restore RCS inventory for Appendix R (station fire) scenarios is 90 minutes. Site Emergency declaration at 30 minutes and 60 minutes for inability to restore feedwater and RCS make-up respectively thus constitutes a conservative action for emergency response. This EAL is based on analysis and actual procedure walk throughs. Licensee Event Report (LER) 50-371/89-009, Rev. 2, (transmitted to the NRC on July 7, 1989) documents the analysis that demonstrates the ability to safely shutdown Unit 1 in accordance with AOP-9. Reference to Note 4 has been added to the CCNPP EAL for consistency with other EALs that include a time duration.

NEI IC#	NEI IC Wording	CCNPP IC#(s)	CCNPP IC Wording	Difference/Deviation Justification
HS3	Other conditions exist which in the judgment of the Emergency Director warrant declaration of a Site Area Emergency. MODE: All	HS6	Other conditions existing that in the judgment of the Emergency Director warrant declaration of a Site Area Emergency MODE: All	Replaced the word "which" with "that" for proper grammar

NEI Ex. EAL #	NEI Example EAL Wording	CCNPP EAL #	CCNPP EAL Wording	Difference/Deviation Justification
1	Other conditions exist which in the judgment of the Emergency Director indicate that events are in progress or have occurred which involve actual or likely major failures of plant functions needed for protection of the public or HOSTILE ACTION that results in intentional damage or malicious acts; (1) toward site personnel or equipment that could lead to the likely failure of or; (2) that prevent effective access to equipment needed for the protection of the public. Any releases are not expected to result in exposure levels which exceed EPA Protective Action Guideline exposure levels beyond the site boundary.	HS6.1	Other conditions exist which in the judgment of the Emergency Director indicate that events are in progress or have occurred which involve: Actual or likely major failures of plant functions needed for protection of the public **OR** HOSTILE ACTION that results in intentional damage or malicious acts; (1) toward site personnel or equipment that could lead to the likely failure of or; (2) that prevent effective access to equipment needed for the protection of the public **ANY** releases are **not** expected to result in exposure levels which exceed EPA Protective Action Guideline exposure levels (1,000 mRem TEDE and 5,000 mRem thyroid CDE) beyond the site boundary.	Reformatted for readability. EPA PAG values have been added for clarification.

NEI IC#	NEI IC Wording	CCNPP IC#(s)	CCNPP IC Wording	Difference/Deviation Justification
HG1	HOSTILE ACTION resulting in loss of physical control of the facility. MODE: All	HG4	Hostile action resulting in loss of physical control of the facility MODE: All	None

NEI Ex. EAL #	NEI Example EAL Wording	CCNPP EAL #	CCNPP EAL Wording	Difference/Deviation Justification
1	A HOSTILE ACTION has occurred such that plant personnel are unable to operate equipment required to maintain safety functions.	HG4.1	A HOSTILE ACTION has occurred such that plant personnel are unable to operate equipment required to maintain **ANY** of the following safety function acceptance criteria: • Reactivity control (RC) • Vital Auxiliaries (VA) • RCS pressure and inventory control (PIC) • Core & RCS heat removal (HR)	Safety functions listed in the NEI basis of this EAL that are applicable to CCNPP are the following and have been included for clarification: • Reactivity control (RC) • Vital Auxiliaries (VA) • RCS pressure and inventory control (PIC) • Core & RCS heat removal (HR)
2	A HOSTILE ACTION has caused failure of Spent Fuel Cooling Systems and IMMINENT fuel damage is likely for a freshly off-loaded reactor core in pool.	HG4.2	A HOSTILE ACTION has caused failure of Spent Fuel Cooling systems **AND** IMMINENT fuel damage is likely	The logic term "**AND**" has been added to the threshold for format consistency. The NEI phrase "for a freshly off-loaded reactor core in pool" has been deleted because any imminent fuel damage caused by hostile action warrants a GE declaration even if it is not from a freshly off-loaded core in pool. This change implements EAL FAQ # 29.

NEI IC#	NEI IC Wording	CCNPP IC#(s)	CCNPP IC Wording	Difference/Deviation Justification
HG2	Other conditions exist which in the judgment of the Emergency Director warrant declaration of a General Emergency. MODE: All	HG6	Other conditions exist that in the judgment of the Emergency Director warrant declaration of a General Emergency MODE: All	Replaced the word "which" with "that" for proper grammar

NEI Ex. EAL #	NEI Example EAL Wording	CCNPP EAL #	CCNPP EAL Wording	Difference/Deviation Justification
1	Other conditions exist which in the judgment of the Emergency Director indicate that events are in progress or have occurred which involve actual or IMMINENT substantial core degradation or melting with potential for loss of containment integrity or HOSTILE ACTION that results in an actual loss of physical control of the facility. Releases can be reasonably expected to exceed EPA Protective Action Guideline exposure levels off-site for more than the immediate site area.	HG6.1	Other conditions exist which in the judgment of the Emergency Director indicate that events are in progress or have occurred which involve: Actual or IMMINENT substantial core degradation or melting with potential for loss of Containment integrity **OR** HOSTILE ACTION that results in an actual loss of physical control of the facility Releases can be reasonably expected to exceed EPA Protective Action Guideline exposure levels (1,000 mRem TEDE and 5,000 mRem thyroid CDE) offsite for more than the immediate site area.	Reformatted for readability. EPA PAG values have been added for clarification.

Category S

System Malfunction

NEI IC#	NEI IC Wording	CCNPP IC#(s)	CCNPP IC Wording	Difference/Deviation Justification
SU1	Loss of all Off-site AC power to emergency busses for 15 minutes or longer. MODE: Power Operation, Startup, Hot Standby, Hot Shutdown	SU1	Loss of **all** offsite AC power to 4 kV vital buses for ≥ 15 min. MODE: 1 - Power Operation, 2 - Startup, 3 - Hot Standby, 4 - Hot Shutdown	"4 kV vital buses" is the CCNPP specific terminology for "emergency busses". The NEI phrase "15 minutes or longer" has been replaced with "≥ 15 min." to reduce EAL-user reading burden. The symbol "≥" means "greater than or equal to" and thus implements the intent of the NEI phrase.

NEI Ex. EAL #	NEI Example EAL Wording	CCNPP EAL #	CCNPP EAL Wording	Difference/Deviation Justification
1	Note: The Emergency Director should not wait until the applicable time has elapsed, but should declare the event as soon as it is determined that the condition has exceeded, or will likely exceed, the applicable time. Loss of all off-site AC power to (site specific emergency busses) for 15 minutes or longer.	SU1.1	Loss of **all offsite** AC power, Table S-1, to 4kV vital buses 11(21) and 14(24) for ≥ 15 min. (Note 4) Note 4: The ED should **not** wait until the applicable time has elapsed, but should declare the event as soon as it is determined that the condition has exceeded, or will likely exceed, the applicable time.	Table S-1 provides a list of onsite and offsite AC power supplies. 4kV vital buses 11(21) and 14(24) are the CCNPP emergency buses. The NEI phrase "15 minutes or longer" has been replaced with "≥ 15 min." to reduce EAL-user reading burden. The symbol "≥" means "greater than or equal to" and thus implements the intent of the NEI phrase. Reference to the NEI note is included in the EAL wording "(Note 4)." Numbering the note facilitates referencing in the EAL matrix.

Table S-1 AC Power Sources	
Onsite	• 1(2)A DG • 1(2)B DG • 0C DG , if aligned
Offsite	• 500kV transmission line 5051* • 500kV transmission line 5052* • 500kV transmission line 5072* • SMECO line , if aligned * A credited 500kV line must have an independent 13kV service transformer

NEI IC#	NEI IC Wording	CCNPP IC#(s)	CCNPP IC Wording	Difference/Deviation Justification
SU2	Inability to reach required shutdown within Technical Specification limits. MODE: Power Operation, Startup, Hot Standby, Hot Shutdown	SU4	Inability to reach required shutdown within Technical Specification limits MODE: 1 - Power Operation, 2 - Startup, 3 - Hot Standby, 4 - Hot Shutdown	None

NEI Ex. EAL #	NEI Example EAL Wording	CCNPP EAL #	CCNPP EAL Wording	Difference/Deviation Justification
1	Plant is not brought to required operating mode within Technical Specifications LCO Action Statement Time.	SU4.1	Plant is **not** brought to required operating mode within Technical Specifications LCO required action completion time	"... required action completion time" is the CCNPP Technical Specification terminology for "... Action Statement Time."

NEI IC#	NEI IC Wording	CCNPP IC#(s)	CCNPP IC Wording	Difference/Deviation Justification
SU3	UNPLANNED loss of safety system annunciation or indication in the control room for 15 minutes or longer. MODE: Power Operation, Startup, Hot Standby, Hot Shutdown	SU5	Unplanned loss of safety system annunciation or indication in the Control Room for ≥ 15 min. MODE: 1 - Power Operation, 2 - Startup, 3 - Hot Standby, 4 - Hot Shutdown	The NEI phrase "15 minutes or longer" has been replaced with "≥ 15 min." to reduce EAL-user reading burden. The symbol "≥" means "greater than or equal to" and thus implements the intent of the NEI phrase.

NEI Ex. EAL #	NEI Example EAL Wording	CCNPP EAL #	CCNPP EAL Wording	Difference/Deviation Justification
1	**Note:** The Emergency Director should not wait until the applicable time has elapsed, but should declare the event as soon as it is determined that the condition has exceeded, or will likely exceed, the applicable time. UNPLANNED Loss of greater than approximately 75% of the following for 15 minutes or longer: a. (Site specific control room safety system annunciation) **OR** b. (Site specific control room safety system indication)	SU5.1	UNPLANNED loss of greater than approximately 75% of safety system annunciation or indication on Control Room panels for ≥ 15 min. (Note 4) Note 4: The ED should **not** wait until the applicable time has elapsed, but should declare the event as soon as it is determined that the condition has exceeded, or will likely exceed, the applicable time.	The NEI phrase "...greater than approximately 75% of the following ...a...b..." has been changed to "...greater than approximately 75% of safety system annunciation or indication on Control Room panels" for simplification. All main control board panels house annunciators and indicators important for control of the plant. A site specific list of Control Room safety system annunciation and indication is not listed in this EAL. Safety-related annunciation and indications are numerous and varied. Just as the Shift Manager is expected to use his/her judgment in assessing the loss of 75% of annunciation and indication, the Shift Manager is best situated to assess the Control Room panel indicators and annunciation that are important for control of the plant. The NEI phrase "15 minutes or longer" has been replaced with "≥ 15 min." to reduce EAL-user reading burden. The symbol "≥" means "greater than or equal to" and thus implements the intent of the NEI phrase. Reference to the NEI note is included in the EAL wording "(Note 6)." Numbering the note facilitates referencing in the EAL matrix.

NEI IC#	NEI IC Wording	CCNPP IC#(s)	CCNPP IC Wording	Difference/Deviation Justification
SU4	Fuel Clad Degradation MODE: Power Operation, Startup, Hot Standby, Hot Shutdown	SU7	Fuel clad degradation MODE: 1 - Power Operation, 2 - Startup, 3 - Hot Standby, 4 - Hot Shutdown	None

NEI Ex. EAL #	NEI Example EAL Wording	CCNPP EAL #	CCNPP EAL Wording	Difference/Deviation Justification
1	(Site specific radiation monitor readings indicating fuel clad degradation greater than Technical Specification allowable limits.)	SU7.2	Letdown Monitor (RY-202-1) high alarm (≥ 1E+06 cpm)	The Letdown Radiation Monitor (1(2)-RY-202-1)) gross radiation channel continuously monitors the activity in a sample drawn from the RCS and actuates an alarm in the Control Room if a predetermined activity level is reached. The sensor is a gross-gamma plus specific isotope (I-135) monitor; the system is designed to detect activity release from the fuel to the reactor coolant within five minutes of a fuel degradation event.
2	(Site specific coolant sample activity value indicating fuel clad degradation greater than Technical Specification allowable limits.)	SU7.1	Coolant activity > **ANY** of the following: • Dose equivalent I-131 0.5 uCi/gm for 100 hrs. continuous • Dose equivalent I-131 acceptable region of T.S. Fig. 3.4.15-1 • Dose equivalent I-131 137.5 uCi/gm • Gross activity 100/E-bar uCi/gm	The specified values are the TS coolant activity limits.

NEI IC#	NEI IC Wording	CCNPP IC#(s)	CCNPP IC Wording	Difference/Deviation Justification
SU5	RCS Leakage MODE: Power Operation, Startup, Hot Standby, Hot Shutdown	SU8	RCS leakage MODE: 1 - Power Operation, 2 - Startup, 3 - Hot Standby, 4 - Hot Shutdown	None

NEI Ex. EAL #	NEI Example EAL Wording	CCNPP EAL #	CCNPP EAL Wording	Difference/Deviation Justification
1	Unidentified or pressure boundary leakage greater than 10 gpm.	SU8.1	Unidentified or pressure boundary leakage > 10 gpm for ≥ 15 min. (Note 4) **OR** Identified leakage > 25 gpm for ≥ 15 min. (Note 4) Note 4: The EC 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.	SU8.1 implements Example EALs #1 and #2. These were combined for improved usability. The NEI phrase "greater than" has been replaced with the symbol ">" to reduce EAL-user reading burden. The symbol means "greater than" and thus implements the intent of the NEI phrase. The phrase "for ≥ 15 min. (Note 4)" has been added to the CCNPP EAL to allow mitigation by operating procedures prior to declaration. This is a deviation from NEI 99-01 Revision 5.
2	Identified leakage greater than 25 gpm,			

NEI IC#	NEI IC Wording	CCNPP IC#(s)	CCNPP IC Wording	Difference/Deviation Justification
SU6	Loss of all On-site or Off-site communications capabilities. MODE: Power Operation, Startup, Hot Standby, Hot Shutdown	SU6	Loss of **all** onsite or offsite communications capabilities MODE: 1 - Power Operation, 2 - Startup, 3 - Hot Standby, 4 - Hot Shutdown	None

NEI Ex. EAL #	NEI Example EAL Wording	CCNPP EAL #	CCNPP EAL Wording	Difference/Deviation Justification
1	Loss of all of the following on-site communication methods affecting the ability to perform routine operations. (site specific list of communications methods)	SU6.1	Loss of **all** Table S-3 onsite (internal) communication methods affecting the ability to perform routine operations **OR** Loss of **all** Table S-3 offsite (external) communication methods affecting the ability to perform offsite notifications to any agency	SU5.1 implements Example EALs #1 and #2. These were combined for improved usability. The NEI example EALs specify site-specific lists of onsite and offsite communications methods. The CCNPP EAL lists these methods in Table S-3 because the number of communications methods is too long to include within the text of the EAL. The adjectives "(internal)" and "(external)" have been added to the CCNPP EAL for clarification. The terms "onsite/offsite" could be interpreted as the location in which the communication originates instead of the location to which communication is directed. Added the words "...to any agency" to clarify the intent of the bases statement: "The availability of one method of ordinary off-site communications is sufficient to inform federal, state, and local authorities of plant issues."
2	Loss of all of the following off-site communication methods affecting the ability to perform offsite notifications. (site specific list of communications methods)			

Table S-3 Communications Systems		
System	**Onsite** (internal)	**Offsite** (external)
Commercial phone system	X	X
Plant page system	X	
Microwave telephone (Hot-Lines) (EOB)	X	X
Dedicated offsite agency telephone system		X
FTS 2001 telephone system		X
CCNPP Radio System	X	X
Satellite Phone System		X
Cellular Phone System	X	X

NEI IC#	NEI IC Wording	CCNPP IC#(s)	CCNPP IC Wording	Difference/Deviation Justification
SU8	Inadvertent Criticality. MODE: Hot Standby, Hot Shutdown	SU3	Inadvertent criticality MODE: 3 - Hot Standby, 4 - Hot Shutdown	None

NEI Ex. EAL #	NEI Example EAL Wording	CCNPP EAL #	CCNPP EAL Wording	Difference/Deviation Justification
1	UNPLANNED sustained positive period observed on nuclear instrumentation. [*BWR*]	N/A	N/A	NEI Example EAL #1 has not been implemented because it applies only to BWR plants. CCNPP is a PWR. PWRs are not equipped with period meters.
1	UNPLANNED sustained positive startup rate observed on nuclear instrumentation. [*PWR*]	SU3.1	An UNPLANNED sustained positive startup rate observed on nuclear instrumentation	None

NEI IC#	NEI IC Wording	CCNPP IC#(s)	CCNPP IC Wording	Difference/Deviation Justification
SA2	Automatic Scram (Trip) fails to shutdown the reactor and the manual actions taken from the reactor control console are successful in shutting down the reactor. MODE: Power Operation, Startup	SA3	Automatic reactor trip failed to shut down the reactor and the manual actions taken from the reactor control console are successful in shutting down the reactor MODE: 1 - Power Operation	The term "reactor" has been added to the phrase "automatic…trip" for clarification. The term "scram" was replaced with "trip" consistent with PWR terminology. The NEI term "fails" has been changed to "failed" for consistency with the example EAL wording. This change implements EAL FAQ #31. The Startup mode has been deleted from the CCNPP EAL. CCNPP Technical Specifications definition of Startup mode is $K_{eff} \geq 0.99$ and rated thermal power ≤ 5%. It is not possible to be in Startup mode with reactor power above 5%.Since the definition of reactor shutdown is reactor power less than or equal to 5% (in accordance with the NEI 99-01 basis for this EAL), this EAL would never be applicable in Startup mode.

NEI Ex. EAL #	NEI Example EAL Wording	CCNPP EAL #	CCNPP EAL Wording	Difference/Deviation Justification
1	a. An automatic scram (trip) failed to shutdown the reactor. **AND** b. Manual actions taken at the reactor control console successfully shutdown the reactor as indicated by (site specific indications of plant shutdown).	SA3.1	An automatic trip failed to shut down the reactor **AND** Manual actions taken at the Control Room panels successfully shut down the reactor as indicated by reactor power ≤ 5%	The term "scram" was replaced with "trip" consistent with PWR terminology. The NEI phrase "reactor control console" has been replaced with "Control Room panels" to use terminology familiar to CCNPP operators. The power range indication above 5% is greater than the decay heat which the shutdown systems (Auxiliary Feed Water and Atmospheric Dump Valves) were designed to remove and is indicative of a condition requiring immediate response to prevent subsequent core damage.

NEI IC#	NEI IC Wording	CCNPP IC#(s)	CCNPP IC Wording	Difference/Deviation Justification
SA4	UNPLANNED Loss of safety system annunciation or indication in the control room with EITHER (1) a SIGNIFICANT TRANSIENT in progress, or (2) compensatory indicators unavailable. MODE: Power Operation, Startup, Hot Standby, Hot Shutdown	SA5	Unplanned loss of safety system annunciation or indication in the Control Room with either (1) a significant transient in progress, or (2) compensatory indicators are unavailable MODE: 1 - Power Operation, 2 - Startup, 3 - Hot Standby, 4 - Hot Shutdown	None

NEI Ex. EAL #	NEI Example EAL Wording	CCNPP EAL #	CCNPP EAL Wording	Difference/Deviation Justification
1	**Note:** The Emergency Director should not wait until the applicable time has elapsed, but should declare the event as soon as it is determined that the condition has exceeded, or will likely exceed, the applicable time. a. UNPLANNED loss of greater than approximately 75% of the following for 15 minutes or longer: • (Site specific control room safety system annunciation) **OR** • (Site specific control room safety system indication) **OR** b. **EITHER** of the following: • A SIGNIFICANT TRANSIENT is in progress. • Compensatory indications are	SA5.1	UNPLANNED loss of greater than approximately 75% of safety system annunciation or indication on Control Room panels for ≥ 15 min. (Note 4) **AND EITHER**: A significant transient is in progress, Table S-2 **OR** Compensatory indications are unavailable (Plant Computer, SPDS) Note 4: The ED should **not** wait until the applicable time has elapsed, but should declare the event as soon as it is determined that the condition has exceeded, or will likely exceed, the applicable time.	The NEI phrase "...greater than approximately 75% of the following ...a...b..." has been changed to "...greater than approximately 75% of safety system annunciation or indication on Control Room panels..." for simplification. All main control board panels house annunciators and indicators important for control of the plant. A site specific list of Control Room safety system annunciation and indication is not listed in this EAL. Safety-related annunciation and indications are numerous and varied. Just as the Shift Manager is expected to use his/her judgment in assessing the loss of 75% of annunciation and indication, the Shift Manager is best situated to assess the Control Room panel indicators and annunciation that are important for control of the plant. The NEI phrase "15 minutes or longer" has been replaced with "≥ 15 min." to reduce EAL-user reading burden. The symbol "≥" means "greater than or equal to" and thus implements the intent of the NEI phrase. The CCNPP compensatory indications are provided by Plant Process Computer and SPDS. Reference to the NEI note is included in the EAL wording "(Note

	unavailable.			4).” Numbering the note facilitates referencing in the EAL matrix. Table S-2 provides the list of events that constitute a “significant transient.” 10% thermal power oscillations have been deleted because it is not possible for CCNPP to have such power oscillations.

Table S-2 Significant Transients
• Automatic turbine runback > 25% thermal power • Electric load rejection > 25% full electrical load • Reactor trip • Safety Injection actuation

NEI IC#	NEI IC Wording	CCNPP IC#(s)	CCNPP IC Wording	Difference/Deviation Justification
SA5	AC power capability to emergency busses reduced to a single power source for 15 minutes or longer such that any additional single failure would result in station blackout. MODE: Power Operation, Startup, Hot Standby, Hot Shutdown	SA1	AC power capability to 4kV vital buses reduced to a single power source for ≥15 min. such that **ANY** additional single failure would result in a complete loss of all 4kV vital bus power MODE: 1 - Power Operation, 2 - Startup, 3 - Hot Standby, 4 - Hot Shutdown	"4kV vital buses" is the CCNPP specific terminology for "emergency busses". The NEI phrase "15 minutes or longer" has been replaced with "≥ 15 min." to reduce EAL-user reading burden. The symbol "≥" means "greater than or equal to" and thus implements the intent of the NEI phrase. The phrase "... any additional single failure would result in station blackout." was replaced with "... **ANY** additional single failure would result in a complete loss of all 4kV vital bus power." This is consistent with the intent that classification be based on a loss of AC power to emergency buses. A Station Blackout involves a loss of all AC power, not just emergency bus power. This change implements EAL FAQ #36.

NEI Ex. EAL #	NEI Example EAL Wording	CCNPP EAL #	CCNPP EAL Wording	Difference/Deviation Justification
1	**Note:** The Emergency Director should not wait until the applicable time has elapsed, but should declare the event as soon as it is determined that the condition has exceeded, or will likely exceed, the applicable time. a. AC power capability to (site-specific emergency busses) reduced to a single power source for 15 minutes or longer. b. Any additional single power source failure will result in	SA1.1	AC power capability to 4kV vital buses 11(21) and 14(24) reduced to a single power source, Table S-1, for ≥ 15 min. (Note 4) **AND** **ANY** additional single power source failure will result in a complete loss of all 4kV vital bus power Note 4: The ED should **not** wait until the applicable time has elapsed, but should declare the event as soon as it is determined	Table S-1 provides a list of onsite and offsite AC power supplies. 4kV vital buses 11(21) and 14(24) are the CCNPP emergency buses. The NEI phrase"...station blackout" has been replaced with "... a complete loss of all 4kV vital bus power " as this describes the intended condition for CCNPP. This change implements EAL FAQ #36. The NEI phrase "15 minutes or longer" has been replaced with "≥ 15 min." to reduce EAL-user reading burden. The symbol "≥" means "greater than or equal to" and thus implements the intent of the NEI phrase. Reference to the NEI note is included in the EAL wording "(Note 4)." Numbering the note facilitates referencing in the EAL matrix.

	station blackout.		that the condition has exceeded, or will likely exceed, the applicable time.	

NEI IC#	NEI IC Wording	CCNPP IC#(s)	CCNPP IC Wording	Difference/Deviation Justification
SS1	Loss of all Off-site and all On-Site AC power to emergency busses for 15 minutes or longer. MODE: Power Operation, Startup, Hot Standby, Hot Shutdown	SS1	Loss of **all** offsite and **all** onsite AC power to 4kV vital buses for ≥ 15 min. MODE: 1 - Power Operation, 2 - Startup, 3 - Hot Standby, 4 - Hot Shutdown	"4kV vital buses" is the CCNPP specific terminology for "emergency busses". The NEI phrase "15 minutes or longer" has been replaced with "≥ 15 min." to reduce EAL-user reading burden. The symbol "≥" means "greater than or equal to" and thus implements the intent of the NEI phrase.

NEI Ex. EAL #	NEI Example EAL Wording	CCNPP EAL #	CCNPP EAL Wording	Difference/Deviation Justification
1	Note: The Emergency Director should not wait until the applicable time has elapsed, but should declare the event as soon as it is determined that the condition has exceeded, or will likely exceed, the applicable time. Loss of all Off-Site and all On-Site AC power to (site specific emergency busses) for 15 minutes or longer.	SS1.1	Loss of **all** offsite and **all** onsite AC power, Table S-1, to 4kV vital buses 11(21) and 14(24) for ≥ 15 min. (Note 4) Note 4: The ED should **not** wait until the applicable time has elapsed, but should declare the event as soon as it is determined that the condition has exceeded, or will likely exceed, the applicable time.	Table S-1 provides a list of onsite and offsite AC power supplies. 4kV vital buses 11(21) and 14(24) are the CCNPP emergency buses. The NEI phrase "15 minutes or longer" has been replaced with "≥ 15 min." to reduce EAL-user reading burden. The symbol "≥" means "greater than or equal to" and thus implements the intent of the NEI phrase. Reference to the NEI note is included in the EAL wording "(Note 4)." Numbering the note facilitates referencing in the EAL matrix.

NEI IC#	NEI IC Wording	CCNPP IC#(s)	CCNPP IC Wording	Difference/Deviation Justification
SS2	Automatic Scram (Trip) fails to shutdown the reactor and manual actions taken from the reactor control console are not successful in shutting down the reactor. MODE: Power Operation, Startup	SS3	Automatic trip and manual actions taken from the reactor control console failed to shut down the reactor 1 - Power Operation	The term "scram" was replaced with "trip" consistent with PWR terminology. The NEI phrase "fails to shutdown the reactor and manual actions taken from the reactor control console are not successful in shutting" has been changed to "and manual actions taken from the reactor control console failed to shut" for clarification. This change implements EAL FAQ #31. The Startup mode has been deleted from the CCNPP EAL. CCNPP Technical Specifications definition of Startup mode is $K_{eff} \geq 0.99$ and rated thermal power ≤ 5%. It is not possible to be in Startup mode with reactor power above 5%.Since the definition of reactor shutdown is reactor power less than or equal to 5% (in accordance with the NEI 99-01 basis for this EAL), this EAL would never be applicable in Startup mode.

NEI Ex. EAL #	NEI Example EAL Wording	CCNPP EAL #	CCNPP EAL Wording	Difference/Deviation Justification
1	a. An automatic scram (trip) failed to shutdown the reactor. **AND** b. Manual actions taken at the reactor control console do not shutdown the reactor as indicated by (site specific indications of reactor not shutdown).	SS3.1	An automatic reactor trip failed to shut down the reactor as indicated by reactor power > 5% **AND** Manual actions taken at the Control Room panels do **not** shut down the reactor as indicated by reactor power > 5%	The term "reactor" has been added to the phrase "automatic…trip" for clarification. The term "scram" was replaced with "trip" consistent with PWR terminology. The phrase "as indicated by reactor power > 5%" has been added to the first contingent and the NEI phrase "do not shutdown the reactor" has been changed to "failed to shut down the reactor" in the second contingent for clarification and consistency of wording. This change implements EAL FAQ #31. The NEI phrase "reactor control console" has been replaced with "Control Room panels" to use terminology familiar to CCNPP operators. The power range indication above 5% is greater than the decay heat which the shutdown systems (Auxiliary Feed Water and Atmospheric

				Dump Valves) were designed to remove and is indicative of a condition requiring immediate response to prevent subsequent core damage.

NEI IC#	NEI IC Wording	CCNPP IC#(s)	CCNPP IC Wording	Difference/Deviation Justification
SS3	Loss of all vital DC power for 15 minutes or longer. MODE: Power Operation, Startup, Hot Standby, Hot Shutdown	SS2	Loss of **all** vital DC power for ≥ 15 min. MODE: 1 - Power Operation, 2 - Startup, 3 - Hot Standby, 4 - Hot Shutdown	The NEI phrase "15 minutes or longer" has been replaced with "≥ 15 min." to reduce EAL-user reading burden. The symbol "≥" means "greater than or equal to" and thus implements the intent of the NEI phrase.

NEI Ex. EAL #	NEI Example EAL Wording	CCNPP EAL #	CCNPP EAL Wording	Difference/Deviation Justification
1	**Note:** The Emergency Director should not wait until the applicable time has elapsed, but should declare the event as soon as it is determined that the condition has exceeded, or will likely exceed, the applicable time. Less than (site specific bus voltage indication) on all (site specific Vital DC busses) for 15 minutes or longer.	SS2.1	< 105 VDC on **all** 125 VDC buses (11, 12, 21 and 22) for ≥ 15 min. (Note 4) Note 4: The ED should **not** wait until the applicable time has elapsed, but should declare the event as soon as it is determined that the condition has exceeded, or will likely exceed, the applicable time.	The NEI IC phrase "less than" has been replaced with "<" to reduce EAL-user reading burden. The symbol "<" means "less than" and thus implements the intent of the NEI phrase. "105 VDC" is the site-specific bus voltage indication. 125 VDC buses (11, 12, 21 and 22) are the CCNPP vital DC buses. The NEI phrase "15 minutes or longer" has been replaced with "≥ 15 min." to reduce EAL-user reading burden. The symbol "≥" means "greater than or equal to" and thus implements the intent of the NEI phrase. Reference to the NEI note is included in the EAL wording "(Note 4)." Numbering the note facilitates referencing in the EAL matrix.

NEI IC#	NEI IC Wording	CCNPP IC#(s)	CCNPP IC Wording	Difference/Deviation Justification
SS6	Inability to monitor a SIGNIFICANT TRANSIENT in progress. MODE: Power Operation, Startup, Hot Standby, Hot Shutdown	SS5	Inability to monitor a significant transient in progress MODE: 1 - Power Operation, 2 - Startup, 3 - Hot Standby, 4 - Hot Shutdown	None

NEI Ex. EAL #	NEI Example EAL Wording	CCNPP EAL #	CCNPP EAL Wording	Difference/Deviation Justification
1	**Note:** The Emergency Director should not wait until the applicable time has elapsed, but should declare the event as soon as it is determined that the condition has exceeded, or will likely exceed, the applicable time. a. Loss of greater than approximately 75% of the following for 15 minutes or longer: • (Site specific control room safety system annunciation) **OR** • (Site specific control room safety system indication) **AND** b. A SIGNIFICANT TRANSIENT is in progress. **AND** c. Compensatory indications are	SS5.1	Loss of greater than approximately 75% of safety system annunciation or indication on Control Room panels for ≥ 15 min. (Note 4) **AND** A significant transient is in progress, Table S-2 **AND** Compensatory indications are unavailable (Plant Computer, SPDS) Note 4: The ED should **not** wait until the applicable time has elapsed, but should declare the event as soon as it is determined that the condition has exceeded, or will likely exceed, the applicable time.	The NEI phrase "...greater than approximately 75% of the following ...a...b..." has been changed to "...greater than approximately 75% of safety system annunciation or indication on Control Room panels ..." for simplification. All main control board panels house annunciators and indicators important for control of the plant. A site specific list of Control Room safety system annunciation and indication is not listed in this EAL. Safety-related annunciation and indications are numerous and varied. Just as the Shift Manager is expected to use his/her judgment in assessing the loss of 75% of annunciation and indication, the Shift Manager is best situated to assess the Control Room panel indicators and annunciation that are important for control of the plant. The NEI phrase "15 minutes or longer" has been replaced with "≥ 15 min." to reduce EAL-user reading burden. The symbol "≥" means "greater than or equal to" and thus implements the intent of the NEI phrase. The CCNPP compensatory indications are provided by Plant Computer and SPDS. Reference to the NEI note is included in the EAL wording "(Note 4)." Numbering the note facilitates referencing in the EAL matrix. Table S-2 provides the list of events that constitute a "significant transient." 10% thermal power oscillations have been deleted

	unavailable.			because it is not possible for CCNPP to have such power oscillations.

Table S-2 Significant Transients
• Automatic turbine runback > 25% thermal power • Electric load rejection > 25% full electrical load • Reactor trip • Safety Injection actuation

NEI IC#	NEI IC Wording	CCNPP IC#(s)	CCNPP IC Wording	Difference/Deviation Justification
SG1	Prolonged loss of all Off-site and all On-Site AC power to emergency busses. MODE: Power Operation, Startup, Hot Standby, Hot Shutdown	SG1	Prolonged loss of **all** offsite and **all** onsite AC power to 4kV vital buses MODE: 1 - Power Operation, 2 - Startup, 3 - Hot Standby, 4 - Hot Shutdown	"4kV vital buses" is the CCNPP specific terminology for "emergency busses".

NEI Ex. EAL #	NEI Example EAL Wording	CCNPP EAL #	CCNPP EAL Wording	Difference/Deviation Justification
1	a. Loss of all off-site and all on-site AC power to (site specific emergency busses). **AND** b. **EITHER** of the following: • Restoration of at least one emergency bus in less than (site specific hours) is not likely. • (Site specific indication of continuing degradation of core cooling based on Fission Product Barrier monitoring.)	SG1.1	Loss of **all** offsite and **all** onsite AC power, Table S-1, to 4kV vital buses 11(21) and 14(24) **AND EITHER:** Restoration of at least one 4kV vital bus within 4 hours is **not** likely **OR** CET readings > 700°F	4kV vital buses 11(21) and 14(24) are the CCNPP emergency buses. The NEI phrase "...of the following: ..." has been deleted. It is evident from the subsequent paragraphs and indentation applied to the CCNPP EAL that they follow the previous paragraph. 4 are the "(site-specific)" hours for station blackout coping. The four-hour interval to restore AC power is based on the blackout coping analysis performed in conformance with 10 CFR 50.63 and Regulatory Guide 1.155. The NEI phrase "...(Site-Specific) Indication of continuing degradation of core cooling based on Fission Product Barrier monitoring" has been replaced with "CET readings > 700°F" for clarification. This threshold represents the NEI conditions consistent with the corresponding fission product barrier Fuel Clad Loss and Potential Loss thresholds.

NEI IC#	NEI IC Wording	CCNPP IC#(s)	CCNPP IC Wording	Difference/Deviation Justification
SG2	Automatic Scram (Trip) and all manual actions fail to shutdown the reactor and indication of an extreme challenge to the ability to cool the core exists. MODE: Power Operation, Startup	SG3	Automatic trip and **all** manual actions fail to shut down the reactor and indication of an extreme challenge to the ability to cool the core exists MODE: 1 - Power Operation	The term "scram" was replaced with "trip" consistent with PWR terminology. The Startup mode has been deleted from the CCNPP EAL. CCNPP Technical Specifications definition of Startup mode is $K_{eff} \geq 0.99$ and rated thermal power ≤ 5%. It is not possible to be in Startup mode with reactor power above 5%.Since the definition of reactor shutdown is reactor power less than or equal to 5% (in accordance with the NEI 99-01 basis for this EAL), this EAL would never be applicable in Startup mode.

NEI Ex. EAL #	NEI Example EAL Wording	CCNPP EAL #	CCNPP EAL Wording	Difference/Deviation Justification
1	a. An automatic scram (trip) failed to shutdown the reactor. **AND** b. All manual actions do not shutdown the reactor as indicated by (site specific indications of reactor not shutdown). **AND** c. **EITHER** of the following exist or have occurred due to continued power generation: • (Site specific indication that core cooling is extremely challenged.) (Site specific indication that heat removal is extremely challenged.)	SG3.1	An automatic reactor trip failed to shut down the reactor as indicated by reactor power > 5% **AND** **All** manual actions fail to shut down the reactor as indicated by reactor power > 5% **AND** **ANY** of the following exist or have occurred: • CET readings > 700°F • RCS pressure > PORV setpoint • RCS subcooling < 25°F	The term "reactor" has been added to the phrase "automatic…trip" for clarification. The term "scram" was replaced with "trip" consistent with PWR terminology. The phrase "as indicated by reactor power > 5%" has been added to the REGNPP EAL for clarification. This change implements EAL FAQ #31. The power range indication above 5% is greater than the decay heat which the shutdown systems (Auxiliary Feed Water and Atmospheric Dump Valves) were designed to remove and is indicative of a condition requiring immediate response to prevent subsequent core damage. The NEI phrase "do not shutdown" has been changed to "fail to shut down" for consistency with the IC wording. This change implements EAL FAQ #31. The NEI phrase "**EITHER** of the following" has been changed to "**ANY** of the following" because the three subsequent conditions are equally weighted; one indicative of challenge to core cooling and

				either of the remaining two indicative of challenge to heat removal. The NEI phrase "due to continued power generation" has been deleted because the extreme challenge to heat removal, equivalent to core cooling red, should not be constrained by requiring it to be caused by continued power generation. This change implements EAL FAQ #37. Site-specific indication that core cooling is extremely challenged is reactor power greater than 5% with indications of CET > 700°F. Site-specific indication that heat removal is extremely challenged is reactor power greater than 5% with RCS pressure > PORV setpoint or RCS subcooling < 25°F. These conditions indicate the core cooling or ultimate heat sink function is under extreme challenge and, therefore, a core melt sequence may exist and rapid degradation of the fuel cladding could begin.

ATTACHMENT (4)

OFFSITE AGENCY APPROVALS

MARYLAND
DEPARTMENT OF
NATURAL RESOURCES

Martin O'Malley, Governor
Anthony G. Brown, Lt. Governor
John R. Griffin, Secretary
Joseph P. Gill, Deputy Secretary

December 17, 2010

Mr. Michael Fick
Director, Emergency Planning
Calvert Cliffs Nuclear Power Plant
1650 Calvert Cliffs Parkway
Lusby, Maryland 20657-4702

RE: Proposed Calvert Cliffs Emergency Action Levels

Dear Mr. Fick:

The Maryland Department of Natural Resources, Power Plant Research Program understands and agrees with the proposed revisions to the Emergency Action Levels and Technical Basis Document, Revision 0, for the Calvert Cliffs Nuclear Power Plant. We appreciate the opportunity to participate and comment on the proposed revision.

Sincerely,

Susan Gray

Susan Gray
Manager, Nuclear Programs
Power Plant Research Program

MARYLAND DEPARTMENT OF THE ENVIRONMENT

MDE

1800 Washington Boulevard • Baltimore MD 21230
410-537-3000 • 1-800-633-6101

Martin O'Malley
Governor

Anthony G. Brown
Lieutenant Governor

Robert M. Summers, Ph.D.
Acting Secretary

December 10, 2010

Mr. Michael Fick
Director, Emergency Planning
Calvert Cliffs Nuclear Power Plant
1650 Calvert Cliffs Parkway
Lusby, Maryland 20657-4702

RE: Proposed Calvert Cliffs Emergency Action Levels

Dear Mr. Fick:

The Maryland Department of the Environment understands and agrees with the proposed revisions to the Emergency Action Levels and Technical Basis Document, Revision 0, for the Calvert Cliffs Nuclear Power Plant. We appreciate the opportunity to participate and comment on the proposed revision.

Sincerely,

Tom Levering
Director, Emergency Preparedness

STATE OF MARYLAND
MILITARY DEPARTMENT

MARTIN O'MALLEY
GOVERNOR

ANTHONY G. BROWN
LIEUTENANT GOVERNOR

JAMES A. ADKINS
BRIGADIER GENERAL
THE ADJUTANT GENERAL

RICHARD G. MUTH
DIRECTOR

MARYLAND EMERGENCY MANAGEMENT AGENCY
State Emergency Operations Center, Camp Fretterd Military Reservation
5401 Rue Saint Lo Drive, Reisterstown, MD 21136
(410) 517-3600 • Fax (410) 517-3610 • Toll Free 1 (877) 636-2872
TTY Users: 1 (800) 735-2258

January 7, 2011

Mr. Michael Fick
Director, Emergency Planning
Calvert Cliffs Nuclear Power Plant
1650 Calvert Cliffs Parkway
Lusby, Maryland 20657-4702

RE: Proposed Calvert Cliffs Emergency Action Levels

Dear Mr. Fick:

The Maryland Emergency Management Agency understands and agrees with the proposed revisions to the Emergency Action Levels and Technical Basis Document, Revision 0, for the Calvert Cliffs Nuclear Power Plant. We appreciate the opportunity to participate and comment on the proposed revision.

Sincerely,

Frederick H Frey for

Richard Muth, Director
Maryland Emergency Management
Agency

ST. MARY'S COUNTY GOVERNMENT
DEPARTMENT OF PUBLIC SAFETY
David D. Zylak, Director
301-475-4200, Ext. 2111 / FAX 301-475-4512

EMERGENCY COMMUNICATIONS
Tommy Mattingly Jr, Communications Manager
301-475-4200, Ext. 2121
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EMERGENCY MANAGEMENT
Jaclyn Shaw, Manager
Main Line: 301-475-4200, Ext. 2125
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ANIMAL CONTROL
Antonio J. Malaspina, Sr., Supervisor
Main Line: 301-475-8018
FAX: 301-475-4924

December 10, 2010

Mr. Michael Fick
Director, Emergency Planning
Calvert Cliffs Nuclear Power Plant
1650 Calvert Cliffs Parkway
Lusby, Maryland 20657-4702

RE: Proposed Calvert Cliffs Emergency Action Levels

Dear Mr. Fick:

The St. Mary's County Emergency Management Agency understands and agrees with the proposed revisions to the Emergency Action Levels and Technical Basis Document, Revision 0, for the Calvert Cliffs Nuclear Power Plant. We appreciate the opportunity to participate and comment on the proposed revision.

Sincerely,

David D. Zylak, Director
Public Safety, St Mary's County

Dorchester County
Emergency Management Agency
M. Wayne Robinson, Director
829 Fieldcrest Road
Cambridge, Maryland 21613

Tel: 410-228-1818 E-Mail: ***dema@docogonet.com*** Fax: 410-228-1216

December 17, 2010

Mr. Michael Fick
Director, Emergency Planning
Calvert Cliffs Nuclear Power Plant
1650 Calvert Cliffs Parkway
Lusby, Maryland 20657-4702

RE: Proposed Calvert Cliffs Emergency Action Levels

Dear Mr. Fick:

The Dorchester County Emergency Management Agency understands and agrees with the proposed revisions to the Emergency Action Levels and Technical Basis Document, Revision 0, for the Calvert Cliffs Nuclear Power Plant. We appreciate the opportunity to participate and comment on the proposed revision.

Sincerely,

Wayne Robinson, Director
Dorchester County Emergency
Management Agency

CALVERT COUNTY
DEPARTMENT OF PUBLIC SAFETY
EMERGENCY MANAGEMENT AND SAFETY DIVISION

175 Main Street
Prince Frederick, Maryland 20678
410-535-1600 • 301-855-1243 • Ext. 2301
Fax: 410-535-3997

Jacqueline K. Vaughan, Director
J.R. Fenwick, Division Chief

December 10, 2010

Mr. Michael Fick
Director, Emergency Planning
Calvert Cliffs Nuclear Power Plant
1650 Calvert Cliffs Parkway
Lusby, Maryland 20657-4702

RE: Proposed Calvert Cliffs Emergency Action Levels

Dear Mr. Fick:

The Calvert County Emergency Management Agency understands and agrees with the proposed revisions to the Emergency Action Levels and Technical Basis Document, Revision 0, for the Calvert Cliffs Nuclear Power Plant. We appreciate the opportunity to participate and comment on the proposed revision.

Sincerely,

John Robert Fenwick, Director
Calvert County Emergency
Management Agency

ATTACHMENT (5)

EAL SUPPORTING DOCUMENTATION (CD)

Calvert Cliffs Nuclear Power Plant, LLC
February 1, 2011

Constellation Energy

Constellation Nuclear Generation Station Administrative Procedure

CNG-OP-1.01-2003

ALARM RESPONSE AND CONTROL

Revision 00100

This Procedure is Applicable for 10 CFR 50.59 / 10 CFR 72.48 Reviews

Tech Spec Related

INFORMATION USE

Applicable To:

- ☒ Calvert Cliffs Nuclear Power Plant, Unit 1 and 2
- ☐ Nine Mile Point Nuclear Station, Units 1 and 2
- ☐ R.E. Ginna Nuclear Power Plant
- ☐ Corporate Offices of CNG

Sponsor: Manager - Operations (CCNPP)

Approval Authority: Manager - Operations (CCNPP)

SUMMARY OF ALTERATIONS

Revision	Change	Summary of Revision or Change
001	00	Converted procedure to the CNG-PR-1.01-1002, Control of Administrative Procedure Format and Content

Immediate Change – Final Approval

Coversheet & Section 1.2.C – Removed R.E. Ginna Nuclear Power Plant and Nine Mile Point Nuclear Station. This procedure revision is applicable ONLY to Calvert Cliffs Nuclear Power Plant.

Section 5.3, Added

- Local panel tests (except Emergency Diesel Generators) should be staggered throughout the week. However, each local panel shall be tested at least once per week (normally tested on the night shift).
- Emergency Diesel Generator remote panels shall be tested daily.

PCR 2008-0219 – to reduce operator burdens and distractions from excessive alarm annunciator checks.

TABLE OF CONTENTS

1.0 INTRODUCTION

1.1 Purpose

A. The purpose of this procedure is to ensure that Constellation Nuclear Generation (CNG) Alarm Response and Control Activities are conducted in a consistent professional manner that contributes to safe and reliable station operation.

1.2 Scope/Applicability

A. This procedure establishes the organizational and individual responsibilities of the Operations Department and provides administrative instructions necessary for the daily conduct of operator response to alarms and alarm control.

B. This procedure applies to all licensed and non-licensed operators and all personnel supporting (directly or indirectly) the Operations Department.

C. This procedure applies to:

- Calvert Cliffs Nuclear Power Plant (CCNPP)

2.0 REFERENCES

2.1 Developmental References

A. CNG-HU-1.01-1001, Human Performance Tools and Verification Practices

B. 10 CFR 50.54, Conditions of Licenses

C. CNG-CA-1.01, Corrective Action Program

D. CNG-HU-1.01, Human Performance Program

E. Technical Specifications (TS)

F. INPO Best Practices Document

G. Station Holds/Safety Tagging

2.2 Performance References

A. None

3.0 DEFINITIONS

3.1 Black Board

All annunciator windows extinguished under full power operations. This is the case at Calvert Cliffs and Nine Mile Point Nuclear Stations. Due to the original design, some Ginna Station annunciators are lit. These alarm windows have been fitted with green covers, and appear green when the alarm window is illuminated.

3.2 Black Dot

The sticker that is placed on an annunciator window or flag to indicate the following:

A. A maintenance activity in the station that causes an alarm on a repeated basis.

B. For identification of a locked in alarm that is caused by a current station configuration due to maintenance in the field.

C. For placement on alarm windows of nuisance alarms with the approval of the Control Room Supervisor (CRS).

3.3 Blue Dot

The sticker placed on an annunciator window to indicate that it is locked-in or annunciator alarm (nuisance) that was taken out of service.

3.4 Compensatory Actions

Actions that are implemented to compensate for an annunciator or recorder point that is non-functional or has one or more non-functional inputs.

3.5 Expected Alarm

An alarm that occurs in the normal course of equipment operation or testing shall be considered expected if the alarm is discussed with the CRS before receipt. Some examples are:

A. Testing of components or equipment calibration.

B. Starting or securing equipment.

C. Alarm as identified in a procedure.

D. Nuisance alarms.

E. Alarms discussed during a pre-evolution brief or prior notification received.

3.6 Nuisance Alarm

Alarms may be considered a nuisance if the alarm is:

- Not valid for existing station, system, or equipment conditions
- A result of a loop, circuit, or equipment failure
- Although valid, repeated actuation of an alarm that distracts operators

3.7 Out of Service (OOS)

An annunciator or recorder point is considered out of service when the alarm circuitry has been disabled or has all inputs removed.

3.8 Priority 1 Alarm

A process computer designator at Ginna Station for computer alarms labeled with priority importance.

3.9 Red Dot

The sticker placed on an annunciator window to indicate that it is part of a Tag Out (according to Site Specific Safety Tagging Procedure).

3.10 Unexpected Alarm

Any alarm that does not meet the criteria of an expected alarm.

3.11 Yellow Dot

The sticker placed on an annunciator window to indicate that one or more inputs to a multiple input annunciator are out of service.

4.0 RESPONSIBILITIES

4.1 The following individuals have been assigned responsibilities within this procedure:

A. Shift Manager (SM)

B. Control Room Supervisor (CRS)

C. Reactor Operator (RPO)

D. Plant Operator/Auxiliary Operator (PO/AO)

E. Shift Technical Advisor (STA)

5.0 PROCESS

5.1 Response to Alarms

A. Operators shall respond promptly to alarm conditions to avoid unwanted or emergency situations, or to mitigate the consequences of an incident or transient.

1. Response to an unexpected alarm – if a Control Room annunciator actuates unexpectedly, then the responding operator performs the following:

    a. Identify the alarm by scanning the annunciator panels.

    b. RPO reports the alarm window to the CRS (or person with Command and Control), paraphrase is acceptable.

    c. Perform Alarm Response actions. Review the associated alarm response even if the alarm clears before the procedure can be completed.

    d. When an alarm annunciated more than once during a shift, it is not required that the alarm response procedure be referenced more than once.

2. Response to expected alarms – when a Control Room annunciator actuates, then the responding operator performs the following:

    a. Identify the alarm by scanning the annunciator panel.

    b. RPO reports the alarm window to the CRS (or person with Command and Control), paraphrase is acceptable.

    c. Referencing the alarm response procedure is not required.

3. Response to Nuisance Alarms 0 if a Control Room annunciator actuates frequently due to station conditions/equipment deficiencies, then the responding person performs the following:

    a. The CRS classifies the alarm as a nuisance alarm.

    b. Ensure appropriate flagging device (black) is installed on annunciator window.

    c. If the alarm repeats, then additional alarm description report to the CRS and reference to the alarm response procedure are not required.

    d. Ensure efforts have been initiated to correct the condition causing the frequent alarm.

4. Response to Plant Process Computer Systems (PPCS) Alarms (Ginna Station)

   a. An Operator or Shift Technical Advisor (STA) reports the PPCS alarm to the CRS (person identified with command and control).

   b. State if the alarm is designated Priority 1.

   c. If the alarm is anticipated as a result of station operating conditions, then the alarm should be called out as expected.

   d. If the alarm is unexpected, the appropriate alarm response procedure (ARP) shall be referenced and the required actions taken. (This may be done by any member of the Control Room staff.)

   e. Inform the CRS or SM of what actions are being taken.

   f. Submit a Condition Report, if required.

5. Response to Multiple Alarms Due to Operational Transients or Emergencies – These alarms may be the result of a station transient, electrical bus malfunction, equipment failure, and so forth. The following steps outline the expected actions:

   a. RPOs announce that multiple alarms have received, stating the cause if it has been diagnosed.

   b. The CRS directs or ensures the RPOs monitor the stations. Typical parameters to monitor are listed below:

      - PWR – Reactor power, RCS temperature, primary pressure, Net Megawatts.

      - BWR – Reactor power, Reactor pressure, Reactor level.

   c. The CRS (or person with Command and Control) directs or ensures the correct procedure is implemented to address the transient, based on their diagnosis of Main Control Board (MCB) alarms, and station monitoring results.

   d. RPOs are expected to take manual actions for automatic actions that did not occur as a result of the malfunction, or are occurring that should not (that is, unexpected control rod movement).

   e. RPOs take manual actions for auto actions that did not occur as a result of the malfunction, or are occurring and should not be (that is, control rod movement).

   f. RPOs periodically evaluate alarm panels to verify active alarms are consistent with station conditions, and implement alarm response procedures for those that are not consistent, as time permits. Results of these evaluations should be included in shift briefings.

g. Reactor operators take manual actions for auto actions

6. Guidance for Official Record entries

a. Unexpected Control Board and/or Priority 1 PPCS (Ginna) alarms received requiring operator action shall be entered in the Station Log except as follows:

- Common alarms from alarm panels located outside the Control Room do not need to be logged.
- Alarms recorded in other procedures do not need to be logged.

b. If multiple alarms occur, only the most significant alarm(s) should be logged. The alarm and follow up actions are to be logged AFTER the operating crew responds to the alarm.

5.2 Annunciator and Recorder Point Control and Flagging

A. The CRS shall determine whether an annunciator or recorder point(s) requires controls due to:

- Maintenance Order (MO)/Work Order (WO)
- Nuisance alarm
- Circuit failure
- Alarm in solid for extended time and NOT providing useful information regarding system status
- Removal of inputs (alarms which receive multiple inputs)
- Safety Tagging
- Condition Report

B. Determine required compensatory actions **[FB0173] [FB0109]**

1. When an annunciator or recorder point is out of service, compensatory actions should be considered if any of the following conditions apply:

- The annunciator/recorder point is required to monitor component/system availability or operability.
- The annunciator/recorder point monitors the performance or condition of operating equipment or equipment that is available for operation.

5.2.B.1 (Continued)

- The annunciator/recorder point is utilized in the Abnormal or Emergency Operating Procedures (AOPs/EOP's/SOP's) for verification or action initiation.
- No other annunciator/recorder point or computer alarm point is available to monitor the affected parameter.
- The annunciator has multiple inputs and is locked-in.

2. When compensatory actions are established, consideration should be given to the following:
   - An alternate means for monitoring the affected parameter.
   - Frequency for monitoring the parameter and identification of parameter limits.
   - Potential for the affected parameter to change and adverse effects on system/plant operation.
   - Temporary log changes to implement the compensatory actions, if required.
   - Actions that may be required during the implementation of an AOP or EOP with the Annunciator/recorder point out of service.

3. If required, implement compensatory actions as follows:
   a. Incorporate the compensatory actions in the appropriate operator logs with a temporary log change, if applicable.
   b. If the compensatory actions are required through shift turnover, indicate that those actions are in effect in the appropriate section of the Shift Turnover Information Sheet.

4. Authorize placing the annunciator/recorder point out of service by documenting compensatory actions and initialing the appropriate block on Attachment 1, Alarm Annunciator/Recorder Point Out of Service Log. If compensatory actions are not required, then document the justification for not implementing those actions on Attachment 1.

C. Place the annunciator/recorder point out of service as follows, if required:

1. Defeat the annunciator/recorder point using the appropriate station process. Examples include:
   - Removing the alarm card
   - Remove fuses

5.2.C.1 (Continued)

- Open Annunciator slide link
- Lifted lead
- DIP (dual in-line package) switch
- Open knife switch
- Pulling Alarm Relay

2. Ensure Attachment 1 is completed up to, not including verification.

NOTE

Yellow dots are not required if an input is removed from any common/repeater panels.

NOTE

CRS discretion can be used to determine if a black dot is required (lower mode, frequency of alarms, and so forth).

D. Place appropriate flagging tool on the Annunciator window:

1. Flagging Tool

- Black – Maintenance or nuisance
- Blue – Locked in or removed from service
- Yellow – One or more inputs removed from service
- Red – Out of service due to safety tagging

2. Inform the CRS when the flagging tool has been installed.

E. The Independent Verifier shall: **[FB0172]**

1. Verify annunciator window has appropriate flagging tool.

2. If the annunciator has been removed from service, then verify the annunciator is OOS by performing or observing an annunciator test.

3. If a recorder point was removed from service, then verify the proper switch was utilized to disable the point.

5.2.E. (Continued)

4. Initial and date as the verifier on Attachment 1.

F. Notify the SM within the same shift of the annunciator status/flagging.

G. Annunciator/recorder points are returned to service as follows:

1. The CRS shall direct the operator to restore the annunciator/recorder point.

2. The Operator shall:

a. If an alarm card was removed, then verify that the alarm card switch or jumper is in its normally open or normally closed position.

(1) A Concurrent Verifier shall be present to verify switch/jumper position.

b. Restore the annunciator/recorder point using the appropriate station process. Examples include:

- Install alarm card
- Install fuses
- Land lead
- Close slide link
- Reset recorder DIP switch
- Close knife switch
- Reinstalling Alarm Relay

c. Observe panel to be tested and note current status of all panel annunciators.

d. Test panel annunciators.

e. Confirm panel annunciators respond as designed.

f. Remove the flagging tool.

g. Complete Attachment 1 documentation.

h. Notify the CRS that the annunciator is restored.

5.2.G. (Continued)

3. Independent Verifier shall: **[FB0172]**

   a. Verify annunciator window flagging tool removed.

   b. If the annunciator has been returned to service, then verify the annunciator is restored by performing or observing an annunciator test.

   c. If a recorder point was returned to service, then verify the proper switch was utilized to restore the point.

   d. Initial and date as the verifier on Attachment 1.

   e. Notify the CRS that the annunciator has been verified.

4. The CRS shall remove applicable compensatory actions by:

   a. Recording the restoration in the appropriate operator log(s).

   b. Deleting the applicable entries from the Shift Turnover Information Sheet.

5.3 Annunciator Test

A. Annunciators will be tested at the following frequency:

- MCB panels – 1/shift
- Local panel tests (except Emergency Diesel Generators) should be staggered throughout the week. However, each local panel shall be tested at least once per week (normally tested on the night shift).
- Emergency Diesel Generator remote panels shall be tested daily.

B. Operators shall inform the CRS or RPO before testing panels outside the Control Room. These alarms will be treated as expected alarms.

C. Operators shall observe panel to be tested and note current status of all panel annunciators prior to testing the panel.

D. During the panel test, operators should confirm panel annunciators respond as designed.

E. A condition report shall be submitted for any discrepancies noted during the annunciator tests.

F. Operators shall inform the Control Room of the completion of local panel tests.

G. *Document completion of test on a station specific attachment or similar form per example on Attachment 2, Panel Test Log.*

H. Replace burned out bulbs with approved replacement.

I. Operators shall inform Control Room upon completion of testing in-plant alarms.

6.0 BASES

[FB0109] SOER 94-02, Boration, Dilution Events in Pressurized Water Reactors; Recommendation 4b.

[FB0172] INPO 85-016/85-031, Section 7/96-008, Chapter 8, Temp Mod control (Installations of Temp Mods be verified independently).

[FB0173] SOER 02-3, Large Power Transformer Reliability, Rec. 3.b.4, appropriate compensatory monitoring practices when alarms are OOS or sealed in for other reasons.

7.0 RECORDS

7.1 The following records are generated by use of this procedure and are controlled by CNG-PR-3.01-1000, Records Management:

A. Alarm Annunciator/Recorder Point OOS Log Sheets, Attachment 1

B. Annunciator Panel Test Log Sheets, Attachment 2

Attachment 1, Alarm Annunciator/Recorder Point Out of Service Log

Date	Window No. or Rcdr. No.	Window Name or Rcdr. Point	Reason for Placing Annunciator Out of Service or Bypassing Rcdr. Point	Compensatory Actions or Reason No Comp Actions are Required	CRS Auth. (Initials/Date)	Alarm Card Switch/Jumper Position NO/NC/NA	Ann. Input/Rcdr. Pt. OOS (Date/Initials)	Ann. Input/Rcdr. Pt. Verified OOS (Date/Initials) **[FB0172]**	Ann. Input/Rcdr. Pt. Returned to Service	Ann. Input/Rcdr. Pt. Verified Returned to Service **[FB0172]**

Notes:

1. Ensure Shift Manager is notified and concurs within the same shift.
2. This log shall be maintained in the Control Room.
3. Forward completed log sheet to GS-Operations.
4. Ginna only Category #3.3.45.

Page 1 of 2

Attachment 2, Panel Test Log

Date														
Shift	1	2	1	2	1	2	1	2	1	2	1	2	1	2
MAIN CONTROL BOARD (RIGHT)														
MAIN CONTROL BOARD (MIDDLE)														
MAIN CONTROL BOARD (LEFT)														
FIRE DETECTION PANELS														
METAL IMPACT MONITOR														
TURBINE PLANT SAMPLE RACK *														
STEAM HEADER SUPPORTS PANEL *														
MAIN TRANSFORMER 11 *														
STATION AUX TRANSFORMER 12A *														
STATION AUX TRANSFORMER 12B *														
STATION AUX XFMR 12A REMOTE *														
STATION AUX XFMR 12B REMOTE *														
DIESEL GEN A ELCP *														
DIESEL GEN A CONTROL PANEL *														

Attachment 2, Panel Test Log (Continued)

Date														
Shift	1	2	1	2	1	2	1	2	1	2	1	2	1	2
DIESEL GEN B CONTROL PANEL *														
AVT CONDST DI CONTROL PANEL *														
HYDROGEN PANEL *														
PRIMARY WATER TREATMENT PANEL *														
PENETRATION TEMP ALARM PANEL *														
WASTE PANEL *														
BORIC ACID PANEL *														
VOLTAGE REGULATOR ALARM PANEL *														
RWST HIGH LEVEL INDICATION LAMP *														
GENERATOR STATOR WINDING TEMP PANEL *														
SECURITY DIESEL GENERATOR ALARM PANEL *														
ANNUNCIATOR PANEL TEST LOG SHEET														

N/A if not in service
* Test once per day normally on night shift, other shifts may be marked N/A.
Ginna only Category #3.3.25.

Constellation Energy

Calvert Cliffs Nuclear Power Plant
Station Administrative Procedure

NO-1-113

CONTROL OF RADIO TRANSMITTERS

Revision 00500

Tech Spec Related

INFORMATION USE

Applicable To:

- ☒ Calvert Cliffs Nuclear Power Plant, Unit 1 and 2
- ☐ Nine Mile Point Nuclear Station, Units 1 and 2
- ☐ R.E. Ginna Nuclear Power Plant
- ☐ Corporate Offices of CNG

Sponsor: General Supervisor-System Engineering (CCNPP)

Approval Authority: General Supervisor - Shift Operations (CCNPP)

SUMMARY OF ALTERATIONS

Revision	Change	Summary of Revision or Change
005	00	Section 1.1 – Deleted information pertaining to Trip Sensitive items. Section 1.2.A - Deleted information pertaining to Trip Sensitive items. **Section 5.6** - Management Expectations for Plant Personnel Entering Trip-Sensitive Areas has been deleted due to the implementation of CNG-OP-1.01-1000 Rev. 00200. RPA 2008-0849 Changed the title of the procedure to delete the Trip Sensitive Area part. Deleted definitions associated with Trip Sensitive Areas and Equipment.

TABLE OF CONTENTS

SECTION TITLE **PAGE**

1.0 INTRODUCTION

1.1 Purpose [B0136]

A. This procedure establishes the requirements and Management's expectations for use of Radio Transmitters (RTs) at Calvert Cliffs. Radio frequency interference (RFI) from RTs can cause induced voltage/current signals in electronic circuitry. Depending on the function of the electronic circuitry, an unexpected plant transient may occur. Restrictions on the type and use of RTs are necessary to prevent plant transients that challenge plant operators and safety systems.

1.2 Scope/Applicability

A. This procedure applies to all users of Radio Transmitters.

B. The established controls are applicable during all operating modes. This procedure also serves as the technical procedure for portable voice radios and cell phones, unless noted otherwise.

2.0 REFERENCES

2.1 Developmental References

A. INPO SER 90-6, Plant Transients and Engineered Safety Feature Actuations Caused by Radio Frequency Interference

B. Federal Communications Commission licenses and regulations

C. Letter, G: PES911111-302; November 11, 1991; closure of POSRC OI 91-081-03

D. CNG-PR-1.01-1011, Control of Station-Specific Procedure Change Process

E. NO-1-100, Conduct of Operations Shift Activities

2.2 Performance References

A. EN-1-100, Engineering Service Process Overview

B. CNG-OP-1.01-1005, Temporary Notes, Operator Aids, and Permanent Labels

C. CNG-PR-1.01, Procedures Program

D. CNG-PR-1.01-1011, Control of Station-Specific Procedure Change Process

E. CNG-PR-3.01-1000, Records Management

3.0 DEFINITIONS

3.1 Engineered Safety Features (ESF) Trip Sensitive Areas

Areas affected by RFI, as defined in NO-1-113 that could actuate Engineered Safety Features Actuation System (ESFAS) components.

3.2 Portable Radio Transmitter (PRT)

A head-set or hand-held device which transmits radio frequency signals. There are two basic types of PRTs used at Calvert Cliffs.

1. Portable Voice Radios

   a. Hand-held radios which are used for 2-way voice communications throughout the plant and surrounding properties (for example: Security, Operations and Emergency Planning Unit). They operate at no more than 6 watts effective radiated power on frequencies licensed to CEG by the FCC. They may utilize radio repeaters connected to an antenna system which consists of a combination of in-plant radiax antenna and outdoor antenna system.

   b. Head-set radios which are used for 2-way voice communications in hands free applications. They may not be licensed and typically operate at no more than 100 milliwatts effective radiated power. They are strictly for portable-to-portable use; no base stations or repeaters can be used. An example of this application is the ALARA radio headsets.

2. Special Purpose Radio Controllers

   a. These devices are used to remotely control equipment (for example: polar cranes). They may operate on licensed or unlicensed frequencies and the power may vary with the application.

   b. The radio controllers are transmitting devices only and their area of coverage is very limited. Care must be taken when specifying frequencies to ensure that there is no co-channel interference with other radio controlled equipment.

   c. Wireless data transmission devices that are used to transmit and receive data in conjunction with the LAN or WAN. They operate at no more than 500 milliwatts effective radiated power. They may utilize repeaters for increased plant coverage.

   d. Cordless telephones, including a head set or hand set radios, which are used for 2-way voice communications. Communications between hand set and base where the base is connected to the plant telephone system.

   e. Cellular telephones used for 2-way voice communications. No license required.

3.3 Portable Radio Transmitter User

An authorized user/owner who is responsible for the conduct and operation of the Portable Radio Transmitter(s), while in their possession, per this procedure.

3.4 Radio Frequency Interference (RFI)

RFI is any radio frequency energy that is generated by a device which may cause adverse effects on the operation of another device. The generating device may be an incidental radiator (d.c. motors, power lines, light switches) or an intentional radiator (radios, remote controls, cellular phones, radar).

3.5 Radio Frequency Interference (RFI) Trip Sensitive Equipment

RFI Trip Sensitive Equipment is equipment that may actuate under the influence of RFI.

3.6 Radio Transmitter (RT)

Any device which is designed to transmit radio frequency signals.

4.0 RESPONSIBILITIES

4.1 General Supervisor – Plant Engineering Section (GS-PES)

A. The GS-PES responsibilities include ownership of the program which controls Radio Transmitters at Calvert Cliffs.

4.2 Engineer Supervisor – Electrical & Control Systems Engineering Unit (ES-E&C SEU)

A. The ES-E&C responsibilities include:

1. Administration of the program which controls use of Radio Transmitters.

2. Appointing the Site Coordinator for Radio Transmitters.

4.3 All General Supervisors

A. All General Supervisors responsibilities include ensuring:

1. Radio Transmitters owned by their section (CEG and contractors) are authorized by the Site Coordinator.

2. Each Radio Transmitter is authorized by the CGG Field Services.

3. All personnel (CEG and contractors) who are issued RTs are familiar with this procedure.

4. Development of implementing procedures/policies for Special Purpose Radio Controllers used by their section.

4.3.A (Continued)

5. All PRTs brought inside the protected area should be used for company business purposes only.

4.4 Site Coordinator for Radio Transmitters

A. The Site Coordinator responsibilities include the following:

1. Maintaining a record of all authorized PRTs with the information found on Attachment 2, Portable Radio Transmitter Assessment .

2. Coordinating the assessment of requests for new PRTs including the definition of restriction for use.

3. Authorizing the use of new PRTs following FCC Licensing and development of an implementing procedure by the user.

4. Providing cross-disciplinary review of implementing procedures/policies for Special Purpose Radio Controllers.

5. Forwarding copies of PRT Assessment Forms to the E&C Design Unit.

4.5 Engineer Supervisor & Controls Design (ES-E&C Design)

A. The ES-E&C Design responsibilities include supporting the Site Coordinator in:

1. Identifying those field installed components which could be affected by RFI from PRTs.

2. Assessing the potential affects of RFI on components from PRTs.

4.6 Supervisor-CGG Field Services

A. The Supervisor –CGG Field Services responsibilities include:

1. Authorizing the procurement of new PRTs and determining what FCC requirements must be met.

2. Supporting those groups requesting new PRTs in determining an acceptable manufacturer and model, and obtaining FCC Licenses.

3. Assisting in the development of implementing procedures/policies for Special Purpose Radio Controllers.

4.7 General Supervisor – Shift Operations (GS-SO)

A. The GS-SO responsibilities include:

1. Providing support to the Site Coordinator in identifying those components which could be affected by RFI from PRTs.

2. Ensuring those areas which have been identified as being susceptible to RFI are clearly posted, according to CNG-OP-1.01-1005, Temporary Notes, Operator Aids, and Permanent Labels.

5.0 PROCESS

5.1 General Restrictions on Radio Transmitters

A. Radio Transmitters mounted on vehicles (for example: radios, cellular phones, CBs, radar) shall be turned off prior to the vehicle entering the protected area except for:

1. Site fire truck and site security vehicles which may transmit within the protected area, but must be outdoors (typically near the tank farm).

2. Radio transmitters mounted on off-site emergency vehicles (fire trucks, ambulance, police).

B. Areas identified as being susceptible to RFI have restrictions on use of PRTs. Rooms which are susceptible to RFI shall be posted according to CNG-OP-1.01-1005, Temporary Notes, Operator Aids, and Permanent Labels.

1. Personnel carrying PRTs shall ensure that it is turned off prior to entering the following areas, unless the PRT is specifically authorized for use in the area:

   a. Control Room.

   b. 500KV Switchyard Control House.

   c. Cable Spreading Rooms.

   d. 45' and 72' Computer Rooms

2. Personnel carrying RTs shall not transmit or receive messages within 10 feet of (unless testing has been performed to establish a less restrictive boundary)

   a. Remote shutdown panels in each Unit's 45' Switchgear Room.

   b. 5' and 27' Auxiliary Building West Penetration Room and 27' Letdown Heat Exchanger Room pressure transmitters.

   c. Containment pressure transmitters located in the 45' Auxiliary Building East and West Penetration Rooms.

   d. Designated locations in the U-1 North and U-2 South, 69' SFP areas.

   e. U-2 SG Feed Pumps.

   f. Any electronic control cabinet or process measurement transmitter.

5.1 (Continued)

C. Use of PRTs in Containment shall be limited as follows, unless specifically authorized by the Site Coordinator:

1. Approval to have a PRT inside containment shall be obtained from the Shift Manager and only if the Reactor Trip Circuit Breakers are open.

2. Any PRT used within containment shall be turned off while transporting it to and from its area of use. **[B0136] [B0568]**

3. On the 45' level of Containment, PRT usage shall be limited to the immediate vicinity (within 5 feet) of the Equipment Hatch or Emergency Hatch. **[B0568]**

D. Use of cellular phones shall be limited as follows, unless specifically authorized by the Site Coordinator:

1. Cell phones may be used in office areas and outside areas where trip sensitive equipment does not exist.

   a. Trip Sensitive Equipment is identified with trip sensitive labels or floor marking per CNG-OP-1.01-1000, Conduct of Operations.

2. Cell phones specifically shall **not** be used in the Turbine Building, Aux. Building, Control Room or Containment and must be turned off when in these areas.

5.2 Use of Portable Radio Transmitters

A. Only authorized PRTs shall be used within the protected area.

1. A controlled list of authorized PRTs shall be maintained by the Site Coordinator (except for cellular phones).

2. This list shall include:

   - Manufacturer
   - Model Number
   - Assigned Calvert Cliffs Number
   - Frequency(ies)
   - Power
   - Owner
   - Any Imposed restrictions, and the implementing procedure

5.2.A (Continued)

3. Cellular phones may be used inside the protected area without approval per NO-1-113, General Restrictions on Radio Transmitters, Section 5.1, shall be followed when using cell phones.

B. If a new PRT is needed within the protected area, the Site Coordinator's authorization shall be obtained as described in 5.3.

C. The General Supervisor of the section using authorized PRTs, shall ensure a procedure/policy which describes the limitations or requirements for using the PRT is implemented, if required by Site Coordinator.

1. For Special Purpose Controllers the procedure shall be prepared according to CNG-PR-1.01-1011, Control of Station-Specific Procedure Change Process.

D. In addition to the general restrictions on PRTs described in 5.1, the following requirements shall also apply to Portable Voice Radio usage:

1. Radio communications should be brief. When possible, use the phone system.

2. Sender and receiver identification shall be included in each message.

3. When taking PRTs inside contaminated areas, the potential for inadvertent transmissions exists due to protective clothing and anti-contamination controls. To prevent inadvertent transmissions, radios should be turned off when not needed, or carried by holding the bag in lieu of gripping the radio within the bag. **[B0568]**

5.3 Authorization of New Portable Radio Transmitters

NOTE

New Portable Radio Transmitters (for example: portable voice radios and special purpose radio controllers) must be approved prior to use within the protected area. Specific approval is necessary to ensure the assumptions made in establishing the general restrictions for RTs (Section 5.1) remain valid. Portable Radio Transmitters with power levels or frequencies which are different then previously assessed could affect equipment/components in areas which were previously identified as not being susceptible to RFI.

A. Organizations requesting permission to use a new PRT shall complete and forward Attachment 1, Request for New Portable Radio Transmitters to the Site Coordinator.

B. The Site Coordinator shall complete shall complete Attachment 2, Portable Radio Transmitter Assessment, to identify any equipment/components which could potentially be affected by the new PRT.

1. Identification should include a review of prints and walkdowns of the areas where the PRT will be used.

5.3.B (Continued)

2. Operations, E&C Design and CGG Field Services shall provide support to the Site Coordinator, as requested.

3. On approval for use, the Site Coordinator shall forward a copy of Attachment 2 to the requesting organization.

C. The requesting organization shall develop a technical procedure, if required by the site coordinator, for new Special Purpose Radio Transmitters according to CNG-PR-1.01-1011.

1. Specific restrictions recommended by the Site Coordinator on Attachment 2 shall be used.

2. The Site Coordinator shall provide cross-disciplinary review according to CNG-PR-1.01, Procedures Program.

D. Following procedure approval, a copy of the procedure shall be submitted to the Site Coordinator.

E. The Site Coordinator shall update the controlled list of authorized PRTs.

5.4 Changes of Limitations/Restrictions of Authorized PRTs

A. All changes to the restrictions contained within this procedure or a Special Purpose Radio Controller's technical procedure shall be reviewed by the Site Coordinator.

1. The assessment of new restrictions shall be documented on Attachment 2.

2. Posted area signs shall be changed according to CNG-OP-1.01-1005, if required.

5.5 Assessment of a New Component's Susceptibility to RFI

A. Each new component shall be assessed for susceptibility to RFI according to EN-1-100, Engineering Service Process Overview.

1. The Site Coordinator's controlled list of PRTs and the location of the component shall be used to determine whether additional restrictions are required for existing PRTs.

B. If additional restrictions are required for PRTs, changes to this procedure shall be initiated according to CNG-PR-1.01-1011.

1. Changes to technical procedure(s) for Special Purpose Radio Transmitters shall be initiated according to CNG-PR-1.01-1011.

6.0 BASES

[B0136] LER 50-318/91-03 and NCR# 12095 – Development of a Site Procedure for Control of RTs. RTs are not allowed inside containment without the Shift Manager's permission. RTs are not allowed in containment. IV Corrective Actions; Actions to Prevent Recurrence; 2.

[B0568] 50.59 Evaluation No. 93-B-999-004-R00.

7.0 RECORDS

A. The following records are generated by the use of this procedure and shall be captured and controlled according to CNG-PR-3.01-1000, Records Management.

1. Completed Attachment 2 for each PRT shall be maintained until use of the PRT is discontinued and it is removed from the protected area.

Attachment 1, REQUEST FOR NEW PORTABLE RADIO TRANSMITTERS

DESCRIBE THE USE/APPLICATION OF THE PORTABLE RADIO TRANSMITTER:
(If a specific radio is required, include all known information for example: manufacturer, model, power, frequency)

____

____

____

____

____

____

____

PLANT AREAS WHERE THE PORTABLE RADIO TRANSMITTER WILL BE USED:

____

____

____

____

____

____

____

Owner of PRT (Section/General Supervisor):______________________

User(s) of the PRT (Sections):______________________

Point of contact:______________________

Phone Number:______________________

Attachment 2, PORTABLE RADIO TRANSMITTER ASSESSMENT

MANUFACTURER____________________ MODEL____________________

POWER (WATTS)____________________ FREQUENCY____________________

OWNERS OF PRT (Sections)____________________

PURPOSE:____________________

LOCATION OF USE:____________________

PLANNED USAGE PERIOD: ____________________

PLANT CONDITIONS:____________________

SENSITIVE COMPONENTS AT PRT USAGE LOCATION: ____________________

SPECIFIC RESTRICTIONS FOR PRT USAGE:____________________

BASIS FOR RESTRICTIONS:____________________

SUPERVISOR-CGG FIELD SERVICES:____________________DATE:__________
CONCURRENCE

SITE COORDINATOR APPROVAL:____________________DATE:__________

CC#:__________

Constellation Energy

Constellation Nuclear Generation Station Administrative Procedure

NO-1-114

CONTAINMENT CLOSURE

Revision 01700

Tech Spec Related

INFORMATION USE

Applicable To:

- ☒ Calvert Cliffs Nuclear Power Plant, Unit 1 and 2
- ☐ Nine Mile Point Nuclear Station, Units 1 and 2
- ☐ R.E. Ginna Nuclear Power Plant
- ☐ Corporate Offices of CNG

Sponsor: General Supervisor-Shift Operations (CCNPP)

Approval Authority: Manager-Operations (CCNPP)

SUMMARY OF ALTERATIONS

Revision	Change	Summary of Revision or Change
017	00	**Section 5.2** – Steps A.3 – Added "Verify personnel are stationed at the COD once per shift during Refueling or if Shutdown Cooling is not in operation per the requirements of TS 3.9.3." This was added per Supervisory Observation. RPA 2009-0278
		Step C.1 - Added "Verify personnel are stationed at the PAL once per shift during Refueling or if Shutdown Cooling is not in operation per the requirements of TS 3.9.3." This was added per Supervisory Observation. RPA 2009-0278
		Changed Maintenance Order to Work Order throughout the procedure. RPA 2008-1300

TABLE OF CONTENTS

1.0 INTRODUCTION

1.1 Purpose

This procedure provides, in addition to the requirements found in operating procedures and Technical Specifications, administrative controls to be used during lower-mode operations to enhance overall nuclear safety with respect to Containment Closure at Calvert Cliffs Nuclear Power Plant (CCNPP). **[B0594] [B0595] [B0138]**

1.2 Scope/Applicability

A. This procedure embodies management expectations regarding the containment closure aspect of shutdown safety at CCNPP. To accomplish safe and controlled outages, management expects full compliance with all procedures. A questioning attitude, coupled with a strong safety-first ethic adopted by all site personnel, will greatly reduce the potential for fission product release during reactor shutdown conditions at CCNPP.

B. This procedure applies to the preparation for, and the tracking and restoration of Containment Closure during Lower Mode Operations at CCNPP. Such preliminary action prior to the onset of core boiling will immediately and effectively reduce the likelihood of a radiological release.

C. Containment Closure considerations shall remain in effect whenever irradiation fuel is located in the Reactor Vessel or the surrounding Refueling Pool. **[B0594]**

2.0 REFERENCES

2.1 Developmental References

A. NO-1-103, Conduct of Lower Mode Operations

B. OM-1-100, Managing Refueling Outages

C. OP-7, Shutdown Operations

D. CCNPP Technical Specifications

E. INPO 92-005, Guidelines for the Management of Planned Outages at Nuclear Power Stations

F. NRC Generic Letter 88-17, Loss of Decay Heat Removal

G. NUMARC 91-06, Guidelines for Industry Actions to Assess Shutdown Management

2.2 Performance References

A. AOP-3B, Abnormal Shutdown Cooling Conditions

B. AOP-4A, Loss of Containment Integrity/Closure

C. AOP-6D, Fuel Handling Incident

D. DOOR-01, Open and Closing of Containment Outage Door

E. EN-4-104, Surveillance Testing

F. ERPIP-3.0, Immediate Actions

G. NO-1-103, Conduct of Lower Mode Operations

H. CNG-OP-1.01-1007, Clearance and Safety Tagging

I. CNG-OP-1.01-2002, Operations Shift Turnover and Relief

J. OP-7, Shutdown Operations

K. STPO-55A, Containment Closure Verification

L. STPO-108 series, Local Leak Rate Test Procedures

M. CCNPP Technical Specifications

N. BGEDRWGs 83018 (U-1) and 83019 (U-2) Sheets 1, Containment Closure Composite Drawing

O. OI-17D, Miscellaneous Waste Processing System

3.0 DEFINITIONS

3.1 Containment Closure

The action or condition that ensures Containment and its associated systems, structures, or components (SSC), as listed in STPO-55A, Containment Closure Verification, provide a functional barrier to fission product release. **[B0594]**

3.2 Containment Closure Deviation

Any penetration not meeting the requirements of STPO-55A, Containment Closure Verification.

3.3 Containment Closure Restoration

The reinstatement of functional barriers to radioactive release per STPO-55A, Containment Closure Verification.

3.4 Containment Closure Tracking

The monitoring of functional barriers to radioactive release providing the flexibility to have the containment building open under appropriate conditions.

3.5 Containment Penetration Status Tracking Board

Board modeled after STPO-55A-1(2) to demonstrate piping and valves for containment penetrations used for containment closure.

3.6 Defueled

The condition when all fuel assemblies have been removed from the reactor vessel and the surrounding refueling pool.

3.7 Lower Mode Operations

Modes 5 and 6 as defined in the Tech Specs or the Defueled condition as defined above.

3.8 Lower Mode Operations Containment Pressure [B0482]

The maximum pressure the Containment may reach with the following conditions:

1. Loss of core cooling due to a Station Black Out (SBO)
2. Initiation of boiling
3. One day after shutdown

   - **12 PSIG** with a one hour time to start DG, align it to a bus, and reestablish Core or Containment Cooling.

3.9 Reduced Inventory

Condition when the reactor vessel contains irradiated fuel assemblies and RCS water level is at or below the 41 foot elevation.

3.10 Restricted Containment Closure Conditions

Those lower mode conditions where higher levels of Containment Closure control are required by this procedure: Movement of irradiated fuel in the containment (T.S. 3.9.3)

- Reduced Inventory (GL 88-17)
- No Shutdown Cooling (SDC) available (T.S. 3.9.4/3.9.5)

3.11 Steam Generator Availability

A S/G may be considered available for heat removal when all of the following conditions exist (additional conditions are required for OPERABILITY):

- Secondary side actual water level is above -40 inches
- Secondary side intact (for example: no openings)
- Associated Atmospheric Dump Valve is available to relieve steam
- A Steam Driven Auxiliary Feed Water train aligned to Main Steam is available for makeup
- The RCS is capable of being pressurized
- It is preferred to have the S/G tubes full but not required. With the tubes not full, the Time to Boil (TTB) would be less and equilibrium temperature reached in a long-term Loss of Shutdown Cooling (SDC) is higher.

3.12 Temporary Containment Closure Device

Any device used to satisfy containment closure requirements in lieu of STPO-55A, Containment Closure Verification, barriers. Temporary Containment Closure Devices shall be capable of withstanding Lower Mode Operations Containment Pressure. **[B0138]**

3.13 Time to Boil(TTB)

Time for the RCS/Refueling Pool mass to reach bulk boiling after loss of SDC as determined by the Figures in OP-7, Shutdown Operations.

3.14 Yellow Caution Tag

Definition per CNG-OP-1.01-1007, Clearance and Safety Tagging.

4.0 RESPONSIBILITIES

4.1 Operations is responsible for:

A. Overall closure of the Containment including inside and outside alignments.

B. Tracking and monitoring Containment Closure status by utilizing NO-1-114, Containment Closure, and CNG-OP-1.01-1007, Clearance and Safety Tagging (STPO-55A, Containment Closure Verification, is an implementing tool).

C. Verifying that all valve alignments necessary to restore Containment Closure within the Time to Boil (TTB), if both S/Gs are not available, and T.S. 3.9.4/3.9.5. This verification can be by an administrative review of Deviation Sheets and/or actual walkdowns.

4.2 Radiation Protection is responsible for:

A. Containment evacuation upon initiation of Containment Closure restoration.

4.3 Mechanical Maintenance is responsible for:

A. The restoring of equipment and personnel hatches, Main Steam Safeties and temporary covers, and other compensatory measures associated with mechanical maintenance on Containment Systems, Structures, or Components (SSC).

4.4 Maintenance Planning and Work Control is responsible for:

A. Identifying all work activities which could affect Containment Closure.

4.5 Outage Management is responsible for:

A. Coordinating closure of the Equipment Hatch Door (EHD), Containment Outage Door (COD) and Personnel Airlocks and ensuring tools, materials, and other equipment such as lighting and rigging are properly staged.

B. Testing the outage organizations' ability to restore Containment Closure by running drills.

C. Ensuring specific responsibilities for the Equipment Hatch and Containment Outage Door are conveyed each shift (liaison with Mechanical Maintenance).

D. Verifying capability exists to close required hatches is *less than* the Time to Boil (TTB) and T.S. 3.9.4/3.9.5 time requirements. (Assume the time to close the Equipment Hatch is approximately 65 minutes with personnel and equipment on station. Add an additional 15 minutes if equipment is stationed and personnel are not.)

E. Being cognizant of all potential hatch/airlock/COD obstructions, deviations, and restoration plans, including COD closure swing path remain clear or swing path can be cleared within required closure times.

F. Ensuring enough manpower is available and coordinating manpower augmentation if necessary.

4.6 Containment Closure Team

A. The team is responsible for restoring Containment Closure in *less than* the Time to Boil (TTB) and T.S. 3.9.4/3.9.5 time requirements.

B. The team consists of those members assigned Containment Closure restoration responsibilities per Deviation Sheets:

- Instrument Maintenance
- Mechanical Maintenance
- Operations
- Outage Management
- Radiation Protection
- System Engineering
- Contract Sponsorship

C. The team's activities are coordinated by the Control Room and Outage Management.

4.7 Personnel involved with Local Leak Rate Testing (LLRT) are responsible for:

A. Continuously attending penetrations where testing is in progress and for being prepared to suspend testing and isolate valves as required to restore Containment Closure.

4.8 All other personnel in Containment and not on the Containment Closure Team are responsible for:

A. Upon notification, leaving their work in a safe condition and evacuating.

5.0 PROCESS

5.1 Discussion

A. The various Shutdown Safety conditions as defined in NO-1-103, Conduct of Lower Mode Operations, are based on relative risk and controlled accordingly with the respective Minimum Essential Equipment Lists in CNG-OP-1.01-2002, Operations Shift Turnover and Relief. Containment Closure conditions as subsets of Shutdown Safety are further simplified by the philosophy of fission product barrier control; the three barriers being fuel cladding, the Reactor Coolant System (RCS), and the Containment. The bases for Containment Closure assume fission products are in the reactor coolant due to some breach of the fuel cladding. Therefore, the RCS and the Containment are the barriers in Modes 5 and 6 that preclude potential fission product release.

B. A prolonged loss of Shutdown Cooling (SDC) or an inadvertent fuel handling incident, are the bases for Containment Closure. Therefore, the ability to close the Containment prior to the Time to Boil (TTB) to withstand Lower Mode Operations Containment Pressure is the major consideration which bounds the above events. If an open penetration *cannot* be adequately controlled in *less than* the TTB and the T.S. 3.9.4/3.9.5 requirements, then the penetration must remain closed. If a penetration *can* be closed in *less than* the TTB and T.S. 3.9.4/3.9.5 requirements, the there are no limits on the duration that the penetration is allowed to be open. **[B0596]**

C. If the RCS is intact (a S/G is available), then Deviation Sheets need to be completed for planned and inadvertent Containment breaches and the ability to restore Containment Closure need only be tracked on the Containment Closure Penetration Status Board.

D. When the RCS is not intact (both S/Gs are not available), the last dependable barrier is the Containment. It is paramount that all Containment penetrations opened to the atmosphere be capable of being closed prior to the Time to Boil (TTB) and T.S. 3.9.4./3.9.5 requirements, in the event of a prolonged loss of Shutdown Cooling (SDC).

1. If a Restricted Containment Closure Condition does not exist, then control of the ability to restore Containment Closure is limited to:
    - Completing Deviation Sheets
    - Tracking penetration status on the Containment Penetration Status Board
    - Monitoring the TTB

2. If a Restricted Containment Closure Condition exists, then strict control of the ability to restore Containment Closure requires:

> **NOTE**
>
> Containment Deviations are not allowed during T.S. 3.9.3/3.9.4 (NOTE 2) conditions.

- Completing Deviation Sheets
- Tracking penetration status on the Containment Penetration Status Board
- Monitoring the TTB
- Performing STPO-55A, Containment Closure Verification
- Hanging yellow caution tags

5.1 (Continued)

E. During Defueled operations, no additional tracking of Containment Closure is required and breaches to containment are allowed.

5.2 Containment Penetrations

A. Equipment Hatch Opening

1. Conditions of Operation:

a. During the movement of irradiated fuel assemblies within the containment (LCO 3.9.3), the equipment hatch shall be shut per STPO-55A, Containment Closure Verification, if the Containment Outage Door (COD) is unable to satisfy its "Closed Condition" and "Operational Requirements". (See Section 5.2.A.3)

b. If ERPIP-3.0, Immediate Actions, Attachment titled, Preparing for Severe Weather, is implemented and requires Containment Closure to be established then the Equipment Hatch shall be installed per STPO-55A, Containment Verification. In addition, The ERPIP-3.0 may require 20 hatch eyebolts.

2. Equipment Hatch Door (EHD)

a. Currently, the Equipment Hatch Door (EHD) cannot be quickly closed when a loss of power occurs. The equipment hatch should, therefore, be opened only for the time necessary to transport material in and out of the containment and should not normally be opened when fuel is in the reactor vessel with S/Gs not available for decay heat removal. If the hatch is required to be opened when a S/G is not available and the Containment Outage Door (COD) will not be utilized to satisfy closure, then:

(1) The amount of time the hatch is open shall be minimized.

(2) The hatch shall only be open when the Time to Boil (TTB) is <u>greater than</u> the time required to close the hatch **(65 minutes closing time plus 15 minutes if personnel are not on station).**

(3) The GS-Mechanical Maintenance shall be responsible to ensure qualified personnel are on-site to close the hatch.

(4) Two on-site and two off-site power supplies should be available to close the equipment hatch. This is not required when the refueling pool level is <u>greater than</u> 57 feet due to the increased TTB. **[B0138] [B0483]**

b. The equipment hatch may be opened, without imposing the requirements of Section 5.2.A.2, Equipment Hatch, if a S/G is available.

c. If utilized for Containment Closure, the containment equipment hatch shall be capable of being closed prior to the TTB and T.S. 3.9.4/3.9.5, held in place by a minimum of 4 eyebolts with no gaps. **[B0138]**

5.2.A.2 (Continued)

d. Deviations for the equipment hatch are not allowed in Reduced Inventory unless:

- The hatch can be closed in *less than* TTB and T.S. 3.9.4/3.9.5.
- The deviation is approved by the GS-SO.
- A Higher Risk Evolution Contingency Plan is prepared per NO-1-103, Conduct of Lower Mode Operations.

3. Containment Outage Door (COD)

a. Verify personnel are stationed at the COD once per shift during Refueling or if Shutdown Cooling is not in operation per the requirements of TS 3.9.3.

b. The COD is designed to be a functional equivalent of the Equipment Hatch Door (EHD), for the purposes of achieving containment closure when required during outage conditions.

c. Two Conditions of Operations:

(1) Standby Condition: Allows the passage of equipment and personnel through an 11' by 17' (nominal) door opening in the COD.

(2) Closed Condition: Capable of retaining a maximum of 12 PSIG within the containment.

d. Operational Requirements:

(1) During an outage, when the plant enters Modes 5 and 6, the COD will only be used for containment closure at least 24 hours after shutdown. This assures that the decay heat in the core will not result in exceeding the 12 PSIG maximum design pressure for the COD.

(2) If ERPIP-3.0, Immediate Actions, Attachment titled; Preparing for Severe Weather, is implemented and requires Containment Closure to be established then the Equipment Hatch shall be installed per STPO-55A, Containment Closure Verification. In addition, the ERPIP-3.0 may require 20 hatch eyebolts.

(3) If the COD is required to be open when a S/G is not available then it shall only be opened when the TTB is *greater than* the time required to close the COD (actual COD closure time plus 10 minutes if personnel are not on station).

(4) The GS-Mechanical Maintenance shall be responsible to ensure qualified personnel are on-site to close the COD.

5.2.A.3.d (Continued)

(5) For Containment Closure the COD shall be capable of being closed prior to the TTB and within the requirements of T.S. 3.9.4/3.9.5.

If the COD is open, Section 5.1.C, 5.1.D or 5.1.E shall be followed, as applicable.

The COD door opening, [1]laydown area, and/or pathway to the COD within the Equipment Hatch Access Building (Butler Building) must remain clear. If equipment or materials must be located in these areas, where as removal of equipment or materials would take longer than 10 minutes, then a separate Attachment 1, Containment Closure Deviation Sheet shall be required.

- The group requiring equipment or materials to be located within the COD door opening, laydown area and/or pathway to the COD within the Equipment Hatch Access Building (Butler Building) is responsible for filling out the separate Attachment 1, Containment Closure Deviation Sheet.
- The purpose of the separate Attachment 1, Containment Closure Deviation Sheet, is to provide methods for ensuring equipment or materials located within the COD door opening, laydown area and/or pathway to the COD within the Equipment Hatch Access Building (Butler Building) can be removed in a prompt manner, such that the COD can be fully shut and dogged prior to the TTB and/or T.S. 3.9.4/3.9.5.

(6) If the time required to close the COD is *less than* 30 minutes and/or during T.S. 3.9.3/3.9.4 (NOTE 2), the COD shall be manned at all times. For T.S. 3.9.3/3.9.4 (NOTE 2) see Section 5.2.A.3.c(8) for specific requirements.

[1] The Laydown area is marked as a designated "restricted area."

5.2A.3.d (Continued)

(7) If the time required to close the COD is *greater than* 30 minutes the COD is not required to be manned, except during T.S. 3.9.3/3.9.4 (NOTE 2).

If the time required to close the COD is *greater than* 30 mintues and T.S. 3.9.3/3.9.4 (NOTE 2) are not applicable, the group responsible for opening/closing the COD shall post a sign, per Door-01, Open and Closing of Containment Outage Door, procedure. The sign will describe that prior to placing unattended equipment and materials located in the COD door opening, laydown area and/or pathway to the COD within the Equipment Hatch Access building (Butler Building), requiring *greater than* 10 minutes to be removed, the owner shall contact the OWC to determine if a separate Attachment 1, Containment Closure Deviation Sheet, is permissible under the current plant conditions and to assist requesting group in determining the desired "Method for Restoration or Closure."

This will ensure that the group responsible for closing the door, once notified, will not be impacted by equipment or materials impeding COD closure. The door shall be fully shut and dogged prior to the TTB or T.S. 3.9.4/3.9.5.

(8) For T.S. 3.9.3/3.9.4 (NOTE 2) the equipment hatch opening may be open if the Containment Outage Door is operable and capable of being closed by a designated individual who is continuously available, stationed near the door. During these conditions, no cables or hoses are permitted to run through the door opening, the door must remain unblocked (for example: capable of being closed within 30 minutes) and capable of being fully shut and dogged. Containment Outage Door grating or truck ramps may be installed if the grating or truck ramps can be removed with the use of a forklift and the door closed within 30 minutes.

- During these conditions the COD will be tracked by an Attachment 1, Containment Closure Deviation Sheet, for tracking purposes only, to be utilized during Abnormal Conditions.

(9) Deviations for the COD are not allowed in Reduced Inventory unless:

- The COD can be fully shut and dogged in *less than* TTB and/or T.S. 3.9.4/3.9.5
- The deviation is approved by the GS-SO
- A Higher Risk Evolution Contingency Plan is prepared per NO-1-103, Conduct of Lower Mode Operations.

5.2.A.3.d (Continued)

(10) As required in Containment Penetrations 5.2.E, temporary pipes and hoses attached to the COD structure must be maintained as closed systems or have the ability to be isolated and/or quickly disconnected to provide a barrier against Lower Mode Operations Containment Pressure.

B. Emergency Air Lock (EAL)

1. For Containment Closure, a minimum of one door in the EAL shall be capable of being closed prior to the TTB and within the time requirements of T.S. 3.9.4/3.9.5. **[B0606]**

2. The Emergency Air Lock temporary door is acceptable to meet T.S. 3.9.3, but it shall have a Containment Closure Deviation Sheet filled out to ensure one Emergency Air Lock door can be closed prior to the TTB and the time requirements of T.S. 3.9.4/3.9.5. This Deviation Sheet shall not be considered a deviation for the movement of irradiated fuel in containment. The temporary door shall also have:

a. Six diametrically opposed C-clamps installed.

b. Each penetration through the temporary door capped; or, if in use, connected to a closed system; or the isolation valve at the penetration is shut.

c. The outer door equalizing valve is shut.

3. The temporary door in the Emergency Air Lock (designed for 5 PSIG) is not capable of withstanding Lower Mode Operations Containment Pressure.

a. If required to perform actions to close all penetrations per T.S. 3.9.4/3.9.5, the Emergency Air Lock temporary closure device cannot be credited for Containment Closure for a Loss of Shutdown Cooling event. At least one door in the Emergency Air Lock must be closed to satisfy these action statements.

(1) During Restricted Containment Closure Conditions: **[B0859]**

- All manually operated valves associated with piping penetrations through the Temporary Door are tagged in accordance with Section 5.3.D, Restricted Containment Closure Conditions.
- When the Temporary Door is installed then, ensure green chain barriers are installed at both locations (inside the Containment at the Emergency Air Lock hatch and outside Containment at the door to prevent unauthorized entry in the area). The chains shall have signs to contact OWC prior to entry.

C. Personnel Air Lock (PAL)

1. Verify personnel are stationed at the PAL once per shift during Refueling or if Shutdown Cooling is not in operation per the requirements of TS 3.9.3.

5.2.C (Continued)

2. For Containment Closure, a minimum of one door in the Personnel Airlock shall be capable of being closed prior to the TTB and within the time requirements of T.S. 3.9.4/3.9.5.

   - During these conditions the PAL will be tracked by an Attachment 1, Containment Closure Deviation Sheet, for tracking purposes only, to be utilized during Abnormal Conditions.

NOTE

Further restrictions apply for the Containment Outage Door, see 5.2.A.3.

D. Equipment shall not be maintained in or through a hatch unless equipment can be removed to support Containment Closure prior to the TTB and within the time requirements of T.S. 3.9.4/3.9.5.

NOTE

Further restrictions apply for the Containment Outage Door, see 5.2.A.3.

E. Temporary pipes and hoses must be maintained as closed systems or have the ability to be isolated and/or quickly disconnected prior to the TTB and within the time requirements of T.S. 3.9.4/3.9.5.

F. During Technical Specification 3.9.3 Conditions of Operation, refer to OI-17D for requirements needed to open the Containment Normal Sump Valves.

G. During Technical Specification 3.9.4 Conditions of Operation (NOTE 2 or Action "A") or 3.9.5 Conditions of Operation (Action "B"), Containment Normal Sump Valves may not be opened. Containment Normal Sump Valves may be open during all other closure conditions as follows: **[B0610] [B0674]**

   1. Refer to OI-17D for draining the Containment Normal Sump.

   2. This is considered a deviation for Reduced Inventory conditions. This will require an Attachment 1, Containment Closure Deviation Sheet to track the valve status.

   3. If the spring return handswitch is key-overridden open, then an Attachment 1 Containment Closure Deviation Sheet shall be completed to track the valve status when Containment Closure is required.

H. Steam Generator instrument sensing lines may be opened to obtain a Steam Generator level reading of the temporarily installed tygon tubing and reclosed upon completion of the level reading.

   1. This is routine, short duration evolution usually conducted during operator containment tours.

   2. This is considered a deviation for all Tech Spec closure requirements <u>unless</u> the Steam Generator is <u>open</u> to the Containment atmosphere with closure established accordingly.

5.2 (Continued)

I. Main Steam Safety Valves may be worked in Reduced Inventory with secondary open to containment if the Safety Valves or temporary covers can be installed prior to TTB and T.S. 3.9.4/3.9.5 requirements, per submitted restoration plans (Deviation Sheets) and GS-SO permission is granted.

J. Containment Purge Valve(s) seats must be visually inspected, at least once per refueling cycle, prior to crediting them for closure. A verification that no visible gaps exist between the valve seat and the valve disc constitutes a satisfactory closure barrier. Adjustments/Repairs may be made during the inspection to eliminate any gaps in order to achieve a satisfactory inspection. Inspection/Verification/Adjustments and repairs are to be performed in accordance with an approved maintenance procedure. **[B0611]**

K. All penetrations as listed in STPO-55A, Containment Closure Verification, which provide access from the Containment atmosphere to the outside atmosphere, shall be capable of being closed by an isolation valve, blind flange, or manual valve prior to the TTB and T.S. 3.9.4/3.9.5 requirements. **[B0594]**

1. All penetrations shall meet all Tech Spec Containment Closure requirements for Tech Spec 3.9.3, 3.9.4, and 3.9.5 conditions.

NOTE

For all closed systems, in or out of service, with Containment penetrations, Containment Closure is met if intact piping and components (even continuous vent valves) can prevent Lower Mode Operations Containment Pressure from relieving to the outside atmosphere.

2. An operating system designed to be a permanent part of the plant meets Containment Closure. **[B0607]**

3. A non-operating system designed to be a permanent part of the plant meets Containment Closure as long as the piping is intact and the penetration is not providing direct access from the Containment to the outside atmosphere. **[B0608]**

4. SG Sluice hoses connected between a COD penetration and a SG with inside closure (for example: SG closed to the containment atmosphere per STPO-55A) may be considered a closed system as allowed by STPO-55A.

5.3 Status Tracking

A. This section provides the administrative means of tracking the status of Containment penetrations during outages. Containment penetration tracking sets the ideal conditions for Containment Closure restoration. Tracking shall be utilized as an aid to ensure a sufficient number of trained individuals are on site to complete restoration of containment openings prior to the TTB if SDC is lost. Tracking shall be accomplished through the use of the Containment Closure Deviation Sheets, the Containment Penetration Status Board (BGEDRWGs 83018 (U-1) and 83019 (U-2) SH0001) and the Shift Turnover Information Sheet.

5.3.A (Continued)

1. Deviation Sheets are used to document personnel notifications and requirements to ensure Containment breaches can be rapidly restored. Deviation Sheets provide the Operator with information describing:
   - Affected penetrations
   - Methods of closure
   - Times required for closure
   - Special protective equipment
   - Required tools and materials
   - Work group contacts responsible for closure **[B0138]**
2. The Containment Penetration Status Board is: **[B0612]**
   - A consolidated, controlled, single-line drawing of all primary (first boundaries back from penetration) Containment inside and outside penetrations.
   - A human factored format to allow big picture, up-to-the-minute status of Containment Closure.
   - Maintained in the Operations Work Control Center (OWC) or Control Room.
3. The Shift Turnover Information Sheet shall be used to track active Containment Closure Deviations including administrative deviations, such as PAL, and COD during the movement of irradiated fuel. The Shift Turnover Information Sheet should include the penetration, the action to restore and the responsible individual.

B. If S/Gs are available, all breaches shall be tracked per Section 5.3.A and 5.1.C above.

C. Whenever the S/Gs are not available for decay heat removal, the following actions shall be taken and conditions met in addition to Section 5.3.A:

1. Prior to proceeding to a condition where both S/Gs are unavailable, the CRS shall verify that all current (open) Containment Closure Deviations shall be capable of being restored within the TTB and T.S. 3.9.4/3.9.5 time requirements (Step 3 of Attachment 1). S/G status shall not be altered until closure is restored or all responsible Work Groups have restoration plans to ensure Containment Closure can be restored prior to TTB or 3.9.4/3.9.5 time requirements.
2. Operations shall calculate Time to Boil (TTB):
   a. Once per shift, based on existing RCS inventory and assuming shutdown cooling is lost.
      (1) A review of T.S. 3.9.4/3.9.5 shall be also conducted while in Mode 6. The lesser time of T.S. 3.9.4/3.9.5 and the TTB, as needed, shall be recorded and updated on the Shift Turnover Information Sheet.

5.3.C.2 (Continued)

b. Before draining the RCS 6 inches or more:

(1) The TTB shall be updated based on the anticipated RCS level following the inventory change and shall be updated on the Shift Turnover Information Sheet as needed.

(2) The updated TTB shall be compared with T.S. 3.9.4/3.9.5 and the time to restore closure in Step 3 of all existing (open) Attachment 1, Containment Closure Deviation Sheets. Inventory level changes shall not be made until all responsible Work Groups have restoration plans to ensure Containment Closure can be established prior to core boiling or 3.9.4/3.9.5.

D. Restricted Containment Closure Conditions

1. Restricted Containment Closure Conditions include: the movement of irradiated fuel in containment, Reduced Inventory as defined in this procedure, and Tech Spec 3.9.4/3.9.5n SDC loop inoperability requiring establishment of containment closure.

a. In addition to the requirements in Section 5.3.C, the status of Containment penetrations shall be verified to be accurate by performing STPO-55A, Containment Closure Verification, within 7 days prior to electively entering Restricted Containment Closure Conditions. The Shift Manager shall determine the requirements to perform STPO-55A in unplanned situations leading to entry into Restricted Containment Closure conditions (for example: unplanned required SDC loop inoperability).

(1) Each time STPO-55A is completed, the person performing the surveillance shall note all closure deviations directly on the STP and on the Containment Penetration Status Board. Subsequent changes to closure need only be captured on the status board. Changes should not be made to STPO-55A after the initial performance.

(2) Each time STPO-55A is performed, the SRO reviewing the STP shall ensure a Deviation Sheet exists for each discrepancy noted on the STP by auditing the Shutdown Control Log. Should a deviation exist which is not current or is improperly entered into the Shutdown Control Log, the SRO shall correct the situation and bring the Log up to date.

(3) For STPO-55As performed for T.S. SR 3.9.3.1, process the original of the STP per EN-4-104, Surveillance Testing. Place a copy of the original STP in the Shutdown Control Log.

(4) For STPO-55As performed for restricted containment closure conditions other than T.S. SR 3.9.3.1, place the original STP in the Shutdown Control Log.

(5) Send previous STPO-55As performed for restricted containment closure conditions (other than T.S. SR 3.9.3.1, which were processed per EN-4-104, Surveillance Testing) and all Containment Closure Deviation Sheets to the Operation's shift office for retention for 2 years.

5.3.D.1.a (Continued)

NOTE

Reference to performing STPO-55A for the initial SG unavailability is included in STPO-55A, Attachment 5, Performance Flowchart for the sake of completeness. This is NOT considered a Restricted Containment Closure condition.

(6) For all elective entries into Restricted Containment Closure Conditions, STPO-55A shall be initially and subsequently executed per the requirements of STPO-55A, Attachment 5, Performance Flowchart. Subsequent required performances of STPO-55A should be tracked on the Shift Turnover Information Sheet per CNG-OP-1.01-2002, Operations Shift Turnover and Relief.

b. All manually operated valves and handswitches (for remotely operated valves) used for Containment Closure shall be yellow caution tagged. **[B0609]**

(1) Implementation of yellow caution tags shall be controlled per CNG-OP-1.01-1007, Clearance and Safety Tagging.

(2) Yellow caution tags for Containment Closure shall be annotated to indicate that equipment is being controlled for Containment Closure and may only be operated with SM/CRS permission.

(3) If a penetration's Containment Closure is being maintained by process flow, then at least one Control Room component or indicator shall be yellow caution tagged (if available) to indicate to Control Room personnel the nature of the closure for that penetration.

(a) If controls that could affect the flow are in various locations within the Control Room, it may be appropriate to yellow caution tag more than one component or indicator for that penetration.

(b) If no appropriate Control Room components or indicators are available, yellow caution tags may be hung locally to provide equivalent information about the nature of Containment Closure for that penetration.

(4) When Containment Closure boundaries are required to be changed, ensure that the new yellow caution tags are hung prior to removing the previous yellow caution tags.

c. If a penetration is controlled by a Safety Tagging Clearance per CNG-OP-1.01-1007, Clearance Safety Tagging, and is aligned differently than STPO-55A, Containment Closure Verification, then:

(1) Review the tagout to be sure it meets closure requirements, including valves within the boundary.

(2) Place a circled comment number in the initials column of the STPO-55A alignment.

5.3D.1.c (Continued)

(3) List the comment number and specific tagout number in the STPO-55A cover sheet Remarks sections.

(4) Ensure yellow caution tags are hung on the credited closure boundaries per the section above in addition to those tags which are a part of the clearance.

(5) Identify these components and closure tags on the Containment Penetration Status Board.

2. Prior to entering and during the movement of irradiated fuel in containment and SDC Maintenance, comply with the following T.S. requirements for Containment Closure:

3.9.3

3.9.4

3.9.5

3. Deviations to Containment Closure during Reduced Inventory shall have their restoration plans approved by the GS-SO.

E. In addition to the current STPO-55A, Containment Closure Verification, a copy of NO-1-114, Containment Closure, and active Deviation Sheets shall be kept in the Shutdown Control Log.

F. Local Leak Rate Testing (LLRT) **[B0610]**

1. Prior to the start of the outage, Operations LLRT personnel shall review every LLRT to performed under the applicable STPO-108, Local Leak Rate Procedure against STPO-55A, Containment Closure Verification, for any change in Containment Closure requirements. This review will result in a document similar to Attachment 3, LLRT Matrix (Example).

a. LLRT personnel shall ensure LLRTs which deviate from Containment Closure ("N"s on the LLRT Matrix) are scheduled by Outage Scheduling for performance during Non-Restricted Containment Closure Conditions. (The GS-SO may approve Deviation Sheets for these LLRTs in Reduced Inventory.)

b. All LLRTs that cannot be performed while maintaining closure ("N"s on the LLRT Matrix) or require draining evolutions prior to setting closure ("Drain-Aheads" on the LLRT Matrix) shall have Deviation Sheets.

5.3.F.1 (Continued)

c. LLRTs that can be performed and still maintained closure can be scheduled in any Mode 5 and 6 condition (including the set-up and post-test valve manipulations: venting, draining, filling, and connection of the Leak Rate Monitor (LRM)).

(1) As long as the LRM is connected to the piping and pressurized above Lower Mode Operations Containment Pressure, Containment Closure is satisfied.

(2) If the LRM is connected to the piping and pressure is <u>less than</u> Lower Mode Operations Containment Pressure or vented, closure shall be maintained by other boundaries or a Deviation Sheet shall exist for the penetration.

2. In-progress LLRTs shall be tracked and status updated per Attachment 4, Containment Closure Status Sheet for LLRTs. There shall be direct communications between the LLRT team and the Operations Work Control Center or Control Room which will include presentation of Attachment 4 by the LLRT personnel.

3. Penetration valve manipulations for LLRTs should only be performed by LLRT, Operations and Instrument and Controls personnel.

4. Failure of the LLRT

a. In order for a degraded boundary, leakage <u>greater than</u> max allowed in accordance with the applicable STPO-108, Local Leak Rate Test Procedure, to serve as a Containment Closure boundary, the valve and test data must be evaluated by Engineering.

b. If the piping will not pressurize or leakage is so significant that the test is obviously failing, DO NOT vent or break the LRM connection until an alternate closure boundary is established and the LRM test connection has been isolated.

5. Upon LLRT completion or while moving test boundaries, ensure Containment Closure is maintained or a Deviation Sheet exists.

G. Transmitter Calibrations

1. Calibration procedures for all transmitters associated with Containment penetrations shall be the controlling procedures and shall meet the closure requirements of this procedure.

2. Per the calibration procedures, transmitter local instrument stops are specifically used to isolate the transmitter from the process fluid during the calibration and shall be credited to maintain closure. Calibration procedures require independent verification of local instrument stops to place the transmitters back in service.

3. There shall be direct communications between the Instrument and Controls personnel and the Operations Work Control Center or Control Room.

5.3.G (Continued)

4. Penetration valve manipulations upstream of the local instrument shop shall be controlled by Operations. Manipulation of transmitter local instrument stops and other downstream valves shall be controlled by Instrument and Controls personnel.

5.4 Deviations

A. As a general operating philosophy, containment breaches should not be allowed. However, when work activities require a deviation to a Containment Closure boundary and the deviation is allowed or an inadvertent breach has occurred, then:

1. The breach shall:

a. Be controlled and recorded according to Section 5.3 of this procedure.

b. Have restoration plans to facilitate closure of the breach at all times in Modes 5 and 6.

(1) Containment Closure restoration plans shall document provisions for the establishment of Containment Closure prior to Tech Spec mandated timelines and the time boiling would occur in the RCS based on existing water inventory and decay heat. This may require, but not be limited to:

- Pre-staging of personnel and material
- Pre-approved Special Work Permits
- Personnel safety equipment

(2) The supervisor of the work group designated to restore closure shall ensure that:

(a) Personnel designated to restore closure have the specific knowledge required for restoration actions.

(b) A walk-through is conducted to ensure the designated personnel are aware of the closure's location, the necessary equipment, and actions required to establish closure.

(c) Trained individuals necessary to establish closure are on station if the TTB is _less than_ one hour (30 minutes for COD per 5.2.A.3).

(d) Trained individuals necessary to establish closure are on-site if the TTB is _greater than or equal to_ one hour.

(e) Manifolds, quick-disconnect devices, or some other means of quickly restoring the penetration's capability to withstand Lower Mode Operations Containment Design Pressure are utilized when temporary hoses, cabling, or other equipment are running through the containment penetration. **[B0138]**

5.4.A.1 (Continued)

c. Have Temporary Containment Closure Devices installed, if necessary, in such a way as to facilitate rapid separation and/or removal and shall be clearly labeled with tags hung on the isolation points identifying:

- Organization
- Contact number
- Purpose/job

B. When a breach in containment will or does exist as a result of maintenance activity or any other unforeseen circumstance, then:

1. The responsible Work Group Designated Contact and Workleader shall complete Steps 1-7 of Attachment 1, Containment Closure Deviation Sheet, per Attachment 2.
2. If the job does not require a tagout, then the Work Group shall present the Deviation Sheet to the Operations Work Control (OWC) Senior Reactor Operator (SRO) or the CRS along with the Work Order when requesting authorization to start work. The SRO shall sign Step 9 of Attachment 1 for review per Attachment 2.
3. If the job requires a tagout, the work group shall forward the Deviation Sheet to Safety Tagging. Safety Tagging shall sign Step 9 of Attachment 1 for review per Attachment 2.
4. The Shift Manager shall sign Step 10 of Attachment 1 for approval per Attachment 2.
5. The GS-SO shall also sign Step11 of Attachment 1 for approval if the deviation exists during Reduced Inventory.
6. The Deviation Sheet shall be filed in the Shutdown Control Log by the OWC SRO or the CRS, and a copy shall be included in the work package. Work Group Workleaders responsible for restoring closure should also retain a copy.
7. The Containment Penetration Status Board shall be updated with the new deviation by the OWC.
8. Once the activity requiring the deviation has been completed, the OWC SRO shall sign for close-out per Attachment 2. The OWC SRO shall then forward the completed sheet to the Shift Office for retention.
9. Information other than the method for restoration may be updated on Attachment 1, Containment Closure Deviation Sheet, in accordance with the guidelines of this procedure. The OWC SRO or the CRS shall initial and date all changes.

   a. If the method of restoration changes, then the existing Attachment 1, Containment Closure Deviation Sheet, shall be closed out, and a new Attachment 1, initiated and approved.

5.4 (Continued)

C. Testing activities, which could affect the ability to close containment, shall be reviewed for:

- Closure controls in place
- Potential loss of power supplies
- Management authorization
- Inadvertent draining
- Compensatory measures

5.5 Closure Restoration

A. To initiate Containment Closure Restoration:

1. Notify personnel and evacuate Containment by initiating ERPIP-3.0, Immediate Actions. **[B0138]**
2. The CRS will perform a follow-up initial notification by communicating with appropriate personnel to ensure expected results:
   - Radiation Protection Supervision
   - Security
   - Outage Management
   - Maintenance
   - Outage Control Center

B. Operations shall:

1. Initially focus on Control Room manipulations and notification of responsible work groups per existing Deviation Sheets. **[B0138]**
2. If necessary, dispatch personnel for component manipulation, coordination of contingency plans, or restoration assistance and verification.
3. Follow-up initial efforts by confirmation of STPO-55A, Containment Closure Verification, requirements by:
   a. Administrative reviews of STPO-55A.

   OR

   b. Actual performance of sections or all of STPO-55A as designated by the Shift Manager/CRS. Consider performing only outside alignments since a harsh Containment environment may exist. **[B0138]**
4. Ensure Containment Purge is secured or an automatic signal is OPERABLE to isolate Containment Purge (depending on Tech Spec requirements).
5. Ensure available Containment Air Coolers and Iodine Removal Units are operating as required.
6. Hang tags on closure boundaries at the Shift Manager's discretion depending upon Containment conditions and the necessity to maintain Containment Closure.

5.5 (Continued)

C. Upon notification, the Containment Closure Team shall report to applicable locations to restore Containment Closure.

D. The LLRT Team shall terminate and isolate all tests in progress and report the status to the Control Room.

E. The Outage Management shall notify the CRS when closure is restored to all Containment hatches.

F. The RCSS shall notify the Control Room that the Containment evacuation is complete (including Closure Team members).

G. Restoration actions required to establish Containment Closure may be terminated when the SDC system, RCS, and fuel matrix have been restored to a controllable and stable condition. The Shift Manager/CRS shall use appropriate judgment when implementing and terminating actions.

5.6 Drills

A. The Containment Closure program shall be periodically assessed by drills.

1. Minimize impact on the outage.
2. *Conduct a drill at least once per refueling outage.*

B. The Outage Manager shall coordinate Containment Closure Restoration drills as necessary to ensure: **[B0138]**

1. Lines of communication are clearly established for both initiation of closure and restoration feedback to the Control Room.
2. Personnel are aware of their assignments.
3. Tools and equipment are staged as required.
4. All closure deviations can be restored within the required time.

5.7 Abnormal Conditions

A. During Restricted Containment Closure Conditions, if a loss of Containment Closure is experienced, enter AOP-4A, Loss of Containment Integrity/Closure, section titled Modes 5 *and* 6. **[B0138]**

B. In the event of a fuel handling incident, enter AOP-6D, Fuel Handling Incident.

C. In the event of an abnormal SDC condition, enter AOP-3B, Abnormal Shutdown Cooling Conditions.

D. In all cases where Containment Closure restoration is required, Containment penetrations to the atmosphere shall be isolated or closed prior to the TTB and within the time requirements of T.S. 3.9.4/3.9.5. **[B0606]**

6.0 BASES

[B0138] NUMARC 91-06 Guidelines for Industry Actions to Assess Shutdown Management, Sections 3.3.5.2, 4.1.1.3, 4.5.1, 4.5.2, and 4.5.3 and Shutdown Safety Task Force, AIT 1U930006, 1U930011, 1U9300008

[B0482] Nuclear Engineering Unit Memo NEU 95-120, dated April 25, 1995

[B0483] NEU Memo 96-288, Time to Boil for Refueling Pool with UGS installed, dated July 16, 1996.

[B0594] NRC Generic Letter 88/17, Section 2.2 Containment Closure

[B0595] BGE Response to NRC Generic Letter 88-17, J.A. Tiernan Letter, dated January 3, 1989

[B0596] Licensing Memo, L95-131, dated August 24, 1995

[B0606] Nuclear Engineering Unit Memo NEU 93-028, dated January 27, 1993

[B0607] Design Engineering Section Commitment Resolution Document, AIT PD9400025

[B0608] Licensing Memo, L94-003, dated January 5, 1994

[B0609] NO-1-112, Safety Tagging, Revision 3, dated November 29, 1995

[B0610] AIT 1F199500548, Rise in IRs indicates a potentially inadequate process for administration of containment closure during periods of high maintenance activities

[B0611] AIT 1F199500655, Refueling Action Plan, ES200001071 and memo dated 3/16/01, from M.A. Junge to J.K. Mills, Subject: Visual Inspection of Containment Purge Valves for Containment Closure

[B0612] BGE Response to Notice of Violation, 89-11, George Creel letter dated August 10, 1989

[B0674] Technical Specification Interpretation # 96-001, ITS 3.9.3 Change Request, Refueling Operations, Containment Penetrations

[B0859] IR-015-307 and IR4-015-735, Maintaining Containment Emergency Air Lock temporary closure requirements during Restricted Containment Closure Conditions

[B1222] Cat 1 RCAR, Tech Spec Requirement for Containment Closure not established during Core Alts (IR200500070).

7.0 RECORDS

7.1 The following records are generated by use of this procedure and shall be controlled according to CNG-PR-3.01-1000, Records Management.

- Containment Closure Deviation Sheets
- STPO-55A, Containment Closure Verification
- Containment Closure Status Sheet for LLRTs

Attachment 1, CONTAINMENT CLOSURE DEVIATON SHEET [B0138] [B1222]

1. Deviation:

   a. Deviation Location:______________________________ (Room Name and Elevation)

   ____ (Penetration/Valve Number)

   b. Reason for Deviation:____

   (WO#, RCR#, etc.) ____

   ____

NOTE
The containment closure deviation sheet shall be closed out whenever the method for closure control is changed. A new containment closure deviation sheet shall be initiated and approved for the "new" method for closure control.

2. Method for Restoration or Closure (for example: Procedures such as: Door-01, Opening and Closing of Containment Outage Door,...) (Detailed instructions): ______________________

   ____

   ____

   ____

3. Estimated Time Required to Physically Establish Closure and/or Exit Containment:___________

4. Time to Boil (TTB) (taken from CNG-OP-1.01-2002, Operations Shift Turnover Sheet):

   ____

   Maximum Restoration Time is within the Time to Boil (TTB) if both S/Gs are not available and/or T.S. 3.9.4, 3.9.5 (4 hours), if applicable.

5. Personnel Protective Equipment necessary to support Restoration or Closure during a Sustained Loss of SDC (required at the penetration if time to boil is *less than or equal to* 1 hour):

   ☐ Respirator ☐ SCBA ☐ Full Anti-C's ☐ RP

   ☐ Scaffolding ☐ Plastic Suit ☐ Rubber Boots ☐ Dose Rate Meter

   ☐ Safety Harness ☐ N/A (Restoration is from outside containment)

   ☐ Other ____

   Other equipment necessary to restore closure (required at the penetration when Time To Boil is *less than* 1 hour):

   ____

   ____

6. Work Group Designated to Restore Closure_______________________________________

Attachment 1, CONTAINMENT CLOSURE DEVIAITON SHEET [B0138] [B1222] (Continued)

7. Work Group Contacts to Restore Closure shall be based on a 24-hour period.

   a. **Designated Contacts** listed shall read, understand and be capable of taking the required actions necessary to restore closure.

   b. **Responsible Workleader** signature certifies that the Workleader has notified and briefed the Designated Contact(s). (NUMARC 4.5.2)

Shift	Start Time/Date	Stop Time/Date	Name	Print Name / Work Group	Contact Number
			Designated Contact:	Print Name: Work Group:	
			Designated Contact:	Print Name: Work Group:	
			Designated Contact:	Print Name: Work Group:	
Responsible Workleader:		Print Name:	Signature:	Work Group:	

			Designated Contact:	Print Name: Work Group:	
			Designated Contact:	Print Name: Work Group:	
			Designated Contact:	Print Name: Work Group:	
Responsible Workleader:		Print Name:	Signature:	Work Group:	

			Designated Contact:	Print Name: Work Group:	
			Designated Contact:	Print Name: Work Group:	
			Designated Contact:	Print Name: Work Group:	
Responsible Workleader:		Print Name:	Signature:	Work Group:	

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Attachment 1, CONTAINMENT CLOSURE DEVIAITON SHEET [B0138] [B1222] (Continued)

8. RWP # for Containment Entry during a Sustained Loss of SDC: ____________________

9. SST, OWC SRO, OR CRS Review: ____________________ / __________
Signature Date

a. If Signature in Step 9 is the SST, then enter the Clearance # : ____________________

10. Shift Manager Approval: ____________________ / __________
Signature Date

11. GS-SO Approval is required if Reduced Inventory Condition Exists: ____________________ / __________
Signature Date

12. OWC SRO Deviation Sheet Close-Out: ____________________ / __________
[B1222] Signature Date

Penetration #	**Valve/Component**	**STP – O-055A Required Position**	**Inside or Outside Closure**	**Verify Position Sign and Date**

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Attachment 2, INSTRUCTIONS FOR CONTAINMENT CLOSURE DEVIATION SHEET [B1222]

NOTE

Attachment 1 may be used for tracking purposes when applicable, for the Containment Outage Door (COD)(Section 5.2.A.3), Emergency Air Lock (EAL) (Section 5.2.B) and Personnel Air Lock (PAL) (Section 5.2.C).

Att. 1 Step	Instruction
1.a.	As a minimum, list the penetration room name and elevation, penetration number and valve number, if applicable.
1.b.	Explanation of job scope. Include WO and/or FCR/MCR number or other document. **[B0610]**
2.	Detailed description of actions required by the work group should the deviation need to be restored. This should consider the unavailability of AC electrical power that could cause the loss of lighting or tools necessary to effect closure. Consideration should be given to having redundant power supplies available for these cases. **[B0138]** Use of plugging, flanging, or other means of restoring closure is acceptable if the Containment penetration can be restored to hold Lower Mode Operations Containment Design Pressure. If the method of restoration changes, then the existing Attachment 1, Containment Closure Deviation Sheet, shall be closed out, and a new Attachment 1 initiated and approved.
3.	Total amount of time from Control Room contact with appropriate work group personnel, to deviation being physically restored. • If possible, the maintenance on the containment penetrations shall be performed such that it is possible to close the penetrations from outside the containment. • When it is not possible to take the required actions from outside the containment, additional time shall be added to the closure time to allow for personnel to safely exit the containment prior to the onset of boiling. Due to the environmental conditions that may exist after the onset of boiling (elevated temperatures, containment pressurization, noise levels) it is important to ensure that personnel will be out of containment prior to the onset of boiling. **[B0138]**
4.	Enter Time to Boil (TTB) from the current CNG-OP-1.01-2002, Operations Shift Turnover Sheet. TTB must be *greater than* the estimated time required to physically establish closure and/or exit containment (Step 3). Time To Boil (TTB) is the time for RCS/Refuel Pool mass to reach bulk boiling after loss of SDC as determined by Figures in OP-7, Shutdown Operations. • T.S. 3.9.4/3.9.5 (Mode 6) has a required action to close all containment penetrations providing direct access from containment atmosphere to outside atmosphere within 4 hours. • Consideration shall be given to ensure required containment closure can be attained within this timeframe. • For the Containment Outage Door (COD), an additional 10 minutes must be applied to the actual COD closure time if personnel are not on station.
5.	Mark or list other appropriate items which would be required to support work group personnel when restoring a deviation inside the Containment during a loss of Shutdown Cooling event. Personnel protective equipment, tools, ladders, flashlights, and materials necessary to support the required work to effect closure of containment openings are immediately available.
6.	Proper name of work group responsible for ensuring the deviation is restored if needed.
7.	During the movement of irradiated fuel assemblies in containment, designated individuals shall be available to close both the Personnel Air Lock (PAL) and the Containment Outage Door (COD). The designated individuals shall be stationed at the Auxiliary Building side of the outer air lock door and at the outside of the Containment equipment hatch. These designated individuals shall be properly trained and knowledgeable of required actions to restore closure.

Attachment 2, INSTRUCTIONS FOR CONTAINMENT CLOSURE DEVIATION SHEET [B1222] (Continued)

Att. 1 Step	Instruction
7. cont.	List all Work Group Designated Contacts which can be contacted immediately by the Control Room to restore the deviation. The Shift, Start Time/Date, and Stop Time/Date shall be recorded. The Contact Name and the Work Group shall be printed legibly. The Contact Number may be a pager, shop or office telephone, etc., but not a home phone number. **[B0138]** The Workleader is responsible for ensuring personnel are aware of the deviation location and the equipment involved for closure. The Responsible Workleaders signature on Attachment 1, certifies that they have notified and briefed the Designated Contact. A walk-through shall be conducted to ensure personnel are aware of the actions required to restore closure. Workleaders are also responsible for ensuring that any changes to work group contacts include their review of the Containment Closure Deviation Sheet. They shall require new work group contacts to sign and date the Attachment 1 in the appropriate section, and inform the Shift Manager of the changes. **[B0138]**
8.	Actual SWP or EWP number which will be used for entry.
9.	<u>If the job requires a tagout, Safety Tagging shall:</u> Ensure all entries on the Sheet are legible. Ensure Steps 1-7 of the Deviation Sheet provide detailed information to promptly and properly execute closure restoration. Sign for review in Step 9 of the Sheet, and enter the Clearance number in Step 9.a. Present the Deviation Sheet to the OWC SRO/CRS when the tagout is presented for approval. <u>If the job does not require a tagout, then the work group shall:</u> a. Present the Deviation Sheet to the OWC SRO or the CRS along with the Work Order when requesting authorization to start work. The OWC SRO or the CRS shall: 1. Obtain the TTB and completion time requirements of T.S. 3.9.4/3.9.5 from the Shift Turnover Information Sheet, and compare to the "Estimated Time Required to Physically Establish Closure and Exit Containment", in Step 3 of the Deviation Sheet. 2. If Step 3 time is <u>*equal to or longer than*</u> TTB and the completion time requirements of T.S. 3.9.4/3.9.5, then the maintenance associated with the deviation shall not be approved until the Work Group provides alternate closure restoration plans to reduce the time associated with restoring closure. 3. Evaluate proposed containment breach against current or expected plant conditions (for example: will breach exist during Reduced Inventory) and inform the Shift Manager of concerns. 4. If a tagout is not required for the job: a. Ensure all entries on the sheet are legible. b. Ensure Steps 1-7 of the Deviation Sheet provide detailed information to promptly and properly execute closure restoration actions. 5. Sign for review in Step 9 of the Sheet.
10.	The Deviation Sheet shall be approved by the Shift Manager authorizing the work. a. The Deviation Sheet shall be filed in the Shutdown Control Log by the OWC SRO or the CRS and a copy shall be included in the work package. b. The OWC SRO or the CRS shall ensure work group leaders responsible for restoring closure have a copy of the signed closure Deviation Sheet. **[B0138]**
11.	If the deviation exists during Reduced Inventory, then the GS-SO shall also approve Attachment 1.
12.	Upon activity requiring deviation completion, the OWC SRO shall close-out the deviation sheet by signing and ensuring that the valves/components affected by the deviation sheet are reconciled in the table. The OWC SRO shall then forward the completed sheet to the Shift Office for retention.

Attachment 3, LLRT MATRIX (EXAMPLE)

Pen.	Equip.	STP O-55A page #	To be worked with 55A set	Tagging arrangement Inner/Outer	Function
1A-1	CV-5464	14	N		RC and PZR sampling
1A-2	CV-5465	14	Y	Tag outside Cont. Alignment (PS-5464 Shut)	RC and PZR sampling
1A-3	CV-5466	14	Y	Tag outside Cont. Alignment (PS-5464 Shut)	RC and PZR sampling
1A-4	CV-5467	14	Y	Tag outside Cont. Alignment (PS-5464 Shut)	RC and PZR sampling
1B-1	CV-2181	16	Y	Tag Inside Cont. Alignment (RCW-340 Shut)	Cont. Vent Header
1B-2	CV-2180	16	Y	Tag Inside Cont. Alignment (RCW-340 Shut)	Cont. Vent Header
1C-1	CV-505	17	Y-Drain Ahead	Tag Inside Cont. Alignment (CVC-506 Shut)	RC Pump Seals
1C-2	CV-506	17	Y-Drain Ahead	Tag Inside Cont. Alignment (CVC-506 Shut)	RC Pump Seals
1D	CV-6529	18	Y	Tag Outside Alignment (PS-6529 L/S Key Rem)	Oxygen Sampling
2A-1	CV-516	39	N		Letdown to purif. demin
2A-2	CVC-103	39	N		Letdown to purif. demin
2A-2	CVC-105	39	N		Letdown to purif. demin
2A-3	CV-515	39	N		Letdown to purif. demin
2B-1	CVC-184	41	N		RC Charging
2B-2	CVC-435	41	N		RC Charging
2B-3	CV-517	41	N		RC Charging
2B-4	CV-519	41	N		RC Charging
2B-5	CV-518	41	N		RC Charging
7A	ILRT-1	20	Y	Tag Inside Cont. Alignment (B1 Flange Installed)	ILRT Test
7B	ILRT-2	21	Y	Tag Inside Cont. Alignment (B1 Flange Installed)	ILRT Test
8-1	MOV-5462	83	Y	Tag Outside Cont. Alignment (EAD-5462 Shut)	Cont. Sump
8-2	MOV-5463	82	Y	Tag Outside Cont. Alignment (Option A)	Cont. Sump
9-1	SI-340	37	Y-Drain Ahead	Tag Inside Cont. Alignment (SI-340 Installed)	Cont. Spray
9-2	SI-326	37	Y-Drain Ahead	Tag Inside Cont. Alignment (SI-340 Installed)	Cont. Spray
10-1	SI-330	53	Y	Tag Inside Cont. Alignment (SI-330 Installed)	Cont. Spray
10-2	SI-316	53	Y	Tag Inside Cont. Alignment (SI-330 Installed)	Cont. Spray
13	Blind Flange	81	N		Cont. Purge
14	Blind Flange	54	N		Cont. Purge
15-1	CV-5292	51	Y	Tag Inside Cont. Alignment (CRM-5291 Shut)	Purge Air Monitor
15-2	CV-5291	51	Y	Tag Inside Cont. Alignment (CRM-5291 Shut)	Purge Air Monitor
16	CV-3832	55	Y-Drain Ahead	Tag Outside Cont. Alignment (CC-3832 Shut)	Component Cooling
18	CV-3833	52	Y-Drain Ahead	Tag Outside Cont. Alignment (CC-3833 Shut)	Component Cooling
19A-1	IA-175	71	N		Instrument Air
19A-2	MOV-2080	71	N		Instrument Air
19B-1	PA-137	73	N		Plant Air
19B-2	PA-1044	73	N		Plant Air

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Attachment 3, LLRT MATRIX (EXAMPLE) (Continued)

Pen.	Equip.	STP-0-55A page #	To be worked with 55A set	Tagging arrangement Inner/Outer	Function
20A-1	N2-347	68	N		Nitrogen Supply
20A-1	CV-612	68	N		Nitrogen Supply
20A-1	CV-622	68	N		Nitrogen Supply
20A-1	CV-632	68	N		Nitrogen Supply
20A-1	CV-642	68	N		Nitrogen Supply
20A-2	CV-612	68	N		Nitrogen Supply
20A-2	CV-622	68	N		Nitrogen Supply
20A-2	CV-632	68	N		Nitrogen Supply
20A-2	CV-642	68	N		Nitrogen Supply
20B-1	N2-395	69	Y	Tag Inside Cont. Alignment (N2-395 Installed))	Nitrogen Supply
20B-2	N2-348	69	Y	Tag Inside Cont. Alignment (N2-395 Installed)	Nitrogen Supply
20C-1	N2-398	70	Y	Tag Inside Cont. Alignment (N2-398 Installed)	Nitrogen Supply
20C-2	N2-349	70	Y	Tag Inside Cont. Alignment (N2-398 Installed)	Nitrogen Supply
21-1	Flanges	102	Y		SG Sec. Manway
21-2	Flanges	102	Y		SG Sec. Manway
22-1	Flanges	102	Y		SG Sec. Manway
22-2	Flanges	102	Y		SG Sec. Manway
23	CV-4260	26	Y-Tank Empty	Tag Inside Cont. Alignment (RCW-303 Shut)	RC Drain Tank
24	SV-6531	19	Y	Tag Outside Alignment (PS-6513 L/S Key Rem.)	Quench Tank
37-1	PSW-1020	36	Y-Drain Ahead	Tag Inside Alignment (PSW-1020 L/Shut)	Plant Water
37-2	PSW-1009	36	Y-Drain Ahead		Plant Water
38	CV-5460	65	N		Demin. Water
39-1	SI-455	35	Y-Drain Ahead	Tag Outside Alignment(SI-463 L/Shut)	SI Test Line
39-2	SI-463	35	Y-Drain Ahead	Tag Outside Alignment (SI-463 L/Shut)	SI Test Line
41	MOV-651	56	N		Shutdown Cooling
41	MOV-652	56	N		Shutdown Cooling
42	Xter tube	94	Y	Before Flooding	Fuel Trans. tube
44-1	FP-145B	66	Y	Tag Inside Alignment (FP-145B Installed)	Fire System
44-2	FP-145-A	66	Y	Tag Inside Alignment (FP-145B Installed)	Fire System
44-3	MOV-6200	66	Y	Tag Inside Alignment (FP-145B Installed)	Fire System
47A-1	SV-6507-A	31	N		Hydrogen Sampling
47A-2	SV-6540A	31	Y	Tag Inside Alignment (PS-6507 L/S Key Rem.)	Hydrogen Sampling
47B-1	SV-6507E	32	Y	Tag Inside Alignment (PS-6540E L/S Key Rem.)	Hydrogen Sampling
47B-2	SV-6540E	32	N		Hydrogen Sampling

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Attachment 3, LLRT MATRIX (EXAMPLE) (Continued)

Pen.	Equip.	STP-0-55A page #	To be worked with 55A set	Tagging arrangement Inner/Outer	Function
47C-1	SV-6507F	33	N		Hydrogen Sampling
47C-2	SV-6540F	33	Y	Tag Inside Alignment (PS-6540 L/S Key Rem.)	Hydrogen Sampling
47D-1	SV-6507G	34	N		Hydrogen Sampling
47D-2	SV-6540G	34	Y	Tag Inside Alignment (PS-6540G L/S Key Rem.)	Hydrogen Sampling
48A-1	MOV-6901	58	N		Hydrogen Sampling
48A-2	MOV-6900	58	N		Hydrogen Purge
48B-1	MOV-6903	59	Y	Tag Inside Alignment (HP-104 Installed)	Hydrogen Purge
48B-2	HP-104	59	Y	Tag Inside Alignment (HP-104 Installed)	Hydrogen Purge
49A-1	SV-6507B	43	N		Hydrogen Sampling
49A-2	SV-6540B	43	Y	Tag Inside Alignment(PS-6540B L/Sh Key Rem)	Hydrogen Sampling
49B-1	SV-6507C	44	N		Hydrogen Sampling
49B-2	SV-6540C	44	Y	Tag Inside Alignment(PS-6540C L/Sh Key Rem)	Hydrogen Sampling
49C-1	SV-6507D	45	N		Hydrogen Sampling
49C-2	SV-6540D	45	Y	Tag Inside Alignment(PS-6540D+E70 L/Sh Key Rem)	Hydrogen Sampling
50	ILRT Pen	67	Y		ILRT Pressurization
59	SFP-178	22	N		Refuel Pool
59	SFP-179	22	N		Refuel Pool
60	ES-142	23	Y	Tag Inside Cont. Alignment (ES-144 L/Shut)	Stm. to Rx HD C/U Area
60	ES-144	23	Y	Tag Inside Cont. Alignment (ES-144 L/Shut)	Stm. to Rx HD C/U Area
61	SFP-180	24	N		Refuel Pool Cooling
61	SFP-181	24	N		Refuel Pool Cooling
61	SFP-182	24	N		Refuel Pool Cooling
61	SFP-186	24	N		Refuel Pool Cooling
62	MOV-6579	79	Y	Tag Outside Cont. Alignment (PH-6579-MOV)	Contain. Heat
64	PH-387	80	Y-Drain Ahead	Tag Inner Alignment (PH-734)	Contain. Heat
67	Eq. Hatch	95	Y		Equipment Hatch
68	PAL	96	Y		Personnel Air Lock
69	EAL	100	Y		Emergency Air Lock
84	ILRT Vent	93	Y	Tag Inner Flange	ILRT Vent (U-2 Only)

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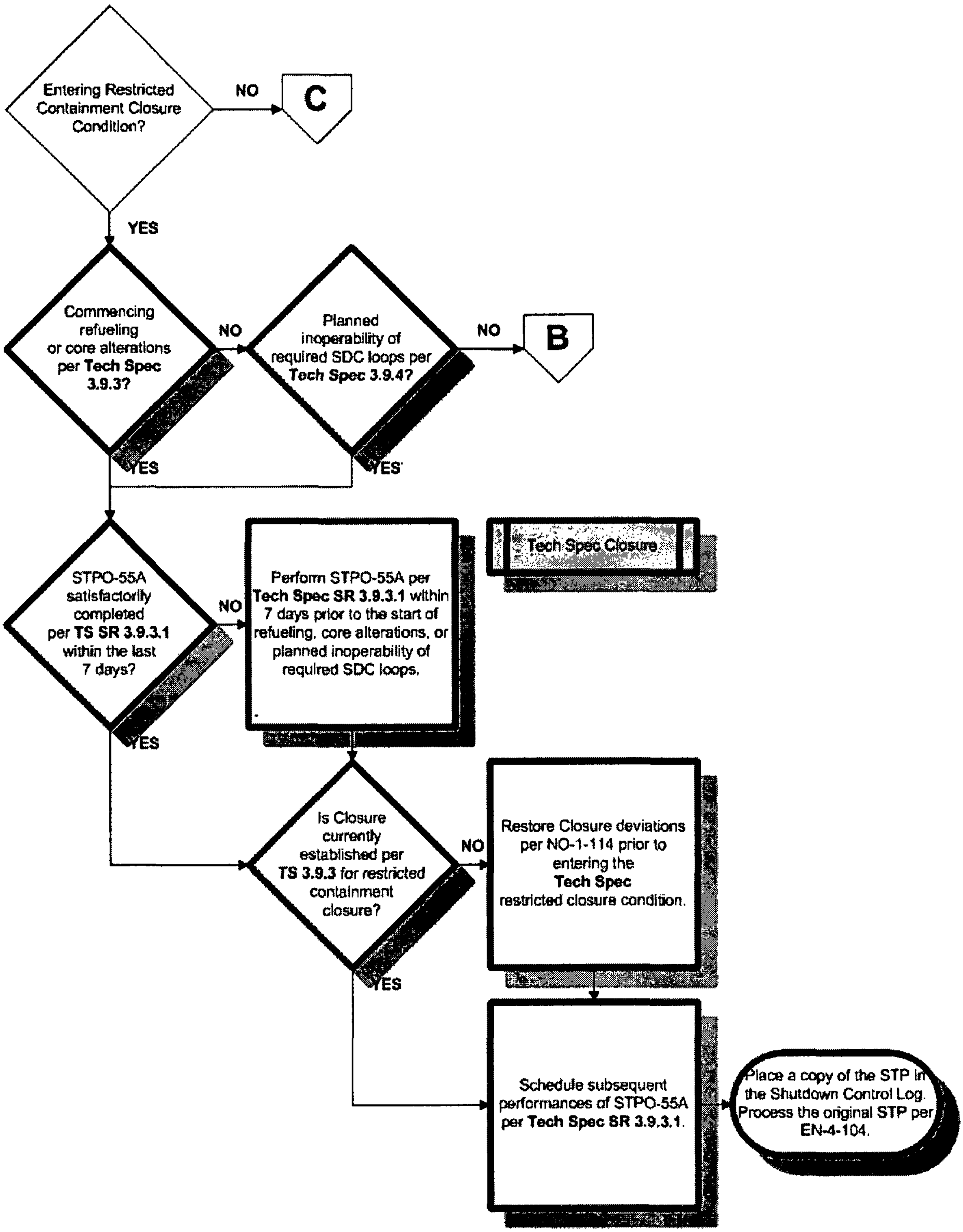
Attachment 4, CONTAINMENT CLOSURE STATUS SHEET FOR LLRTS

When an LLRT begins, enter the Penetration No. and Name of the responsible personnel. Indicate whether Containment Closure is established for the penetration or a deviation exists. Enter the Start Time and Date. When the LLRT is finished, enter the Finish Time and Date.

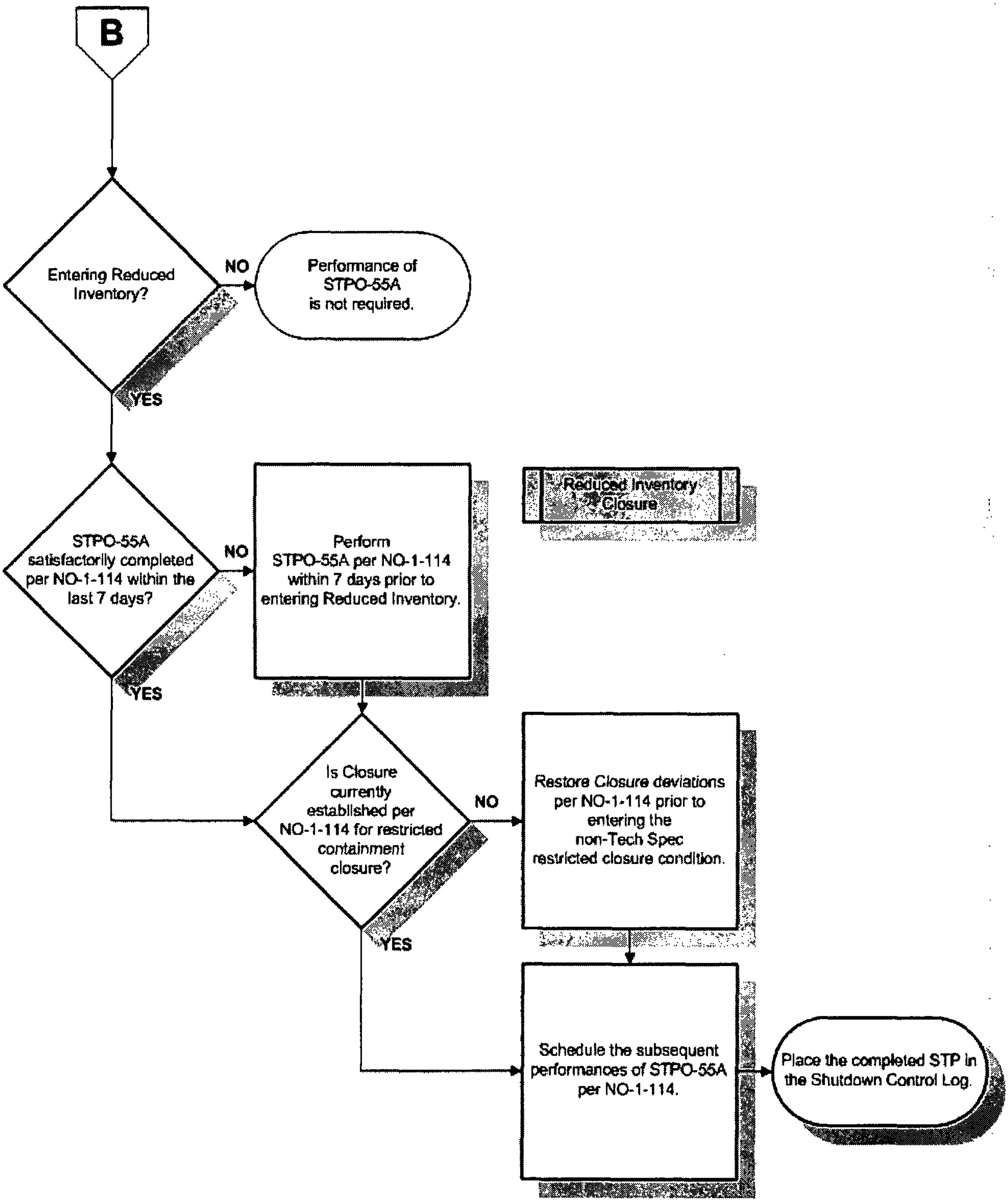
INTERFACE DIRECTLY WITH THE CONTROL ROOM OR THE OPERATIONS WORK CONTROL CENTER.

Penetration No.	Name	Cntmt. Closure (Yes/Dev.)	Start Time	Date	Finish Time	Date

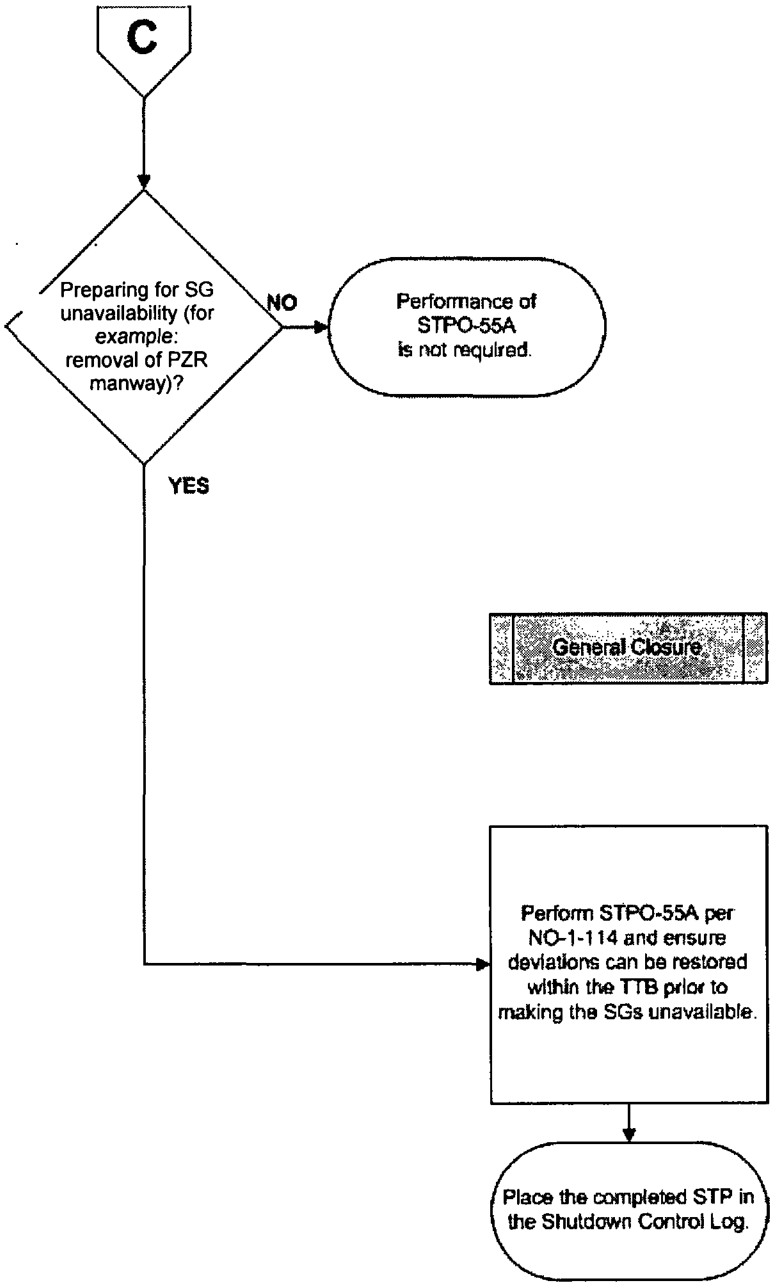
Attachment 5, STPO-55A PERFORMANCE FLOWCHART



Attachment 5, STPO-55A PERFORMANCE FLOWCHART (Continued)



Attachment 5, STPO-55A PERFORMANCE FLOWCHART (Continued)



FORM 1, ECP COVER SHEET

ECP Supp No.: **ECP-10-000208** **Rev. No.:** **0000**

A. INITIATION – ENGINEERING SERVICE REQUEST (ESR)

ESR No.: ESR-09-003787 ☐ N/A

Originator/Ext/Date: ______________________
Printed Name

Site (check one): ☒ CCNPP ☐ NMP ☐ REG

☒ UNIT 1 ☒ UNIT 2 ☐ COMMON ☐ ISFSI

Requested Due Date and Reason:

System: 009 Priority: Routine

Equip ID: 1LIT1100, 1LIT1101A, 1LIT1101B, 1LIT1201A, 1LIT1201B, 1LIT1301B, 1LIT1401B, 1LIT1501B, 1LIT1601B, 2LIT2101B, 2LIT2201B, 2LIT2301B, 2LIT2401B, 2LIT2501A, 2LIT2501B, 2LIT2601A, 2LIT2601B, 2LIT2100, 1LI1100, 2LI2100

ER Component Classification ☐ Critical ☒ Significant ☐ Economic ☐ RTF ☐ N/A

WO No.: CR No. CR Category:

Is this a request for a Temporary Change? ☒ Yes ☐ No

Is this a request for Generic Engineering? ☐ Yes ☒ No

Reasons for Request/Problem Statement/Proposed Changes:

This change installs nine (9) radar probes per unit with local indication to monitor intake structure level. At each unit intake structure, six (6) probes are installed downstream of the traveling screens, one (1) is installed upstream of the trash rakes, and two (2) are installed between the traveling screens and the trash rakes. This change provides a temporary local indication of level to meet a plant commitment. A permanent change installed per ECP-10-000209 uses radar probes installed in this change to provide remote indication of water levels to the control room.

☐ Check if additional sheets are attached (mark sheets with ECP No., Supplement and Rev. No., as applicable)

APPROVAL: ______________________ Date: __________
Originator's Supervisor / Workgroup Lead

TRB APPROVAL (New MODS and some EQVs (Section 5.2.B) require TRB approval.) ☐ N/A

TRB Approval Meeting Date:

☐ Standby 20/40 List Date:

☒ Active 20/40 List Date: 9/16/2009

Account No.: N02711

ECP Supp No.: **ECP-10-000208** **Rev. No.:** **0000**

B. INITIAL ENGINEERING SCREEN AND ASSIGNMENT ☐ N/A

PRELIMINARY SERVICE TYPE DETERMINATION

☐	Engineering Response		☐	Administrative Document Change	
☐	Engineering Evaluation		☐	Equivalent Change	
☐	Commercial Change	☐ DC ☐ EQV	☒	Design Change	☒ Modification ☐ Setpoint ☐ Document Change

RESPONSIBLE ENGINEER: SUPERVISOR:

SYSTEM ENGINEERING REVIEW/APPROVAL ☐ N/A

System Engineer/Supervisor Brad Wright ☐ Approved ☐ Disapproved
(Printed Name/Signature)

☐ Check if additional sheets are attached (mark sheets with ECP No., Supplement and Rev. No., as applicable)

C. FINAL SERVICE

SERVICE CLASSIFICATION: ☐ SR ☒ NSR ☐ Augmented Quality

☐	Generic Engineering	☒	Temporary Change

SERVICE PERFORMED (check one)

☐	Engineering Response		☐	Administrative Document Change	
☐	Engineering Evaluation		☐	Equivalent Change	
☐	Commercial Change	☐ DC ☐ EQV	☒	Design Change	☒ Modification ☐ Setpoint ☐ Document Change

COMMENTS

ORIGINAL FORM 1 WITH SIGNATURES ON PG 1 + 2 COULD NOT BE LOCATED

ECP Supp No.: **ECP-10-000208** **Rev. No.:** **0000**

ECP RISK SCREENS

Technical Task Risk Rigor Assessment (CNG-CM-1.01-2000) :

Assessment required: ☒ YES ☐ NO (If "NO" proceed to QRT Risk Ranking)

Consequence Risk Factors Identified: ☒ YES ☐ NO

If "YES," complete Human Performance and Process Risk Evaluations and any additional actions required by CNG-CM-1.01-2000. Include summary of results in ECP. – **Attachment 12**

If "NO," proceed to QRT Risk Ranking

Independent Third Party Review Screen Results: ☐ 1 ☐ 2 ☐ 3 ☒ 4

Third Party Review Required: ☐ YES ☒ NO IF YES, TYPE:

Third Party Review Completed: ☐ YES ☐ NO ☒ NOT REQUIRED

QRT Risk Ranking (CNG-CM-1.01-1000): ☐ Red ☐ Yellow ☐ Green ☒ N/A

ECP SUPPLEMENT INVENTORY

Inventory all Forms and attachments, issued with this ECP Supplement, which do not have unique document identification numbers. Ensure all products issued with unique document identification numbers are electronically linked to the ECP Supplement in FCMS.
Form # refers to CNG-FES-015 form numbers unless otherwise specified.

	Form #	Title	Number of Pages	Comments
☒	Attach. 12	Design Inputs and Change Impact Screen	14	CNG-CM-1.01-1003, Att 12
☐	6	EQV Technical Evaluation		
☒	7	Design Change Technical Evaluation	2	
☒	7A	DC Technical Evaluation Continuation	6	
☐	7B	DC Technical Evaluation Continuation		
☒	8	Operational Impact Statement	2	
☒	9	Installation and Testing Requirements	10	
☒	11	ECP Material List	1	
☐	12	Engineering Evaluation		
☒	13	Record of Walkdown	2	
☒	16	Fire Protection/Appendix R Review Electrical Design Features Checklist	2	
☒	17	Fire Protection/Appendix R Review Fire Protection Design Features Checklist	2	
☐	18	Design Review For ALARA Good Practices		
☐		10 CFR 50.59/72.48 Screen		CNG-NL-1.01-1011, Att 2
☐				
☐				
☐				
☐				
☐				
☐				
☐				

TO/CO Plan No.: 201000229

FORM 1, ECP COVER SHEET

ECP Supp No.: **ECP-10-000208** **Rev. No.:** **0000**

REVIEW AND APPROVAL – Sargent & Lundy LLC:

Responsible Engineer: David J. Goode (S&L) [signature] — 5/3/10
(Printed Name and Signature) Date:

(If Required) Professional Engineer: ☒ N/A
(Printed Name, Signature, and Requal Date) Date:

Is Design Verification Required? ☐ Yes ☒ No

If yes, Design Verification Form is ☐ Attached ☐ Filed with:

Independent Reviewer: Angelo A. Emanuele (S&L) [signature] — 5/3/10
(Printed Name and Signature) Date:

Additional Preparer	Jeff J. Grajewski (S&L) [signature]	5/3/10
	Printed Name and Signature	Date
Additional Reviewer	Carol D. Frantilla [signature]	5/3/10
	Printed Name and Signature	Date
Additional Reviewer	N/A	
	Printed Name and Signature	Date
Approved	Dan M. Wright	5/5/2010
	Printed Name and Signature	Date

ECP Cause Code (Section 5.1.1): O

FORM 1, ECP COVER SHEET

ECP Supp No.: ECP-10-000208 **Rev. No.: 0000**

IMPLEMENTATION REVIEW/APPROVALS ☒ N/A (if not required)

Engineering Manager: SR SANDERS for TJ Stanley Date: 5/6/10
(Printed Name and Signature)

PORC/PLANT GENERAL MANAGER REVIEW/APPROVALS ☒ N/A (if not required)

PORC Meeting No.: ____________ Date: ____________
(Printed Name and Signature)

PORC Chairman: ____________ ☐ Approved ☐ Disapproved
(Printed Name and Signature)

Plant General Manager: ____________ ☐ Approved ☐ Disapproved
(Printed Name and Signature)

ECP Supp No.: **ECP-10-000208** **Rev. No.:** **0000**

Sect	Design Input or Change Impact	Applicable Yes	Applicable No	Action Tracking
CNG-FES-007, Preparation of Design Inputs and Change Impact Screen, shall be used in preparation of Attachment 12. CNG-FES-007 provides detailed screening questions for each section of this screen to assist in determining the correct overall screening result for each topic. The column titled "Sect" refers to the applicable section of CNG-FES-007. CNG-FES-007 also provides recommended actions for any "Yes" answers. Review and approval of attachment 12 is indicated by review and approval of the ECP.				
5.4.1	**Mechanical/Civil/Structural Design:** Does the Engineering Change involve any Mechanical System characteristics where design limits are placed on the mechanical properties of a system or components? Does the Engineering Change involve any Civil/Structural requirements where limits are placed on the structural properties of an SSC such as equipment foundations and component supports? **[FB0241]**	☒	☐	
A mechanical assembly is required for radar probe mounting in existing penetrations. In addition, since the man holes are located in the open, the electronics modules protrude above ground and are exposed to weather and other hazards from personnel working in the area. For this reason, a protective structure is also designed around the probe to prevent damage. See Form 7/7A for evaluation of this design. To install conduit for radar probes 1LIT1100, 1LIT1101B, 1LIT1201A, 2LIT2501A, 2LIT2601A, and 2LIT2100, concrete must be cut in the intake structure. See Form 7/7A for an evaluation of the structural integrity of the intake structure per this change.				
5.4.1.1	**Containment Sump Recirculation Issue (CCNPP and Ginna, only):** Does the Engineering Change affect inputs or assumptions made in containment sump testing and analysis? Does the Engineering Change affect or impact commitments in response to Generic Letter 2004-02?	☐	☒	
5.4.1.2	**Managing Gas Accumulation in Emergency Core Cooling, Decay Heat Removal, and Containment Spray Systems:** Does the Engineering Change involve the potential introduction of gas into the Emergency Core Cooling System, the Decay Heat Removal System, or the Containment Spray System or involve the potential failure of managing gas accumulation in the subject systems? **[FB0509]**	☐	☒	
5.4.2	**Valves:** Does the Engineering Change involve or impact motor operated valves, air operated valves, relief valves or check valves? Does the Engineering Change involve or affect valve packing?	☐	☒	
5.4.3	**Flooding:** Does the Engineering Change impact or affect internal or external flooding analysis?	☐	☒	

ECP Supp No.: ECP-10-000208 Rev. No.: 0000

Sect	Design Input or Change Impact	Applicable Yes	Applicable No	Action Tracking
5.4.4	**HVAC:** Does the Engineering Change impact or affect HVAC or HVAC loading?	☐	☒	
5.4.5	**Thermal Fatigue:** Does the Engineering Change affect SSCs or piping in the Thermal Fatigue Program?	☐	☒	
5.4.6	**Control Room Habitability:** Does the Engineering Change impact Control Room Habitability? **[FB0243]**	☐	☒	
5.4.7	**Seismic:** Does the Engineering Change add, relocate, or alter Seismic Category I mechanical and/or electrical components that impact the Seismic Qualification?	☒	☐	
This change installs non-safety related components in the intake structure. The radar probe mountings were evaluated for Seismic Category II over I criteria in Form 7/7A, Design Change Technical Evaluation.				
5.4.8	**Structural Barriers:** Does the Engineering Change impact or change the functional performance of any plant structural barrier?	☐	☒	
5.4.9	**Coatings:** Does the Engineering Change require the application or removal of a protective coating or change any coating specifications or procedures?	☐	☒	
5.4.10	**Electrical/I&C Design:** Does the Engineering Change involve any Electrical requirements where limits are placed on the electrical properties of a system or components? Does the Engineering Change involve any Instrument Control requirements, including digital technology requirements? **[FB0242] [FB0255]**	☒	☐	
This change installs a total of 18 radar level probes (9 per unit) at penetrations in the intake structure. See Form 7/7A for technical evaluation of these components. Power to all radar level probes is supplied via temporary 24Vdc power supplies installed centrally to probe locations. Junction boxes are installed to distribute power wiring. See Form 7/7A for power supply evaluation. Each radar probe comes with a local indicator that relies on firmware for setup. A Cyber Security Assessment shall be performed in accordance with 201000229 Form 10, Turnover Closeout Plan. Temporary conduit is installed per this change for power cable for radar probes that are installed upstream of the traveling screens. See Form 7/7A for further evaluation of conduit.				

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Sect	Design Input or Change Impact	Applicable Yes	Applicable No	Action Tracking
5.4.10.1	**Transmission System Impacts:** Does the Engineering Change impact the transmission system?	☐	☒	
5.4.11	**10CFR50.49 Environmental Qualification:** Is the Environmental Qualification (EQ) of equipment is affected?	☐	☒	
5.4.12	**Human Factors:** Are there any Human Factors requirements introduced by the Engineering Change?	☐	☒	
5.4.13	**Station Blackout:** Does the Engineering Change impact any station blackout analysis or procedures?	☐	☒	
5.4.14	**Reactivity, Nuclear Fuels:** Is there any impact on nuclear fuel, core components, core design, reactivity management, criticality control and accountability of nuclear materials as well as transient and /or accident analysis? **[FB0247] [FB0252]**	☐	☒	
5.4.15	**[NMP Only] BWR Vessel Internals Program (BWR VIP):** Does the Engineering Change impact the BWR Vessel Internal Program?	☐	☒	
5.4.16	**Fire Protection/Appendix R:** Does the Engineering Change affect any fire protection/suppression equipment, change combustible loading, or impact the Fire Protection Program? Does the Engineering Change impact the plant's ability to safely shutdown in the event of an Appendix R Fire?	☐	☒	
All components installed per this change are located outside the plat and do not have any fire protection requirements. However, cable installed should comply with IEEE 383 standards at the recommendation of the fire protection engineer. See forms 16 & 17 as required.				
5.4.17	**PRA:** Does the Engineering Change affect the existing Probabilistic Risk Assessment (PRA), Mitigating System Performance Index (MSPI) Basis Document PRA content, and shutdown risk models?	☐	☒	
5.4.18	**NFPA 805:** Does the Engineering Change impact the analysis or assumptions of the NFPA-805 Program?	☐	☒	

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Sect	Design Input or Change Impact	Applicable Yes	Applicable No	Action Tracking
5.4.19	**Safety Classification:** Does the Engineering Change require a change in safety classification or category for any SSC?	☒	☐	
This change adds new components as defined in Form 7/7A Design Change Technical Evaluation.				
5.4.20	**Equipment Database(s):** Does the Engineering Change require changes to the equipment technical database(s)?	☒	☐	
Radar probes installed are given CompIDs and added to the FCMS database per this change. See form 7/7A for a list of CompIDs and descriptions. Cables installed are temporary and not scheduled. Therefore, this change does not impact CRS.				
5.4.21	**Equipment Reliability**: Does the engineering Change add, remove or modify equipment with critical functions or require classification of new equipment to determine equipment reliability classification?	☒	☐	
The radar level probe system was chosen based on successful demonstration at CCNPP. However, since these are new components, an equipment reliability classification shall be performed at project closeout per 201000229 Form 10, Turnover Closeout Plan.				
5.4.22	**SPV:** Does the Engineering Change impact single point vulnerability (SPV) so as to add to the potential to cause an unplanned reactor SCRAM?	☐	☒	
5.4.23	**Maintenance Rule:** Does the Engineering Change impact maintenance rule SSCs or impact the Maintenance Rule Program?	☐	☒	
5.4.24	**EPIX:** Does the Engineering Change affect the existing Equipment Performance Information Exchange (EPIX) database?	☐	☒	
5.4.25	**License Renewal:** Does the Engineering Change potentially impact the results or conclusions of a License Renewal Aging Management Review, Aging Management Program, or Time Limited Aging Analysis or affect any License Renewal Commitment?	☐	☒	
5.4.26	**IST/ISI Program:** Does the Engineering Change result in changes in plant configuration, calculations, or safety analyses that may create or eliminate safety functions which fall within the scope of the IST/ISI program, or require changing surveillance test/acceptance criteria?	☐	☒	

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Sect	Design Input or Change Impact	Applicable Yes	Applicable No	Action Tracking
5.4.27	**FAC:** Does the Engineering change have the potential to impact the Flow Accelerated Corrosion program?	☐	☒	
5.4.28	**Thermal Performance:** Does the Engineering Change plant efficiency or electrical megawatt output or their measurement in a way that may impact thermal performance monitoring?	☐	☒	
5.4.29	**Appendix J:** Does the Engineering Change modify any containment penetration or containment leakage criteria in a manner that will impact the Appendix J Program?	☐	☒	
5.4.30	**Boric Acid Program Impact:** Will any new component or system by added or installed near a boric acid system or a credible boric acid leak path?	☐	☒	
5.4.31	**Periodic and Surveillance Testing:** Does the Engineering Change necessitate changes to periodic or surveillance testing?	☐	☒	
5.4.32	**System Engineering:** Does the Engineering Change require active involvement on the part of System Engineering?	☒	☐	
This change adds new components for level monitoring. The system engineer has been engaged in the design and operation of these components installed in the intake structure.				
5.4.33	**ALARA:** Are Radiation Protection/ALARA programs impacted by the Engineering Change that affect any of the following during normal or post accident conditions: Radiation sources, changes affecting controlled radiation areas, primary coolant fluid systems (Cobalt Materials); contaminated systems; radiation monitoring systems; HVAC Systems which could transport airborne contaminants; change or alter shielding?	☐	☒	
This change is installs equipment located outside and not exposed to radiation.				
5.4.34	**Environmental Impacts:** Are there environmental conditions and impacts affected by the Engineering Change?	☐	☒	

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Sect	Design Input or Change Impact	Applicable Yes	Applicable No	Action Tracking
5.4.35	**Emergency Preparedness:** Does the Engineering Change potentially impact the existing Emergency Plan or environmental or discharge monitoring that are used to prevent undue risk to public health and safety?	☐	☒	
5.4.36	**FSAR/Tech Spec/TSB/Regulatory Commitments:** Does the change alter any statement or commitment in the UFSAR (A thorough review of the UFSAR is required)? Does the change result in non-compliance with the Technical Specifications or the Technical Specifications' Bases? Does the change modify or delete any regulatory commitment?	☐	☒	
5.4.37	**Security:** Does the Engineering Change involve any Security procedures or requirements such as site monitoring, alarm systems, vehicle barrier systems, security and security lighting?	☐	☒	
5.4.38	**NEIL:** *Does the Engineering Change have any impact on the* requirements of any applicable Nuclear Electric Insurance Limited (NEIL) Insurance Standard, or other appropriate insurance standards? **[FB0244] [FB0245]**	☐	☒	
5.4.39	**Information Technology:** Does the Engineering change affect plant systems computer hardware, firmware, software, or data or analytical software used in the design or analysis of plant systems, structures, or components? Does it modify or add a digital device in a plant system?	☒	☐	
Cyber Security Assessment shall be performed for firmware on each probe per 201000229 Form 10, Turnover Closeout Plan.				
5.4.40	**Industrial Safety:** Are there any Industrial Safety requirements such as restricting the use of dangerous materials, hazardous chemicals, escape provisions from enclosures, pertinent OSHA requirements, and grounding of electrical systems?	☒	☐	
Radar probes are installed in penetrations that are in traffic areas on the intake structure. The probes protrude up from the cement structure and create a walking hazard. A protective cover is designed to protect the radar probes, but shall also be externally marked to bring attention to personnel in the area to avoid a tripping hazard. In addition, installation requires opening of metal grating upstream of the traveling screens. Fall protection requirements shall be implemented during this installation.				

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Sect	Design Input or Change Impact	Applicable Yes	Applicable No	Action Tracking
5.4.41	**Margin:** Does Engineering Change reduce design or operating margin or address an existing margin issue identified on the station Low Margin List?	☐	☒	
5.4.42	**Operating Experience (OE):** Are there any applicable Operating Experience listed on the INPO internet site or equivalent sources that are applicable to this Engineering Change? **[FB0242] [FB0246] [FB0247]**	☐	☒	
An OE search on the INPO website resulted in several hits relating to intake structure level. However, all of these events were associated with the circulation pump control and SCRAM aspect due to decreased level. This change only provides local indication of intake structure level and does not include any control or reactor SCRAM functions. Operating Experience OE27386 is for spurious alarms caused by Ohmart-Vega VEGAPULS 65 radar transmitter installed to measure Caustic Storage Tank Level. The cause of this event was splashing the radar probe with sodium hydroxide during filling. A solution from the vendor was to adjust the signal-to-noise ratio for the transmitter so it remains stable during splashing or unstable tank event. For this change, the vendor has demonstrated the performance on-site for this application dealing with foaming and it has been found to be acceptable by CCNPP personnel. An additional result of the OE is the vendor discovered multiple echoes due to the fact that the measurement is in an enclosed tank. Since this change is not an enclosed tank, this is not a concern with relation to the transmitter location and mounting.				
5.4.43	**Training:** Are there any changes to or additional training requirements required by the Engineering Change? **[FB0252]**	☒	☐	
Ohmart/Vega radar probes are new components to CCNPP. Personnel shall be trained for the operation and setup of these components per TRR CCNPP-2010-351 as required in 201000229 Form 10, Turnover Closeout Plan.				
5.4.44	**Simulator:** Does the Engineering Change necessitate any changes to simulator hardware, programming, labels, programming, or training requirements?	☐	☒	
5.4.45	**Procedures:** Are there any procedure changes caused by the Engineering Change?	☒	☐	
This change installs new components for monitoring intake structure level. Impacts to existing plant procedures are to be determined by CCNPP per 201000229 Form 10, Turnover Closeout				
5.4.46	**Operational Impact:** Does the Engineering Change potentially change any Operational Requirements?	☐	☒	
5.4.47	**Load Handling:** Does the Engineering Change impact any load handling procedures or load path analysis; are there any specific load handling requirements for installation, implementation, or removal associated with the Change?	☐	☒	

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Sect	Design Input or Change Impact	Applicable Yes	Applicable No	Action Tracking
5.4.48	**Personnel:** Are there any Personnel Requirements and Limitations associate with the Engineering Change, such as the need for trade specialists and engineering experts as well as support personnel, such as Radiation Chemistry technicians, welding technicians with special expertise, use of specific contractor or station procedures for installation or the need for mock-ups for training, installation, or operation?	☐	☒	
5.4.49	**Special Procedures & Specifications:** Are there any special procedures and installation specifications that apply, but are not part of the normal installation procedural direction?	☐	☒	

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Sect	Design Input or Change Impact	Applicable Yes	Applicable No	Action Tracking
5.4.50	**Impacted Organizations:** Does the Engineering Change impact any interfacing departments such as Operations, System Engineering, Training (including Plant Simulator), Maintenance, Reactor Engineering, Radiation Protection and others?	☒	☐	

Indicate the organizations that may be impacted by the proposed activity by checking the appropriate boxes. Use blanks for unlisted organizations or individuals.

Engineering

☒	Mechanical/Civil Design Engineering	☒	Systems Engineering
☒	Electrical/I&C Design Engineering	☐	Engineering Programs
☐	Nuclear Fuel Services	☐	Fire Protection Engineer
☒	Engineering Equipment Reliability	☐	Appendix R Engineer
☒	Procurement Engineering	☐	
☐		☐	

Other Organizations

☒	Operations	☒	Maintenance – E&C
☐	Chemistry	☐	Planning – Mechanical
☐	Licensing	☒	Planning – E&C
☐	Rad Protection	☒	PDU
☒	Maintenance – Mechanical	☒	Nuclear Training
☐	Maintenance Support	☐	
☐	Outage Management	☐	
☐		☐	

Sect	Design Input or Change Impact	Applicable Yes	Applicable No	Action Tracking
5.4.51	**Walkdown Requirements:** Determine the need for walkdowns to look at accessibility to the work area(s) and any special installation considerations that need to be addressed during design development	☒	☐	

☒ 10% (pre-design)	☐ Turnover
☐ 50%	☐ Other (Specify)________
☒ 90% (post-design; prior to ECP approval)	☒ Other (Specify) Designer Walkdown for Layout

Sect	Design Input or Change Impact	Applicable Yes	Applicable No	Action Tracking
5.4.52	**Design Progress Meetings:** Determine the need for design progress meetings during design development	☒	☐	

☒ 10% (pre-design)	☐ Other (Specify)________
☐ 50%	☐ Other (Specify)________
☒ 90% (post-design; prior to ECP approval)	☐ Other (Specify)________

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Sect	Design Input or Change Impact	Applicable Yes	Applicable No	Action Tracking

Waiver of Required Walkdowns or Design Progress Meetings: Step 5.3F.5.f allows for waiver of any Walkdown or Design Progress Meeting required by Step 5.3F. Rationale for and approval of such waivers shall be documented in this space:

Approval: ______________________ ______________

Printed Name & Signature **Date**

The space below may be used for documenting any additional results of the Responsible Engineer/Engineering Supervisor Initial Meeting (Record notes, decisions, and actions)

Technical Task/Risk-Rigor Assessment

Page 1 of 1

Attachment 1, Determination of Consequence Factors

HIGH CONSEQUENCE – If the change is not correctly evaluated and addressed, can it result in:	YES	NO
Personal injury, safety issue • Hot environment / heat stress • Diving activities • Hazardous materials • Radiological Hazards >1 Rem for job, Dose rate > 1 Rem/hr. • Any unmonitored release	☐	☒
Reactivity Management Event	☐	☒
Scram	☐	☒
Operability issue involving multiple trains of a safety related system operability	☐	☒
Unplanned Safety System Actuation/Loss	☐	☒

MEDIUM CONSEQUENCE – If the change is not correctly evaluated and addressed, can it result in:	YES	NO
Regulatory non-compliance (environmental, NRC, State)	☐	☒
Lost/limited Generation (≥5%)	☐	☒
Adverse impact on outage (≥2 hours) or project critical path	☐	☒
Operator Workaround or Challenge created or not addressed	☐	☒
Introduction of foreign material	☐	☒
Reactor coolant or steam generator chemistry transient outside of acceptable band.	☐	☒
Operability Issue involving one train of safety related equipment	☐	☒
Tech Spec violation	☐	☒
Reportable environmental consequence or violation	☐	☒

LOW CONSEQUENCE – If the change is not correctly evaluated and addressed, can it result in:	YES	NO
Unplanned Component Unavailability	☐	☒
Unbudgeted financial consequences (≥$50K)	☐	☒

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Sect	Design Input or Change Impact	Applicable Yes	Applicable No	Action Tracking
Unplanned entry into a Tech Spec shutdown LCO		☐	☒	
Unplanned Security vulnerability		☐	☒	
Aggregate review: Are there any activities, conditions, or situations that, when combined with this activity, could cause undesirable consequences?		☐	☒	
Repeat functional failure of Maintenance Rule systems, structures or components with potential to create new (a)(1) system.		☐	☒	
Unplanned Fire Protection Vulnerability • Emergency Plan affected • High sensitivity issue with public or Regulator		☐	☒	
Other unacceptable consequence not listed • Security compensatory actions • Fire protection compensatory actions • Emergency plan affected • NPDES permit affected • High sensitivity issue with public or regulator. • Potential adverse reduction in safety or production margins • Other		☐	☒	

Attachment 2, Determination of Risk Factors

HUMAN PERFORMANCE RISK FACTORS – Is the likelihood of a technical error increased by:	*YES*	*NO*
Overconfidence/complacency, "can-do" attitude	☐	☒
Mental state (such as stress, illness, fatigue)	☐	☒
Conflicts (personality)	☐	☒
Knowledge/experience, low proficiency lack of skills/training/qualification.	☐	☒
First time or non-routine evolution	☐	☒
Method changed or new process/procedure	☐	☒
Infrequently performed	☐	☒
Frequently performed (habit intrusion), repetitive actions or monotony	☐	☒
High Complexity	☐	☒
Inadequate information available/problem not clearly understood	☐	☒
Availability/complexity of tools	☐	☒
Group think, lack of independence	☐	☒
High workload/schedule pressure	☒	☐
Distraction/interruptions	☐	☒
Availability of resources (people)	☐	☒
Unclear goals, roles, responsibilities	☐	☒
Lack of or unclear standards	☐	☒
Omission/failure to revise required document	☐	☒
Other issues:	☐	☒
PROCESS RISK FACTORS	**YES**	**NO**
Is the exact scope of the task NOT completely understood?	☐	☒
Are there parts of the task process/procedure that cannot be followed? Are we Out-Of-Process (OOPS)? Are there parts of the task the current process does not address?	☐	☒
Are parts of the process or task not understood, unclear, or controversial	☐	☒
Is task on a fast track?	☒	☐

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Sect	Design Input or Change Impact	Applicable Yes	Applicable No	Action Tracking
Is an outside organization, including external design organizations and equipment vendors, providing significant inputs?		☒	☐	
Are the critical parameters NOT known?		☐	☒	
Are all the tools, programs, and procedures necessary for the task NOT available and useable?		☐	☒	
Is design basis NOT known?		☐	☒	
Are multiple parties or disciplines involved such that errors may be introduced via communication channels or coordination?		☐	☒	
Is this a Station first-time action, configuration change or process:		☐	☒	
Is this a First of a Kind task? Will the product or process result in operation outside of industry Operating Experience? Is it the first application of this technology or methodology in the nuclear industry?		☐	☒	
Other factors not listed:		☐	☒	
TOTAL number of applicable Risk Factors		3		

(Medium Consequence) and (<=3 Risk Ranking Factors) = Requires normal review by station.

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Attachment 4, Technical Task Pre-Job Brief Form

Supervisor Performing Brief: Brad Wright	**Date of Brief: 4/8/2010**
Participants: David Dvorak – CCNPP **David Goode – S&L**	**Document/Task ID:** **ECP-10-000208**
Task Description: **Intake Structure Level Radar Probe Installation – Temporary Installation**	
Resources and Estimated Time to Complete:	**Task Due Date/Time: 4/30/2010**
Minimum Briefing Expectations	**Key Briefing Points**
Define Scope: Clearly define the task and what the task entails (scope). Discuss how the scope of the task was validated.	Install 18 radar level probes (9 per unit) at the intake structure for level monitoring. Each unit will have 6 installed downstream of the traveling screens and 1 upstream of the trash rakes. Additionally, a probe is installed on the bay side of traveling screens 11B and 12A (Unit 1) and 25B and 26A (Unit 2). All radar probes will provide local level indication, except those installed upstream of the trash rakes. These probes will have a remote indicator mounted near the transmitter.
Roles and Responsibilities: Clearly define roles and Responsibilities (such as performer, preparer, checker, independence of verifier, project coordinator, corporate, Non Station Personnel).	Design to be prepared by S&L In-house RE is Dave Dvorak
Critical Parameters: Assumptions, inputs, or requirements that if allowed to be untrue or not met, would adversely affect the task outcome.	None
Procedure/Standards: Discuss and ensure proper understanding of the procedures and standards applicable to the task (such as	CNG-CM-1.01-1003 Rev 02 (Design and Config.) CNG-CM-1.01-1004 Rev 00 (Temp Change Process)

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Sect	Design Input or Change Impact	Applicable		Action Tracking
		Yes	No	

Equipment Reliability, Configuration Control, Standards, and Industry Codes & Standards) Bring copy (copies) of governing process procedure for the task to the brief.	CNG-CM-1.01-2003, Rev 00 (Owner Acceptance) List all output products (ECN, etc) RE to develop Forms
Training and Qualification: Review personnel qualifications. Establish appropriate mentoring and oversight of appropriate.	None
Lessons Learned: Discuss previous lessons learned and experience (Operating Experience, Corrective Action Program & individual) that may be applicable to this task, particularly those involving human performance errors.	Clearly identify mounting details and ensure hardware specified on BOM is consistent with mounting.
Fundamentals: Discuss applicable fundamentals.	

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Attachment 4, Technical Task Pre-Job Brief Form (Continued)

Additional Briefing Topics validate the risk factors chosen with the briefing members.			
Consequence Mitigation: For each consequence factor identified in Attachment 1, list the factor and the actions to be employed.			
Consequence Factors	Compensating Action	Owner	Due Date
NONE			
Human Performance Risk Mitigation: For each human performance risk factor identified in Attachment 2, list the factor and the actions to be employed to mitigate that risk.			
Risk Factors	Compensating Action	Owner	Due Date
High workload/schedule pressure **(late design start – required for 6/1 commitment)**	**1. S&L to use resources in Chicago** **2. Coordinate CCNPP project team for design inputs and review.**	**1. Goode** **2. Wright**	**1. 4/30/2010** **2. 4/30/2010**
Process Risk Mitigation: For each process risk factor identified in Attachment 2, list the factor and the actions to be employed to mitigate that risk.			
Risk Factors	Compensating Action	Owner	Due Date

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Sect	Design Input or Change Impact	Applicable Yes	Applicable No	Action Tracking
Project on a fast track	**1. Parallel reviews by S&L and CENG** **Supplemental reviewers** **2. Owner Acceptance Review process**	**1. Goode** **2. Wright**		**1. 4/30/2010** **2. 4/30/2010**
Outside Design Organization	**Communicate with vendor to confirm operation and setup of radar probes**	**Goode**		**4/30/2010**

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Attachment 4, Technical Task Pre-Job Brief Form (Continued)

Required?	Follow-up Action	Owner	Date	Tracking Mechanism
Yes	Progress update	**S&L**	**Weekly**	**Project Schedule**
Yes	10% / 50% / 90% reviews	**S&L**	**As scheduled**	**Project Schedule**
No	ITPR or CRB			
No	Additional Pre-job Brief			
No	Other (specify)			

FORM 7, DESIGN CHANGE TECHNICAL EVALUATION

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1.0 PURPOSE: (CNG-FES-007, SECTION 5.3.1)

- *Provide an overview of the proposed activity and its interfaces.*

This change installs eighteen radar level probes (9 per unit) in the intake structure to monitor level. Twelve radar level probes (6 per unit) are installed downstream of the traveling screens in existing penetrations located on the intake structure. Four radar probes (2 per unit) are installed between the trash rakes and the traveling screens. These components each contain local indication for monitoring. An additional two radar probes (1 per unit) are located upstream of the trash rakes. These components are located under the intake structure grating and are not easily accessible for local monitoring due to safety requirements in proximity to the bay. As a result, remote indication modules are connected to these radar probes and installed on power racks located between traveling screens for monitoring. All radar probes are powered from 24Vdc power supplies (one per unit), distributed to each probe from local junction boxes also installed per this change.

Installation of this change is a temporary configuration for the radar probes. The permanent installation will provide remote indication back to the control room per ECP-10-000209.

2.0 FUNCTIONS: (CNG-FES-007, SECTION 5.3.2)

- *Basic functions of each structure, system, and component. (NQA-1, Question 1)*

The basic function of the radar probes is to provide local indication to monitor intake structure level. All components installed per this change are expected to operate during all plant modes. However, instrumentation is for monitoring purposes only and is not credited for any function during off-normal, abnormal, or emergency operation.

- *Interface requirements including definition of the functional and physical interfaces involving structures, systems, and components: (NQA-1, Question 7)*
  - *The effect on existing plant equipment capability, such as DC battery loads, AC bus capacity, available stored water inventory, service instrument air capacity, water systems capability (intake, service and component cooling water), and HVAC capability;*

    Instrumentation installed per this ECP is powered through temporary 24Vdc power supplies that receive 120Vac power from local welding carts located in closed, ventilated rooms near screens 14B (Unit 1) and 25A (Unit 2). Welding carts are to be supplied by CCNPP prior to implementation of this change. See Form 7A for evaluation of the power supply.

  - *The effect of cumulative tolerances in the design;*

    The changes associated with this ECP do not impact cumulative design tolerances.

  - *The effect on design and safety analyses to ensure the analytical bases remain valid;*

    Changes associated with this ECP do not impact the design of safety analyses.

  - *The compatibility with unimplemented design changes to specify any required sequence for implementation;*

    Changes associated with this ECP are not impacted by unimplemented design changes.

  - *Compatibility with Technical Specification requirements.*

    Changes associated with this ECP do not have any Technical Specification requirements.

3.0 FAILURE EFFECTS: (CNG-FES-007, SECTION 5.3.3)

- *Failure effects requirements of SSCs including a definition of those events and accidents which they must be designed to withstand. (NQA-1, Question19)*

Instrumentation installed per this change is not credited for monitoring in any design basis event or accident. A failure of a radar probe would only result in the loss of local intake structure level indication. Separate instrumentation is relied upon for control

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room indication of intake structure level and circulation pump control. Therefore, no failure effects requirements are associated with this change.

- *Reliability requirements of structures, systems, and components including their interactions, which may impair functions important to safety (NQA-1, Question 30)*

Instrumentation installed per this change is relied on for non-safety related local monitoring and does not interface with any systems or components providing control related to safety. Therefore, no equipment reliability requirements are associated with this change.

4.0 CODES, STANDARDS, REGULATORY REQUIREMENTS AND CLASSIFICATION: (CNG-FES-007, SECTION 5.3.4)

- *Codes and standards, regulatory requirements and commitments or responses to Federal, State, and Local Regulations (NQA-1, Question 3).*

1. IEEE STD. 383 – Qualifying Class 1E Electric Cables and Field Splices for nuclear Generating Stations, Rev. 2008
2. NFPA 70: NEC Handbook 2008
3. AWS D1.1, Structural Design and Welding
4. AISC Manual of Steel Construction, 7th Edition or later
5. ACI 318-63 Building Code Requirements for Reinforced Concrete

5.0 REFERENCES:

1. VTM 12335-030 - Ohmart-Vega Vendor Manual
   a. Tab 1 – VEGAPULS 65 Operating Instructions
   b. Tab 2 – PLISCOM Operating Instructions
   c. Tab 3 – VEGADIS61 Operating Instructions
2. Phoenix Contact Data Sheet 102050_01_en – Power Supply Unit, Primary Switch Mode

CONTINUATION

Design Change Technical Evaluation must include FORM 7A or FORM 7B or both.

☒ FORM 7A is attached

☐ FORM 7B is attached

☐ Additional Pages attached

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6.0 DESIGN OPERATION AND PERFORMANCE REQUIREMENTS:

- *Performance requirements such as capacity, rating, system output (NQA-1, Question 2).*

  The maximum measurable range of the radar probes installed per this change is approximately 99 ft (Ref 1). Each radar probe is mounted around the 10' elevation and required to measure intake structure level which reaches a maximum depth at the (-) 26' (below sea level) elevation corresponding to the bottom of the intake structure at the traveling screens. Therefore, the maximum range needs to encompass at least 40' to allow for some margin in the measurement. Therefore, the range of the probes is acceptable with respect to the process.

  To satisfy a plant commitment, the probes must provide local indication of intake structure level. All radar probes are equipped with an integral display mounted on the top of each module. The exception is the two probes installed under the front of the intake structure upstream of the trash rakes. Due to safety requirements required to access these components for monitoring, each has an indicator routed back to power racks between the nearest set of traveling screens. This location does not have any special safety requirements and is easier for personnel to view. As a result, this change satisfies the plant commitment to install local level indication at the intake structure.

  No existing accuracy requirements are placed on intake structure level monitoring. However, resolution is 0.039 in. with a deviation of ± 0.394 in. per (Ref. 1). This shall be used for operations to determine impact of local indicators on procedures.

- *Design conditions such as pressure, temperature, fluid chemistry, and voltage. (NQA-1, Question 4)*

  Radar probes are 2-wire (loop powered) components powered by an external 24Vdc power supply. Each probe provides a 4-20mA output signal to its local indicator.

- *Loads such as seismic, wind, thermal, and dynamic; the cumulative effect of design changes on the analytical design basis. (NQA-1, Question 5)*

  See the structural requirements section below for an evaluation of the effects on loading due to this change.

- *Material requirements including such items as compatibility, electrical insulation properties, protective coatings, and corrosion resistance. (NQA-1, Question 8)*

  All components installed per this change, including mounting assemblies, junction boxes and cables are located outdoors on the intake structure and exposed to the elements. Stainless steel shall be used to fabricate mounting assemblies to prevent rusting. Junction boxes shall have a rating of NEMA 4 or better, and cables shall be outdoor type to withstand exposure to weather.

- *Mechanical requirements such as vibration, stress, shock, and reaction forces. (NQA-1, Question 9)*

  Each radar probe located downstream of the traveling screens is installed in an existing penetration using a mounting assembly designed per this change. The mounting assembly is made up of three sub-assemblies that are bolted together. The design of the assembly secures the radar probe to the intake structure, allows easy cable terminations and view of indication, and protects the radar probes from damage. Detailed drawings of the assembly and sub-assemblies can be found in Form 9, Installation and Testing Instructions.

- *Structural requirements covering such items as equipment foundations and pipe supports. (NQA-1, Question 10)*

  Water level monitoring instrumentation is to be mounted to the intake structure. Since the intake structure concrete is safety-related, these mountings will be analyzed per Seismic Category II over I criteria. The SSE loading condition will be conservatively considered with normal operating allowable loads. Unless noted otherwise, the peak acceleration is conservatively used to qualify the instrumentation supports, with 1% damping considered for SSE. The SSE accelerations are 1.875 x (6.20g) = 11.625g (use 11.7) horizontally and 1.875 x (0.51g) = 0.956g (use 1.0) vertically. Accelerations are taken from the response spectrum curves in Calculation No. CA04085 Attachment F, for the intake structure.

  Form 9 Installation sketches 001 and 002 depict the mounting of the VEGAPULS 65 radar sensor and PLICSCOM adjustment module through penetrations in the concrete. The combined weight of the radar sensor and the module is about 7 lbs. The critical element of the support is the 1" x 1" x 1/8" angle in the penetration. The total weight of the support is approximately 50 lb, and the actual natural frequency of the support is about 39 Hz, which is in the rigid zone of the response spectra curves. The corresponding ZPA for all the curves is below unity, which is much less than the conservatively assumed peak accelerations. Therefore, all components of the support are considered acceptable by Engineering Judgement based on the small magnitude of the seismic loads.

Form 9 Installation sketch 003 depicts the mounting of the VEGAPULS 65 radar sensor and PLICSCOM adjustment module to the wall of the intake structure. The total weight of the equipment and mounting is approximately 17 lbs. Dead load plus peak SSE seismic load results in applied bending moments of Mx = 1000 in-lbs and My = 1000 in-lbs in the plane of the wall, and Mz = 826 in-lbs as a torsional moment perpendicular to the plane of the wall. These moments result in a maximum plate bending stress of 7.9 ksi, which is much less than the 22.5 ksi bending stress allowed for a stainless steel plate under normal operating loads. Thus, the plates are adequate for supporting the equipment. The 3/16" all-around fillet weld has a capacity of 2.98 kips/in, which is much greater than the applied dead plus SSE seismic weld stress of 0.73 kips/in. Therefore, the all-around fillet weld is acceptable. The dead plus SSE seismic tension in a single anchor is computed to be 427 lbs, while the dead plus SSE seismic shear in a single anchor is computed to be 140 lbs. Given the allowable tension load of 683 lbs and an allowable shear load of 1261 lbs, the linear interaction ratio for the anchors is 0.73 which is less than 1.0 and is therefore acceptable.

Form 9 Installation sketch 004 depicts the mounting of the VEGADIS 61 level indicator. The total weight of the equipment and mounting is approximately 30 lbs. The equipment is mounted to a ¼" thick backing plate. Applying dead load and peak SSE seismic loads and considering the backing plate acting in one-way bending results in a plate bending stress of 14.4 ksi, which is less than the 22.5 ksi allowed for a stainless steel plate under normal operating loads, and therefore the plate is adequate. The anchors used are the same as in Installation Sketch 003 while the anchor forces here are less than those in Form 9 Installation Sketch 003. Thus the anchors are acceptable by comparison.

Form 9 Installation sketch 008 depicts the mounting of the VEGAPULS 65 radar sensor and PLICSCOM adjustment module underneath grating. The total weight of the equipment and mounting is approximately 15 lbs. The actual natural frequency of the support is approximately 50 Hz, which is in the rigid zone of the response spectra curves. The corresponding ZPA for all the curves is below unity, which is much less than the conservatively assumed peak accelerations. Therefore, all components of the support are considered acceptable by Engineering Judgement based on the small magnitude of the seismic loads.

In addition to the installation of instrument supports, a 3" deep trench will be excavated in the EL 10'-0" floor slab of the intake structure to accommodate buried conduit. The top 1 ½" of the concrete is concrete topping, and has no structural purpose. Thus, only 1 ½" of the concrete excavated is structural concrete. With the removal of the 1 ½" of structural concrete, the moment capacity of the beam is reduced by approximately 15%, and this trenched portion of the slab will not see its design live load during this time. When the trenches are filled with concrete, after 28 days, the slab will be at full design capacity. Therefore, the removal of this concrete has an insignificant impact on the structural integrity of the EL 10'-0" slab.

- *Hydraulic requirements such as pump Net Positive Suction Head (NPSH), allowable pressure drops, and allowable fluid velocities. (NQA-1, Question 11)*

  The changes associated with this ECP have no hydraulic requirements.

- *Chemistry requirements including provisions for system flushing, batch sampling and in-line sampling. Power plant water chemistry treatment for primary systems, steam generator, and plant limitations on water chemistry. (NQA-1, Question 12)*

  The changes associated with this ECP have no chemistry requirements.

- *Electrical requirements such as source of power, load profile voltage, electrical insulation, motor requirements, physical and electrical separation of circuits and equipment; the effect of cable routing or rerouting on the cable tray system (loading, seismic capability, and capacity l imitations) (NQA-1, Question 13)*

  The radar probes are two-wire devices that are loop powered from two temporary 24 Vdc power supplies (one per unit, 9 probes each). Each radar probe has a maximum current of 22 mA. Therefore, each power supply must be able to supply the maximum current draw from each radar probe all at once or 248 mA (125% max). The Phoenix Contact power supply specified on Form 11, Bill of Materials has a maximum output current output of 1.5A (Ref 2). In addition, the power supply complies with DIN VDE 0106 for protection against electric shock (Ref. 2) as required by the radar probe (Ref. 1).

  Power is supplied via 2/C shielded 20 AWG cable for all radar probes. Power cable is distributed at junction boxes located inside each equipment where the power supplies are installed. Junction boxes are also installed at the far north end and far south end of the traveling screens to distribute power to radar probes upstream of the traveling screens. See Form 9, Installation and Testing Instructions for a detailed junction box layout. Wiring for the two remote indicators for 1LIT1100 and 2LIT2100 is via a 4/C shielded 20 AWG cable routed in conduit along with its power cable.

Embedded conduit is installed to support the power cable for radar probes mounted upstream of the traveling screens and below the 10' elevation surface of the intake structure. Conduit below grade shall be encased in concrete per Form 9, Installation and Testing Instructions, and shall be Polystyrene Type I conduit per E-406 Section 105.3; all other conduit shall be rigid steel. The route for the conduit starts between traveling screens 11B/12A for 1LIT1100, 1LIT1101B and 1LIT1201A and between 25B/26A for 2LIT2100, 2LIT2501A and 2LIT2601A. Conduit is not scheduled per this change. See Form 9, Installation and Testing Instructions for detailed conduit routing. All other power cable installed per this change is unscheduled and field routed open air between junction boxes and radar probes.

All cable shall be procured as non-safety related cable and takes exceptions to specification SP-317 as this change is to a non-safety related system. Cable selected contains PVC insulation and jacket; however, it meets the specifications for Class 1E cable per IEEE 383. Therefore, it meets the recommendations from Fire Protection in Form 17, Fire Protection/Appendix R Review Fire Protection Design Features Checklist for flame retardant cable and is qualified to be installed in nuclear power plants. In addition, the cable is only rated to 300V rather than 600V. This is acceptable however, since this is a low voltage application (24 Vdc) that will never challenge the voltage rating of the cable. Since the permanent modification will route cable through conduit rather than free-air, all conduits are dedicated conduit and will not contain any other cables. Therefore, it is acceptable to procure cable as non-safety related cable as it will not be routed with any safety related cable upon permanent installation.

- *Operational requirements under various conditions, such as startup, normal operation, shutdown, maintenance, abnormal or emergency operation, special or infrequent operation including installation of design changes, and the effect of system interaction. (NQA-1, Question 15)*

  The changes associated with this ECP are temporary and provide local indication of intake structure level during normal operation. They are not required to operate during any other plant condition.

- *Instrumentation and control requirements including indicating instruments, controls, and alarms required for operation, testing, and maintenance. Other requirements such as the type of instrument, installed spares, range of measurement, and location of indication are included (NQA-1, Question 16)*

  This change installs radar level probes at the Unit 1 and Unit 2 Intake structure. Each radar probe is assigned a CompID per this change as follows:

CompID	Description
1LIT1100	Unit 1 Trash Rake Upstream Radar Level
1LIT1101A	Traveling Screen 11 Upstream Radar Level
1LIT1101B	Traveling Screen 11 Downstream Radar Level
1LIT1201A	Traveling Screen 12 Upstream Radar Level
1LIT1201B	Traveling Screen 12 Downstream Radar Level
1LIT1301B	Traveling Screen 13 Downstream Radar Level
1LIT1401B	Traveling Screen 14 Downstream Radar Level
1LIT1501B	Traveling Screen 15 Downstream Radar Level
1LIT1601B	Traveling Screen 16 Downstream Radar Level
2LIT2100	Unit 2 Trash Rake Upstream Radar Level
2LIT2601A	Traveling Screen 26 Upstream Radar Level
2LIT2601B	Traveling Screen 26 Downstream Radar Level
2LIT2501A	Traveling Screen 25 Upstream Radar Level
2LIT2501B	Traveling Screen 25 Downstream Radar Level
2LIT2401B	Traveling Screen 24 Downstream Radar Level
2LIT2301B	Traveling Screen 23 Downstream Radar Level
2LIT2201B	Traveling Screen 22 Downstream Radar Level
2LIT2101B	Traveling Screen 21 Downstream Radar Level

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An integral indicator is mounted on top of the electronics module for each radar probe. Components 1LIT1100 and 2LIT2100 are located inside a safety area that requires personal floatation gear for entry. Therefore, these probes are connected to indicators that are mounted on existing power racks between traveling screens that have no safety requirements to view. The level indicators are assigned the following CompIDs:

CompID	Description
1LI1100	Unit 1 Trash Rake Upstream Radar Level Indicator
2LI2100	Unit 2 Trash Rake Upstream Radar Level Indicator

The integral indicator also serves as the configuration/adjustment module. The initial setup and configuration of the radar probes is to be performed by the vendor with oversight from CCNPP. This setup shall be documented and maintained to support documentation required for the permanent installation of these transmitters per ECP-10-000209. Also see Form 10, Turnover Closeout Plan for this requirement.

- *Redundancy, diversity, and separation requirements of structures, systems, and components. (NQA-1, Question 18)*

  Since the power supply and current signal are carried on the same two-wire cable, reliable separation shall exist between the supply circuit and the mains circuit per DIN VDE 0106 part 101.

- *Fire protection or resistance requirements: Safe shutdown analyses, the introduction of safe shutdown equipment into fire areas; Routing of piping and electrical cables and the necessity for cable fireproofing and/or fire stops; Fire detection and fire suppression capability; Fire barrier capability including fire door installation; Fire dampers; Access to fire fighting and emergency equipment; Use of non-combustible materials; Introducing combustible materials into safe shutdown areas by design or during installation or operation; Smoke and toxic gas generation. (NQA-1, Question24)*

  The changes associated with this ECP do not have and Fire protection or resistance impacts. Cable installed at radar probes is approved per IEEE 383 to be flame retardant as recommended by CCNPP Fire Protection. See Forms 16 and 17 for Fire Protection and Appendix R screenings.

- *Requirements for criticality control and accountability of nuclear materials (NQA-1, Question 32)*

  The changes associated with this ECP do not have any control or accountability of nuclear materials requirements.

7.0 APPLICATION ENVIRONMENT: (CNG-FES-007, SECTION 5.3.3)

- *Environmental conditions anticipated during storage, construction, operation, and accident conditions, such as pressure, temperature, humidity, corrosiveness, site elevation, wind direction, exposure to weather, flooding, nuclear radiation, electromagnetic radiation, and duration of exposure; qualification test requirements; shelf or service life limitations. (NQA-1, Question 6)*

  All components installed per this change are located outside at the intake structure. It can be conservatively assumed that the outdoor temperature at the intake structure has a maximum range of 0°F to 110°F. All components installed per this change, except the power supply, are required to withstand exposure to rain, sleet and snow. In addition to the ambient (outside) temperature, the radar probes measure water level from the Chesapeake Bay that is conservatively assumed to have a temperature range of 32°F to 90°F.

  The radar level probes have an ambient rating of -40°F to 176°F (Ref. 1) for their electronics unit, which is acceptable for the expected operating and storage range. In addition, each probe can measure a medium with a process temperature between -40°F and 266°F (Ref. 1), which is acceptable for the expected process temperature. In addition, each probe comes with an integral or remote indicating module with an ambient temperature rating of 5°F to 158°F (Ref. 3 and 4), which is just outside the lower end of the expected operating range. However, due to the nature of this change (non-safety related), the fact that the modules only provide local indication, and the unlikely probability that local indication would be desired during these extreme temperatures; the difference in rated and expected temperatures is acceptable.

  The power supply has an ambient temperature of 5°F to 158°F (Ref. 2), which is also just outside the lower end of the expected operating range. This is acceptable for the same reasons as the indicators. In addition, the power supply is mounted inside a structure, which is likely to be warmer than the outside temperature.

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8.0 FIELD INTERFACE: (CNG-FES-007, SECTION 5.3.3)

- *Layout and arrangement requirements. (NQA-1, Question 14)*

  The changes associated with this ECP do not have layout or arrangement requirements.

- *Test requirements including pre-operational and subsequent periodic tests and the conditions under which they will be performed. (NQA-1, Question 20)*

  Preliminary and Post-installation testing shall be performed as required per Form 9, Installation and Testing Instructions. No periodic testing is required per this change.

- *Accessibility, maintenance, repairs and pre-service and in-service inspection requirements for the plant including the conditions under which these will be performed (NQA-1, Question 21).*

  The changes associated with this ECP do not have accessibility, maintenance, repairs or pre-service and in-service inspection requirements. Per VTM 12335-030, radar probes do not require periodic maintenance; however, it is suggested that a check of the instrument measurement against a field measurement be performed at intervals in accordance with the Equipment Reliability Classification established per 201000229 Form 10, Turnover Closeout Plan.

- *Personnel requirements and limitations including the qualification and number of personnel available for operation, maintenance, testing and inspection, and radiation exposures to the public and facility personnel. (NQA-1, Question 22)*

  The changes associated with this ECP do not have personnel requirements or limitations.

- *Safety requirements for preventing personnel injury including such items as radiation safety, minimizing radiation exposure to personnel, criticality safety, restricting the use of dangerous materials, escape provisions from enclosures, and grounding of electrical systems (NQA-1, Question 28).*

  Installation of 1LIT1100 and 2LIT2100 require entry into a safety area that requires personal floatation gear. This should be worn when applicable in accordance with Form 9 Installation and Testing Instructions.

- *Materials, processes, parts, and equipment suitable for application. (NQA-1, Question 27)*

  Material used for this application shall be per Form 11, ECP Materials List.

- *Quality and quality assurance requirements (NQA-1, Question 29)*

  The components installed per this change are non-safety related. Therefore, NQA-1 requirements are not specifically invoked.

- *Interface requirements between equipment and operation and maintenance personnel (NQA-1, Question 31)*

  Local indication at each radar probe and 1LI1100 and 2LI2100 may be incorporated into procedures as necessary. This is to be determined by operations per 201000229 Form 10, Turnover Closeout Plan.

- *Load path requirements for installation, removal, and repair of equipment and replacement of major components (NQA-1, Question 33)*

  The changes associated with this ECP have no load path requirements.

9.0 OTHER (NQA-1 REQUIREMENTS): (CNG-FES-007, SECTION 5.3.3)

- *Security requirements to include access and administrative control requirements and system design requirements including redundancy, power supplies, support system requirements, emergency operational modes, and personnel accountability(NQA-1, Question 17)*

  Each radar probe indicating and adjustment module has the option to require a PIN number to change the settings. If the PIN option is enabled, only read functions are permitted without entering the PIN. This option is a permanent setting on the module.

- *Transportability requirements such as size and shipping weight, limitations, ICC regulations. (NQA-1, Question 23)*

  The changes associated with this ECP do not have transportability requirements.

- *Other requirements to prevent undue risk to the health and safety of the public. (NQA-1, Question 26)*

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The changes associated with this ECP do not have other requirements related to public health and safety.

- *Handling, storage, cleaning, and shipping requirements. (NQA-1, Question 25)*

  The radar probes installed per this change shall be shipped in accordance with instruction notes included on packaging. Storage shall be in un-opened original packaging until time of use. Storage shall be under the following conditions:

  - Not in the open
  - Dry and dust free
  - Not exposed to corrosive media
  - Protected against solar radiation
  - Avoiding mechanical shock and vibration

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A. INSTRUCTIONS:

Describe in Section B the **operational impact** of the change, if any. Include the following, as applicable,

- How the system operates differently from what it did.
- Changes to the functional and operational characteristics of the system or component.
- Impact of this change on the simulator software, e.g., flow coefficients on valves, and hardware
- Electrical, interlocks, power supply, control schemes, alarm and indication changes, etc.
- Mechanical, capacity changes, effect on system parameters, setpoint changes, equipment additions or deletions, cross-connects, etc.
- Valves, changes in failure mode, capacity drive mechanism, actuation signal alarm and control function or location, etc.
- Technical Specification, "How must operation change?
- Are new or changes to current operator movements required?
- Potential for this modification to inadvertently cause a reactor trip, including:
  - Is any affected equipment trip sensitive or located in trip sensitive areas?
  - Will construction activities impact trip sensitive equipment or occur in trip sensitive areas?
- Will post modification testing impact trip sensitive equipment?
- Potential for this modification to impact the control or monitoring of reactivity
- Potential for this modification to affect changes in operating limits and changes in operating margins and strategies.
- For core reloads, evaluate and determine if there will be significant changes in core physics or reactivity coefficients.
- Recommendations for special post-modification testing, e.g., testing for potential side effects due to changes in system operation, if any
- Recommendations for special Operator or Maintenance training, if any
- Recommendations for unique configuration documentation or logistic support, if any.
- Other recommended compensatory measures, if any, e.g., mock-ups, simulation
- Discuss potential for MOD to impact control of core monitoring, operating limits, core physics, and reactivity

B. DESCRIPTION OF IMPACT:

This change installs radar level probes at the Intake Structure to provide local level indication only. Each probe provides local level indication from an electronic display module on the top of the component. The exception is the indication for 1LIT1100 and 2LIT2100, which is located on the east intake structure wall at new remote indicators 1LI1100 and 2LI2100 respectively. The radar probes do not provide control room indication or alarms and have no input to control related functions. Refer to the following ECNs for location of instrumentation for level monitoring.

CompID	Description	ECN
1LIT1201B	Traveling Screen 12 Downstream Radar Level	ECP-10-000208 61348-0039
1LIT1301B	Traveling Screen 13 Downstream Radar Level	ECP-10-000208 61349-0042
1LIT1401B	Traveling Screen 14 Downstream Radar Level	ECP-10-000208 61349-0042
1LIT1501B	Traveling Screen 15 Downstream Radar Level	ECP-10-000208 61349-0042
1LIT1601B	Traveling Screen 16 Downstream Radar Level	ECP-10-000208 61349-0042
2LIT2501B	Traveling Screen 25 Downstream Radar Level	ECP-10-000208 63351SH001-0031
2LIT2401B	Traveling Screen 24 Downstream Radar Level	ECP-10-000208 63350-0040
2LIT2301B	Traveling Screen 23 Downstream Radar Level	ECP-10-000208 63350-0040
2LIT2201B	Traveling Screen 22 Downstream Radar Level	ECP-10-000208 63350-0040
2LIT2101B	Traveling Screen 21 Downstream Radar Level	ECP-10-000208 63350-0040

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CompID	Description	ECN
1LIT1101B	Traveling Screen 11 Downstream Radar Level	ECP-10-000208 61348-0039
2LIT2601B	Traveling Screen 26 Downstream Radar Level	ECP-10-000208 63351SH001-0031
1LIT1101A	Traveling Screen 11 Upstream Radar Level	ECP-10-000208 61348-0039
1LIT1201A	Traveling Screen 12 Upstream Radar Level	ECP-10-000208 61348-0039
2LIT2601A	Traveling Screen 26 Upstream Radar Level	ECP-10-000208 63351SH001-0031
2LIT2501A	Traveling Screen 25 Upstream Radar Level	ECP-10-000208 63351SH001-0031
1LIT1100	Unit 1 Trash Rake Upstream Radar Level	ECP-10-000208 61348-0039
2LIT2100	Unit 2 Trash Rake Upstream Radar Level	ECP-10-000208 63351SH001-0031
1LI1100	Unit 1 Trash Rake Upstream Radar Level Indicator	ECP-10-000208 61348-0039
2LI2100	Unit 2 Trash Rake Upstream Radar Level Indicator	ECP-10-000208 63351SH001-0031

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A. INSTALLATION INSTRUCTIONS:

This installation section addresses the changes associated with the installation of new radar level probes at the Units 1 & 2 intake structure. All work associated with this ECP is classified as non-safety related and shall be planned and controlled in accordance with the applicable procedures and specifications.

General Notes:

Installation steps may be performed in any order as determined by the field.

Radar probes are being installed in temporary configuration by this modification. The level indicator numbers have been assigned in FCMS but the junction box numbers, cable numbers and conduit numbers will not be assigned due to this temporary configuration. Conduits will be installed in the concrete and will remain in place but will not be numbered until the next modification ECP-10-000209 is installed.

Precautions:

When working on the waterfront, ensure site procedures for fall protection, Personnel Floatation Devices and other safety procedures are followed.

Pre-Installation Instructions

NOTE: Additional assemblies shall be fabricated at the direction of the System Engineer

1. Fabricate 6"/8" penetration mounting sub-assemblies (10 each) per Installation Sketch #1
2. Fabricate 10" penetration mounting sub-assemblies (2 each) per Installation Sketch #2
3. Fabricate anchor mounting brackets (4 each) per Installation Sketch #3
4. Fabricate bay level mounting brackets (2 each) per Installation Sketch #8
5. Fabricate Level Indicator Brackets (2 each) per Installation Sketch #4
6. Build junction boxes per Installation Sketch #6
7. At penetration locations, top of penetration shall be ground flush with concrete to provide flush mounting surface.
8. Replace rail mount attachment with wall mount attachment on 1LI1100 and 2LI2100

Installation Instructions

NOTE: Sub-section installations can be performed in any order.

Junction Box Installation

1. Install junction boxes with supports for transmitters located *upstream* of the traveling screens to intake structure deck per E-406 Section 104.3 Sht 65 (Behind 11B/12A for 1LIT1100, 1LIT1101A and 1LIT1201A and behind 25B/26A for 2LIT2100, 2LIT2501A, and 2LIT2601A per Installation Sketch #5). Mount per Installation Sketch #6. Verify box locations with Project Manager (PM) prior to installation.

2. Install junction boxes with supports for power distribution inside equipment rooms behind screens 14B and 25A per E-406 Section 104.3 Sht 61 (Reference Installation Sketch #5). Mount per Installation Sketch #6. Verify box locations with PM prior to installation.

Embedded Conduit Installation

3. Scan area where conduit is to be installed between screens 11B/12A and 25B/26A, extending out to trash rakes, to identify any existing conduits or rebar (Ref. ECP-10-000208 61348-0039, ECP-10-000208 63351SH001-0031 and Installation Sketch #5).

   NOTE: The following step requires cutting of concrete.

4. Lay out conduit route using Installation Sketches #5 and #9 as a guide to cut concrete from junction boxes to instrument locations for 1LIT1100, 1LIT1101A and 1LIT1201A on the north side of the intake structure and 2LIT2100, 2LIT2501A, and 2LIT2601A on the south side. Remove upper 3" of concrete along conduit route. **CAUTION:** DO NOT cut any rebar or remove concrete more than 3" deep.

5. Install PVC conduit below grade for radar probes upstream of the traveling screens per E-406 and Installation Sketch #9. Conduit length shall be adequate to stub out of concrete once poured.

6. Field route rigid steel conduit from junction boxes to PVC conduits and connect using CG fittings to prevent water intrusion (Reference Installation Sketch #5).

7. Inspect and check conduit to assure continuity and correct position in accordance with E-406 Section 105.3.

8. Encase conduit in concrete per E-406 Section 105.3 and C-0010. Roughen the sides and the bottom of the trench before pouring to ensure adequate bond. Fill trench with structural concrete that conforms to Bechtel Specification 6750-C-9 and shall be class C-1 concrete (4000 psi min at 28 days).

Remote Indicators

9. Field locate remote indicator 1LI1100 either on east intake structure wall or racks located between traveling screen sets. Verify location with PM prior to installation.

10. Install remote indicator mounting brackets for 1LI1100 per Installation Sketch #4 and VTM 12335-030 Tab 3.

11. Field locate remote indicator 2LI2100 either on east intake structure wall or racks located between traveling screen sets. Verify location with PM prior to installation.

12. Install remote indicator mounting brackets for 2LI2100 per Installation Sketch #4 and VTM 12335-030 Tab 3.

Power Supply Installation

NOTE: The following steps assume welding carts are located inside structure intake equipment rooms.

13. Inside intake structure room located behind traveling screen 14B, attach temporary DIN rail to welding cart. Use DIN rail as template to drill holes in cart and attach using machine screws.

14. Attach 24Vdc power supply to DIN rail.

15. Wire electrical pigtail cord to 120VAC input terminals L(+) & N(-).

16. Inside intake structure room located behind traveling screen 25A, attach temporary DIN rail to welding cart. Use DIN rail as template to drill holes in cart and attach using machine screws.

17. Attach 24Vdc power supply to DIN rail.

18. Wire electrical pigtail cord to 120VAC input terminals L(+) & N(-).

Cable Install

NOTE: Cables shall be routed along the existing fiberglass pipes for support and protection. Cables to be attached using tie-wraps or other approved means. Ensure all cables are properly supported and attached.

19. Field route cable between power supplies and corresponding junction boxes inside equipment rooms using E-406 Section 102 as guidance per Installation Sketches #5, #6, and #7. At power supplies, 24Vdc power is connected across terminals 24V and 0V.

20. Field route cables from equipment room junction boxes to radar probe locations for the following components using Installation Sketch #5 and E-406 Section 102 for guidance. Cables shall be routed through existing penetrations in equipment rooms.

CompID	Description	Power Supply Location
1LIT1201B	Traveling Screen 12 Downstream Radar Level	14B Equipment Room
1LIT1301B	Traveling Screen 13 Downstream Radar Level	14B Equipment Room
1LIT1401B	Traveling Screen 14 Downstream Radar Level	14B Equipment Room
1LIT1501B	Traveling Screen 15 Downstream Radar Level	14B Equipment Room
1LIT1601B	Traveling Screen 16 Downstream Radar Level	14B Equipment Room
2LIT2501B	Traveling Screen 25 Downstream Radar Level	25A Equipment Room
2LIT2401B	Traveling Screen 24 Downstream Radar Level	25A Equipment Room
2LIT2301B	Traveling Screen 23 Downstream Radar Level	25A Equipment Room
2LIT2201B	Traveling Screen 22 Downstream Radar Level	25A Equipment Room
2LIT2101B	Traveling Screen 21 Downstream Radar Level	25A Equipment Room

21. Field route power cable from junction box in equipment room behind traveling screen 14B to junction box located at the north end of the intake structure per Installation Sketch #6.

22. Field route power cable from junction box in equipment room behind traveling screen 25A to junction box located at the south end of the intake structure per Installation Sketch #6.

23. Field route cables from junction boxes to radar probe locations for the following components using Installation Sketch #5 and E-406 Section 102 for guidance.

CompID	Description	Junction Box Location
1LIT1101B	Traveling Screen 11 Downstream Radar Level	Intake Structure North End
2LIT2601B	Traveling Screen 26 Downstream Radar Level	Intake Structure South End

24. Wire at junction boxes per Installation Sketches #6 and #7.

    24.1 Cable may have single communication wire, this wire is not used and can be left floating at both ends. Ensure wire does not contact any terminals.

25. Route cable through conduit previously installed between junction boxes and radar probes upstream of the traveling screens as follows:

25.1 Route 20 AWG T.S.P. through ¾" conduit for 1LIT1101A and 1LIT1201A on the north side of the intake structure and 2LIT2501A and 2LIT2601A on the south side.

25.2 Route 20 AWG T.S.P and 4/C 20 AWG cable through ¾" conduit to 1LIT1100 on the north side of the intake structure and 2LIT2100 on the south side.

25.3 Terminate cable in junction box per Installation Sketch #7

26. Route 4/C 20 AWG cable from junction box to 1LI1100 and 2LI2100 respectively per Installation Sketch #5. Terminate cable at junction box per Installation sketch #7 and at indicator per VTM 12335-030 Tab 3.

Radar Probe Installation

NOTE: Radar Probe Installation can be performed in any order

27. Perform the following steps for each radar probe listed below:

CompID	Description	ECN
1LIT1201B	Traveling Screen 12 Downstream Radar Level	ECP-10-000208 61348-0039
1LIT1301B	Traveling Screen 13 Downstream Radar Level	ECP-10-000208 61349-0042
1LIT1401B	Traveling Screen 14 Downstream Radar Level	ECP-10-000208 61349-0042
1LIT1501B	Traveling Screen 15 Downstream Radar Level	ECP-10-000208 61349-0042
1LIT1601B	Traveling Screen 16 Downstream Radar Level	ECP-10-000208 61349-0042
2LIT2501B	Traveling Screen 25 Downstream Radar Level	ECP-10-000208 63351SH001-0031
2LIT2401B	Traveling Screen 24 Downstream Radar Level	ECP-10-000208 63350-0040
2LIT2301B	Traveling Screen 23 Downstream Radar Level	ECP-10-000208 63350-0040
2LIT2201B	Traveling Screen 22 Downstream Radar Level	ECP-10-000208 63350-0040
2LIT2101B	Traveling Screen 21 Downstream Radar Level	ECP-10-000208 63350-0040

27.1 Locate radar probe per the applicable ECN in the table above.

27.2 Install 6"/8" lower mounting assembly in penetration per Installation Sketch #1.

27.2.1 Rotate the lower assembly to the desired position (based on detector head orientation).

27.2.2 Thread a nut on the 3/8" bolt and run close to the bolt head, thread the bolt into the welded nut on the angle leg and tighten the jacking bolts firmly against the penetration side wall. Once tight, lock the bolt in place using the jam nut on the bolt.

27.2.3 Perform above step for all jacking bolts.

27.3 Install 6"/8" upper mounting assembly per Installation Sketch #1. Position the upper plate assembly on top of the lower assembly plate and secure using (4) ¼-20 x 3/4" bolts. Note that the radar probe may be installed in the upper plate prior to installing upper plate to insure correct orientation.

27.4 Install radar probe in threaded coupling of upper mounting assembly and attach electronics module. Verify that the probe orientation is correct, if not rotate the probe or the lower assembly to obtain desired orientation.

27.5 Label radar probe.

27.6 Wire transmitter per installation Sketches #6 and #7 and VTM 12335-030 Tab 1.

27.6.1 Prior to cable entry into gland seal, a drip loop shall be installed to drain moisture away from radar probe electronics and junction box.

27.6.2 Cable has single communication wire, this wire is not used and can be left floating at both ends. Ensure wire does not contact any terminals.

27.7 Install protective cage assembly per Installation Sketch #1. Verify the probe orientation is correct and cables are installed. Attach the guard assembly to the upper plate using (4) ¼-20 SS bolts and nuts. If interference is encountered contact System Engineer and Engineering for resolution.

28. Perform the following steps for each radar probe listed below:

CompID	Description	ECN
1LIT1101B	Traveling Screen 11 Downstream Radar Level	ECP-10-000208 61348-0039
2LIT2601B	Traveling Screen 26 Downstream Radar Level	ECP-10-000208 63351SH001-0031

NOTE: To minimize interference with the penetration, 3" long angle may be trimmed as necessary.

NOTE: Welded nuts on lower assemblies may be lowered up to 2" if needed based on penetration conditions.

28.1 Locate radar probe per the applicable ECN in the table above.

28.2 Install 10" lower mounting assembly in penetration per Installation Sketch #2.

28.2.1 Rotate the lower assembly to the desired position (based on detector head orientation).

28.2.2 Thread a nut on the 3/8" bolt and run close to the bolt head, thread the bolt into the welded nut on the angle leg and tighten the jacking bolts firmly against the penetration side wall. Once tight, lock the bolt in place using the jam nut on the bolt.

28.2.3 Perform above step for all jacking bolts.

28.3 Install 10" upper mounting assembly per Installation Sketch #2. Position the upper plate assembly on top of the lower assembly plate and secure using (4) ¼-20 x 3/4" bolts. Note that the radar probe may be installed in the upper plate prior to installing upper plate to insure correct orientation.

28.4 Install radar probe in threaded coupling of upper mounting assembly and attach electronics module. Verify that the probe orientation is correct, if not rotate the probe or the lower assembly to obtain desired orientation.

28.5 Label radar probe.

28.6 Wire transmitter per installation Sketches #6 and #7 and VTM 12335-030 Tab 1.

28.6.1 Prior to cable entry into gland seal, a drip loop shall be installed to drain moisture away from radar probe electronics and junction box.

28.6.2 Cable has single communication wire, this wire is not used and can be left floating at both ends. Ensure wire does not contact any terminals.

28.7 Install protective cage assembly per Installation Sketch #2. Verify the probe orientation is correct and cables are installed. Attach the guard assembly to the upper plate using (4) ¼-20 SS bolts and nuts. If interference is encountered contact System Engineer and Engineering for resolution.

29. Perform the following steps for each radar probe listed below:

CompID	Description	ECN
1LIT1101A	Traveling Screen 11 Upstream Radar Level	ECP-10-000208 61348-0039
1LIT1201A	Traveling Screen 12 Upstream Radar Level	ECP-10-000208 61348-0039
2LIT2601A	Traveling Screen 26 Upstream Radar Level	ECP-10-000208 63351SH001-0031
2LIT2501A	Traveling Screen 25 Upstream Radar Level	ECP-10-000208 63351SH001-0031

29.1 Locate radar probe per the applicable ECN in the table above.

29.2 Install anchor mounting bracket per Installation Sketch #3. The components shall be installed below grating in front of the traveling screens. Centerline of mounting bracket shall be at the 9'-0" elevation. Verify location and elevation for each bracket prior to drilling anchors. Install mounting bracket using Hilti Kwik Bolt anchors per site procedure and Hilti installation procedure.

29.3 Install radar probe in threaded coupling on mounting bracket and attach electronics module.

29.4 Label radar probe.

29.5 Wire transmitter per installation Sketches #6 and #7 and VTM 12335-030 Tab 1.

29.5.1 Prior to cable entry into gland seal, a drip loop shall be installed to drain moisture away from radar probe electronics and junction box.

29.5.2 Cable has single communication wire, this wire is not used and can be left floating at both ends. Ensure wire does not contact any terminals.

30. Perform the following steps for each radar probe listed below

CompID	Description	ECN
1LIT1100	Unit 1 Trash Rake Upstream Radar Level	ECP-10-000208 61348-0039
2LIT2100	Unit 2 Trash Rake Upstream Radar Level	ECP-10-000208 63351SH001-0031

NOTE: Prior to removing grating or accessing location for Bay Level detectors 1LIT1100 and 2LIT2100, ensure all requirements for fall protection and personal floatation are met.

30.1 Locate radar probe per the applicable ECN in the table above.

30.2 Use conduit route as guide to locate radar probe in grating, cut 8" (±1") square hole in grating and mesh per Installation Sketch #8 (Reference Dwg 61843).

CAUTION: Ensure that cutting does not create instability in grating. Grating shall be banded prior to cutting to ensure stability is maintained during cutting.

30.3 Prior to installation, wire transmitter per installation Sketches #6 and #7, and VTM 12335-030 Tab 1. Ensure adequate cable exists to provide a drip loop when probe position is finalized.

30.4 Install radar probe per Installation Sketch #8. Verify probe and lower assembly are level and plumb.

30.5 Ensure a drip loop exists prior to cable entry in the transmitter gland seal to drain moisture away from radar probe electronics and junction box.

30.6 Cable has a single communication wire, this wire is not used and can be left floating at both ends. Ensure wire does not contact any terminals.

30.7 Tighten bolts and secure instrument in lower mounting assembly and tighten lower assembly to upper assembly.

30.8 Attach upper mounting assembly to grating using grating clips and stainless steel hardware as required.

30.9 Label radar probe.

Radar Probe Configuration

31. Configure Radar probes per vendor instruction and VTM 12335-030 Tab 2.

32. Document As-Left configuration for individual probes as necessary and submit to Lead Design Engineer (LDE). NOTE: A configuration document can be shared between probes with identical configuration, but the applicable components shall be listed on the document.

Power Up System

33. At equipment room located behind traveling screen 14B, plug power supply into 120 Vac welder utility receptacle.

34. At equipment room located behind traveling screen 25A, plug power supply into 120 Vac welder utility receptacle.

Removal Instructions

NOTE: This change does not remove radar probes, level indicators, or junction boxes in support of implementation of ECP-10-000209 for permanent installation.

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1. Inside intake structure rooms located behind traveling screen 14B and 25A, unplug power supply from welder outlet.

2. On power supply, de-terminate 120Vac wiring at terminals L & N and 24Vdc wiring at terminals 24V and 0V.

3. Determinate all cable between power supply and radar probes.

4. Remove cable and place in stock if desired. Cable may be re-used for permanent installation under ECP-10-000209.

5. Inside intake structure rooms located behind traveling screens 14B and 25A, detach power supply from DIN rail mounted on welding cart. These components should be placed back in stock.

6. Remove DIN rail from welding cart. DIN rail and mounting screws can be placed in stock if desired.

B. TESTING REQUIREMENTS AND ACCEPTANCE CRITERIA:

Post-Installation Testing

1. Power supply testing

   Acceptance Criteria: 24Vdc ±1%

2. Radar probe configuration

   Acceptance Criteria: Shall be tested with vendor upon installation.

3. New cables shall be continuity tested in accordance with E-406

   Acceptance Criteria: Test values shall be per E-406

4. Concrete

   Acceptance Criteria: Concrete shall be tested in accordance with Specification C-0011B

5. Radar probe functional test

   Radar probes do not fall out of calibration and either function as designed, or do not function at all. Therefore, ensuring that the radar probe is measuring level is enough to verify functionality.

   Acceptance Criteria: Radar probe measurement compares to field measurement

FORM 9, INSTALLATION AND TESTING INSTRUCTIONS
(Sheet 9 of 10)

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C. TURNOVER/CLOSEOUT REQUIREMENTS – The following steps shall be included in the implementing work order(s) to ensure turnover and/or closeout requirements are satisfied.

TO/CO Plan No: 201000229

Final Turnover: ☐	Partial Turnover: ☐

Description of components/system (work scope) to be turned over to Operations:

Work Orders being turned over: ____________

Does this work complete this modification?	☐ YES	☐ NO
If "NO," does the 10 CFR 50.59/72.48 review cover the interim plant configuration created by these work orders?	☐ YES	☐ NO

If this is a partial turnover, list the Work Orders required to complete modification.

N/A	Action	Responsible Individual	INITIALS
	PRIOR TO TESTING:		
☐*	All work Orders associated with this ECP that are required to be completed prior to turnover are complete	Resp Eng$^{\nabla}$	______
☒*	Final Walkdown Complete with Operations and Maintenance Document "Open Items" on FORM 13 and attach to work order	Resp Eng$^{\nabla}$	______
☒*	Engineering for all At-Risk Activities has been approved	Resp Eng$^{\nabla}$	______
☒*	Final Walkdown "Open Items List" has been reviewed. They will neither prevent safe testing nor reduce the safety of plant operations	Resp Eng$^{\nabla}$	______
	AFTER TESTING:		
☐*	Test Results have been reviewed and are satisfactory ☐ with anomalies listed below: ☐ without anomalies	Resp Eng$^{\nabla}$ System Eng	______ ______
☒*	Final Walkdown "Open Items List" has been reviewed and does not adversely impact safe operation or the design intent	Resp Eng$^{\nabla}$	______
☐*	All Labels are Installed (Valve, Component, Alarm)	Ops Shift Mngr	______
☐*	Any CRs impacting operability have been dispositioned	Resp Eng$^{\nabla}$	______

FORM 9, INSTALLATION AND TESTING INSTRUCTIONS
(Sheet 10 of 10)

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☒	Engineering for all At-Risk Activities has been approved	Resp Eng$^\nabla$	
☐	All Critical CDs and associated ECNs are Statused as APPROVED/DESIGN BASIS	Resp Eng$^\nabla$	
☐	All Required Changes to Operations Procedures are Complete	Ops Shift Mngr	
☐	All TO/CO actions required prior to Turnover are complete	Resp Eng$^\nabla$	
☐	Training has been notified of turnover	Resp Eng$^\nabla$	
☐*	Training to Support Turnover Complete		
☐	• Operations	Ops Shift Mngr	
☐	• All others	Resp Eng$^\nabla$	
☒*	Applicable Technical Specification Changes are Completed and UFSAR or USAR Changes are Submitted?	Resp Eng$^\nabla$	
☐*	Plant Change Description provided to the Shift Manager NOTE: Description is generally a copy of the ECP coversheet and the Operational Impact Statement, if required)	Resp Eng$^\nabla$	
☐*	Acceptance of Change: Manager – Operations (or Designee)	Printed Name/ Signature	Date

* These requirements are generally not applicable for EQVs

$^\nabla$ Responsible Engineer may be the responsible engineer or project manager assigned to the ECP or an alternate assigned by their organization.

Attachment 4, TC Installation and Testing Instructions Addenda

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A. PRE-INSTALLATION REQUIREMENTS – The following steps shall be included in the implementing work order(s) to ensure pre-installation requirements are satisfied.

WARNING! – Do Not Install After ____________________

Date is 30 days after TCP approval date. Return to the Responsible Engineer if installation is delayed beyond this date.

Applicable Modes: ☒ 1 ☒ 2 ☒ 3 ☒ 4 ☒ 5 ☒ 6 ☐ Defueled

N/A	Action		Responsible Individual	INITIALS
☒	Engineering for all At-Risk Activities has been approved		Resp Eng[1]	
☒	Pre-Installation Training Complete		Resp Eng[1]	
☒	Pre-Installation Procedure Changes Complete		Resp Eng[1]	
☐	Installation Approval	Date:	Shift Manager	

B. TEMPORARY CHANGE INSTALLATION AND VERIFICATION – The following steps shall be included in the implementing work order(s) to ensure TC tags are hung and installation is verified.

Installer: Name (Print) ____________________ Initials:______ Ext.______

Verifier: Name (Print) ____________________ Initials:______ Ext.______

Tag Config Seq#	Equip ID/Description/Location	INSTALLATION		RESTORATION	
		Installer Initial/Date	Verifier Initial/Date	Restorer Initial/Date	Verifier Initial/Date
~~N/A~~	~~None~~ 5/5/10				

Note any TC Tag placement restrictions, for example, TC tag not hung on SSC in containment:

[1]Responsible Engineer may be the Responsible Engineer, System Engineer, or Engineering Services designee.

Attachment 4, TC Installation and Testing Instructions Addenda

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C. POST-INSTALLATION TURNOVER REQUIREMENTS – The following steps shall be included in the implementing work order(s) to ensure post-installation turnover requirements are satisfied.

N/A	Action	Responsible Individual	**INITIALS**
	BEFORE POST-INSTALLATION TESTING:		
☐	Installation Steps of Work Order(s) that are required to be completed prior to post-installation testing are complete	Resp Eng[1]	______
	AFTER POST-INSTALLATION TESTING:		
☐	Test Results have been reviewed and are satisfactory	Resp Eng[1]	______
☐	All TC Tags or Labels are Installed	Ops Shift Mngr	______
☐	All ECNs are Statused as TC INSTALLED	Resp Eng[1]	______
☐	TCP status changed to TC INSTALLED	Resp Eng[1]	______
☐	All TO/CO actions required prior to Turnover are complete	Resp Eng[1]	______
☐	Training has been notified of TCP Installation	Resp Eng[1]	______
☐	Post-Installation Training Complete	Resp Eng[1]	______
☐	Post-Installation Procedure Changes Complete	Resp Eng[1]	______
	Acceptance of Temporary Change: Manager – Operations ______________________ (or Designee) Printed Name/ Signature		______ Date

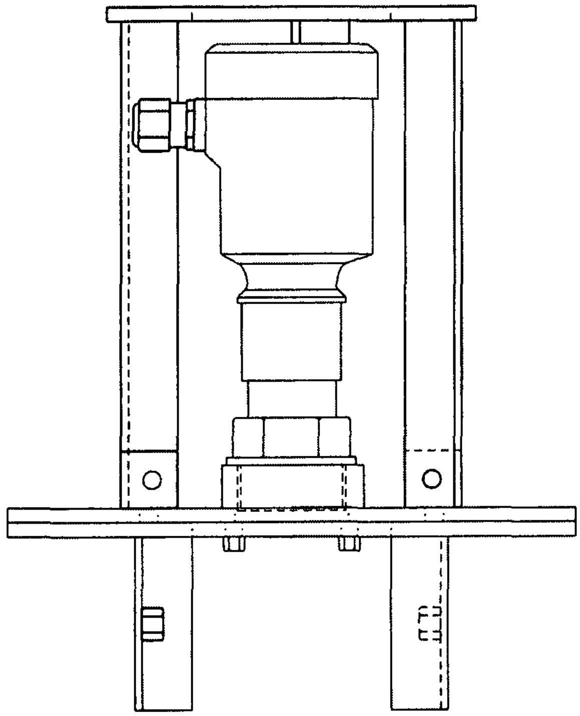
D. TEMPORARY CHANGE RESTORATION – The following steps shall be included in the implementing work order(s) to ensure pre-restoration requirements are satisfied and TC removal is verified.

	Restoration Approval	Date:	Shift Manager	______

The Restorer and Verifier shall initial in the table of Tags (Section B) to indicate restoration and verification.

Restorer:	Name (Print)	______________________	Initials:______	Ext.______
Verifier:	Name (Print)	______________________	Initials:______	Ext.______

Other Restoration Requirements and Post Restoration Actions, including Testing, Training, Procedure Revisions, and Operations Turnover are addressed on the ECP Installation and Testing Instructions, CNG-FES-015, Form 10.



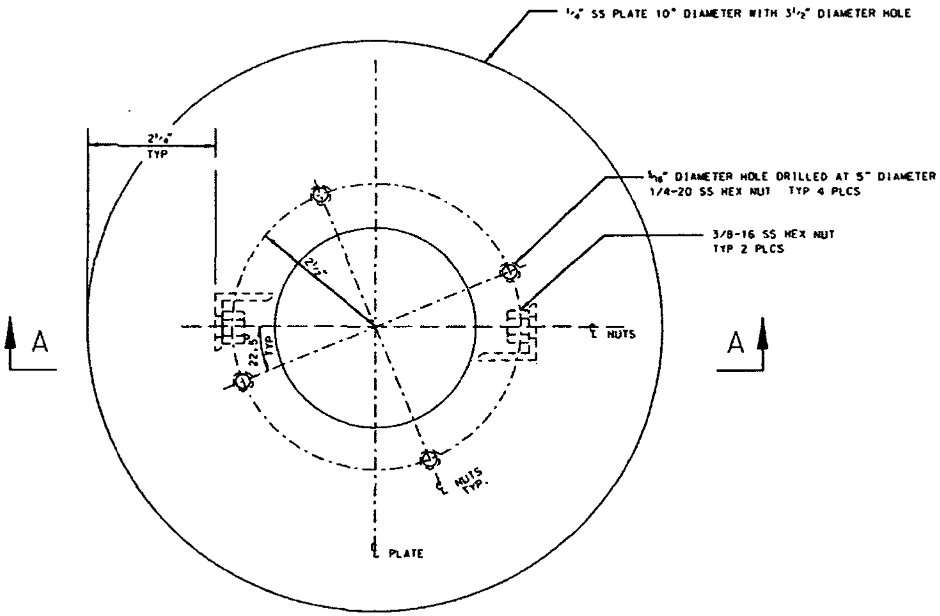
6"& 8" PENETRATION MOUNTING ASSEMBLY
ELEVATION VIEW

NOTES:

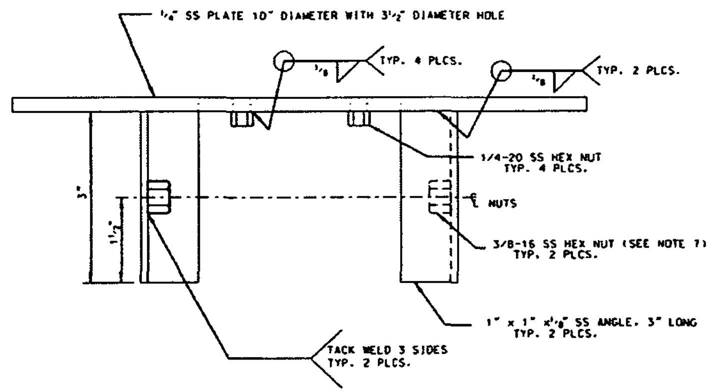
1. ALL MATERIAL TO BE TYPE 316 OR 304 SS OR EQUAL
2. ALL CORNERS AND EDGES TO BE GROUND SMOOTH
3. ANGLE LEGS MAY BE TRIMMED AS NECESSARY
4. GRIND TOP OF RISER PIPE FLUSH WITH CONCRETE
5. TIGHTEN ALL BOLTS TO A SNUG-TIGHT CONDITION
6. WELD ELECTRODE SHALL BE E316-XX OR E304-XX AS REQ.
7. DRILL 5/16" DIAMETER HOLE IN ANGLE PRIOR TO WELDING NUT.

6" & 8" ASSEMBLY BILL OF MATERIALS

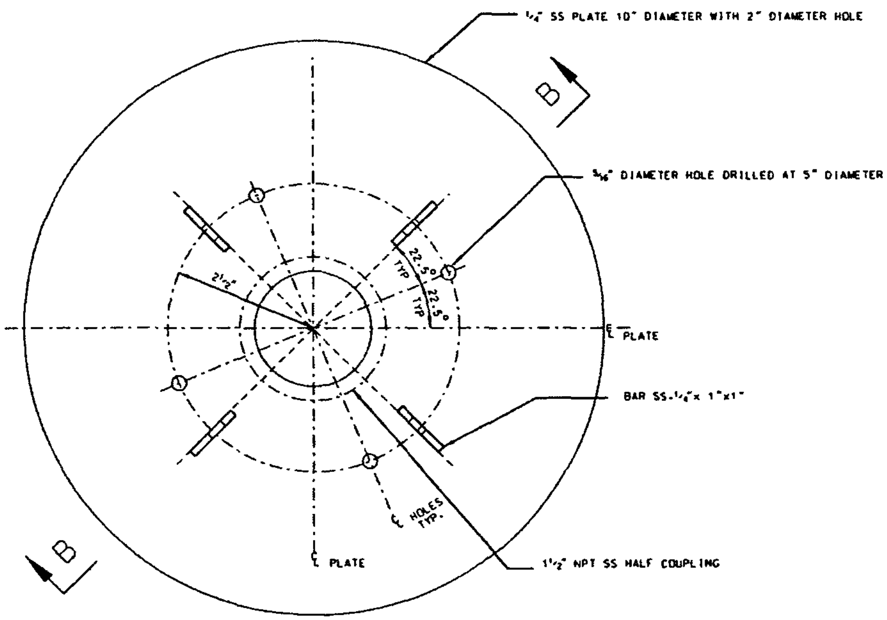
QTY	DESCRIPTION
4	ANGLE, TYPE 316 SS, 1/8" x1" x 1", 8 1/2" LONG
2	ANGLE, TYPE 316 SS, 1/8" 1"x 1", 3" LONG
1	PLATE, TYPE 316 SS, 1/4", 10" DIAMETER WITH 3 1/2" DIAMETER HOLE
1	PLATE, TYPE 316 SS, 1/4", 10" DIAMETER WITH 2" DIAMETER HOLE
1	PLATE, TYPE 316 SS, 1/4", 6 1/2" DIAMETER WITH 3 1/2" DIAMETER HOLE
4	BAR, TYPE 316 SS, 1/4" x 1"x1"
1	HALF COUPLING, SS, 1 1/2" NPT
2	HEX NUT, TYPE 316 SS, 3/8-16
8	HEX NUT, TYPE 316 SS, 1/4-20
2	BOLT, HEX HEAD, TYPE 316 SS, 3/8-16, LENGTH AS REQ.
8	BOLT, HEX HEAD, TYPE 316 SS, 1/4-20, LENGTH AS REQ.
4	WASHER, FLAT, TYPE 316 SS, 1/4"
4	WASHER, LOCK, TYPE 316 SS, 1/4"



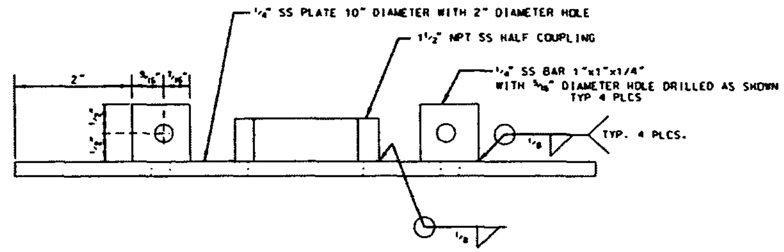
6"& 8" PENETRATION LOWER MOUNTING ASSEMBLY
PLAN VIEW



6"& 8" PENETRATION LOWER MOUNTING ASSEMBLY
SECTION VIEW
A-A

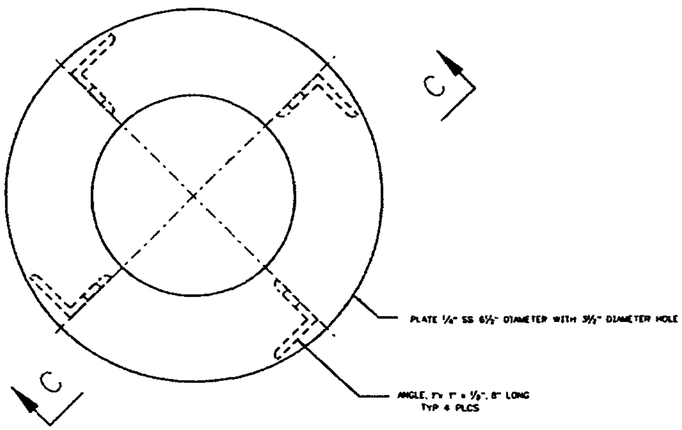


6"& 8" PENETRATION UPPER MOUNTING ASSEMBLY
PLAN VIEW

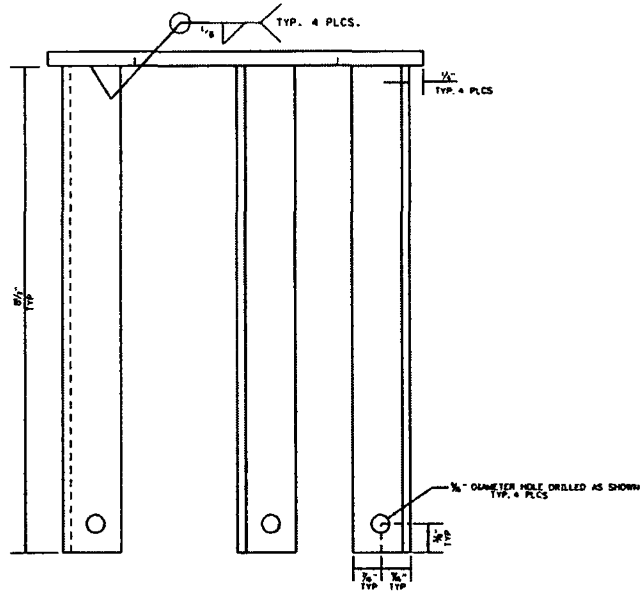


6"& 8" PENETRATION UPPER MOUNTING ASSEMBLY
SECTION VIEW
B-B

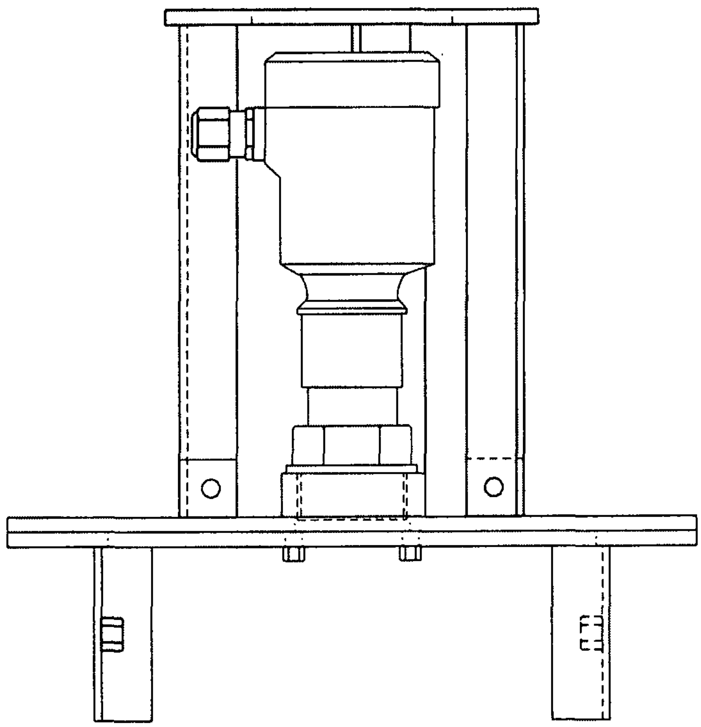
ECP-10-000208 INSTALLATION SKETCH 001 SHEET 4 OF 4



PROTECTIVE CAGE ASSEMBLY
PLAN VIEW



PROTECTIVE CAGE ASSEMBLY
SECTION VIEW
C-C



10" PENETRATION MOUNTING ASSEMBLY
ELEVATION VIEW

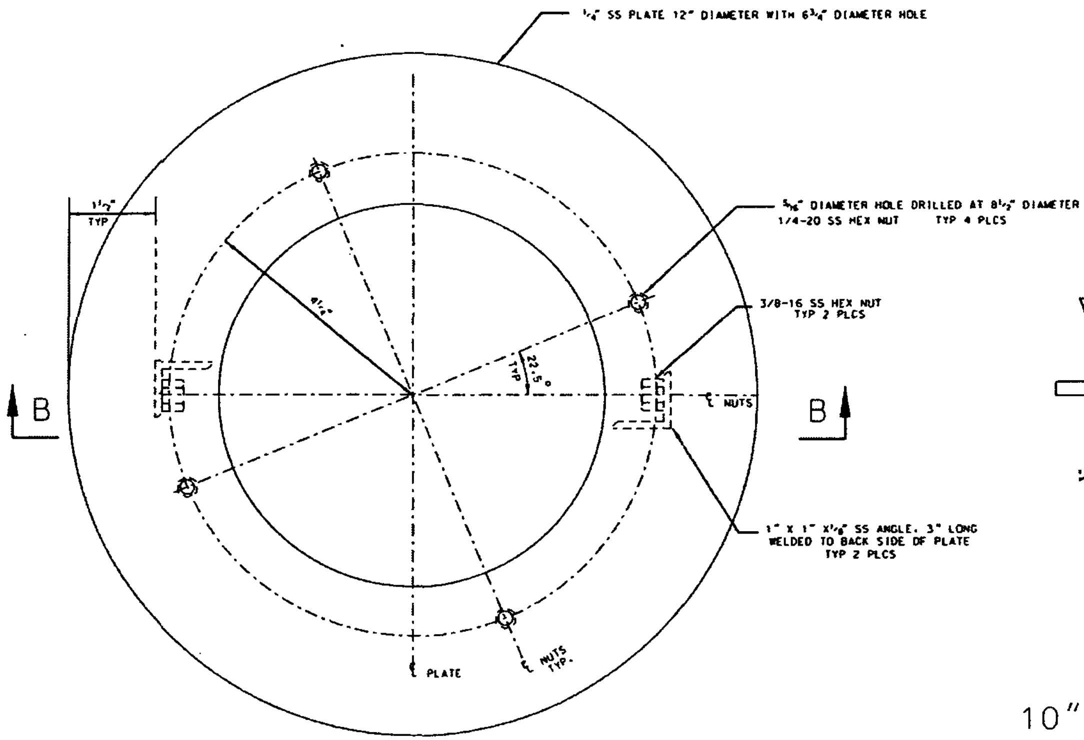
NOTES:

1. ALL MATERIAL TO BE TYPE 316 OR 304 SS OR EQUAL
2. ALL CORNERS AND EDGES TO BE GROUND SMOOTH
3. ANGLE LEGS MAY BE TRIMMED AS NECESSARY
4. GRIND TOP OF RISER PIPE FLUSH WITH CONCRETE
5. TIGHTEN ALL BOLTS TO A SNUG-TIGHT CONDITION
6. WELD ELECTRODE SHALL BE E316-XX OR E304-XX AS REQ.
7. DRILL 7/16" DIAMETER HOLE IN ANGLE PRIOR TO WELDING NUT.

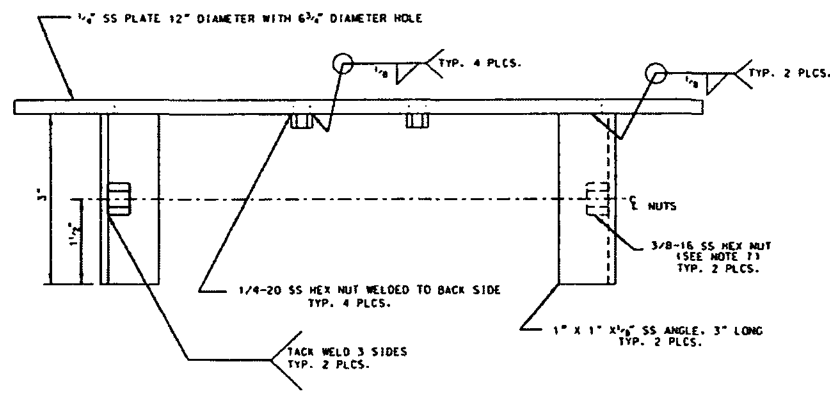
10" ASSEMBLY BILL OF MATERIALS

QTY	DESCRIPTION
4	ANGLE, TYPE 316 SS, 1/8" x 1" x 1", 8 1/2" LONG
2	ANGLE, TYPE 316 SS, 1/8" x 1" x 1", 3" LONG
1	PLATE, TYPE 316 SS, 1/4", 12" DIAMETER WITH 6 3/4" DIAMETER HOLE
1	PLATE, TYPE 316 SS, 1/4", 12" DIAMETER WITH 2" DIAMETER HOLE
1	PLATE, TYPE 316 SS, 1/4", 6 1/2" DIAMETER WITH 3 1/2" DIAMETER HOLE
4	BAR, 316 SS, 1/4" x 1" x 1"
1	HALF COUPLING, SS, 1 1/2" NPT
4	HEX NUT, TYPE 316 SS, 3/8-16
8	HEX NUT, TYPE 316 SS, 1/4-20
2	BOLT, HEX HEAD, TYPE 316 SS, 3/8-16, LENGTH AS REQ.
8	BOLT, HEX HEAD, TYPE 316 SS, 1/4-20, 3/4" LENGTH AS REQ.
4	WASHER, FLAT, TYPE 316 SS, 1/4"
4	WASHER, LOCK, TYPE 316 SS, 1/4"

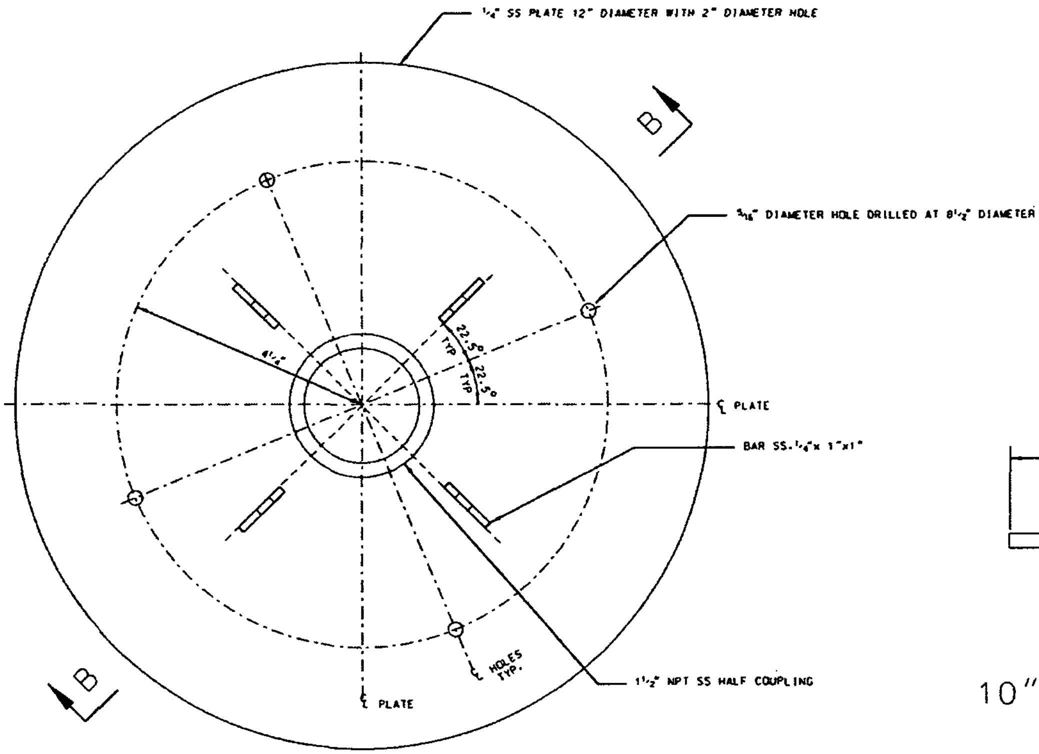
INSTALLATION SKETCH 002



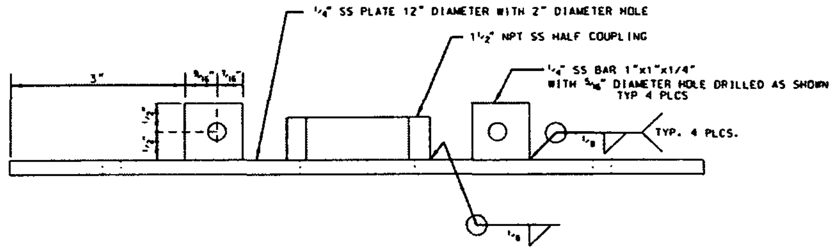
10" PENETRATION LOWER MOUNTING ASSEMBLY
PLAN VIEW



10" PENETRATION LOWER MOUNTING ASSEMBLY
SECTION VIEW
B-B



10" PENETRATION UPPER MOUNTING ASSEMBLY
PLAN VIEW

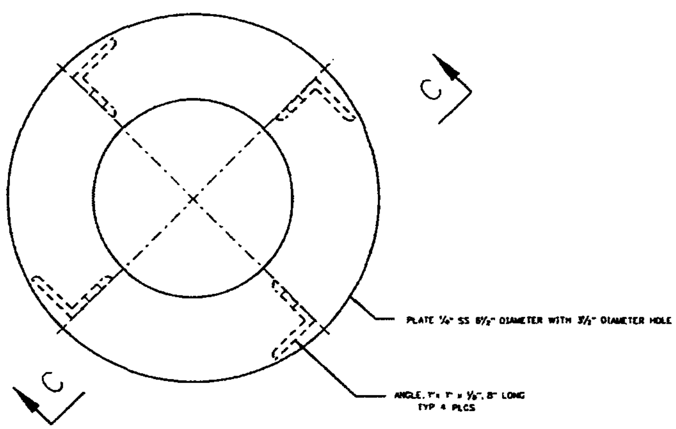


10" PENETRATION UPPER MOUNTING ASSEMBLY
SECTION VIEW
B-B

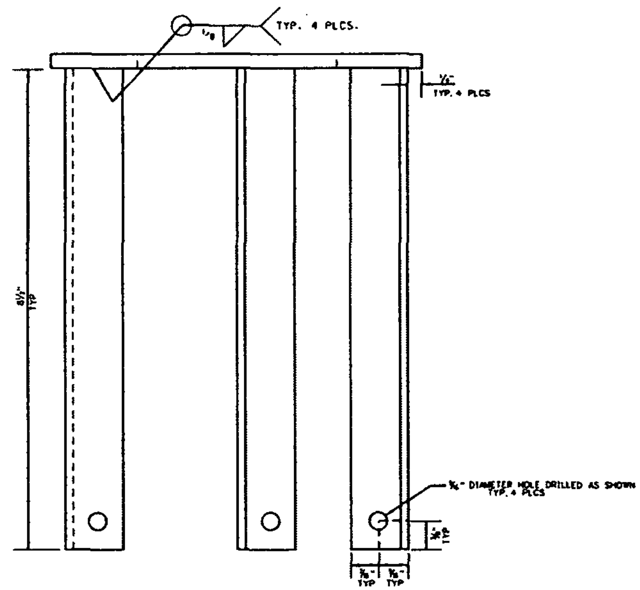
ECP-10-000208

INSTALLATION SKETCH 002

SHEET 4 OF 4

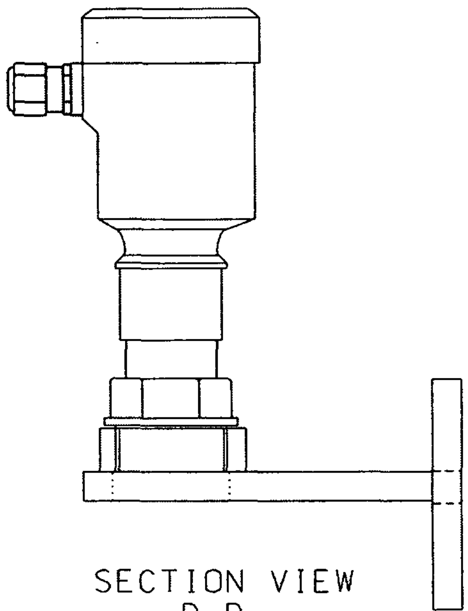


PROTECTIVE CAGE ASSEMBLY
PLAN VIEW

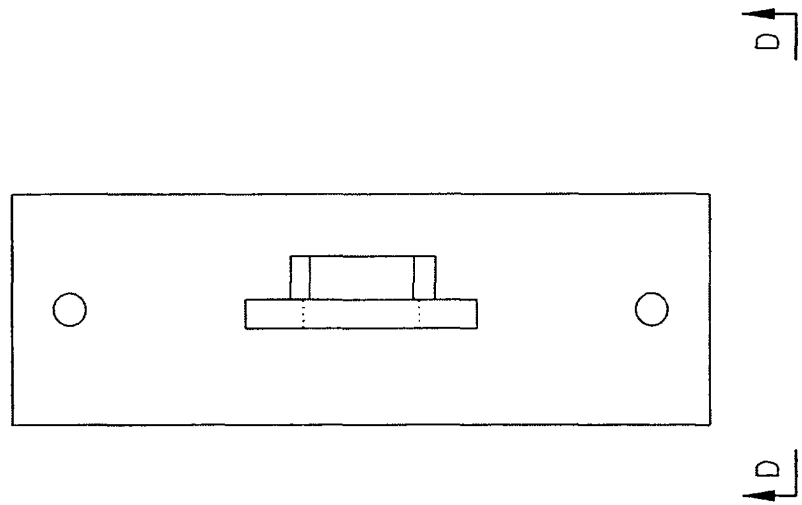


PROTECTIVE CAGE ASSEMBLY
SECTION VIEW
C–C

INSTALLATION SKETCH 003



SECTION VIEW
D-D



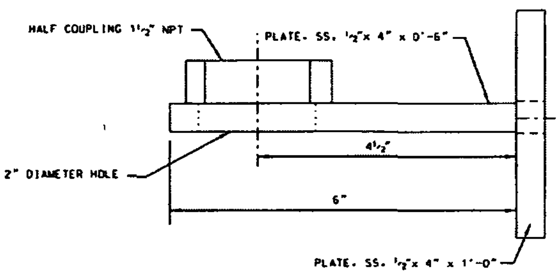
ANCHOR BRACKET MOUNTING ASSEMBLY
ELEVATION VIEW

ANCHOR BRACKET ASSEMBLY BILL OF MATERIALS

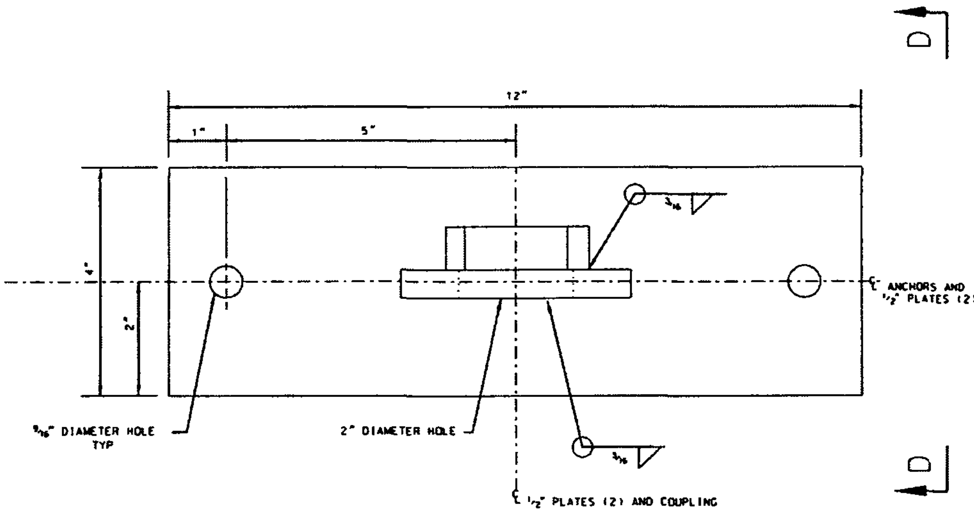
QTY	DESCRIPTION
1	PLATE, TYPE 316 SS, 1/2" x 4" x 1'-0"
1	PLATE, TYPE 316 SS, 1/2" x 4" x 0'-6"
1	HALF COUPLING, SS, 1 1/2" NPT
2	ANCHOR BOLT, HILTI KB III, SS, 3/8" DIAMETER x 3" LONG

NOTES:

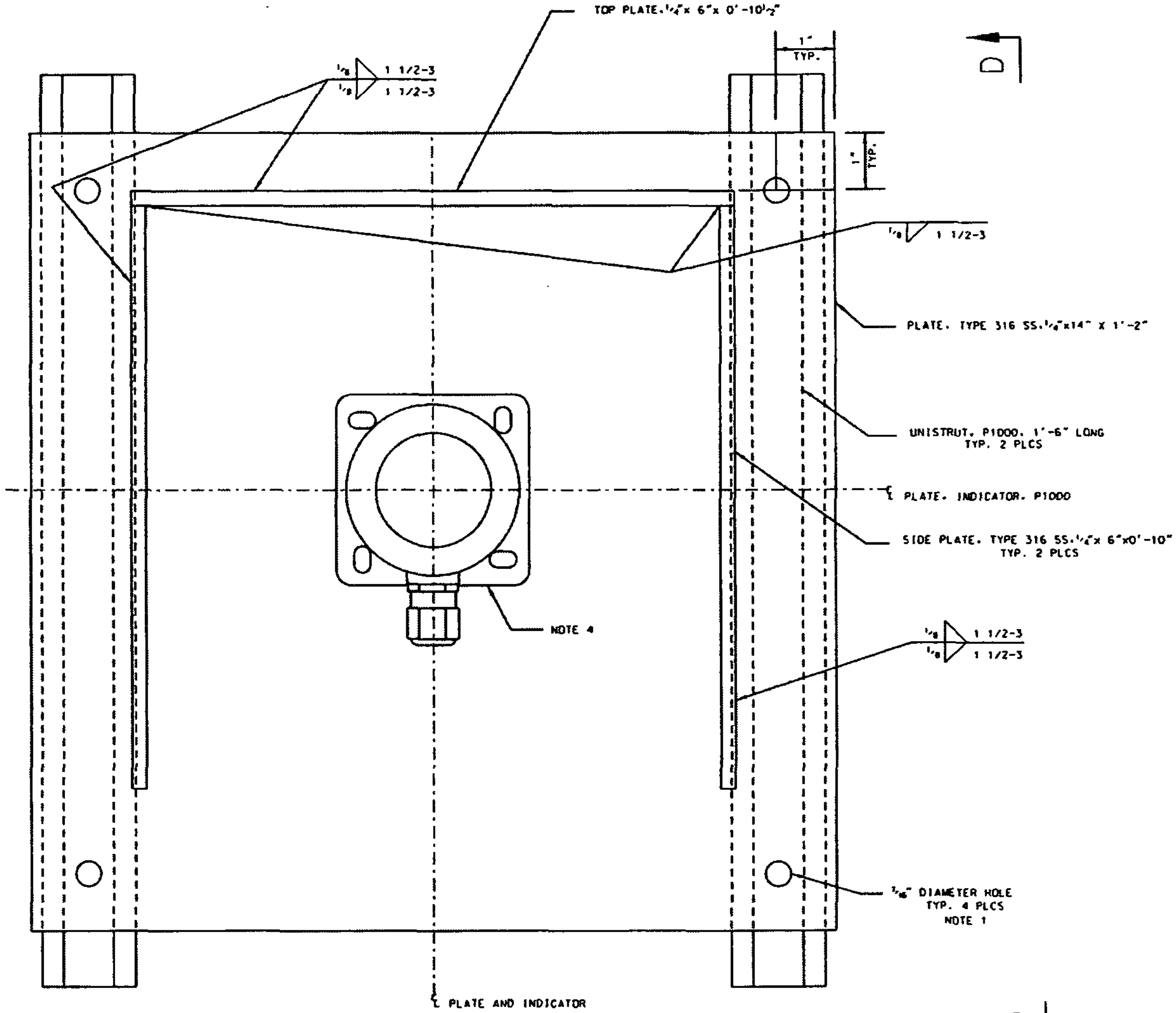
1. ALL MATERIAL TO BE TYPE 304 OR 316 SS OR EQUAL
2. ALL CORNERS AND EDGES TO BE GROUND SMOOTH
3. BRACKET TO BE MOUNTED USING 3/8" DIAMETER SS HILTI KWIK BOLT III ANCHORS. MIN EMBED TO BE 1 5/8"
4. NO REBAR SHALL BE CUT FOR INSTALLATION OF CEA'S. RELOCATE BRACKET AS NECESSARY TO AVOID REBAR.
5. WELD ELECTRODE SHALL BE E316-XX OR E304-XX AS REQ.



SECTION VIEW
D-D



ANCHOR BRACKET MOUNTING ASSEMBLY
ELEVATION VIEW

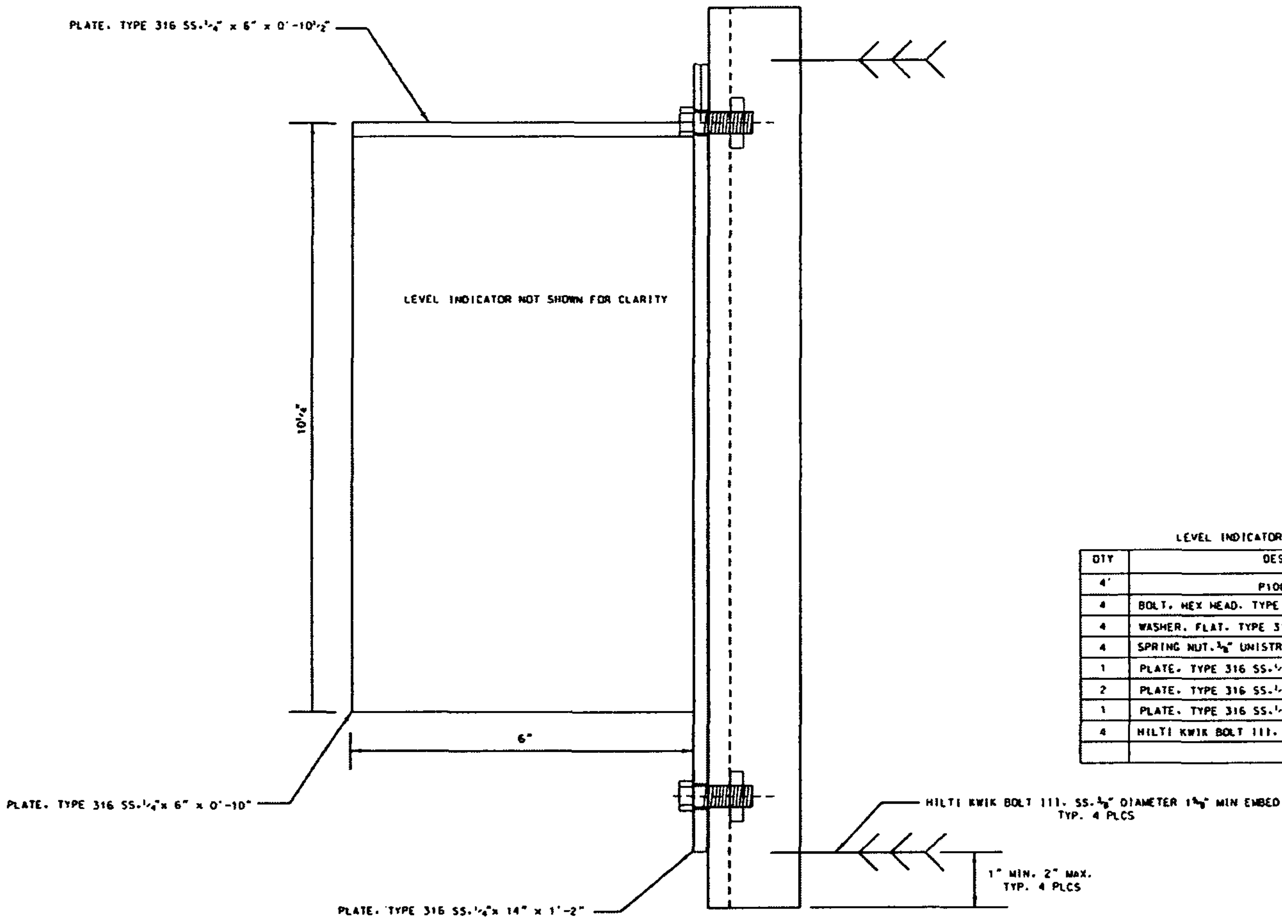


LEVEL INDICATOR MOUNTING ASSEMBLY
ELEVATION VIEW

NOTES:

1. MOUNT PLATE TO UNISTRUT CHANNEL USING 3/8" SS BOLTS AND UNISTRUT SPRING NUTS
2. MOUNT UNISTRUT P1000 CHANNELS USING HILTI KWIK BOLT III ANCHORS 3/8" DIAMETER SS MIN EMBED 1 5/8"
3. ALL MATERIAL TO BE TYPE 304 OR 316 SS UNLESS NOTED.
4. MOUNT INSTRUMENT TO BACK PLATE USING 1/4-20 SS BOLTS HOLES TO BE DRILLED AND TAPPED IN BACK PLATE PER VENDOR TEMPLATE
5. UNISTRUT CHANNELS MAY BE RUN VERTICAL OR HORIZONTAL EXACT ORIENTATION TO BE DETERMINED BY FIELD
6. TIGHTEN ALL BOLTS TO A SNUG TIGHT CONDITION
7. ANCHOR BOLTS AND HOLES NOT SHOWN FOR CLARITY
8. NO REBAR SHALL BE CUT IN THE INSTALLATION OF THE CEA'S.
9. WELD ELECTRODE SHALL BE E316-XX OR E304-XX AS REQ.

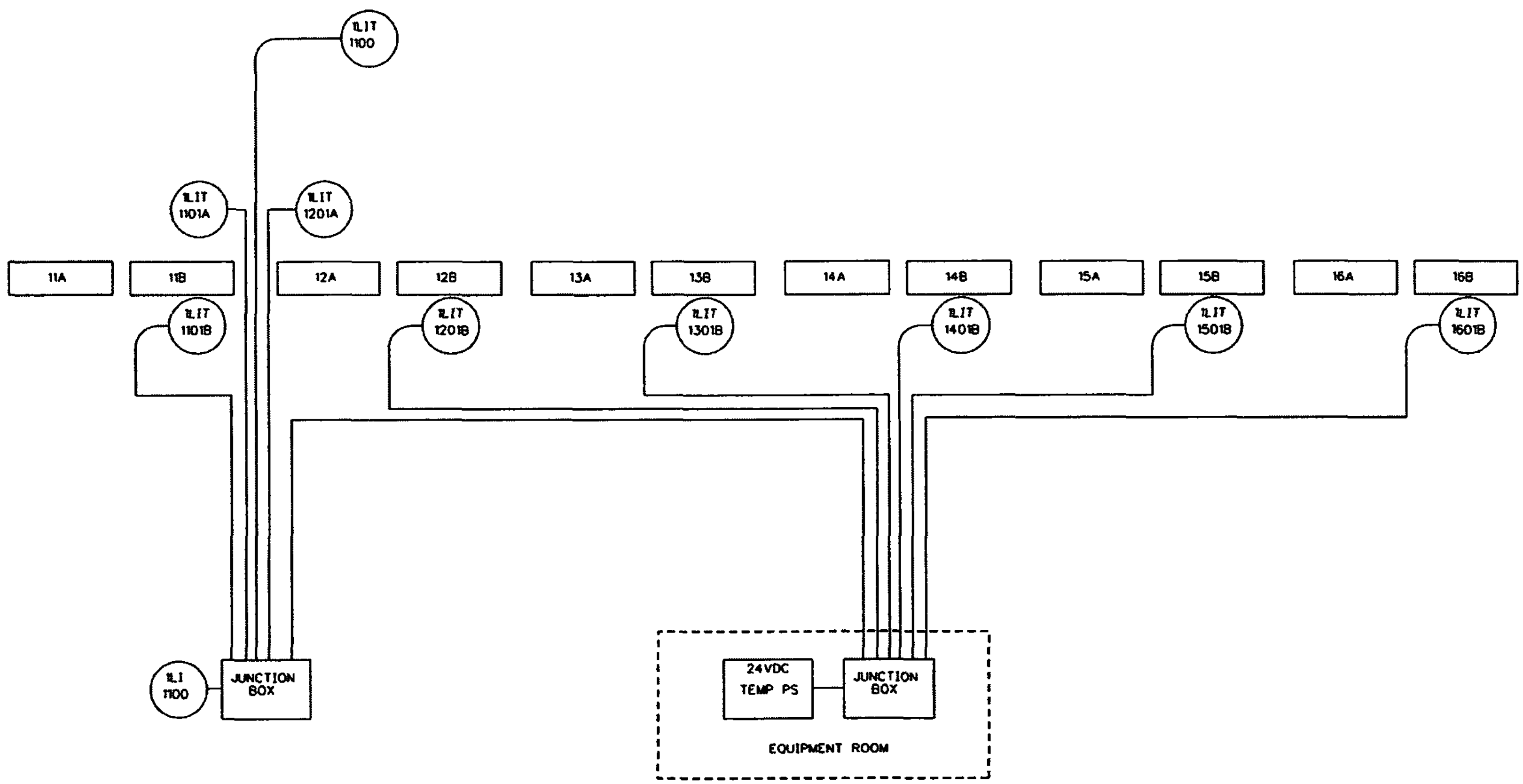
ECP-10-000208 INSTALLATION SKETCH 004



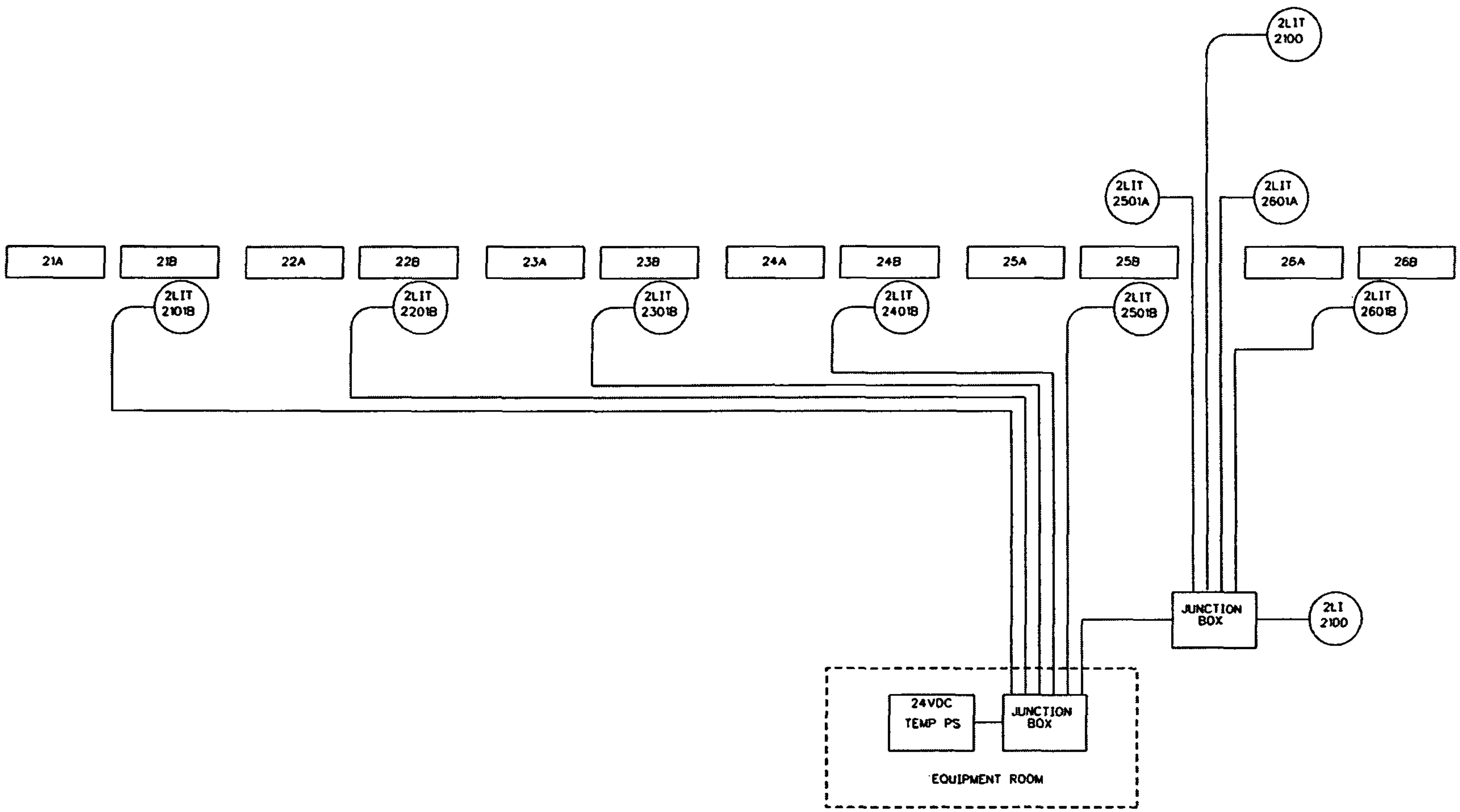
SECTION VIEW
D-D

LEVEL INDICATOR MOUNTING DETAIL BILL OF MATERIALS

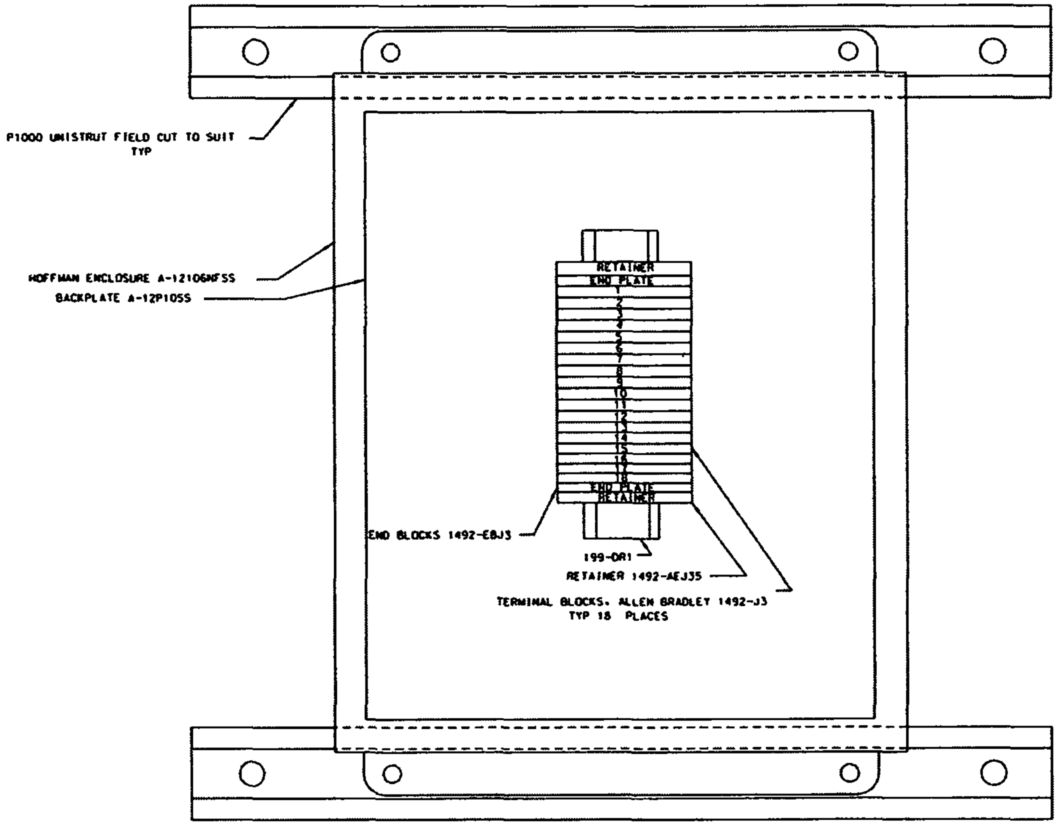
QTY	DESCRIPTION
4'	P1000 UNISTRUT
4	BOLT, HEX HEAD, TYPE 316 SS, 3/8-16, 1 " LONG
4	WASHER, FLAT, TYPE 316 SS, 3/8"
4	SPRING NUT, 3/8" UNISTRUT
1	PLATE, TYPE 316 SS, 1/4" x 14" x 1'-2"
2	PLATE, TYPE 316 SS, 1/4" x 6" x 0'-10"
1	PLATE, TYPE 316 SS, 1/4" x 6" x 0'-10 1/2"
4	HILTI KWIK BOLT III, SS, 3/8" DIAMETER 2 1/4" LONG


1LIT 1100
1LIT 1101A
1LIT 1201A
11A
11B
12A
12B
13A
13B
14A
14B
15A
15B
16A
16B
1LIT 1101B
1LIT 1201B
1LIT 1301B
1LIT 1401B
1LIT 1501B
1LIT 1601B
1LI 1100
JUNCTION BOX
24VDC TEMP PS
JUNCTION BOX
EQUIPMENT ROOM

ECP-10-000208 INSTALLATION SKETCH 005 SHEET 2 OF 2



INSTALLATION SKETCH 006



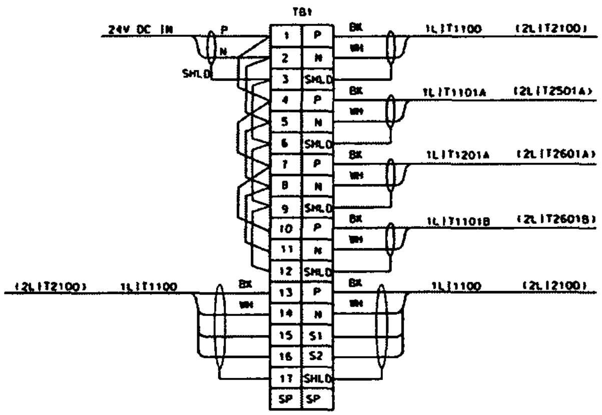
JUNCTION BOX
(COVER REMOVED)

NOTES

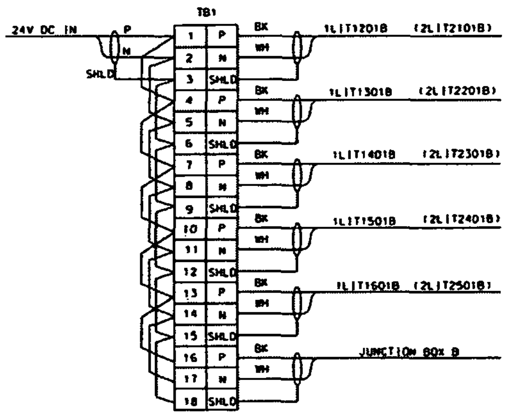
1. JUNCTION BOXES TO BE LOCATED BY FIELD.
2. JUNCTION BOXES TO BE MOUNTED PER SKETCH AND E-406
3. JB TO BE ATTACHED USING SS 1/4-20 BOLTS, WASHERS AND SPRING NUTS

JUNCTION BOX MOUNTING DETAIL BILL OF MATERIALS

QTY	DESCRIPTION
6"	DIN RAIL 199-DR1, ALLEN BRADLEY OR EQUAL
18	TERMINAL BLOCKS, ALLEN BRADLEY 1492-J3
2	TERMINAL BLOCK END BLOCKS, ALLEN BRADLEY 1492-EBJ3
1	HOFFMAN ENCLOSURE A-12106NFSS
1	HOFFMAN ENCLOSURE, BACKPLATE A-12P10SS
4'	P1000 UNISTRUT FIELD CUT TO SUIT
4	BOLT, HEX HEAD, 316 SS, 1/4-20, 3/4" LONG
4	WASHER, FLAT, 316 SS, 1/4"
4	SPRING NUT, 1/4-20 UNISTRUT P1006-1420
2	TERMINAL BLOCK END RETAINER, ALLEN BRADLEY 1492-EAJ35



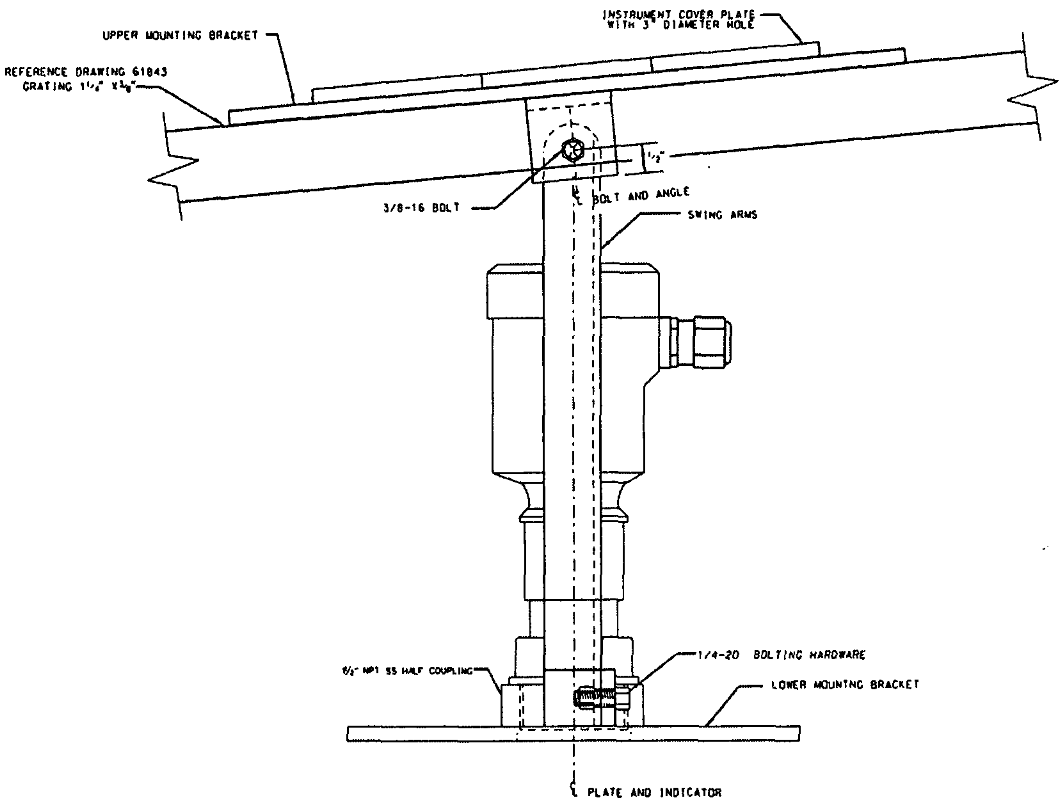
CONNECTION DETAIL
JUNCTION BOX TYPE B



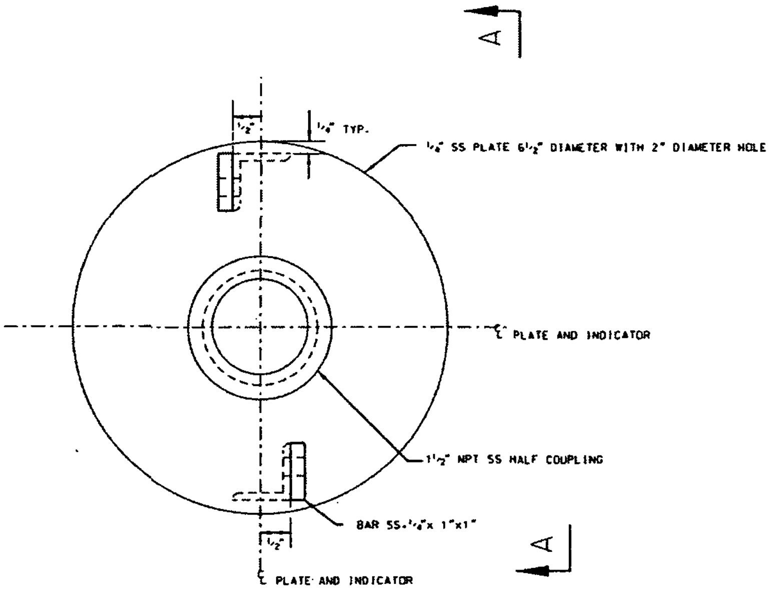
CONNECTION DETAIL
JUNCTION BOX TYPE A

NOTES:

1. JUNCTION BOX TO BE LOCATED BY FIELD
2. INSTALL DIN RAIL TO ALLOW FOR MORE BLOCKS IN THE FUTURE

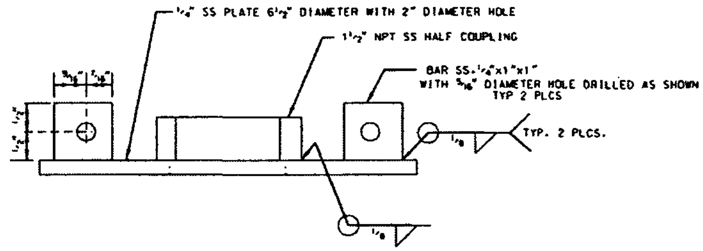


ELEVATION VIEW
BAY LEVEL
MOUNTING BRACKET ASSEMBLY

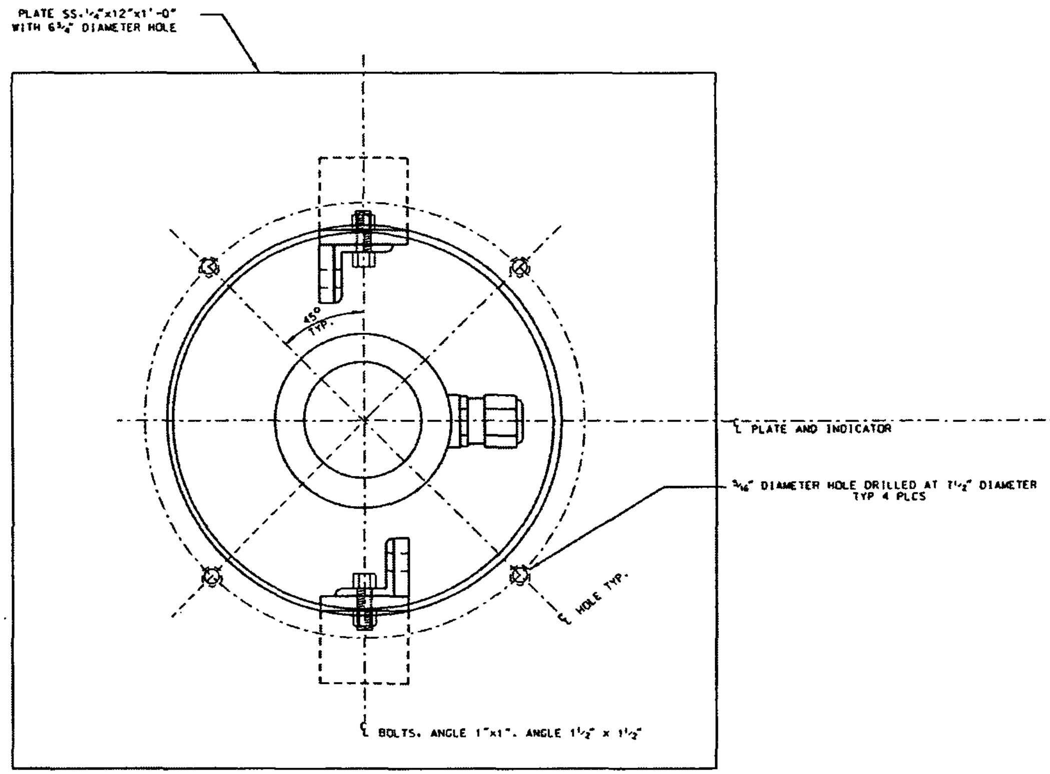


PLAN VIEW

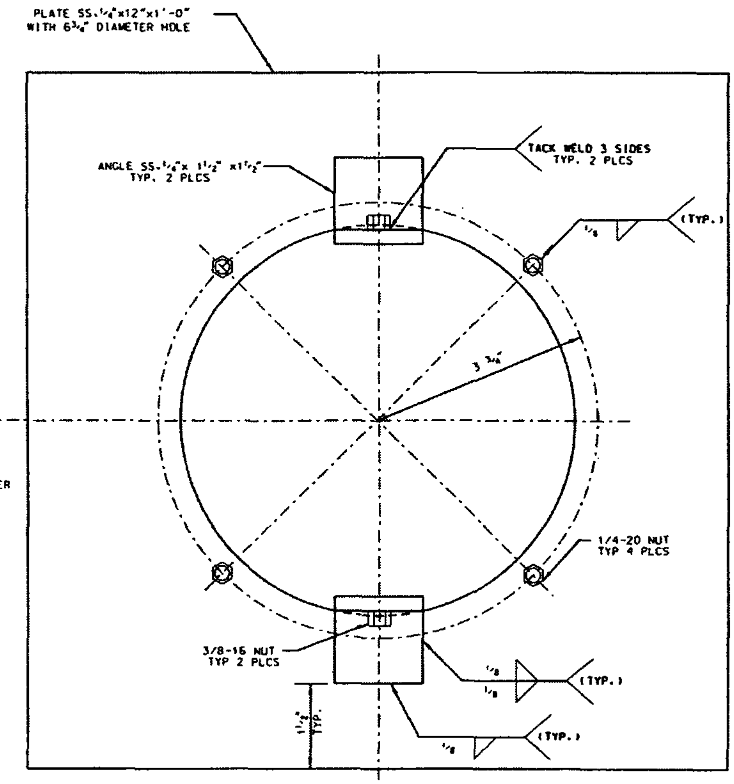
BAY LEVEL
LOWER MOUNTING BRACKET



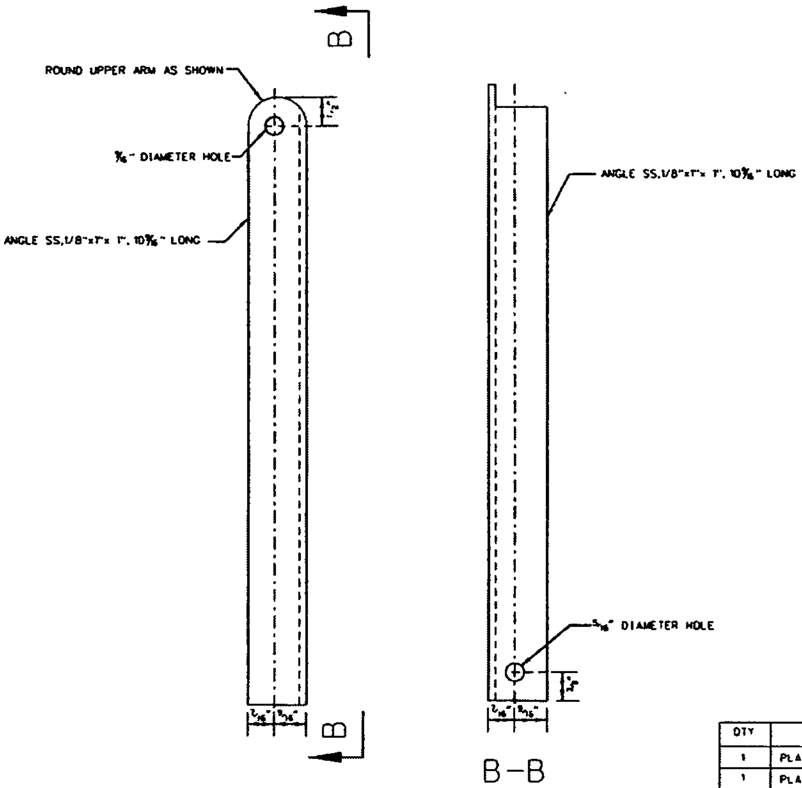
SECTION VIEW
A-A



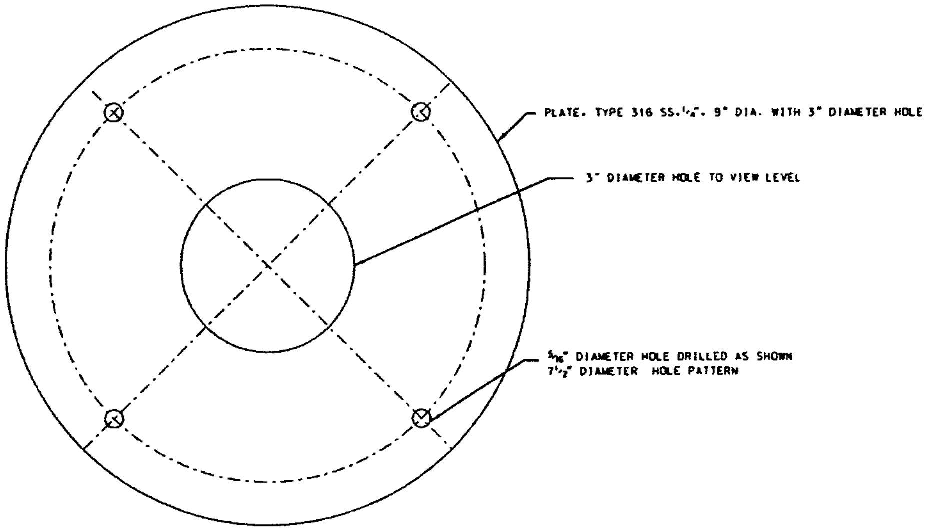
PLAN VIEW
BAY LEVEL
UPPER MOUNTING BRACKET
(INSTRUMENT COVER NOT SHOWN)



PLAN VIEW BOTTOM SIDE
BAY LEVEL
UPPER MOUNTING BRACKET



BAY LEVEL INSTRUMENT
SWING ARM DETAIL
2 REQ.



BAY LEVEL INSTRUMENT
INSTRUMENT COVER PLATE

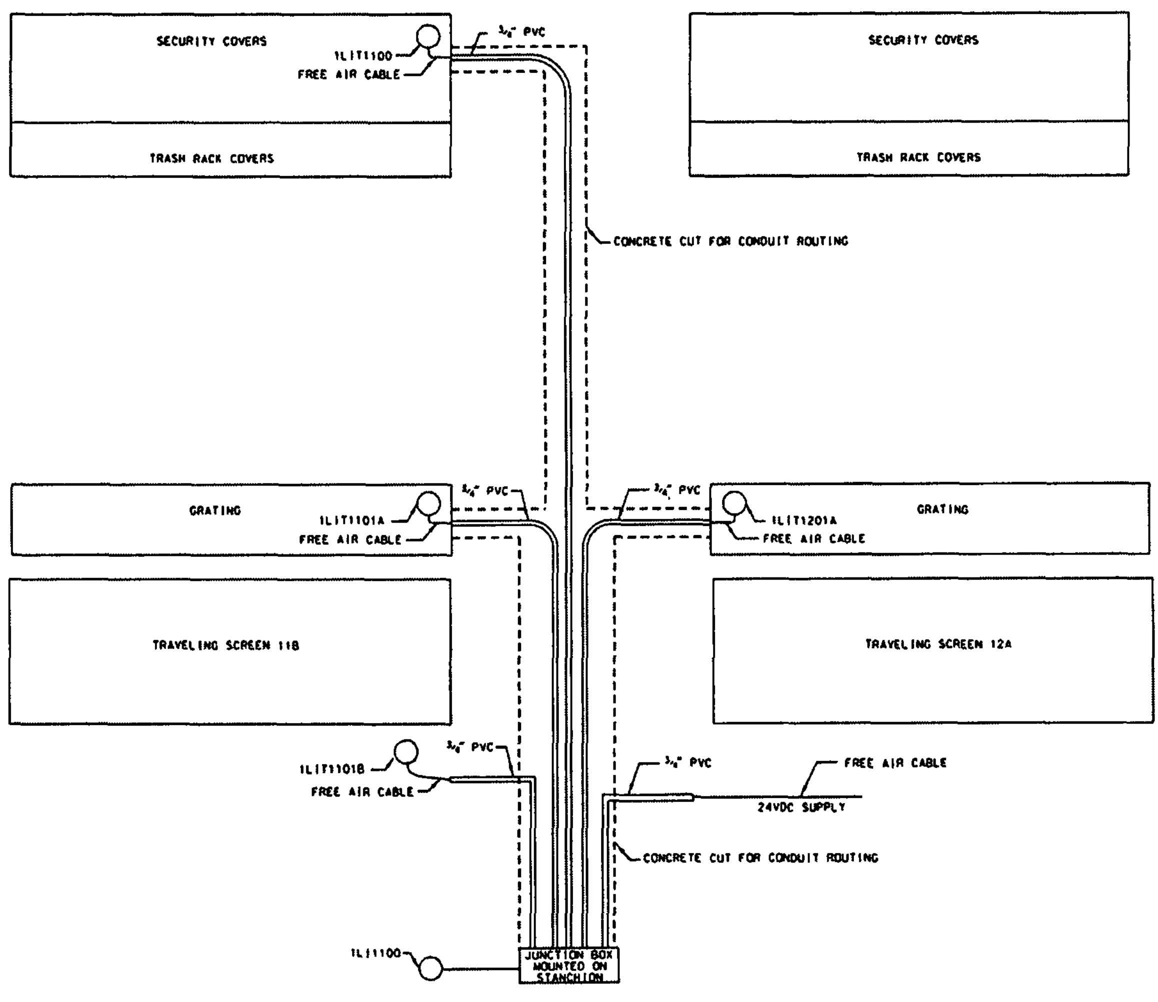
BAY LEVEL BRACKET ASSEMBLY BILL OF MATERIALS

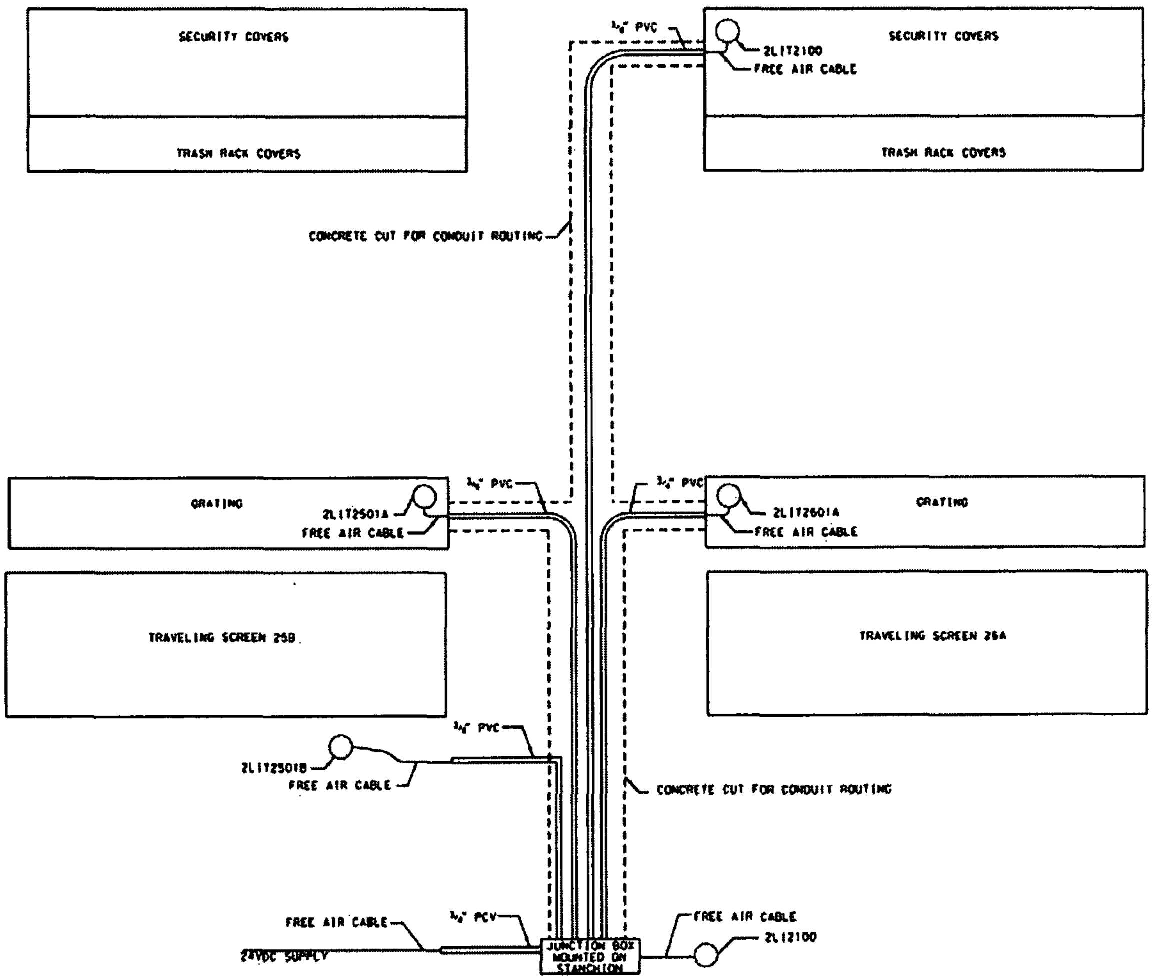
QTY	DESCRIPTION
1	PLATE, TYPE 316 SS, 1/4"x12"x 1'-0" WITH 6 3/4" DIAMETER HOLE
1	PLATE, TYPE 316 SS, 1/4", 6 1/2" DIAMETER WITH 2" DIAMETER HOLE
1	BAR, TYPE 316 SS, 1/4"x1"x1"
6	HEX HEAD BOLTS, SS, 1/4-20 LENGTH AS REQ.
6	HEX NUT, SS, 1/4-20
2	ANGLE, SS, 1/8"x1"x1", 10 9/16" LONG
2	ANGLE, SS, 1/4"x 1 1/2"x1 1/2", 1 1/2" LONG
2	HALF COUPLING, SS, 1 1/2" NPT
4	GRATING CLIPS AND SS BOLTING HARDWARE
2	HEX NUT, SS, 3/8-16
2	HEX HEAD BOLTS, SS, 3/8-16 x 1"
1	PLATE, TYPE 316 SS, 1/4", 9" DIAMETER WITH 3" DIAMETER HOLE

NOTES:

1. ALL MATERIAL TO BE TYPE 304 OR 316 SS OR EQUAL
2. ALL CORNERS AND EDGES TO BE GROUND SMOOTH
3. UPPER MOUNTING BRACKET PLATE TO BE ATTACHED TO GRATING USING GRATING CLIPS AND SS BOLTS AND HARDWARE AS REQUIRED.
4. VERIFY THE LEVEL PROBE IS PLUMB AND LEVEL PRIOR TO TIGHTENING BOLTS
5. SECURE INSTRUMENT CABLE TO ENSURE IT DOES NOT AFFECT POSITION OF INSTRUMENT AND IS PROPERLY SUPPORTED
6. ENSURE THE GRATING CUTOUT DOES NOT EXCEED 96 SQUARE INCHES MINIMIZE SIZE OF OPENING, SHALL BE 8" x 8" +/- 1". GRATING SHALL BE BANDED AS REQUIRED PER SITE PROCEDURES.
7. INSTRUMENT SHALL BE PLACED TO ENSURE IT DOES NOT IMPACT HINGED GRATING COVERS.
8. BOLT COVER PLATE TO UPPER MOUNTING PLATE AS SHOWN USING 1/4-20 BOLTS
9. TIGHTEN ALL BOLTS TO A SNUG TIGHT CONDITION
10. WELD ELECTRODE SHALL BE E316-XX OR E304-XX AS REQ.

ECP-10-000208 INSTALLATION SKETCH 009 SHEET 1 OF 2




SECURITY COVERS
TRASH RACK COVERS
3/4" PVC
2LIT2100
FREE AIR CABLE
SECURITY COVERS
TRASH RACK COVERS
CONCRETE CUT FOR CONDUIT ROUTING
GRATING
2LIT2501A
FREE AIR CABLE
3/4" PVC
3/4" PVC
2LIT2601A
FREE AIR CABLE
GRATING
TRAVELING SCREEN 25B
TRAVELING SCREEN 26A
3/4" PVC
2LIT2501B
FREE AIR CABLE
CONCRETE CUT FOR CONDUIT ROUTING
FREE AIR CABLE
3/4" PCV
24VDC SUPPLY
JUNCTION BOX MOUNTED ON STANCHION
FREE AIR CABLE
2LI2100

ECP-10-000208
FORM 11, ECP MATERIALS LIST

COMPONENT/PART	MAKE/ MODEL or TYPE	QUANTITY/ UNITS	PO/SRI NO./ REQ NO. or SRI NO.	STOCK ID NO.	Q-LIST CLASS	PRINT/ CODE or STAND.	ON ORDER (Y/N)	REMARKS
Radar Level Probe	Ohmart-Vega/PS65. UXMNDH KNAX	18			NSR		Y	
Remote Display w/ Wall Mount Adapter	Ohmart-Vega/DIS61 .UFKNB	2			NSR		Y	
24VDC Power Supply 1.5A	Phoenix Contact/MINI-SYS-PS-100-240AC/24DC/1.5	2			NSR			Or Equivalent
2/C 20 AWG T.S.P	Belden/1033A	2000 ft			NSR			Or Equivalent
4/C 20 AWG w/ shield	Belden/3016A	100 ft			NSR			Or Equivalent
Pigtail power cord – 4ft	FarmTek/WF4629	2			NSR			Or Equivalent

Refer to Installation Sketches for additional BOM items.

State as appropriate that when a Purchase Order (P.O.) is specified on the Bill of Materials, the item purchased shall be from that P.O. and no substitution is allowed without the concurrence of the Responsible Engineer.

FORM 13, RECORD OF WALKDOWN

ECP Supp No.: ECP-10-000208 **Rev. No.:** 0000

Type of Walkdown Completed: ☒ **CONCEPTUAL or PRE-DESIGN** ☐ **POST DESIGN/PRE-INSTALLATION** ☐ **POST CONSTRUCTION/INSTALLATION** ☐ **TURNOVER/CLOSEOUT** ☐ **OTHER:**	**Walkdown Lead:** Brad Wright **Date Completed:** 3/17/2010 **System/structure/component/location:** Unit 1 & 2 Intake Structure **Objective:** Walkdown general layout for radar probe locations and identify any issues

REQUIRED Walkdown Participants – Check Box for All that Apply
Identify the Involved Personnel by Name; Indicate Agreement with Walkdown Conclusions

☐ Change Requester:	☐ Design Engineer:	☐ System Engineer:	☐ Project Manager:
☒ Designer (Mech/Civil): Dave Earp	☐ Designer (E&C)	☐ Planner (Mech.):	☒ Planner (E&C): Donald Debuhr
☐ Maintenance (Mech.):	☐ Maintenance (E&C):	☐ Procurement Engineer:	☐ Health Physics:
☐ Design Engr (Elec):	☐ Design Engr (I&C):	☒ Design Engr (Mech): Charlie LaRue (S&L)	☐ Design Engr (Civil):
☐ Operations:	☐ ISI:	☐ Fire & Safety:	☐ Test Coordinator:
☐ Simulator or Training:	☐ Nuclear Training:	☐ As-needed participant:	☐ As-needed participant:

OPEN ISSUES AND EXCEPTIONS LISTING:

Description of Open Item or Issues Identified	Resolution Actions	Resp Org	ECD/AI & Actions
10" penetrations have interference issues with piping	Install probes in 6" penetrations	S&L	

USE ADDITIONAL SHEETS AS REQUIRED — Check if additional sheets used ☐

FORM 13, RECORD OF WALKDOWN
(Page 2 of 2)

ECP Supp No.: ECP-10-000208 **Rev. No.: 0000**

Type of Walkdown Completed:	Walkdown Lead: Brad Wright
☒ CONCEPTUAL or PRE-DESIGN	Date Completed: 3/24/2010
☐ POST DESIGN/PRE-INSTALLATION	System/structure/component/location: Unit 1 & 2 Intake Structure
☐ POST CONSTRUCTION/INSTALLATION	
☐ TURNOVER/CLOSEOUT	Objective: Locate radar probe power supply location, locate bay level locaions, and identify conduit routing
☐ OTHER:	

REQUIRED Walkdown Participants – Check Box for All that Apply
Identify the Involved Personnel by Name; Indicate Agreement with Walkdown Conclusions

☐ Change Requester:	☐ Design Engineer:	☐ System Engineer:	☐ Project Manager:
☐ Designer (Mech/Civil):	☐ Designer (E&C)	☐ Planner (Mech.):	☐ Planner (E&C):
☐ Maintenance (Mech.):	☐ Maintenance (E&C):	☐ Procurement Engineer:	☐ Health Physics:
☐ Design Engr (Elec):	☐ Design Engr (I&C):	☒ Design Engr (Mech): Charlie LaRue (S&L)	☐ Design Engr (Civil):
☐ Operations:	☐ ISI:	☐ Fire & Safety:	☐ Test Coordinator:
☐ Simulator or Training:	☐ Nuclear Training:	☐ As-needed participant:	☐ As-needed participant:

OPEN ISSUES AND EXCEPTIONS LISTING:

Description of Open Item or Issues Identified	Resolution Actions	Resp Org	ECD/AI & Actions
None			

USE ADDITIONAL SHEETS AS REQUIRED   Check if additional sheets used ☐

FORM 16, FIRE PROTECTION/APPENDIX R REVIEW ELECTRICAL DESIGN FEATURES CHECKLIST
(Page 1 of 2)

ECP Supp No.: ECP-10-000208 Rev. No.: 0000

Does the proposed engineering in its final form, as well as during interim installation procedures:			
A. Does the ECP directly or indirectly involve Appendix R equipment?	☐ YES	☒ NO	☐ N/A
B. Does the ECP add, relocate or change equipment whose operation would be required to achieve the safe shutdown functions identified in the Interactive Cable Analysis (ICA)? (Include equipment that provides additional safe shutdown systems boundary isolation, electrical distribution equipment changes (such as addition or deletion of loads, changes to relay settings) and electrically powered instruments.)	☐ YES	☒ NO	☐ N/A
C. Is Appendix R equipment being deleted?	☐ YES	☒ NO	☐ N/A
D. Does the ECP change the characteristics of Appendix R equipment?	☐ YES	☒ NO	☐ N/A
E. Does the ECP involve existing or adding new power, control, or instrumentation cables for Items A through D above?	☐ YES	☒ NO	☐ N/A
F. Does the ECP create a new potential associated circuit of concern?	☐ YES	☒ NO	☐ N/A
G. Does the ECP involve or add cables associated with Auxiliary Shutdown Panel isolation from the control room?	☐ YES	☒ NO	☐ N/A
H. Does the ECP affect compliance with requirements of, or affect drawings, statements, or analysis in the site Fire Protection Program, such as changes to fire rated compartmentation as shown on the drawings?	☐ YES	☒ NO	☐ N/A
Does the ECP involve changes to the automatic suppression system or standpipe and hose systems for safety-related structures?	☐ YES	☒ NO	☐ N/A
I. Does the ECP affect plant Technical Specifications, Technical Requirements Manuals, or FSAR positions related to Fire Protection?	☐ YES	☒ NO	☐ N/A
J. Does the ECP involve emergency lighting required for operation of safe shutdown equipment and access and egress routes thereto or require additional emergency lighting?	☐ YES	☒ NO	☐ N/A
K. Does the ECP affect communication systems that are being credited for use in shutting the plant down during a fire?	☐ YES	☒ NO	☐ N/A
Or does the modification introduce a new manual action that will require communication from an area of the plant for which the communication system has not been previously analyzed?	☐ YES	☒ NO	☐ N/A
L. Does the ECP require a revision to the ICA Database?	☐ YES	☒ NO	☐ N/A
M. Does the ECP require a change to the ASSARF packages?	☐ YES	☒ NO	☐ N/A
N. Does the ECP affect criteria for which a NRC-approved technical exemption request has been granted according to the site Fire Protection Program?	☐ YES	☒ NO	☐ N/A
O. Does the ECP affect cable trays with marinate tray covers required for Appendix R? For CCNPP, this refers specifically to trays ZD1CF07, ZE1CF19, ZF1CL17, and ZG1CL10 in Unit 1 containment and ZF2CL07, ZD2CF08, and ZE2CF18 in Unit 2 containment.	☐ YES	☒ NO	☐ N/A
P. Does the ECP affect calculations or AOPs associated with the ICA? (If, "YES," review required by Nuclear Engineering and Nuclear Operations.)	☐ YES	☒ NO	☐ N/A

FORM 16, FIRE PROTECTION/APPENDIX R REVIEW ELECTRICAL DESIGN FEATURES CHECKLIST

ECP Supp No.: ECP-10-000208 **Rev. No.: 0000**

Discuss "YES" responses below. Indicate if the change is justified. Identify information in the site Fire Protection Program that will require update and for which changes are unacceptable. When comments are resolved, the resolution, along with appropriate justification, should be included.

NONE

The change ☐ does/ ☐ does not adversely affect the ability to achieve and maintain safe shutdown in the event of a fire.

Fire Protection Electrical Engineer: ______________________ ______________

(Printed Name and Signature) Date

FORM 17, FIRE PROTECTION/APPENDIX R REVIEW FIRE PROTECTION DESIGN FEATURES CHECKLIST

ECP Supp No.: **ECP-10-000208** **Rev. No.:** **0000**

Does the proposed engineering in its final form, as well as during interim installation procedures:			
A. Add, delete or relocate combustible materials within the plant (any quantity of flammable/combustible liquid or combustible solid)?	☐ YES	☒ NO	☐ N/A
B. Affect Appendix R or controlled fire rated barriers (walls, floors, doors, dampers, penetrations). Add new barriers, remove or relocate existing barriers, or modify the design of existing barriers in ways not previously approved? Use penetration seal designs which have not been previously approved? Affect the integrity of structural steel? Affect fire propagation requirements, including areas free of intervening combustible and radiant energy shields?	☐ YES	☒ NO	☐ N/A
C. Affect installed fire protection SSCs (sprinkler, halon, hose stations, smoke detection, heat detection, flame detection, etc.). Install new components within 36 inches of sprinklers/detectors/etc. or that block sprinkler spray patterns, or isolate portions of a room from normal circulation?	☐ YES	☒ NO	☐ N/A
D. Affect fire fighting equipment, fire suppression efforts, or post fire operations in any area of the plant?	☐ YES	☒ NO	☐ N/A
E. Add or relocate Safety Related equipment (specifically equipment which might be adversely affected by water spray) in an area of the plant protected by a sprinkler system?	☐ YES	☒ NO	☐ N/A
F. Add/modify fire hazards within the plant in any way?	☐ YES	☒ NO	☐ N/A
G. Alter the physical arrangement or configuration of an area that affects fire protection equipment/systems, access and egress routes, emergency lighting, etc?	☐ YES	☒ NO	☐ N/A
H. Utilize construction/modification techniques that may result in an inadvertent operation of fire protection equipment or systems.	☐ YES	☒ NO	☐ N/A
I. Alter the location, function, design or material requirements of fire protection equipment and/or systems?	☐ YES	☒ NO	☐ N/A
J. Affect the Reactor Coolant Pump/Motor Lube Oil Collection System?	☐ YES	☒ NO	☐ N/A
K. Affect compliance with requirements of source documents as applicable such as the SER, UFSAR, TRM, etc.?	☐ YES	☒ NO	☐ N/A
L. Affect compliance with guidelines provided in applicable National Fire Protection Association Codes and Standards?	☐ YES	☒ NO	☐ N/A
M. Affect compliance with guidelines provided in applicable Nuclear Electric Insurance Limited Loss Control Standards?	☐ YES	☒ NO	☐ N/A
N. Alter or affect what has previously been stated in the site Fire Protection Program such as drawings, statements, or analysis (combustible loading or fire hazards)?	☐ YES	☒ NO	☐ N/A
O. Affect criteria for which an NRC-approved technical exemption request has been granted?	☐ YES	☒ NO	☐ N/A

FORM 17, FIRE PROTECTION/APPENDIX R REVIEW FIRE PROTECTION DESIGN FEATURES CHECKLIST

ECP Supp No.: ECP-10-000208 **Rev. No.: 0000**

"YES" responses shall be discussed and appropriately evaluated below. Identify information in the Fire Protection Program that will require update and/or for which the activity is considered unacceptable.

This change installs approximately 600ft of cable routed free-air on the intake structure deck. This installation is outside the intake structure and does not have any fire protection requirements. It has been recommended that cable used be flame retardant in accordance with IEEE 383. Cable specified on Form 11 , ECP Material List, meets the specifications of IEEE 383 for Class 1E flame retardant cable.

The change ☐ does/ ☒ does not adversely affect the ability to achieve and maintain safe shutdown in the event of a fire.

Fire Protection Engineer: N/A per Chris Dobry

(Printed Name and Signature) Date

ACTION VALUE BASIS DOCUMENT

ESP No.:	1995-01326	Supp. No.:		Rev. No.:		Page __ of __

INITIATION

DOC. ID: EOP-13.02 Rev: 0 Page 1 of 3

Responsible Group: DBS ICU

Responsible Engineer: J. P. McQuighan

BASIS DOCUMENT

Use Summary: This action value is used to indicate degrading conditions in containment, and to ensure that the Containment Combustible Gas Control Safety Function Acceptance Criteria are satisfied.

Parameter: Hydrogen Concentration

Value: 4.0%

Comments: None

REVIEW AND APPROVAL:

Responsible Engineer: [signature] Date: 11-16-95

Independent Reviewer: KR Nelson Date: 11-17-95

Approval: BGE Date:

OR (for Vendor Documents)

Owner Acceptance Reviewer(s):

NBU: [signature] Date: 11/17/95

PD&MAU: [signature] Date: 1-2-96

ICU: James McQuighan Date: 11-17-95

entered into Notes 9/9/96 DBM

ACTION VALUE BASIS DOCUMENT

Basis Reference(s):

1. BGE Emergency Operating Procedures:

EOP #	Revision	Unit 1 Change #	Unit 2 Change #
4	2	12	9
5	2	16	15
8	2	21	25

2. IR0-033-206, Revision 1.

Attachment:

1. Excerpt from NFPA 325M, pg. 59, "Fire Hazard Properties of Flammable Liquids, Gases, and Volatile Solids", 1991 Edition.

2. IR0-033-206, Revision 1.

Documentation Level: 3

Function of Equipment and Use:

The control room indication for hydrogen concentration is used by the operators to indicate degrading conditions exist in containment. This indication is precautionary and ensures that the Containment Combustible Gas Control Safety Function Acceptance Criteria are satisfied.

Action Value Basis:

Following an Excess Steam Demand Event (ESDE) inside containment, LOAF or LOCA, containment hydrogen concentration is expected to increase as a result of the metal-water reactions involving zircaloy or stainless steel at high RCS temperatures, the radiolysis of water by fission product decay, or the corrosion of aluminum and zinc by containment spray. The purpose of this action value is to warn the operators that the possibility of hydrogen ignition exists if containment hydrogen concentration reaches 4.0% by volume. In addition, caution statements in the EOPs advise against the use of any equipment in containment if the hydrogen concentration is greater than 4.0% by volume to reduce the possibility of hydrogen ignition.

Two control room recorders (AR-6519 and AR-6527) indicate the hydrogen concentration in containment, and follow-on grab samples provide redundant indication. Corroborative indications of degrading containment conditions are provided by containment temperature indication, containment pressure indication, and containment radiation monitors.

The current action value of 4.0% by volume hydrogen concentration, defined as the Lower Explosive Limit of hydrogen mixed in air (Attachment 1), is conservative because the moisture content in containment following a LOCA, LOAF, or ESDE would inert the hydrogen-air mixture (Reference 2). When the hydrogen concentration exceeds 0.5%, all available Hydrogen Recombiners will be operating to maintain/reduce the hydrogen concentration to a non-explosive level. If the Containment Environment continues to deteriorate, the EOPs direct the operator to ERPIPs. Subsequent Hydrogen Purge System operation is directed by the Plant Technical Support Center based on grab sample results. For the following reasons, this application is a Level 3 application and consideration of instrument uncertainty is not required:

- The action value is very conservative considering hydrogen-air mixture inerting by moisture.
- No equipment manipulations are made by operators as a result of this indication.
- The action value is only precautionary to remind operators of the possibility of ignition following a LOCA, LOAF, or ESDE.
- Corroborative indications of containment environmental conditions are available.
- Redundant indication of hydrogen concentration via grab samples is available.

Remaining Actions:

Chemistry will verify that uncertainty is applied to grab samples as needed. This will be from an EOP perspective and will be tracked by BGE AIT # ES199501326, Milesone 8.

ACTION VALUE BASIS DOCUMENT

ESP No.:	ES199801680	Supp. No.:	001	Rev. No.:	0000	Page of

INITIATION

DOC. ID: EOP-23.02 Rev: 1 Page 1 of 3

Responsible Group: Bechtel I&C RDO

Responsible Engineer: S. Kapur

BASIS DOCUMENT

Use Summary: This action value is used to verify that plant parameters are in the normal or expected post-trip range, to ensure that Inventory Control and RCS/Heat Removal Safety Function Acceptance Criteria are satisfied, to provide indirect support of a safety function, and to consider parameters in the decision making process.

Parameter: Subcooling Margin (SCM)

Value: 25° F Subcooled

Comments: None

REVIEW AND APPROVAL:

Responsible Engineer: Sushant Kapur Date: 3.6.03

Independent Reviewer: MSG [illegible] (Bechtel) Date: 3/6/03

Approval: Dlaneja (Bechtel) Date: 3/10/03

OR (for Vendor Documents)

Owner Acceptance Reviewer(s): J. P. McQuighan JPM 9/5/03

NEU: ______ Date: ______

PD&MAU: ______ Date: ______

ICU: ______ Date: ______

ACTION VALUE BASIS DOCUMENT

Basis Reference(s):

1. BGE Emergency Operating Procedures:

EOP #	Revision	Unit 1 Change #	Unit 2 Change #
6	2	13	12
8	2	21	25

2. CE-NPSD-928-NP, Section 4, "Possible Solutions to Margin Loss", Revision 1.

3. Deleted

4. BGE Action Value Basis Document AVB # EOP-23.01, "Subcooling Margin", Revision 1.

5. Deleted

6. BGE Emergency Operating Procedures:

Attachment #	Revision	Unit 1 Change #	Unit 2 Change #
1	2	10	8

7. Westinghouse Calculation 8067-ICE-36372, Rev. 01, "Uncertainty Calculation for the Subcooled Margin Monitor System for the BG&E PAMS Upgrade".

Documentation Level: 2

Function of Equipment and Use:

Subcooling Margin (SCM) is used to verify that plant parameters are in the normal or expected post-trip range, to ensure that Inventory Control Safety Function Acceptance Criteria are satisfied, to provide indirect support of a safety function, and to consider parameters in the decision making process.

Action Value Basis:

This action value is used to provide a minimum control limit of SCM for a Steam Generator Tube Rupture (SGTR) and for the Functional Recovery Procedure (FRP), since it can include an SGTR. This value was chosen lower than 30 degrees in order to reduce the primary-to-secondary pressure difference which will reduce flow out of the break. This action value is classified as a level 2 application, which requires that instrument uncertainties be addressed, either in a formal calculation or by engineering judgment.

The control room indication for subcooling margin is obtained from PAMS displays 1CRT1C05A/B and 2CRT2C05A/B. PAM displays on C04 and C06 are part of an integrated system and can provide identical data. These indicators are classified as Category 1 PAM instruments.

The 25 degrees Subcooled action value is intended to ensure that subcooling of the RCS remains above zero when instrument uncertainties are addressed (this is set lower than the 30 degrees subcooled action value to allow RCS and S/G pressures to be more closely equalized to minimize flow out the break). During non-harsh containment conditions (which apply to a SGTR), the 25 degrees subcooling will accommodate instrument uncertainties when PAMS displays are used (Reference 7) at RCS pressure greater than or equal to 250 psia. In addition to this action value, several corroborative indications of adequate SCM are available to the operators, including pressurizer level, pressurizer pressure, and RVLMS.

This action value is also used as one of the many Reactor Coolant Pump (RCP) starting criterion in EOP 6. The purpose of this is to ensure a single phase, subcooled RCS fluid, thereby limiting the expected decrease in Pressurizer level once the pumps are started. This function is classified as a level 2 application, which requires that instrument uncertainties be addressed, either in a formal calculation or by engineering judgment. Based on the uncertainty discussions in Reference 4, the 25 degree value is sufficient during normal containment conditions. This is acceptable since RCP restart is prohibited by CIS closing component cooling containment Isolation Valves during harsh containment conditions. In addition, the RCP starting criteria also requires that RCS pressure be within the RCP operating curves (Reference 6), which is more limiting than the SCM requirement.

Remaining Actions:

None

INITIATION (Control Doc Type - AVBASIS)

DOCID: EOP-24.34 (DEVICE# - CHANGE#) REV.NO.: 0001 (LETTER, IF APPLICABLE)

RESPONSIBLE GROUP: Bechtel I&C (RDO)

RESPONSIBLE ENGINEER: S. Kapur

BASIS DOCUMENT

Use Summary: This action value is used to establish the Core and RCS Heat Removal Safety Function Status check Criteria for EOP-3, 4, 6, and 7. The value is based on assessing the effectiveness of heat removal via the steam generators.

PARAMETER: CET Temperature

VALUE: < 600 degrees F

COMMENTS:

REVIEW AND APPROVAL:

RESPONSIBLE ENGINEER:	Sushant Kapur	DATE:	3/5/03
INDEPENDENT REVIEWER:	[signature] (Bechtel)	DATE:	03/05/2003
APPROVAL:	[signature] (Bechtel)	DATE:	3/10/03

OR (for Vendor Documents)

OWNER ACCEPTANCE REVIEWER(S): P. McQuighan [initials] 9/5/03

NEU:		DATE:	
PD&MAU:		DATE:	
ICU:		DATE:	

BASIS Reference(s):

1. CEN-152 Rev 4, "Emergency Procedure Guidelines"
2. CA02090, "Uncertainty Calculation for Pressurizer Pressure"
3. System 058 Setpoint File
4. HCI Record of Conversation (attached)
5. System 083 Setpoint File
6. DCALC CA01311, "Instrument Uncertainty Calculation for CET Indication"
7. Westinghouse Calculation 8067-ICE-36370, Rev. 01, "Uncertainty Calculation for the Core Exit Thermocouple System for the BG&E PAMS Upgrade" (CCN PP Document CA01311-0002.)

JMC 9/8/2003

Documentation Level: 2

Function of Equipment and Use:

The Control Room CET indications are used by the operators for Safety Function Status Checks for EOP-3, 4, 6, and 7 to ensure that Steam Generator heat removal is adequate.

Action Value Basis:

The purpose of this action value is to demonstrate adequate heat removal via the Steam Generators. The Main Steam Safety Valves (MSSVs) will lift as necessary to limit S/G pressure and will remove energy from the primary as a result. If S/G pressure increases above the MSSV lift setpoints, then it can be assumed that S/Gs are not effective in removing heat from the RCS and therefore the core. The engineering limit is based on the saturation temperature corresponding to the lift setpoint of the PORVs. Although no specific plant manipulations are required by this engineering limit, uncertainty will be considered to ensure that a proper diagnosis of plant status can be made by operators. Therefore, the AVBD is a category 2.

The nominal setpoint for the PORVs is 2385 psia per Reference (3). The total loop uncertainty associated with the PORV setpoint is + 77.69 psi / -67.78 psi per reference (2). Therefore the lowest actual pressure that could lift a PORV is 2385 – 67.78 = 2317.22 psia. The saturation temperature corresponding to this pressure is 657 degrees F. Therefore as long as RCS temperature remains below 657 degrees, then the S/G's are adequately removing heat from the core. CET indication uncertainty should be considered. For temperatures less than 700 degrees F, an indication bias of –4.1 degrees F exists per references (6 and 7). Other uncertainties associated with CET indication are random and will not specifically be accounted for since eight (8) CET indications are available to the operator. Per reference (4), operators are expected to use the highest reading valid CET indication for EOP applications. With eight indicators, the probability that one of the indicators will read high is approximately 96%. Therefore, as long as the highest indicated CET temperature remains below approximately 653 degrees F (657 – 4 degrees bias), then the S/G's are adequately removing heat from the core. In order to provide additional margin, an engineering limit of 600 degrees F is recommended. This value is sufficiently above the saturation temperature (approx. 548 degrees F) corresponding to the highest MSSV relief pressure (1030 psia per reference 5) to avoid prematurely failing the Safety Function Status Check . (The worst case positive uncertainty of the CETs is 45.70 degrees F per reference 7).

Remaining Actions:

None

EOP-24.34 ATTACHMENT

HCI RECORD OF CONVERSATION

Parties Involved: Jim Willis (HCI), Bob Bleacher (BGE Operations)

Subject: CET Temperature Indications

Date: 20 February 1996

Discussion: I spoke with Bob Bleacher of BG&E Operations regarding the CET temperature indications which are used during Emergency Operating Procedures. Specifically, I asked Bob which of the eight CET temperature indications are used to determine the CET temperature. Bob stated that the operators are instructed to use the most conservative of the valid CET temperature readings. If a temperature reading is not valid, the operator will switch to a valid indication. So it is probable that the operator will have available eight valid CET temperature readings to choose the most conservative value from.

Signature of HCI Representative: ________________

cc. Kirk Melson
Bob Hunter
Bob Bleacher
Frank Barich
BGEAVB File

Calvert Cliffs Units 1 and 2
TECH SPEC ACTION VALUE BASIS DOCUMENT
MODULE 1 - SPENT FUEL POOL LEVEL

ESP No.:	ES199800829	Supp. No.:	000	Rev. No.:	0000	
TS-05.01	Rev. 0					

INITIATION:

Responsible Group: DES ICU

Responsible Engineer: Stephen Keefe

BASIS DOCUMENT:

System: FUEL STORAGE
Component: SPENT FUEL POOL
Parameter: LEVEL
T.S. Value(s): > or = 21.5 FT above top of irradiated fuel in the fuel racks

Mode(s): All

Comment: Margin is satisfactory.

REVIEW AND APPROVAL:

ABB:
Verification Status: Complete
The information contained in this document has been verified to be correct by means of design review.

Cognizant Engineer: J.R.Congdon / Joseph R. Congdon Date: 11/29/99

Independent Reviewer: H. F. Shamro / Howard F. Shamro Date: 11/30/99

Management Approver: C. J. Gimbrone / Date: 12/1/99

BGE: Owner Acceptance Reviewer(s):

NEU: Kim S R Knippel Date: 1/13/00

ICU: S.E. KEEFE Date: 1/13/00

ESP No.:	ES199800829	Supp. No.:	000	Rev. No.:	0000	
TS-05.01	Rev. 0					

Part I - Action Value Bases

References

	Unit	Document
1.	1, 2	Calvert Cliffs Nuclear Power Plant Improved Technical Specifications. Unit 1 Amendment No. 230, Unit 2 Amendment No. 206
2.	1, 2	Calvert Cliffs UFSAR, Revision 25
3.	1, 2	BG&E Calculation Number C-91-224, Rev. 0, Spent Fuel Pool Water Level Evaluation Per NCR 8936/9960, 8/26/91
4.	1, 2	Calvert Cliffs Master Calibration Data Sheet for Spent Fuel Pool Level, 0-LIA-2001 and 0-LIA-2002
5.	1, 2	BGE DCALC CA 04048, Rev. 0, "Fuel Handling Accident During Reconstitution," 3/10/98
6.		CE-NPSD-925, revision 00, "Guidelines for Addressing Instrument Uncertainties in Emergency Operating Procedures and Technical Specifications" January 1994
7.	1, 2	CCNPP Unit One and Two Control Room Logsheet Mode 1 and 2
8.	1, 2	CCNPP Surveillance Test Procedure, STP 0-87-1(2), Rev. 12, Borated Water Source 7 Day Operability Verification, 4/29/98

T.S. Value Use and Application Summary:
LCO 3.7.13 – Limiting condition for operation for spent fuel pool water level.
SR 3.7.13.1 – OPERABILITY surveillance requirement for spent fuel pool water level.

Technical Specification Value:	Engineering Limit:	Documentation Level Category:
> or = 21.5 ft above top of irradiated fuel in the fuel racks	23 ft above the top of the fuel pins of a ruptured fuel assembly standing on the floor of the SF Pool	2

Action Value Basis:
The engineering limit for minimum spent fuel pool level is 23 feet of water covering a ruptured fuel assembly. If a fuel handling incident were to occur while handling fuel in the spent fuel pool area, the following assumptions apply:

- The activity release to the spent fuel pool water is identical to that assumed in the case of a fuel handling incident in containment.
- The overall spent fuel pool decontamination factor for iodine is 100. This is based upon a water level of at least 23 ft covering a ruptured fuel assembly.
- Because a fuel handling accident is shown only to occur by the dropping of a fuel assembly onto the floor of the spent fuel pool, the Technical Specification requirements that 21.5 ft of water cover the spent fuel racks preserves at least 23 ft above a damaged fuel assembly.
- The release of activity to the air above the spent fuel pool is identical to that released to the containment air in the event of a fuel handling incident in containment.

The analytical limit is 23 ft above the top of a fuel pin. The "top of a fuel pin" is inferred from the treatment of a fuel assembly on top of a spacer in the SF fuel rack (Ref. 5, pg. 13) where the height of 7.266" from the top of the fuel assembly to the top of a fuel rod is credited. The basis for the analytical limit is to have 23 ft of water available for scrubbing in the event of the rupture of a fuel assembly sitting upright on the floor of the Spent Fuel Pool (SFP).

Technical Specifications (TS) allows meeting the 23 ft criterion by requiring 21.5 ft above fuel in the SFP racks, as this is stated to ensure 23 ft over a bundle resting on the bottom of the SFP. The surveillance procedure criterion is 65' 8.5" (Ref. 8, pg. 14). This is most likely the result of the walkdown performed for C-4477.0 (Ref. 3, pg. 13) which states that the top of the fuel assembly in the storage rack is at 44'2 1/2". Adding 21' 6" gives 65' 8 1/2".

Therefore, the analytical limit based on SFP level is:

Elevation of the bottom of the SFP	=	+ 30' 0" (Ref. 3, SH 20)
Height of a fuel bundle	=	+ 13' 1.241" (Ref. 3, SH 20)
Height of water column assumed in analysis	=	+ 23' 0" (Ref. 5, pg. 13)
Distance from top of the fuel ass'y to top of fuel pin	=	- 7. 266" (Ref. 5, pg. 13)
Analytical limit based on SFP level	=	+ 65' 5.98" = 65' 6" or 65.5 ft

Potential effects of indication error:
If spent fuel pool level is less than 23 ft of water covering a ruptured fuel assembly, then, the safety analysis is not valid and more iodine may be released following a fuel handling accident (dropped fuel assembly) than calculated in the safety analysis.

Supporting Reference Excerpts:
Ref. 2 -- 14.18.3.2 -- Fuel Handling Incident in the Spent Fuel Pool Area
If a fuel handling incident were to occur while handling fuel in the spent fuel pool area, the following assumptions would apply:

a. The activity release to the spent fuel pool water is identical to that assumed in the case of a fuel handling incident in containment. The overall spent fuel pool decontamination factor for iodine is 100. This is based upon a water level of at least 23 ft covering a ruptured fuel assembly on the floor of the SF Pool. Because a fuel handling accident is shown only to occur by the dropping of a fuel assembly onto the floor of the spent fuel pool, the Technical Specification requirements that 21.5 ft of water cover the spent fuel in the racks preserves at least 23 ft above a damaged fuel assembly. The release of activity to the air above the spent fuel pool is identical to that released to the containment air in the event of a fuel handling incident in containment.

ESP No.:	ES199800829	Supp. No.:	000	Rev. No.:	0000	
TS-05.01	Rev. 0					

Part II - Instrument Uncertainties Assessment

Spent Fuel Pool Level Instrument ID:	0-LIA-2001 (64 FT 8 IN- 67 FT 8 IN)	0-LIA-2002 (64 FT 8 IN- 67 FT 8 IN)
TLU Calc Complete?:	No	No
Calibration data:	Yes (Ref. 4)	Yes (Ref. 4)
TLU, % Span:	-	-
Accuracy:	± 1.0%	± 1.0%
TLU, Eng. Units:	-	-
Setting Tolerance:	± 1.4% / ± 0.5 IN	± 1.4% / ± 0.5 IN

Instrument Uncertainties Assessment:
This application of Spent Fuel Pool level indication is Category 2 per CE-NPSD-925 (reference 9, pg. 21). This category is reserved for instrument uses that possess a moderate to low degree of nuclear safety significance, or are corroborative in nature. Engineering judgement may be used when determining and applying instrument uncertainties.

The engineering limit for minimum spent fuel pool (SFP) level of 23 feet of water covering a ruptured fuel assembly is 65' 6". The minimum TS limit is ≥ 21.5 ft above fuel in the SFP racks is 65' 8.5". The surveillance (Ref. 8, pg. 14) acceptance criterion is also 65' 8.5". Therefore, there is 2.5" of margin between the analytical limit and the TS limit / surveillance procedure criterion.

The Control Room operator is required to log the lowest level of LIA-2001 or LIA-2002 every 12 hours (66.5 – 67.25 ft) (Ref. 7, pg. 5). Spent Fuel Pool Level is also checked locally once a day using a tape measurement attached to the side of the Spent Fuel Pool. The remote Control Room SF Pool level instrument is the primary means of monitoring SF Pool level. The local indication is a back-up. TS compliance is achieved by performance of STP 0-87-1/2 every seven days. The surveillance acceptance criterion is 65' 8.5" (Ref. 8, pg. 14).

There is no TLU calculation for SFP level indication, LIA-2001 or LIA-2002. However, these instruments are calibrated periodically to a tolerance of ± 1.4% (± 0.5") (Ref.4) and the error allowance for readability is less than 1%. The 2.5" analytical margin will accommodate this instrument inaccuracy. Use of calibration data is judged to be acceptable instead of requiring calculating the TLU because if it were calculated, the TLU value would likely still be less than the available margin. In other words, in order to exceed the 65' 6" analytical minimum level with an indicated minimum level of 65' 8.5", the total loop inaccuracy would have to be greater than ± 2.5" or ± 8.9%. Loop inaccuracy of this magnitude is not likely.
The 2.5" available margin between the analytical limit and the TS limit / surveillance procedure criteria can accommodate the ± 0.5" instrument inaccuracy.

Evaluation:
The margin is satisfactory.

Calvert Cliffs Units 1 and 2
TECH SPEC ACTION VALUE BASIS DOCUMENT
MODULE 1 – REFUELING CANAL WATER LEVEL

ESP No.:	ES199800829	Supp. No.:	000	Rev. No.:	0000	
TS-06.01	Rev. 0					

INITIATION:

Responsible Group: DES ICU

Responsible Engineer: Stephen Keefe

BASIS DOCUMENT:

System: REFUELING
Component: WATER
Parameter: LEVEL
T.S. Value(s): > or = 23 ft over top of fuel seated in reactor vessel
< 23 ft over top of fuel seated in reactor vessel
Mode(s): 6

Comment: Margin is satisfactory.

1. During the course of this evaluation, it was determined that LI-4140 was currently referenced for use in OP-07. It should not be, since LI-4138 and LI-4139 replaced it. BG&E has initiated an Issue Report (IR3-000-608) to revise OP-7 to remove all reference to LI-4140.

REVIEW AND APPROVAL:

ABB:
Verification Status: Complete
The information contained in this document has been verified to be correct by means of design review.

Cognizant Engineer: J.R.Congdon / Joseph Congdon Date: 11/29/99

Independent Reviewer: H. F. Shamro / Howard F. Shamro Date: 11/30/99

Management Approver: C. J. Gimbrone / [signature] For CJG Date: 12.1.99

BGE: Owner Acceptance Reviewer(s):

NEU: Kim R Knipp Date: 1/13/00

ICU: S.E. KEEFE [signature] Date: 1/13/00

Calvert Cliffs Units 1 and 2
TECH SPEC ACTION VALUE BASIS DOCUMENT
MODULE 1 – REFUELING CANAL WATER LEVEL

ESP No.:	ES199800829	Supp. No.:	000	Rev. No.:	0000	
TS-06.01	Rev. 0					

Part I - Action Value Bases

References

	Unit	Document
1.	1, 2	Calvert Cliffs Nuclear Power Plant Improved Technical Specifications. Unit 1 Amendment No. 230, Unit 2 Amendment No. 206
2.	1, 2	Calvert Cliffs UFSAR, Revision 25
3.	2	BGE Procedure "Calvert Cliffs Nuclear Power Plant Unit Two OP-7 Shutdown Operations," 5/12/99
4.	1, 2	CE-NPSD-925, revision 00, "Guidelines for Addressing Instrument Uncertainties in Emergency Operating Procedures and Technical Specifications" January 1994
5.	1, 2	CCNPP Unit One and Two Control Room Logsheet Mode 6
6.	1, 2	BG&E Calculation No. I-92-39, Rev. 0, Instrument Loop Uncertainty Estimate, RCS Mid-Loop Wide Range Level Monitor Loop

T.S. Value Use and Application Summary:
APP, 3.9.4 –Applicability condition for Shutdown cooling and Coolant Circulation-High Water Level
APP, 3.9.5 – Applicability condition for Shutdown cooling and Coolant Circulation-Low Water Level
LCO, 3.9.6 – Limiting condition for operation for Refueling Pool Water Level
SR, 3.9.6.1 – Surveillance requirement for Refueling Pool Water Level

Technical Specification Value:	Engineering Limit:	Documentation Level Category:
> or = 23 FT, over top of fuel seated in reactor vessel during CORE ALTERATION or movement of irradiated fuel or when only one shutdown cooling loop is OPERABLE and in operation. < 23 FT, over top of fuel seated in reactor vessel is permitted if no CORE ALTERATION or fuel movement is taking place and two SDC loops are OPERABLE	• 23 ft over top of fuel seated in reactor vessel if only one shutdown cooling loop OPERABLE and in operation • < 23 ft over top of fuel is allowed if two shutdown cooling loops are OPERABLE with one loop in operation	2

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TS-06.01	Rev. 0					

Action Value Basis:
The engineering limit for minimum refuel pool level is 23 ft of water covering a fuel assembly seated in the reactor vessel. The top of the core (and assembly) is 32.9 ft. Therefore, the analytical limit is 55.9 ft (32.9 + 23). The TS Refueling Pool Level is 56.70 ft (Ref. 3, Table 1). The height of water above the top of a fuel assembly at the TS limit is (56.7' – 32.9') = 23.8'. Refueling level is surveilled to (56.7 – 67.25) in Mode 6 using the Control Room Logs (Ref. 5, pg. 2).

Per ref. 1 – The engineering limits for refueling canel water level is a function of the number of shutdown cooling loops OPERABLE and in operation, thus:

Ref. 1, Bases 3.9.4, Applicable Safety Analysis summary: If only one shutdown cooling loop is OPERABLE (and in operation), then refuel pool level must be > or = 23 feet of water above the top of fuel seated in the reactor vessel. Although no specific analysis has been located, it is believed that high water level and the resulting higher heat capacity would permit sufficient time to take compensatory actions in case the operating SDC loop were to fail.

Ref. 1, Bases 3.9.5, Applicable Safety Analysis summary: If two shutdown cooling loops are OPERABLE, with at least one loop in operation, then, refuel pool level may be <23 feet of water above the top of fuel seated in the reactor vessel.

A minimum refueling pool water level of 23 ft above the irradiated fuel assemblies seated in the reactor vessel is required during fuel movement to ensure that the radiological consequences of a postulated fuel handling accident inside containment are within the acceptable limits given in the UFSAR (Ref. 2).

Potential effects of indication error:
If refueling pool level is less than 23 feet of water covering a ruptured fuel assembly, then, the safety analysis is not valid and more iodine may be released following a fuel handling accident (dropped fuel assembly) than calculated in the safety analysis.

Supporting Reference Excerpts:
Ref. 2 – 14.18.3.1 – Fuel Handling Incident in Containment
Because iodine is readily absorbed by water and the fuel being handled is under water, much of the iodine released from the damaged rods would be retained in the refueling pool water. To account for this preferential retention of iodine by the pool water, a decontamination factor of 10^2 is assumed, which corresponds to the value suggested in Regulatory Guide 1.25. No additional credit is taken for plate-out of iodine on surfaces within the containment.

Ref. 2 – 14.18.3.2 – Fuel Handling Incident in the Spent Fuel Pool Area
The release of activity to the air above the spent fuel pool is identical to that released to the containment air in the event of a fuel handling incident in containment.

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Part II - Instrument Uncertainties Assessment

RCS Water Level Instrument ID:	Refueling Level Cart Indicator (Wide Range) 1(2) LI-4139	Refueling Level Cart Recorder (Wide Range) 1(2) LR-4138	Refueling Level Cart Alarm (Wide Range) 1K184	Local Refueling Level Indicator (Wide Range) 1(2) LG-4139	Local Tygon (WR)
Is TLU Calc Complete?: **Calibration data:**	Yes Reference 6	Yes Reference 6	Yes Reference 6	No	N/A
TLU, % Span:	N/A	N/A	N/A	N/A	N/A
TLU, Eng. Units:	± 7.48 in.	±8.21 in.	±11.57 in.	N/A	N/A

Instrument Uncertainties Assessment:
This application of Refueling Pool level indication is Category 2 per CE-NPSD-925 (reference 4, pg. 21). This category is reserved for instrument uses that possess a moderate to low degree of nuclear safety significance, or are corroborative in nature. Engineering judgement may be used when determining and applying instrument uncertainties.

LT-4138(NR) and 4139(WR) are temporary instruments located on the refueling instrumentation cart, which is placed in service during refueling. They are read remotely in the Control Room. If these instruments are not available, LG-4139 or a tygon are available for wide range RCS/Refueling Pool level determination in containment.

The analytical limit is 55.9 ft (32.9 + 23). The TS Refueling Pool Level in 56.70 ft (Ref. 3, Table 1). Refueling level is surveilled to (56.7 – 67.25) in Mode 6 using the Control Room Logs (Ref. 5, pg. 2). The height of water above the top of a fuel assembly is at the TS limit is (56.7' – 32.9') = 23.8'. There exists a (23.8' – 23.0') = 0.8' = 9.6" of margin.

The above table list TLUs for wide RCS range level instrumentation. The TLU for 1(2) LI-4139 is ± 7.48 inches and the TLU for 1(2) LI-4138 is ± 8.21 inches. Since these values are less than the 9.6" of available margin, the margins are satisfactory.

Evaluation:
Margin evaluation is satisfactory. The above discussion demonstrates that LI-4139 and 4138 provide sufficient margin to the analytical value.

Calvert Cliffs Units 1 and 2
TECH SPEC ACTION VALUE BASIS DOCUMENT
MODULE 12 – RMS ALARM, CONTAINMENT HIGH RANGE

ESP No.:	ES199800829	Supp. No.:	000	Rev. No.:	0000	
TS-76.01	Revision 1					

INITIATION:

Responsible Group: DES ICU

Responsible Engineer: Stephen Keefe

BASIS DOCUMENT:

System: RMS
Component: AREA
Parameter: RADIATION
T.S. Value(s): Operable (CONTAINMENT HIGH RANGE)

Mode(s): 1, 2 and 3

Comment: Margin evaluation is not applicable.

RECORD OF REVISIONS

REVISION	PAGE NO.	DESCRIPTION OF CHANGE
0	All	Initial Issue
1	All	Miscellaneous clarifications and edits

REVIEW AND APPROVAL:

CE Nuclear Power:
Verification Status: Complete
The information contained in this document has been verified to be correct by means of design review.

Cognizant Engineer: L. A. Wild / L O Wild Date: 7/28/00

Independent Reviewer: J. R. Congdon / Joseph R Congdon Date: 8/1/00

Management Approver: R. O. Doney / R O Doney Date: 8-1-00

BGE: Owner Acceptance Reviewer(s):

NEU: Kimberly R Knipp Date: 9/26/00

ICU: S.E. KEEFE / Stephen Keefe Date: 9/26/00

Calvert Cliffs Units 1 and 2
TECH SPEC ACTION VALUE BASIS DOCUMENT
MODULE 12 – RMS ALARM, CONTAINMENT HIGH RANGE

ESP No.:	ES199800829	Supp. No.:	000	Rev. No.:	0000	
TS-76.01	Revision 1					

Part I - Action Value Bases

References

	Unit	Document
1.	1, 2	Calvert Cliffs Nuclear Power Plant Technical Specifications: Unit 1 Amendment No. 233, Unit 2 Amendment No. 209
2.	1, 2	Calvert Cliffs UFSAR, Revision 26
3.		CE-NPSD-925, Revision 00, "Guidelines for Addressing Instrument Uncertainties in Emergency Operating Procedures and Technical Specifications", January 1994
4.	1	STP 0-98-1, Containment High Range Monitors Monthly Functional Test, Revision 4, January 4, 1997
5.	2	STP 0-98-2, Containment High Range Monitors Monthly Functional Test, Revision 4, January 4, 1997
6.	1	STP-M-562-1, Containment High Range Radiation Monitor Alignment Check, Revision 4, May 28, 1992
7.	1, 2	Calvert Cliffs Unit 0, 1, and 2 Setpoint File, NEQR440, 12/16/99, for Systems 077

Use and Application Summary:

SR 3.3.10.1- Operability Channel Check for PAM Instrumentation
SR 3.3.10.3- Operability Channel Calibration for PAM Instrumentation

Technical Specification Value:	Engineering Limit:	Documentation Level / Category
Operable – PAMI	None	6

Action Value Basis:

Reference 1, Technical Specification Bases B 3.3.10 (pg. B 3.3.10-2, B 3.3.10-7)
APPLICABLE SAFETY ANALYSES -- The PAM instrumentation ensures the OPERABILITY of Regulatory Guide 1.97 Type A variables, so that the control room operating staff can:

- Perform the diagnosis specified in the emergency operating procedures. These variables are restricted to preplanned actions for the primary success path of DBAs; and
- Take the specified, preplanned, manually controlled actions, for which no automatic control is provided, that are required for safety systems to accomplish their safety functions.

The PAM instrumentation also ensures OPERABILITY of Category I, non-Type A variables. This ensures the control room operating staff can:

- Determine whether systems important to safety are performing their intended functions;
- Determine the potential for causing a gross breach of the barriers to radioactivity release;
- Determine if a gross breach of a barrier has occurred; and
- Initiate action necessary to protect the public as well as to obtain an estimate of the magnitude of any impending threat.

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TS-76.01	Revision 1					

Containment area radiation detectors are provided to monitor for the potential of significant radiation releases and to provide release assessment for use by operations in determining the need to invoke site emergency plans.

Containment area radiation instrumentation consists of two radiation detectors with displays and alarm in the Control Room. The radiation detectors have a measurement range of 1 to 10^8 R/hr.

Reference 2, UFSAR, Section 7.5.8 (pg. 7.5-17)
Variables and systems can be monitored under accident conditions. Included is the instrumentation required for the operators to take the plant to hot shutdown from outside the Control Room and to monitor for radiation released following a postulated accident. Certain instrumentation in the Control Room and on the Auxiliary Safe Shutdown Panels used for normal plant operations is designated for post-accident monitoring (PAM) use.

Potential effects of indication error:
None

Supporting Reference Excerpts:
None

Part II - Instrument Uncertainties Assessment

	Containment High Range Radiation Monitor
Instrument ID:	1(2)-RI-5317A, 1(2)-RI-5317B (1 to 10^8 R/hr)
TLU Calc Complete: **Calibration Data:**	No Yes (Ref. 6)
% Span: **Alarm Tolerance:**	- ± 4 R/Hr (Ref. 7, Pg. 8 and 13)
Eng. Units: **High Alarm Setpoint:** **High High Alarm Setpoint:**	(R/Hr) 6 R/Hr 20 R/Hr (Ref. 7, Pg. 8 and 13)

Instrument Uncertainties Assessment:
Per CE-NPSD-925 (Reference 3, page 23), this parameter application is Category 6 (not applicable) because it is a Technical Specification OPERABILITY requirement only. No specific value is given in the referenced section of TS; therefore instrument uncertainties can not be evaluated. These instrument applications are covered elsewhere when the instruments are actually used.

Evaluation:
Margin evaluation is not applicable.

Calvert Cliffs Nuclear Power Plant

TECHNICAL REQUIREMENTS MANUAL (TRM)

(NORMS DOC ID: NO-TRM)

Revision 01502

Effective Date 4/27/10

Sponsor: General Supervisor - Shift Operations

Approval Authority: Manager - Operations

RECORD OF REVISIONS AND CHANGES

Revision	Change	Summary of Revision or Change
015	02	Section 15.3.6 Applicability Deleted (Unit-2 only). PCR-10-02030. ECP-09-000147 Now also applicable to Unit-1. Section 15.3.6.B Changed from 4-hour TO 16-minute. Deleted Note explaining that the Nonconformance can be exited when either the 2-minute or 16-minute are restored. PCR-10-02030. ECP-09-000147 The system gives bad quality when the 2-minute goes bad. The Nonconformance can be exited when restored. No explanation is needed.

TABLE OF CONTENTS

SECTION TITLE **PAGE**

15.9.4 CRANE TRAVEL - SPENT FUEL POOL 85

LIST OF TABLES

LIST OF FIGURES

LIST OF ACRONYMS

APD	Amplitude Probability Distribution
ASME	American Society of Mechanical Engineers
BAST	Boric Acid Storage Tank
CEA	Control Element Assembly
COLR	Core Operating Limits Report
ECCS	Emergency Core Cooling System
LEFM	Leading Edge Flow Measurement
NRC	Nuclear Regulatory Commission
RCS	Reactor Coolant System
RWT	Refueling Water Tank
TNC	Technical Normal Conditions
TVR	Technical Verification Requirement
SA	Spectral Analysis
SFP	Spent Fuel Pool

BASES

[B0627] Licensing Renewal Aging Management Basis Document: Snubber Visual Inspection Program (AMBD-0028, Rev. 0000)

[B0650] UCR00191, Letter from Atomic Energy Commission (AEC), dtd 1/31/74. CCNPP agreed to changes in this letter, but was not put into Tech Spec revision 4/1/74 as requested: Add verification requirements for inventory of sealed sources.

[B0653] Memo dtd 11/20/00 from A. J. O'Donnell to Steve McCord regarding basis for Calvert Cliffs Meteorological Parameter – Delta T (ERPIP 825).

[B0703] ASME, Section XI, Subsection IWL, Requirements for Class CC Concrete Components of Light-Water Cooled Plants, 1992 Edition with 1992 Addenda as modified and amended by 10 CFR 50.55a.

[B0772] Steam Generator Project – Design Engineering, Memo SGP-EN-01-297, Basis for change to TRM 15.4.1, Reactor Coolant System Chemistry, dtd 12/14/01 to S. C. McCord from T. W. Reed.

[B0783] Memo dtd 3/28/94 from J. Alvey to M. Navin; Subj. Watertight Door Tickets.

[B0800] Memo dtd 4/27/02 from J. Calle to S. McCord: Subj. Unit-1 Steam Generator Replacement Project TRM 15.7.1 Steam Generator Pressure/Temperature Limitation.

[B0889] ES200400058/IR4-016-662 (1E200300078, MS #2): Technical Evaluation of the Licensing Basis and Design Basis for the RCP Pump Bay Heat Detectors.

[B0898] Memo dtd 12/4/03 from Jack DeSando, Stephen Keefe and Jay Robinson to C. Jones: Subj: Operability of New Intake Building Fire Detection System.

[B0909] ES200400030, Modify the Restoration Times of the Boric Acid Flow Paths contained in Section 15.1.2

[B2369] EPRI PWR Primary Water Chemistry Guidelines

15.0 TECHNICAL REQUIREMENTS MANUAL

15.0.1 BACKGROUND

This manual contains requirements that were removed from the Technical Specifications during the conversion from the standard format Technical Specifications to the improved standard Technical Specifications. This conversion was approved by the Nuclear Regulatory Commission (NRC) on May 4, 1998 (Amendment Nos. 227/201), and placed in Revision 22 of the UFSAR.

UCR No. 00110, approved on November 9, 1999, permitted removal of the TRM from the UFSAR and replaced it with a summary. This change endorsed in NEI 98-03, Guidelines for Updating Final Safety Analysis Reports, Revision 1, and a NEI letter dated June 30, 1999. The NEI letter answers questions about NEI 98-03. The letter was reviewed and concurred with by the NRC. NEI and the NRC has said the TRM can and should be a document separate from the UFSAR and controlled by the 10 CFR 50.59 process. The 10 CFR 50.59 revision process will be controlled by NO-1-118, Control of the Technical Requirements Manual (TRM).

To minimize the impact of associated procedures that reference the Technical Requirements Manual (TRM), the manual will continue to retain the same chapter numbering and title that was used when it was maintained within the UFSAR, Revision 24.

15.0.2 USE

The following terms are used in this manual and have the same definition as the equivalent term in the Technical Specifications.

- channel calibration
- channel check
- channel functional test
- core operating limits report (COLR)
- mode
- operable/operability
- rated thermal power
- shutdown margin
- staggered test basis
- thermal power

The use of logical connectors (AND, OR) is the same as their use in the Technical Specifications.

Technical Normal Conditions (TNCs) are intended to be met during the times they are applicable. If a TNC is not met, the contingency measures must be taken within the specified time, unless otherwise noted. If a Restoration Time requires periodic performance on a "once per…" basis, a frequency extension (1.25 times allowance) applies to each performance after the initial performance. If a TNC is no longer applicable, the contingency measures do not have to be completed.

Technical Verification Requirements (TVRs) are included as a means to determine equipment operability. They must be performed within the frequency (with a 1.25 times allowance) noted for equipment to meet the TNC. The measuring of the frequency is from the time of the previous performance or from the time a specified condition of frequency is met.

15.0.3 FAILURE TO MEET A TNC OR TVR

When it is discovered that a TNC has not been met and the associated contingency measures are not satisfied (or an associated contingency measure is not provided), the equipment subject to the TNC is in a nonconforming condition, subject to the requirements of CNG-OP-1.01-1002, Conduct of Operability Determinations/Functionality Assessments (e.g., Attachment 6, Functionality Assessments), 10 CFR Part 50, Appendix B, Criteria XV and XVI, Generic Letter 91-18 and Generic Letter 91-18, Revision 1. In this situation, appropriate actions shall be taken as necessary to provide assurance of continued safe plant operations. In addition, an issue report shall be written and assessment of reasonable assurance of safety shall be conducted. Items to be considered for this assessment include the following:

- Availability of redundant or backup equipment;
- Compensatory measures, including limited administrative controls;
- Safety function and events protected against;
- Probability of needing the safety function;
- Conservatism and margins; and
- Probabilistic Risk Assessment or Individual Plant Evaluation results that determine how operating the plant in the manner proposed will impact core damage frequency.

If this assessment concludes that safety is sufficiently assured, the facility may continue to operate while prompt corrective action is taken.

Entry into a Mode or other specified condition in the Applicability, where a TNC is not met, may be made after completing the above assessment and it concludes safety is is sufficiently assured; exceptions to this Requirement are stated in the individual Requirement.

- This Requirement shall not prevent changes in Modes or other specified conditions in the Applicability that are required to comply with the Contingency Measures or that are part of a shutdown of the unit.

When it is discovered that a TVR frequency (including the 1.25 times extension) has not been met, the equipment subject to the TVR is in a nonconforming condition. In this situation, an issue report shall be written and, if indicated, determination to evaluate the impact on plant safety shall be performed in a timely fashion and in accordance with plant procedures.

Actions should be taken to restore conformance with the TNCs/TVRs in a timely fashion.

If equipment has been removed from service or declared inoperable, it may be returned to service under administrative control to perform testing required to demonstrate its operability.

15.1 REACTIVITY CONTROL SYSTEMS

15.1.1 BORON DILUTION

NORMAL CONDITION — TNC 15.1.1 Reactor Coolant System (RCS) flow rate shall be ≥ 3,000 gpm.

APPLICABILITY — Modes 1, 2, 3, 4, 5, and 6, whenever a reduction in RCS boron concentration is being made from a source whose boron concentration is less than the present Shutdown Margin requirements (Refueling Boron for Mode 6) per COLR.

CONTINGENCY MEASURES

Nonconformance	Contingency Measures	Restoration Time
A. The RCS flow rate is not within limits.	A.1 All operations involving a reduction in RCS boron concentration must be suspended.	Immediately

VERIFICATION REQUIREMENTS

<table>
<tr><th>TVR</th><th>Verification</th><th>Frequency</th></tr>
<tr><td rowspan="2">15.1.1.1</td><td>Verify RCS flow rate is ≥ 3,000 gpm.
NOTE: This technical verification requirement is not required to be performed in Modes 1 and 2 (i.e., RCP's operating).</td><td>Within 1 hour prior to the start of a reduction in RCS boron concentration

AND

Every hour during a reduction in RCS boron concentration</td></tr>
<tr><td colspan="2">This includes verifying at least one reactor coolant pump is in operation or verifying that at least one low pressure safety injection pump is in operation and supplying the required amount of flow through the RCS.</td></tr>
</table>

15.1.2 BORATION FLOW PATHS - OPERATING

NORMAL CONDITION

TNC 15.1.2 Two boron injection flow paths shall be operable.

Each boration flow path consists of a boric acid storage tank (BAST) connected to the RCS via a boric acid pump or gravity feed connection, and a charging pump; or a refueling water tank (RWT) connected to the RCS via a charging pump. Charging pumps are required to be powered from separate emergency buses. Each boric acid pump is required to be powered from an emergency bus. Each BAST must have its associated heat tracing systems. Each operable boron injection flow path must have at least one heat tracing circuit operable.

No actuation signals are required for operability.

APPLICABILITY Modes 1, 2, 3, and 4.

CONTINGENCY MEASURES [B0909]

Nonconformance	Contingency Measures	Restoration Time
A. One required boron injection flow path is inoperable due to Charging Pump inoperability.	A.1 Restore the required boron injection flow path to operable status	72 hours
B. One required boron injection flow path is inoperable for reasons other than Condition A.	B.1 Restore the required boron injection flow path to operable status.	7 days
C. Contingency measure and associated restoration time of Nonconformance A or B are not met.	C.1 See Section 15.0.3.	

VERIFICATION REQUIREMENTS

TVR	Verification	Frequency
15.1.2.1	Verify the temperature of the heat traced portion of the flow path from the concentrated BASTs is above the temperature limit line shown on Figure 15.1.2-1.	7 days
15.1.2.2	Verify the BAST boron concentration is as specified in Figure 15.1.2-1, but limited to ≤ 8%.	7 days
15.1.2.3	Verify the BAST borated water volume is as specified in Figure 15.1.2-1.	7 days

15.1.2 BORATION FLOW PATHS - OPERATING - Continued

VERIFICATION REQUIREMENTS - Continued

TVR	Verification	Frequency
15.1.2.4	Verify the BAST borated water solution temperature is as specified in Figure 15.1.2-1.	7 days
15.1.2.5	Verify that each manual, power-operated, or automatic valve in the flow path that is not locked, sealed, or otherwise secured in position, is in its correct position.	31 days

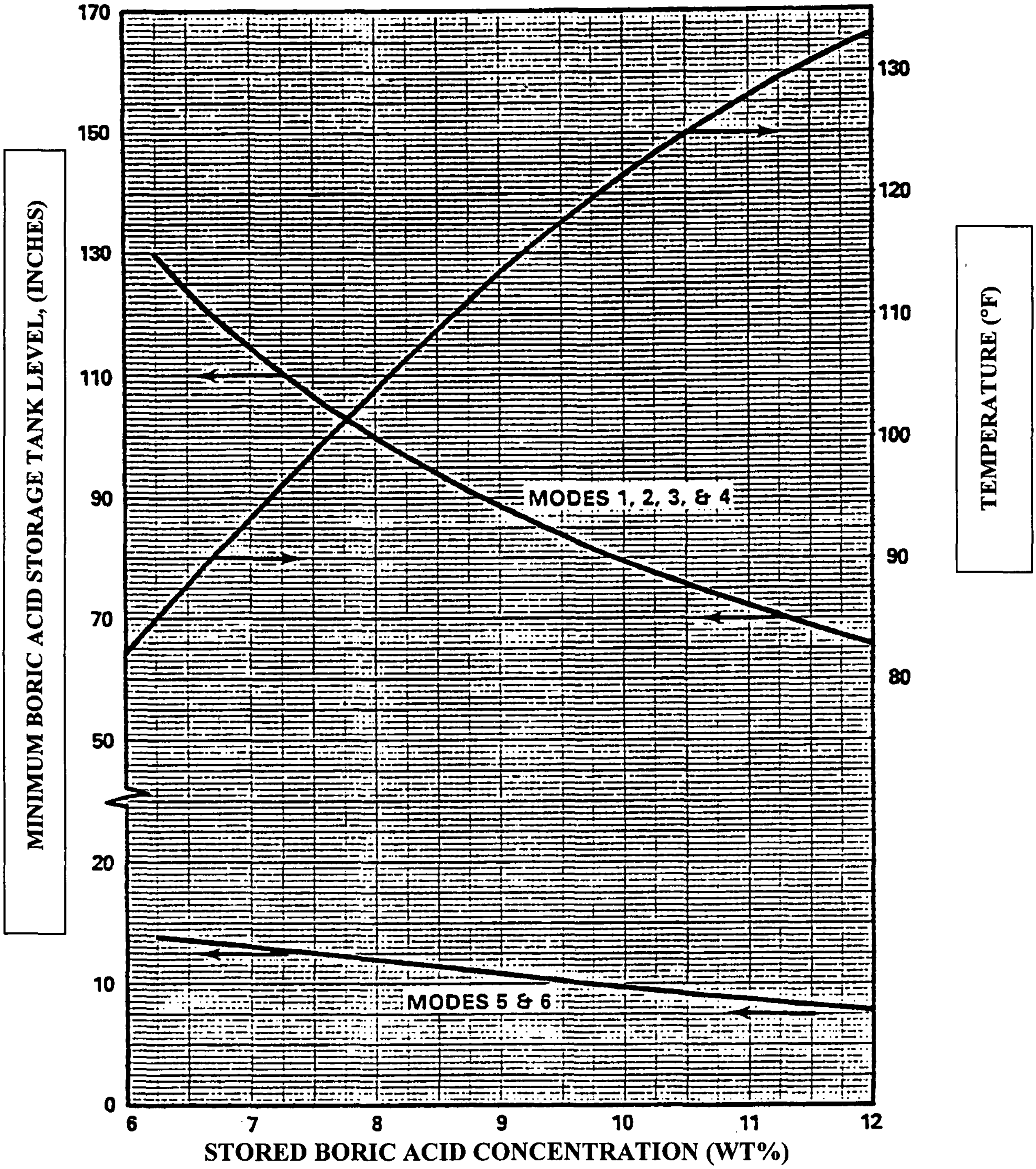

FIGURE 15.1.2-1
MINIMUM BAST VOLUME AND TEMPERATURE
AS A FUNCTION OF STORED BORIC ACID CONCENTRATION

15.1.3 BORATION FLOW PATHS - SHUTDOWN

NORMAL CONDITION

TNC 15.1.3 One boron injection flow path shall be operable.

The boration flow path must consist of a BAST connected to the RCS via a boric acid pump or gravity feed connection, and a charging pump; or an RWT connected to the RCS via a charging pump or high pressure safety injection pump. Each flow path is also required to contain associated heat tracing systems. The operable boric acid pump, charging pump, or high pressure safety injection pump shall be capable of being powered from an operable emergency bus.

APPLICABILITY Modes 5 and 6.

CONTINGENCY MEASURES

Nonconformance	Contingency Measures	Restoration Time
A. The required boron injection flow path is inoperable.	A.1 Suspend positive reactivity additions.	Immediately

VERIFICATION REQUIREMENTS

TVR	Verification	Frequency
15.1.3.1	Verify temperature of the heat traced portion of the flow path is above the temperature limit line shown on Figure 15.1.2-1.	7 days, if a flow path from the BAST is used
15.1.3.2	Verify RWT borated water temperature ≥ 35°F.	24 hours, if the RWT is used as a borated water source and the outside air temperature is < 35°F
15.1.3.3	Verify the BAST borated water volume is as specified in Figure 15.1.2-1.	7 days, if the BAST is the borated water source
15.1.3.4	Verify the RWT borated water volume is ≥ 9,844 gallons.	7 days, if the RWT is the borated water source
15.1.3.5	Verify the BAST borated water solution temperature is as specified in Figure 15.1.2-1.	7 days, if the BAST is the borated water source
15.1.3.6	Verify the RWT borated water solution temperature is ≥ 35°F.	7 days, if the RWT is the borated water source
15.1.3.7	Verify the BAST boron concentration is as specified in Figure 15.1.2-1.	7 days, if the BAST is the borated water source
15.1.3.8	Verify the RWT boron concentration is ≥ 2,300 ppm.	7 days, if in Mode 5 and the RWT is the borated water source

15.1.3 BORATION FLOW PATHS - SHUTDOWN - Continued

VERIFICATION REQUIREMENTS - Continued

TVR	Verification	Frequency
15.1.3.9	Verify the RWT boron concentration in Mode 6 exceeds the larger of 2300 ppm or the Refueling boron concentration limit specified in the COLR.	7 days, if in Mode 6 and the RWT is the borated water source
15.1.3.10	Verify that each manual, power-operated, or automatic valve in the flow path that is not locked, sealed, or otherwise secured in position, is in its correct position.	31 days

15.1.4 CONTROL ELEMENT ASSEMBLY (CEA) POSITION INDICATION

NORMAL CONDITION

TNC 15.1.4 Two CEA position indicator channels shall be operable for each shutdown and regulating CEA.

Any two of the following three CEA position indication channels are allowed to be operable to satisfy this TNC:

a. Control element assembly voltage divider reed switch position indicator channel;

b. Control element assembly "Full Out" or "Full In" reed switch position indicator channel as verified by actuation of the applicable position indicator;

c. Control element assembly pulse counting position indicator channel.

The only time the CEA "Full In" or "Full Out" reed switch position indicator channels can be considered operable for one of the three CEA Position Indicator Channels is when the CEAs are either fully withdrawn or fully inserted.

APPLICABILITY Modes 1 and 2.

CONTINGENCY MEASURES

Nonconformance	Contingency Measures	Restoration Time
A. One or more CEA(s) per group having its voltage divider reed switch position indicator channel inoperable and either the "Full out" OR "Full in" reed switch position indicator channel inoperable.	A.1 Restore the position indicator channel(s) to operable status.	6 hours
	OR	
	A.2.1 Reduce thermal power to ≤ 70% rated thermal power. If negative reactivity insertion is required to reduce thermal power, boration shall be used.	6 hours
	AND	
	after power is reduced ≤70% rated thermal power.	
A. continued	A.2.2.1 Fully withdraw CEA group(s) with inoperable position indicator(s) and verify the CEA(s) to be fully withdrawn via a "Full Out" indicator.	10 hours
	OR	

15.1.4 CONTROL ELEMENT ASSEMBLY (CEA) POSITION INDICATION - Continued

CONTINGENCY MEASURES - Continued

Nonconformance	Contingency Measures	Restoration Time
A. continued	A.2.2.2 Fully insert CEA group(s) with inoperable position indicator(s) and verify the CEA(s) to be fully inserted via a "Full In" indicator. <u>OR</u>	10 hours
	A.3 If the failure existed before entry into Mode 2 or occurs prior to an "all CEAs out" configuration, the CEA group(s) with an inoperable position indicator channel(s) must be moved to the "Full Out" position and verified to be fully withdrawn via a "Full Out" indicator.	Within 10 hours of entry into Mode 2 <u>AND</u> Prior to exceeding 70% of rated thermal power.
B. One or more CEA(s) per group having its CEA pulse counting position indicator channel inoperable and either the "Full Out" or "Full In" reed switch position indicator or the voltage divider reed switch position indicator channel inoperable.	B.1 Verify that either the CEA voltage divider reed switch position indicator channel or the “Full Out” or “Full In” reed switch position indicator channel for the affected CEAs is operable. <u>AND</u> B.2 Restore the position indicator channel(s) to operable status	1 hour 24 hours
C. Contingency measures and associated restoration times of Nonconformance A or B are not met.	C.1 See Section 15.0.3.	

15.1.4 CONTROL ELEMENT ASSEMBLY (CEA) POSITION INDICATION - Continued

VERIFICATION REQUIREMENTS

<table>
<tr><th>TVR</th><th>Verification</th><th>Frequency</th></tr>
<tr><td rowspan="2">15.1.4.1</td><td>Verify CEA position indicator channels agree within 4.5 inches.</td><td>Every 12 hours
<u>AND</u>
Every 4 hours when deviation circuit is inoperable</td></tr>
<tr><td colspan="2">This shall be accomplished by the following:
(1) Verifying the CEA pulse counting position indicator channels and the CEA voltage divider reed switch position indicator channels agree within 4.5 inches; <u>or</u>
(2) Verifying the CEA pulse counting position indicator channels and the CEA "Full Out" or "Full In" reed switch position indicator channels agree within 4.5 inches; <u>or</u>
(3) Verifying the CEA voltage divider reed switch position indicator channels and the CEA "Full Out" or "Full In" reed switch position indicator channels agree within 4.5 inches.</td></tr>
</table>

TABLE 15.1.4-1

CEA POSITION INDICATION APPLICABILITY

One or more rod(s) per group affected	Affected rod(s) partially inserted	Affected rod(s) fully withdrawn (or fully inserted)	Voltage divider position indicator channel	Pulse counting position indicator channel	"Full out"/ "Full in" position indicator	Applicable Non-conformance
X	X		I		I*	A
X	X		I	I	I*	C (See Section 15.0.3)
X	X			I	I*	B
X		X	I	I		B
X		X	I		I	A
X		X		I	I	B

I = Inoperable

I* = Inoperable due to rods partially inserted

15.2 NOT USED

15.3 INSTRUMENTATION

15.3.1 RADIATION MONITORING INSTRUMENTATION

NORMAL CONDITION

TNC 15.3.1 One Main Vent Wide Range Noble Gas Effluent Radiation Monitor Channel and two Main Steam Header Noble Gas Effluent Radiation Monitor Channels shall be operable with their alarm setpoints as specified in the setpoint control manual.

The measurement range of the main vent wide range noble gas effluent monitors is 10^{-7} to 10^{5} μCi/cc. The measurement range of the main steam header noble gas effluent is 10^{-2} to 10^{5} R/hr.

APPLICABILITY Modes 1, 2, 3, and 4.

CONTINGENCY MEASURES

Nonconformance	Contingency Measures	Restoration Time
A. A radiation monitoring channel alarm setpoint is not within limits.	A.1 Restore the setpoint to within limits. OR A.2 Declare the affected radiation monitoring channel inoperable.	4 hours 4 hours
B. Contingency measure and associated restoration time of Nonconformance A are not met. OR One or more required radiation monitoring channels are inoperable.	B.1 Initiate the preplanned alternate method of monitoring the appropriate parameter. AND B.2.1 Restore the inoperable channel to operable status. OR B.2.2 See Section 15.0.3.	72 hours 7 days

VERIFICATION REQUIREMENTS

TVR	Verification	Frequency
15.3.1.1	Perform a channel check.	12 hours
15.3.1.2	Perform a channel functional test.	31 days
15.3.1.3	Perform a channel calibration.	18 months

15.3.2 METEOROLOGICAL INSTRUMENTATION

NORMAL CONDITION

TNC 15.3.2 The following meteorological monitoring instrumentation channels shall be operable.

a. wind speed - 10 meter nominal elevation,

b. wind speed - 60 meter nominal elevation,

c. wind direction - 10 meter nominal elevation,

d. wind direction - 60 meter nominal elevation, and

e. air temperature - delta T (60 m - 10 m). **[B0653]**

APPLICABILITY At all times.

CONTINGENCY MEASURES

Nonconformance	Contingency Measures	Restoration Time
A. One or more required meteorological monitoring channels are inoperable.	A.1 Restore required channels to operable status.	7 days
B. Contingency measure and associated restoration time of Nonconformance A are not met.	B.1 See Section 15.0.3.	

VERIFICATION REQUIREMENTS

TVR	Verification	Frequency
15.3.2.1	Perform a channel check.	24 hours
15.3.2.2	Perform a channel calibration.	6 months

15.3.3 INCORE DETECTOR SYSTEM

The instruments covered by the verification requirements are discussed in the UFSAR Section 7.5.4.3.

NORMAL CONDITION

TNC 15.3.3 The Incore Detection System must be operable as follows:

A. For base functionality of the Incore Detection System,

1. At least one detector segment in each core quadrant at each of the four axial elevations must be operable.

B. For monitoring the azimuthal power tilt with the Incore Detection System,

1. At least two quadrant symmetric detector segments groups must be operable at each of the four axial elevations in the outer 184 fuel assemblies.

C. For recalibration of the Excore Neutron Flux Detection System,

1. At least 75% of all incore detector segments must be operable.

2. A minimum of nine incore detector segments must be operable at each of the four axial elevations.

3. A minimum of two detector segments in the inner 109 fuel assemblies must be operable at each of the four axial elevations.

4. A minimum of two detector segments in the outer 108 fuel assemblies must be operable at each of the four axial elevations.

D. For monitoring total planar radial peaking factor, total integrated radial peaking factor, or linear heat rate,

1. At least 75% of all incore detector string locations must be operable (i.e., at least three of four segments are considered operable).

2. A minimum of nine incore detector segments must be operable at each of the four axial elevations.

3. A minimum of two detector segments in the inner 109 fuel assemblies must be operable at each of the four axial elevations.

4. A minimum of two detector segments in the outer 108 fuel assemblies must be operable at each of the four axial elevations.

5. All 5 x 5 arrays of fuel assemblies that contain 25 fuel assemblies must contain at least one operable detector segment on any axial level.

15.3.3 INCORE DETECTOR SYSTEM - Continued

NORMAL CONDITIONS - Continued

NORMAL CONDITION

E. For post-refueling startup testing and power ascension,

1. Meet the requirements of (B) and (D.1 through D.4) above for azimuthal power tilt monitoring and for monitoring total planar radial peaking factor, total integrated radial peaking factor, or linear heat rate,

AND

2. either Criteria I, II, III, **OR** IV

a. Criterion I

1. All incore detector string locations must have at least one operable detector segment in any of the four axial elevations.

b. Criterion II

1. At least 75% of all incore detector string locations in a quadrant have at least one operable detector segment.

2. All 5x5 arrays of fuel assemblies that contain 25 fuel assemblies must contain at least one operable detector segment on any axial level.

c. Criterion III

1. Symmetry checks must be performed on all CEA groups

d. Criterion IV

1. Perform an evaluation of the ability of the incore detector system to detect core power symmetry with the actual operable incore detector pattern prior to exceeding 30% power.

2. Implement symmetry checks as identified in the evaluation.

3. Implement penalties on the total planar radial peaking factor, total integrated radial peaking factor, and linear heat rate as identified in the evaluation.

APPLICABILITY When the Incore Detection System is used as described above in items A through E.

15.3.3 INCORE DETECTOR SYSTEM - Continued

CONTINGENCY MEASURES

Nonconformance	Contingency Measures	Restoration Time
A. TNC 15.3.3.A, B, C, D.1 through D.4, or E not satisfied.	A.1 Stop using the Incore Detection System for that associated function.	Immediately on discovery
B. TNC 15.3.3.D.5 not satisfied	B.1 Stop using the Incore Detection System for monitoring total planar radial peaking factor, total integrated radial peaking factor, or linear heat rate. Perform an evaluation of the ability of the incore detector system to detect average power asymmetry of at least 10% between quadrant 4x4 groups of assemblies with the actual operable incore detector distribution, and if not, implement penalties as identified in the evaluation to allow use of the Incore Detection System for monitoring total planar radial peaking factor, total integrated radial peaking factor, and linear heat rate.	Within 14 EFPDs, either: Perform an evaluation and implement any required penalties, OR reduce power to less than or equal to 50%.

VERIFICATION REQUIREMENTS

TVR	Verification	Frequency
15.3.3.1	Perform a channel check.	Once within 24 hours prior to use <u>AND</u> 7 days thereafter
15.3.3.2	Perform a channel calibration. Neutron detectors are excluded from the channel calibration, but all electronic components are included.	24 months
15.3.3.3	Perform a calibration of the neutron detectors in the reactor core.	Prior to installation

15.3.4 SEISMIC MONITORING INSTRUMENTATION

NORMAL CONDITION — TNC 15.3.4 Each instrument listed in Table 15.3.4-1 must be operable.

APPLICABILITY At all times.

CONTINGENCY MEASURES

Nonconformance	Contingency Measures	Restoration Time
A. One or more instruments are inoperable	A.1 Restore the instrument(s) to operable status	30 days
B. Contingency measure and associated restoration time of Nonconformance A not met	B.1 See Section 15.0.3	
C. The system is activated during a seismic event	C.1 Restore to an operable status. AND	24 hours
	C.2 Retrieve and analyze data from the activated instruments to determine the magnitude of the vibratory ground motion.	Following the event

VERIFICATION REQUIREMENTS

TVR	Verification	Frequency
15.3.4.1	Perform channel check.	31 days
	This channel check includes verifying the seismic monitoring instruments are energized except for the seismic triggers for the Triaxial Time-History Strong Motion Accelographs.	
15.3.4.2	Perform channel functional test.	6 months
15.3.4.3	Perform channel calibration.	Within 5 days following a seismic event AND 24 months

TABLE 15.3.4-1

SEISMIC MONITORING INSTRUMENTATION

	INSTRUMENTS AND SENSOR LOCATIONS	MEASUREMENT RANGE
1.	Triaxial Time-History Strong Motion Accelographs	
a.	0-YE-001 Unit 1 Containment Base	0-1g
b.	0-YE-002 Unit 1 Containment 69'	0-1g
c.	0-YE-003 Auxiliary Building Base	0-1g
d.	0-YE-004 Intake Structure	0-1g
e.	0-YE-005 Free Field	0-1g
2.	Triaxial Seismic Switches	
a.	0-YS-001 Unit 1 Containment Base	NA
b.	0-YS-002 Unit 1 Containment 69'	NA
3.	Seismic Acceleration Recorder	
a.	0-YRC-001 Control Room	NA
b.	0-YR-001 Control Room	NA

15.3.5 FIRE DETECTION INSTRUMENTATION

NORMAL CONDITION — TNC 15.3.5 The Fire Detection Instrumentation for each fire detection zone shall be operable.

APPLICABILITY — Whenever equipment in the fire detection zone is required to be operable.

CONTINGENCY MEASURES

Nonconformance	Contingency Measures	Restoration Time
A. One or more fire detection instruments outside Containment are inoperable. AND Instrument(s) is located in a fire detection zone(s) not equipped with an automatic wet pipe sprinkler system alarmed and supervised to the Control Room.	A.1 Establish an hourly fire watch to inspect the zone(s) with the inoperable instrument(s). AND A.2 Restore the instrument to operable status.	Within 1 hour and once per hour thereafter 14 days
B. One or more fire detection instruments located inside Containment are inoperable.	B.1.1 Inspect in the Containment. OR B.1.2 Monitor containment air temperature at the containment dome and containment reactor cavity locations. AND B.2 Restore instrumentation to operable status.	Once per 8 hours Within 1 hour AND Once per 1 hour thereafter 14 days

15.3.5 FIRE DETECTION INSTRUMENTATION - Continued

CONTINGENCY MEASURES - Continued

Nonconformance	Contingency Measures	Restoration Time
C. One or more fire detection instruments outside Containment are inoperable. AND The instrument(s) is located in fire detection zone(s) equipped with an automatic wet pipe sprinkler system alarmed and supervised to the Control Room.	C.1.1 Establish an hourly fire watch. OR C.1.2.1 Inspect the zone(s) with inoperable instruments. AND C.1.2.2 Perform a channel functional test on the automatic sprinkler system, including the water flow alarm and supervisory system, in the zone with the inoperable instrument. AND C.2 Restore instrumentation to operable status.	Within 1 hour Within 1 hour AND Once per 24 hours thereafter Within 1 hour AND Once per 24 hours thereafter 14 days
D. Contingency measure and associated restoration time of Nonconformance A, B, or C are not met.	D.1 See Section 15.0.3.	
E. Fire detection instrument found inoperable during Technical Verification Requirement (TVR) 15.3.5.3.	E.1 For each detector found inoperable, perform TVR 15.3.5.3 on an additional 10% of all fire detection instruments accessible during plant operation. Note that TVR 15.3.5.3 is only required to be performed on either 10% of or 10 fire detection instruments accessible during plant operation, whichever is less	Within frequency associated with TVR 15.3.5.3 that identified inoperable instruments

15.3.5 FIRE DETECTION INSTRUMENTATION - Continued

VERIFICATION REQUIREMENTS

TVR	Verification	Frequency
15.3.5.1	Verify each non-supervised circuit associated with detection alarms between the instrument and Control Room is operable.	31 days
15.3.5.2	Verify each National Fire Protection Association Code 72D Class B supervised circuit's supervision associated with the detector alarms is operable.	6 months
15.3.5.3	Perform a channel functional test on the fire detection instruments accessible during plant operation.	6 months
	This is only required on 25% of the fire detection instruments selected on a rotating basis such that all fire detection instruments accessible during plant operation will be tested over a 24 month period. If in any detection zone there are less than four detectors, at least one different detector in that zone shall be tested every 6 months.	
15.3.5.4	Perform a channel functional test on the fire detection instruments inaccessible during plant operation. This is not required if performed in the previous 6 months.	Each Mode 5 entry exceeding 24 hours

TABLE 15.3.5-1

FIRE DETECTION INSTRUMENTS -- UNIT 1

ROOM/AREA AUX BLDG.	INSTRUMENT LOCATION	MINIMUM INSTRUMENTS OPERABLE HEAT	FLAME	SMOKE
100/103/ 104/116	Corridors - Elevation (-)10'0"			5
110	Coolant Waste Receiving & Monitoring Tank Pump Room			2
111	Waste Processing Control Room			1
112/114	Coolant Waste Receiving Tank		4	
113	Miscellaneous Waste Receiver Tank Room			1
115	Charging Pump Room			3
118/122	Emergency Core Cooling System (ECCS) Pump Room			7
119/122	ECCS Pump Room			7
200/202	Corridors, &			
209/210	Corridors &			
212/219	N/S Corridor & Personnel Elevator Access Vestibule			13
207/208	Waste Gas Equipment Room			3
216	Reactor Coolant Make-up Pumps			1
217	BAST & Pump Room			2
218	Volume Control Tank Room			1
220	Degasifier Pump Room			1
221/326	West Piping Penetration Room		2	3
222	Hot Instrument Shop			2
223	Hot Machine Shop			4
224	East Piping Area			10
225	Radiation Exhaust Vent Equipment Room			4
226	Service Water Pump Room		3	6
227/316	East Piping Penetration Room		3	5
228	Component Cooling Pump Room			8
301/304/300	Battery Room & Corridor			3
306/1C	Cable Spreading Room & Cable Chase[(a)]	2		10
308	N/S Corridor			6
315	Main Steam Piping Area			6
317	Switchgear Room, Elevation 27'-0"[(a)]			6
318	Purge Air Supply Room			2

TABLE 15.3.5-1 (Continued)

FIRE DETECTION INSTRUMENTS -- UNIT 1

ROOM/AREA AUX BLDG.	INSTRUMENT LOCATION	MINIMUM INSTRUMENTS OPERABLE HEAT	FLAME	SMOKE
319/325	West Passage and Vestibule			6
320	Spent Fuel Heat Exchanger Room			3
323	Passage 27' Valve Alley & Filter Room			3
324	Letdown Heat Exchanger Room			1
Elevation 27'-0"	Switchgear Vent Duct	1		
1A	Cable Chase 1A			1
1B	Cable Chase 1B			1
405	Control Room			6
410	N/S Corridor			4
417/418	Solid Waste Processing		2	3
413/419/420	Cask and Equip Loading Area &			
424/425/426	Cask and Equip Loading Area		3	22
421	Diesel Generator No. (1B)[a]	2		
423	West Electrical Penetration Room			3
428	East Piping Area			7
429	East Electrical Pen Room			3
430	Switchgear Room Elevation 45'-O"[a]			8
439	RWT Pump Room			2
441	Spent Resin Metering Tank Room			1
Elevation 45'-0"	Switchgear Vent Duct	1		
Elevation 69'-0"	Control Room Vent Duct "A"			1
Elevation 69'-0"	Cable Spreading Room Vent Duct			1
512	Control Room Heating, Ventilation, and Air Conditioning Equipment			4
586/588/589/590	Radiation Chemistry Area,			
592/593	Radiation Chemistry Area,			
595/596/597	Radiation Chemistry Area,			
587	Frisker Area,			
591	Clothing Disposal, and			
523/594	N/S Corridor & Dressout/Frisker Area			20
520	Spent Fuel Pool (SFP) Area Vent Equipment Room			2
524	Main Plant Exhaust Equipment Room			8
525	Containment Access Area			3

TABLE 15.3.5-1 (Continued)

FIRE DETECTION INSTRUMENTS -- UNIT 1

ROOM/AREA AUX BLDG.	INSTRUMENT LOCATION	MINIMUM INSTRUMENTS OPERABLE HEAT	FLAME	SMOKE
529	Electrical Equipment Room			3
530/531/533	SFP Area		5	17
536/537	Miscellaneous Waste Evaporator & Equipment Room			3
Elevation 83'-0"	Cable Tunnel			4
603	Auxiliary Feedwater Pump Room			2
Containment Bldg.				
U-1	Reactor Coolant Pump Bay East[b][d]	16		
U-1	Reactor Coolant Pump Bay West[b][d]	16		
U-1	East Electric Penetration Area[b]	[c]		
U-1	West Electric Penetration Area[b]	[c]		
Intake Structure	Elevation 3'-0" Unit 1 Side **[B0898]**			1
1A DG Bldg.				
Zone 1[a]	Diesel Generator Room, Oil Separator Room, 1A Diesel Generator Building Trench, Fan Room, Maintenance Shop, and Hallway	33		
Zone 2[a]	Battery Room, Non-IE Electric Panel Room, Control Room, 1-E Switchgear Room, Future Expansion Room	1		11
Zone 3[a]	Fuel Oil Storage Tank Room	8		
Zone 4	General Area, Third Room	17		
Zone 5	Heating, Ventilation, and Air Conditioning Duct, Second and Third Floor			2

(a) Detectors that automatically actuate Fire Suppression Systems.

(b) Detection Instruments located within the Containment are not required to be operable during the performance of Type A Containment Leakage Rate Tests.

(c) Monitored by four protecto wires.

(d) RCP bay heat detectors are not required to be operable in Modes 5 and 6 if **all** of the following conditions are met: **[B0889]**

1. The RCP motors in the associated bay are de-energized, and
2. The RCS temperature is 200°F or less

TABLE 15.3.5-1 (Continued)

FIRE DETECTION INSTRUMENTS -- UNIT 2

ROOM/AREA AUX BLDG.	INSTRUMENT LOCATION	MINIMUM INSTRUMENTS OPERABLE[b] HEAT	FLAME	SMOKE
101/120	ECCS Pump Room			7
102/120	ECCS Pump Room			7
105	Charging Pump Room			3
106	Miscellaneous Waste Monitor Tank			1
107/109	Coolant Waste Monitor Tank		4	
108	Pump Room-Elevation (-)10'-0"			1
201	Component Cooling Pump Room			9
203	East Piping Area			10
204	Radiation Exhaust Vent, Equipment Room			4
205	Service Water Pump Room		3	6
206/310	East Piping Penetration Room		3	5
211/321	West Piping Penetration Room		2	3
213	Degasfier Pump Room			1
214	Volume Control Tank Room			1
215	BAST & Pump Room			2
216A	Reactor Coolant Make-up Pumps			2
302/2C	Unit 2 Cable Spreading Room & Cable Chase[a]	2		10
305/307/303	Unit 2 Battery Room & Corridor			3
309	Main Steam Piping Area			6
311	Switchgear Room, Elevation 27'-0"			6
312	Purge Air Supply Room			2
322	Letdown Heat Exchanger Room			1
Elev. 27'-0"	Switchgear Vent Duct	1		
2A	Cable Chase 2A			1
2B	Cable Chase 2B			1
407	Switchgear Room, Elev. 45'-0"[a]			8
408	East Piping Area			7
409	East Electrical Penetration Room			3
414	West Electrical Penetration Room			3
416	Diesel Generator No. (2B)[a]	2		
422	Diesel Generator No. (2A)[a]	2		
440	RWT Pump Room			2
Elev. 45'-0"	Switchgear Vent Duct	1		

TABLE 15.3.5-1 (Continued)

FIRE DETECTION INSTRUMENTS -- UNIT 2

ROOM/AREA		MINIMUM INSTRUMENTS OPERABLE[b]		
AUX BLDG.	INSTRUMENT LOCATION	HEAT	FLAME	SMOKE
526	Main Plant Exhaust Equipment Room			8
527	Containment Access			3
532	Electrical Equipment Room			3
Elev. 69'-0"	Cable Spreading Room Vent Duct			1
Elev. 83'-0"	Cable Tunnel			4
605	Auxiliary Feedwater Pump Room			2
Containment Bldg.				
Unit 2	Reactor Coolant Pump Bay East[b][d]	16		
Unit 2	Reactor Coolant Pump Bay West[b][d]	16		
Unit 2	East Electric Penetration Area[b]	[c]		
Unit 2	West Electric Penetration Area[b]	[c]		
Intake Structure	Elevation 3'-0" Unit 2 Side **[B0898]**			1

(a) Detectors that automatically actuate Fire Suppression Systems.

(b) Detection instruments located within the Containment are not required to be operable during the performance of Type A Containment Leakage Rate Tests.

(c) Monitored by four protecto wires.

(d) RCP bay heat detectors are not required to be operable in Modes 5 and 6 if **all** of the following conditions are met: **[B0889]**

1. The RCP motors in the associated bay are de-energized, and
2. The RCS temperature is 200°F or less

15.3.6 FEEDWATER FLOW MEASUREMENT INSTRUMENTATION

NORMAL CONDITION	TNC 15.3.6	The LEFM CheckPlus system shall be operable, with the Plant Computer available to perform the secondary calorimetric calculation.

APPLICABILITY Mode 1 > 2700 MWt.

CONTINGENCY MEASURES

-- NOTE --
15.0.3 is not applicable to the LEFM CheckPlus system to raise Reactor Power >2700 MWt within the Restoration Time of the Contingency Measures.

Nonconformance		Contingency Measures		Restoration Time
A.	The LEFM CheckPlus system is inoperable.	A.1.1	Maintain Reactor Power within 10% Rated Thermal Power (RTP) of the initial power level when the system was declared inoperable.	Immediately
			OR	
		A.1.2	Ensure Reactor Power is ≤ 2700 MWt (98.6% RTP).	Immediately
		AND		
		A.2.1	Shift to the compensated venturi feedwater flow measurement input to the secondary calorimetric.	1 hour
			OR	
		A.2.2	Ensure Reactor Power is ≤ 2700 MWt (98.6% RTP).	1 hour
		AND		
		A.3.1	Restore the LEFM CheckPlus system to operable status.	72 hours
			OR	
		A.3.2	Ensure Reactor Power is ≤ 2700 MWt (98.6% RTP).	72 hours

15.3.6 FEEDWATER FLOW MEASUREMENT INSTRUMENTATION - Continued

CONTINGENCY MEASURES - Continued

Nonconformance	Contingency Measures	Restoration Time
B. The Plant Computer is unavailable, OR An input from other than the LEFM CheckPlus system to the secondary calorimetric calculation has failed, AND resulted in bad quality of the 16-minute average calculation.	B.1 Restore the Plant Computer and all inputs to the secondary calorimetric calculation. OR B.2 Ensure Reactor Power is ≤ 2700 MWt (98.6% RTP).	Prior to the next required performance of Tech. Spec. SR 3.3.1.2 OR 24 hours, whichever is less Prior to the next required performance of Tech. Spec. SR 3.3.1.2 OR 24 hours, whichever is less
C. Contingency measure and associated restoration time of Nonconformance A or B are not met.	C.1 See Section 15.0.3.	

VERIFICATION REQUIREMENTS

TVR	Verification	Frequency
15.3.6.1	Perform applicable maintenance checks contained in VTD 15799-004-1002: • General examination of the LEFM cabinet • Power supply inspection • Central processing unit inspection • Acoustic processing unit checks • Analog input checks • Ethernet communication test • Transducer checks • Spool piece metering section checks	24 months
	Analog output checks, relay output checks, watchdog timer checks and the serial communication test contained in VTD 15799-004-1002 are not applicable to Calvert Cliffs.	

15.4 REACTOR COOLANT SYSTEM

15.4.1 CHEMISTRY

NORMAL CONDITION TNC 15.4.1 The RCS chemistry shall be maintained within limits. **[B2369]**

Parameter	Action Level 2	Action Level 3
Dissolved Oxygen	>0.100 ppm	> 1 ppm
Chloride	> 0.15 ppm	> 1.50 ppm
Fluoride	> 0.15 ppm	> 1.50 ppm
Sulfate	> 0.15 ppm	> 1.50 ppm
Hydrogen	< 15 cc/kg	< 5 cc/kg

H_2 limits only apply when RX is critical.

The dissolved oxygen limits are not applicable with $T_{avg} \leq 250°F$.

APPLICABILITY At all times.

CONTINGENCY MEASURES

Nonconformance	Contingency Measures	Restoration Time
A. One or more chemistry parameter values in excess of Action Level 3 with $T_{avg} \geq 250°F$.	A.1 Initiate unit shutdown and cooldown to $T_{avg} < 250°F$.	Immediately
	If a shutdown is initiated due to RCS parameter(s) exceeding Action Level 3, and RCS chemistry improves to within Action Level 3, the shutdown and cooldown per this Contingency Measure may be terminated.	
B. One or more chemistry parameter values in excess of Action Level 2.	B.1 Restore the parameter(s) to below Action Level 2 limits.	24 hours
C. The contingency measure and associated restoration time of Nonconformance B are not met.	C.1 Initiate unit shutdown and cooldown to $T_{avg} < 250°F$.	Immediately
	If a shutdown is initiated due to RCS parameter(s) exceeding Action Level 2, and RCS chemistry improves to within Action Level 2, the shutdown and cooldown per this Contingency Measure may be terminated.	

15.4.1 CHEMISTRY - Continued

CONTINGENCY MEASURES - Continued

Nonconformance	Contingency Measures	Restoration Time
D. The contingency measure and associated restoration time of Nonconformance A or C are not met. OR One or more chemistry parameter values exceed Action Level 3 with T_{avg} <250°F.	D.1 See Section 15.0.3.	

VERIFICATION REQUIREMENTS

TVR	Verification	Frequency
15.4.1.1	--- **NOTE** TVR is not required to be performed when all of the following requirements are met: **[B0772]** • RCS chemistry sampling is not possible due to low RCS level. • The applicable Unit (1 or 2) is defueled. • RCS temperature is less than 145°F. • RCS chemistry is verified to be within TRM 15.4.1 limits prior to reducing RCS level. --- Verify RCS chemistry to be within the TNC limits.	72 hours OR 8 hours prior to exceeding 250^{0}F in the RCS

15.4.2 PRESSURIZER PRESSURE/TEMPERATURE LIMITS

NORMAL CONDITION — TNC 15.4.2 The pressurizer temperature shall be limited to:

a. A maximum heatup of 100°F in any one hour period,

b. A maximum cooldown of 200°F in any one hour period, and

c. A maximum spray water temperature differential of 400°F.

APPLICABILITY At all times.

CONTINGENCY MEASURES

Nonconformance	Contingency Measures	Restoration Time
A. Pressurizer temperature in excess of any of the above limits.	A.1 Restore temperature to within limits. AND A.2 Perform an engineering evaluation to determine the effects of the out-of-limit condition on the fracture toughness properties of the pressurizer and to determine that the pressurizer is acceptable for continued operation.	30 minutes None
B. Contingency measure and associated restoration time of Nonconformance A are not met.	B.1 See Section 15.0.3.	

VERIFICATION REQUIREMENTS

TVR	Verification	Frequency
15.4.2.1	Verify the pressurizer temperature is within the limits during system heatup or cooldown.	At least once per 30 minutes
15.4.2.2	Verify the spray water temperature differential is within the limits during auxiliary spray operation.	At least once per 12 hours

15.4.3 AMERICAN SOCIETY OF MECHANICAL ENGINEERS (ASME) CODE CLASS 1, 2, AND 3 COMPONENTS

NORMAL CONDITION — TNC 15.4.3 The structural integrity of ASME Code Class 1, 2, and 3 components shall be within the limits of the Inservice Inspection Program.

APPLICABILITY — Modes 1, 2, 3, 4, 5, and 6.

CONTINGENCY MEASURES

Nonconformance	Contingency Measures	Restoration Time
A. Structural integrity of ASME Class 1 component(s) is not within the limit.	A.1 Restore structural integrity of the affected component(s) to within the limit.	Prior to increasing the RCS temperature more than 50°F above the minimum temperature required by nil-ductility temperature considerations
	OR	
	A.2 Isolate the affected component(s).	Prior to increasing the RCS temperature more than 50°F above the minimum temperature required by nil-ductility temperature considerations
B. Structural integrity of ASME Class 2 component(s) is not within the limit.	B.1 Restore the structural integrity of the affected component(s) to within the limit.	Prior to increasing the RCS temperature above 200°F
	OR	
	B.2 Isolate the affected component(s).	Prior to increasing the RCS temperature above 200°F
C. Contingency Measures and associated restoration times of Nonconformances A or B not met.	C.1 See Section 15.0.3.	
D. Structural integrity of any ASME Class 3 component(s) is not within the limit.	D.1 Restore the structural integrity of the affected component(s) to within the limit.	None
	OR	

15.4.3 AMERICAN SOCIETY OF MECHANICAL ENGINEERS (ASME) CODE CLASS 1, 2, AND 3 COMPONENTS - Continued

CONTINGENCY MEASURES - Continued

Nonconformance	Contingency Measures	Restoration Time
	D.2 Isolate the affected component(s) from service.	None

VERIFICATION REQUIREMENTS

TVR	Verification	Frequency
15.4.3.1	Verify structural integrity of ASME Class 1, 2, and 3 components are within limits specified in the Inservice Inspection Program.	In accordance with the Inservice Inspection Program
15.4.3.2	Verify structural integrity of main steam and main feedwater piping is within limits specified in the augmented Inservice Inspection Program.	In accordance with the augmented Inservice Inspection Program.
	<u>Augmented Inservice Inspection Program for Main Steam and Main Feedwater Piping</u> The unencapsulated welds greater than four inches in nominal diameter in the main steam and main feedwater piping runs located outside the Containment and traversing safety-related areas or located in compartments adjoining safety-related areas shall be inspected per the following augmented inservice inspection program using the applicable rules, acceptance criteria and repair procedures of the ASME Boiler and Pressure Vessel Code, Section XI, Endorsed in the Inservice Inspection Program, for Class 2 components.	
	Each weld must be examined in accordance with the above ASME Code requirements, except that 100% of the welds must be examined, cumulatively, during each ten year inspection interval. The welds to be examined during each inspection period shall be selected to provide a representative sample of the conditions of the welds. If these examinations reveal unacceptable structural defects in one or more welds, an additional 1/3 of the welds shall be examined and the inspection schedule for the repaired welds shall revert back as if a new interval had begun. If additional unacceptable defects are detected in the second sampling, the remainder of the welds shall also be inspected. Alternatively, a Risk-Informed process for piping outlined in EPRI Topical Report 1006937 revision 0-A may be used for the weld selections and the determination of required additional examinations when defects are discovered.	

15.4.4 DELETED

15.4.5 LETDOWN LINE EXCESS FLOW

NORMAL CONDITION	TNC 15.4.5	The bypass valve for the excess flow check valve in the letdown line shall be closed.

APPLICABILITY Modes 1, 2, 3, and 4.

CONTINGENCY MEASURES

Nonconformance	Contingency Measures	Restoration Time
A. Bypass valve open.	A.1 Close bypass valve.	4 hours
B. Contingency measure and associated restoration time of Nonconformance A are not met.	B.1 See Section 15.0.3.	

VERIFICATION REQUIREMENTS

TVR	Verification	Frequency
15.4.5.1	Verify bypass valve for the excess flow check valve in the letdown line is closed.	4 hours prior to entry into Mode 4 from Mode 5

15.4.6 RCS VENTS

NORMAL CONDITION

TNC 15.4.6 Two RCS vent paths shall be operable.

The two RCS vent paths consist of two closed solenoid valves in series at the reactor vessel head and the pressurizer vapor space.

APPLICABILITY Modes 1 and 2.

CONTINGENCY MEASURES

Nonconformance	Contingency Measures	Restoration Time
A. Reactor vessel head vent path is inoperable.	A.1 Maintain the inoperable vent path closed with power removed from the actuator of the solenoid valves. AND	Immediately
	A.2 Restore the Inoperable reactor vessel head vent path to operable status.	30 days
B. Pressurizer vapor space vent path is inoperable.	B.1 Maintain the inoperable vent path closed with power removed from the actuator of the solenoid valves. AND	Immediately
	B.2.1 Verify one power-operated relief valve and its associated flow path is operable. AND	72 hours
	B.2.2 Restore the inoperable pressurizer vapor space vent path to operable status. OR	Prior to entering Mode 2 following the next Mode 3 of sufficient duration
	B.3 Restore inoperable pressurizer vapor space vent path to operable status.	30 days
C. Reactor vessel head vent path is inoperable. AND Pressurizer vapor space vent path is inoperable.	C.1 Restore both inoperable vent paths to operable status.	72 hours
D. Contingency measure and associated restoration time of Nonconformances A, B, or C are not met.	D.1 See Section 15.0.3.	

15.4.6 RCS VENTS - Continued

VERIFICATION REQUIREMENTS

TVR	Verification	Frequency
15.4.6.1	Verify manual isolation valves in each vent path are locked in the open position.	24 months
15.4.6.2	Verify flow through the RCS vent path with the vent valves open.	24 months

15.5 NOT USED

15.6 CONTAINMENT SYSTEMS

15.6.1 CONTAINMENT STRUCTURAL INTEGRITY

NORMAL CONDITION — TNC 15.6.1 The structural integrity of the Containment shall be maintained at a level consistent with the acceptance criteria of the ASME Boiler and Pressure Vessel Code Section XI, Subsection IWL, Requirements for Class CC Concrete Components of Light-Water Cooled Plants, 1992 Edition with 1992 Addenda, as modified and amended by 10 CFR 50.55a. **[B0703]**

APPLICABILITY Modes 1, 2, 3 and 4.

CONTINGENCY MEASURES

Nonconformance	Contingency Measures	Restoration Time
A. The Acceptance Standards of IWL-3000 are not met.	A.1 Perform actions as required by IWL-3310.	Immediately
B. Contingency measure and associated restoration time of Nonconformance A are not met <u>OR</u> Containment structure exhibits possible evidence of abnormal degradation.	B.1 See Section 15.0.3.	
C. Containment Structural Integrity not conforming to acceptance criteria of TVR 15.6.1.2.	C.1 Restore structural integrity. <u>OR</u> C.2 Complete an engineering evaluation that assures structural integrity.	Prior to increasing RCS temperature above 200°F.

VERIFICATION REQUIREMENTS

TVR	Verification	Frequency
15.6.1.1	Perform the examinations required by IWL-2000.	In accordance with the Concrete Containment Tendon Surveillance Program.
15.6.1.2	Conduct an inspection of the pre-selected concrete crack patterns adjacent to containment tendon end anchorage's during the Type A Containment Leakage Rate Tests with Containment at maximum test pressure.	In accordance with the Containment Leakage Rate Testing Program.

15.6.2 CONTAINMENT CLOSEOUT

NORMAL CONDITION — TNC 15.6.2 The Containment shall remain free of loose debris (rags, trash, clothing, etc.) which could be transported to the containment sump and cause restriction of the pump suctions during LOCA conditions.

APPLICABILITY Modes 1, 2, 3, and 4.

CONTINGENCY MEASURES

Nonconformance	Contingency Measures	Restoration Time
A. NONE	A.1 NONE	NONE

VERIFICATION REQUIREMENTS

TVR	Verification	Frequency
15.6.2.1	A visual inspection shall be performed for all accessible areas of the Containment.	Prior to establishing Containment integrity.
15.6.2.2	A visual inspection shall be performed of the areas affected within Containment.	Upon completion of the containment entry when Containment integrity has been established.

15.7 PLANT SYSTEMS

15.7.1 STEAM GENERATOR PRESSURE/TEMPERATURE LIMITATION

NORMAL CONDITION	TNC 15.7.1	The temperatures of both the primary and secondary coolant in the steam generators shall be > 80°F when the pressure of either the primary or secondary coolant in the steam generator is > 200 psig.
APPLICABILITY	At all times.	

CONTINGENCY MEASURES

Nonconformance	Contingency Measures	Restoration Time
A. Steam generator pressure/ temperature not within limits.	A.1 Reduce the steam generator pressure on the applicable side to ≤ 200 psig.	30 minutes
	AND	
	A.2 Perform an engineering evaluation. The engineering evaluation shall determine that the steam generator remains acceptable for continued operation.	Prior to increasing steam generator temperatures above 200°F

VERIFICATION REQUIREMENTS

TVR	Verification	Frequency
15.7.1.1	Verify each steam generator primary and secondary coolant pressure is < 200 psig. This is only required to be performed when steam generator primary or secondary coolant temperature is < 80°F.	1 hour, if the following conditions apply: **[B0800]** Both manways are installed on both the primary and secondary side of the steam generator, OR Both manways are installed on the primary side (hot leg/cold leg) of the steam generator AND either one or both manways are removed from the secondary side. In this case, the surveillance need only be performed on the primary side, OR Both manways are installed on the secondary side of the steam generator AND either one or both manways are removed from the primary side (hot leg/cold leg). In this case, the surveillance need only be performed on the secondary side.

15.7.2 SNUBBERS

NORMAL CONDITION — TNC 15.7.2 The safety-related snubbers shall be operable.

Safety-related snubbers include those snubbers installed on safety-related systems and snubbers on non-safety related systems if their failure or the failure of the system on which they are installed would have an adverse effect on any safety-related system.

APPLICABILITY — Modes 1, 2, 3, 4, 5, and 6.

In Modes 5 and 6 the only snubbers required to be operable are those snubbers located on systems required to be operable.

CONTINGENCY MEASURES

Nonconformance	Contingency Measures	Restoration Time
A. One or more snubbers inoperable.	A.1 See Technical Specification LCO 3.0.8.	

VERIFICATION REQUIREMENTS

TVR	Verification	Frequency
15.7.2.1	Verify snubbers are operable per the snubber inspection program. As used here, type of snubber shall mean snubbers of the same design and manufacturer, irrespective of capacity.	In accordance with the snubber inspection program

15.7.2 SNUBBERS - Continued

VERIFICATION REQUIREMENTS - Continued

TVR	Verification	Frequency
	Snubber Inspection Program	

a. *Visual Inspections*

Visual inspections shall be performed in accordance with the schedule determined by Table 15.7.2.1. Snubbers are categorized as inaccessible or accessible during reactor operation. Each of these categories (inaccessible and accessible) may be inspected independently or jointly according to the schedule determined by Table 15.7.2.1. The visual inspection interval for each population or category of snubbers shall be determined based upon the criteria provided in Table 15.7.2.1. **[B0627]**

b. *Visual Inspection Acceptance Criteria*

Visual inspections shall verify (1) that there are no visible indications of damage or impaired operability, and (2) that the snubber installation exhibits no visual indications of detachment from foundations or supporting structures. **[B0627]** Snubbers that appear inoperable as a result of visual inspections may be determined operable for the purpose of establishing the next visual inspection interval, provided that: (1) the cause of the rejection is clearly established, remedied and functionally tested for that particular snubber and for other snubbers that may be generically susceptible; or (2) the affected snubber is functionally tested in the as found condition and determined operable per the Hydraulic Snubbers Functional Test Acceptance Criteria, as applicable. When the fluid port of a hydraulic snubber is found to be uncovered, the snubber shall be determined inoperable unless it can be determined operable via functional testing for the purpose of establishing the next visual inspection interval.

For the snubber(s) found inoperable, an engineering evaluation shall be performed on the component(s) that are supported by the snubber(s). The scope of this engineering evaluation shall be consistent with the licensee's engineering judgment and may be limited to a visual inspection of the supported component(s). The purpose of this engineering evaluation shall be to determine if the component(s) supported by the snubber(s) were adversely affected by the inoperability of the snubber(s) in order to ensure that the supported component remains capable of meeting the designed service.

c. *Functional Tests*

At least once per 24 months, a representative sample of 10% of each type of snubbers in use in the plant shall be functionally tested either in-place or in a bench test. For each snubber that does not meet the functional test acceptance criteria of the Hydraulic Snubbers Functional Test Acceptance Criteria, an additional 5% of that type snubber shall be functionally tested until no more failures are found or until all snubbers of that type have been functionally tested.

Snubber Inspection Program – Continued

Snubber Inspection Program - Continued

c. *Functional Tests*

Snubbers identified as "Especially Difficult to Remove" or in "High Exposure Zones" shall also be included in the representative sample (permanent or other exemptions from functional testing for individual snubbers in these categories may be granted by the NRC only if a justifiable basis for exemption is presented and/or snubber life destructive testing was performed to qualify snubber operability for all design conditions at either the completion of their fabrication or at a subsequent date).

In addition to the regular sample, snubbers that failed the previous functional test shall be retested during the next test period. If a spare snubber has been installed in place of a failed snubber, then both the failed snubber (if it is repaired and installed in another position) and the spare snubber shall be retested during the next test period. Failure of these snubbers shall not entail functional testing of additional snubbers.

If any snubber selected for functional testing either fails to lock up or fails to move, i.e., frozen in place, the cause will be evaluated and if caused by manufacturer or design deficiency all generically susceptible snubbers of the same design subject to the same defect shall be functionally tested. This testing requirement shall be independent of the requirements stated above for snubbers not meeting the functional test acceptance criteria.

For the snubber(s) found inoperable, an engineering evaluation shall be performed on the component(s) that are supported by the snubber(s). The scope of this engineering evaluation shall be consistent with the licensee's engineering judgment and may be limited to a visual inspection of the supported component(s). The purpose of this engineering evaluation shall be to determine if the component(s) supported by the snubber(s) were adversely affected by the inoperability of the snubber(s) in order to ensure that the supported component remains capable of meeting the designed service.

d. *Hydraulic Snubbers Functional Test Acceptance Criteria*

The hydraulic snubber functional test shall verify that:

1. Activation (restraining action) is achieved within the specified range of velocity or acceleration in both tension and compression.
2. Snubber bleed, or release rate, where required, is within the specified range in compression or tension. For snubbers specifically required to not displace under continuous load, the ability of the snubber to withstand load without displacement shall be verified.

Snubber Inspection Program – Continued

	Snubber Inspection Program - Continued e. *Snubber Service Life Monitoring* A record of the service life of each snubber, the date at which the designated service life commences and the installation and maintenance records on which the designated service life is based shall be maintained. At least once per 24 months, the installation and maintenance records for each safety-related snubber shall be reviewed to verify that the indicated service life has not been exceeded or will not be exceeded prior to the next scheduled snubber service life review (including the 1.25 times extension). If the indicated service life will be exceeded prior to the next scheduled snubber service life review, the snubber service life shall be reevaluated or the snubber shall be replaced or reconditioned so as to extend its service life beyond the date of the next scheduled service life review. This reevaluation, replacement or reconditioning shall be indicated in the records.

TABLE 15.7.2-1

SNUBBER VISUAL INSPECTION INTERVAL

Population or Category (Notes 1 and 2)	NUMBER OF INOPERABLE SNUBBERS Column A Extend Interval (Notes 3 and 6)	Column B Repeat Interval (Notes 4 and 6)	Column C Reduce Interval (Notes 5 and 6)
1	0	0	1
80	0	0	2
100	0	1	4
150	0	3	8
200	2	5	13
300	5	12	25
400	8	18	36
500	12	24	48
750	20	40	78
1000 or greater	29	56	109

Note 1: The next visual inspection interval for a snubber population or category size shall be determined based upon the previous inspection interval and the number of inoperable snubbers found during that interval. Snubbers may be categorized, based upon their accessibility during power operation, as accessible or inaccessible. These categories may be examined separately or jointly. However, the licensee must make and document that decision before any inspection and shall use that decision as the basis upon which to determine the next inspection interval for that category.

Note 2: Interpolation between population or category sizes and the number of inoperable snubbers is permissible. Use next lower integer for the value of the limit for Columns A, B, or C if that integer includes a fractional value of inoperable snubbers as determined by interpolation.

Note 3: If the number of inoperable snubbers is equal to or less than the number in Column A, the next inspection interval may be twice the previous interval but not greater than 48 months.

Note 4: If the number of inoperable snubbers is equal to or less than the number in Column B but greater than the number in Column A, the next inspection interval shall be the same as the previous interval.

TABLE 15.7.2-1 (Continued)

SNUBBER VISUAL INSPECTION INTERVAL

Note 5: If the number of inoperable snubbers is equal to or greater than the number in Column C, the next inspection interval shall be two-thirds of the previous interval. However, if the number of inoperable snubbers is less than the number in Column C but greater than the number in Column B, the next interval shall be reduced proportionally by interpolation, that is, the previous interval shall be reduced by a factor that is one-third of the ratio of the difference between the number of inoperable snubbers found during the previous interval and the number in Column B to the difference in the numbers in Columns B and C.

Note 6: An extension of 1.25 times the inspection interval is applicable for all inspection intervals up to and including 48 months.

15.7.3 SEALED SOURCE CONTAMINATION

NORMAL CONDITION — TNC 15.7.3 — Each sealed source containing radioactive material either in excess of 100 microcuries of beta and/or gamma emitting material or 5 microcuries of alpha emitting material shall be free of ≥ 0.005 microcuries of removable contamination.

APPLICABILITY — At all times.

CONTINGENCY MEASURES

Nonconformance	Contingency Measures	Restoration Time
A. Sealed source removable contamination is not within the limit.	A.1 Withdraw the sealed source from use.	Immediately
	Whenever a sealed source is withdrawn from use, the sealed source must either be decontaminated and repaired, or disposed of in accordance with the regulations.	

VERIFICATION REQUIREMENTS

Startup sources and fission detectors previously subjected to core flux do not require verification tests.

TVR	Verification	Frequency
15.7.3.1	Verify leakage and/or contamination levels for sealed sources in use. This is only required to be performed on sealed source containing radioactive material with a half-life greater than 30 days (excluding Hydrogen-3), and in any form other than gas.	6 months
	The leakage and contamination test may be performed by licensee, or other persons specifically authorized by the NRC or an agreement state. The test method shall have a detection sensitivity of at least 0.005 microcuries per test sample.	

15.7.3 SEALED SOURCE CONTAMINATION - Continued

VERIFICATION REQUIREMENTS - Continued

TVR	Verification	Frequency
15.7.3.2	Verify leakage and/or contamination levels for sealed sources not in use.	Prior to use or transfer to another licensee. This is only required to be performed if not tested in the previous 6 months.
	Sealed sources transferred without a certificate indicating the last test date shall be tested prior to being placed into use.	
15.7.3.3	Verify leakage and/or contamination levels for startup sources and fission detectors.	Within 31 days prior to being subjected to core flux or installed in the core <u>AND</u> Following repair or maintenance to the source or detector.
15.7.3.4	Perform inventory of Sealed sources. **[B0650]**	Annual

15.7.4 DELETED

15.7.5 FIRE SUPPRESSION WATER SYSTEM

NORMAL CONDITION

TNC 15.7.5 The Fire Suppression Water System shall be operable.

The Fire Suppression Water System shall consist of:

a. Two high pressure pumps, each with a capacity of 2500 gpm, with their discharge aligned to the fire suppression header,

b. Two water supplies, each with a minimum contained volume of 300,000 gallons, and

c. An operable flow path capable of taking suction from the Pretreated Water Storage Tank Nos. 11 and 12 and transferring the water through distribution piping with operable sectionalizing control or isolation valves to the yard hydrant curb valves required to be in operation in Section 15.7.9 and the first valve ahead of the water flow alarm device on each sprinkler, hose standpipe, or spray system riser required to be operable per TNCs 15.7.6, 15.7.8, and 15.7.9.

Opening the cross-connect valves to the owner-controlled loop of the fire fighting system does not affect the operability of the plant fire suppression system. However, the owner-controlled loop of the fire fighting system cannot be used to make the plant fire suppression system operable.

Shutting fire suppression water system isolation valve 0-FP-137(PIV) for the purpose of performing surveillance testing (e.g., TVR 15.7.5.10) does not make the fire suppression water system inoperable provided:

- An operator is dedicated to immediately open the valve and return it to its normal position (e.g., locked open) and is in direct communication with the Control Room.

- An inspection of the affected sprinkler area(s) has been conducted to verify no ignition source activities are in progress (e.g., welding, cutting, grinding, stress relieving of pipe welds, heat treatment of metals, and open flames).

APPLICABILITY At all times.

15.7.5 FIRE SUPPRESSION WATER SYSTEM – Continued

CONTINGENCY MEASURES

Nonconformance	Contingency Measures	Restoration Time
A. One pump inoperable. <u>OR</u> One water supply inoperable.	A.1 Restore pump or water supply to operable status.	7 days
B. Fire Suppression Water System inoperable for reasons other than Nonconformance A.	B.1 Establish a backup Fire Suppression Water System. AND See Section 15.0.3.	24 hours
C. Contingency measure and associated restoration time of Nonconformance A is not met.	C.1 See Section 15.0.3.	

VERIFICATION REQUIREMENTS

TVR	Verification	Frequency
15.7.5.1	Verify that each required contained water supply volume contains ≥ 300,000 gallons.	7 days
15.7.5.2	Verify diesel fire pump starting battery bank electrolyte level is above the plates.	7 days
15.7.5.3	Verify diesel fire pump starting battery bank overall voltage is ≥ 24 Volt.	7 days
15.7.5.4	Verify electric motor driven fire pump operates for ≥ 15 minutes. <u>AND</u> Verify diesel fire pump starts from ambient condition and operates for ≥ 30 minutes.	31 days on a staggered test basis
15.7.5.5	Verify each manual, power-operated, and automatic valve in the flow path that is not locked, sealed, or otherwise secured in position is in the correct position.	31 days

15.7.5 FIRE SUPPRESSION WATER SYSTEM – Continued

VERIFICATION REQUIREMENTS - Continued

TVR	Verification	Frequency
15.7.5.6	Verify the diesel fire pump fuel oil day storage tank contains ≥ 174 gallons.	31 days
15.7.5.7	Verify diesel fire pump fuel oil storage tank sample is within limits.	92 days
	The sample is required to be obtained in accordance with American Society for Testing and Materials D270-65 and results of sampling is required to be within the limits specified in Table 1 of the American Society for Testing and Materials D975-81 when checked for viscosity, water, and sediment.	
15.7.5.8	Verify diesel fire pump starting battery bank specific gravity is appropriate for continued service of the battery.	92 days
15.7.5.9	Perform a system flush of the filled portions of fire suppression water system.	12 months
15.7.5.10	Verify each testable valve in the flow path can be cycled, by cycling it through at least one complete cycle of full travel.	12 months
15.7.5.11	Perform a system functional test on the fire suppression water system.	18 months
	The verification will consist of: a. simulating automatic actuation of the system throughout its operating sequence, b. verifying each automatic valve in the flow path actuates to its correct position, c. verifying each pump develops at least 2500 gpm at a discharge pressure of 125 psig, and d. verifying each high pressure pump starts (sequentially) to maintain the fire suppression water system pressure ≥ 80 psig.	
15.7.5.12	Verify the diesel fire pump starts from ambient conditions on the auto-start signal and operates for ≥ 20 minutes, while loaded with the fire pump.	18 months

15.7.5 FIRE SUPPRESSION WATER SYSTEM - Continued

VERIFICATION REQUIREMENTS - Continued

TVR	Verification	Frequency
15.7.5.13	Perform an inspection of the fire pump diesel.	18 months
	This inspection is required to be performed in accordance with procedures prepared in conjunction with its manufacturer's recommendations for the class of service.	
15.7.5.14	Verify fire pump diesel starting batteries, cell plates, and battery racks show no visual indication of physical damage or abnormal deterioration.	18 months
15.7.5.15	Verify fire pump diesel starting battery-to-battery and terminal connections are clean, tight, free of corrosion, and coated with anti-corrosion material.	18 months
15.7.5.16	Perform a system flow test on the fire suppression water system.	24 months
	The test must be performed in accordance with the Fire Protection Handbook. [Fire Protection Handbook, 14th Edition, Section 11, Chapter 5 (published by The National Fire Protection Association)]	
15.7.5.17	Perform a system functional test on the fire suppression water system.	24 months
	The verification will consist of: a. simulating automatic actuation of the system throughout its operating sequence, and b. cycling each valve in the flow path that is not testable during plant operation through at least one complete cycle of full travel.	

15.7.6 SPRAY AND SPRINKLER SYSTEM

NORMAL CONDITION

TNC 15.7.6 The fire suppression spray and/or sprinkler system shall be operable.

The spray and/or sprinkler systems identified in Tables 15.7.6-1 and 15.7.6-2 are required to be operable.

Shutting spray and sprinkler system isolation valve(s) for the purpose of performing surveillance testing (e.g., TVR 15.7.6.2, 15.7.5.10) does not make the spray and sprinkler system inoperable provided:

- An operator is dedicated to immediately open the valve and return it to its normal position (e.g., locked open) and is in direct communication with the Control Room.
- An inspection of the affected sprinkler area has been conducted to verify no ignition source activities are in progress (e.g., welding, cutting, grinding, stress relieving of pipe welds, heat treatment of metals, and open flames).

APPLICABILITY Whenever equipment in the spray/sprinkler protected areas are required to be operable.

CONTINGENCY MEASURES

Nonconformance	Contingency Measures	Restoration Time
A. One or more spray or sprinkler systems inoperable in areas where redundant safe shutdown systems or components could be damaged.	A.1 Establish a continuous fire watch with backup fire suppression equipment. AND A.2 Restore system(s) to operable status.	1 hour 14 days
B. One or more spray or sprinkler systems inoperable in areas other than in Nonconformance A.	B.1 Establish an hourly fire watch patrol. AND B.2 Restore system(s) to operable status.	1 hour 14 days
C. Contingency measures A.2 and/or B.2 and associated restoration time cannot be met.	C.1 See Section 15.0.3.	

15.7.6 SPRAY AND SPRINKLER SYSTEM - Continued

VERIFICATION REQUIREMENTS

<table>
<tr><th>TVR</th><th>Verification</th><th>Frequency</th></tr>
<tr><td>15.7.6.1</td><td>Verify each manual, power-operated, and automatic valve in the flow path that is not locked, sealed, or otherwise secured in position is in its correct position.</td><td>31 days</td></tr>
<tr><td>15.7.6.2</td><td>Cycle each valve in the flow path through at least one complete cycle of full travel.</td><td>12 months</td></tr>
<tr><td rowspan="2">15.7.6.3</td><td>Perform a system functional test on the fire suppression spray and sprinkler system.</td><td>18 months</td></tr>
<tr><td colspan="2">The verification will include:
a. simulating automatic actuation of the system, and
b. verifying that the automatic valves in the flow path actuate to their correct positions on a simulated test signal.</td></tr>
<tr><td>15.7.6.4</td><td>Verify by visual inspection of the area in the vicinity of each nozzle that the nozzle spray pattern is unobstructed.</td><td>18 months</td></tr>
</table>

TABLE 15.7.6-1

SPRINKLER LOCATIONS -- UNIT 1

SPRINKLER LOCATION	CONTROL VALVE ELEVATION
1B Diesel Generator	45'-0"
Unit 1 East Pipe Penetration Room 227/316[(a)]	5'-0"
Unit 1 Auxiliary Feedwater Pump Room 603[(a)]	12'-0"
Unit 1 East Piping Area Room 428[(a)]	45'-0"
Unit 1 East Electrical Penetration Room 429[(a)]	45'-0"
Unit 1 West Electrical Penetration Room 423[(a)]	45'-0"
Unit 1 Main Steam Piping Room 315[(a)]	45'-0"
Unit 1 Component Cooling Pump Room 228[(a)]	5'-0"
Unit 1 East Piping Area 224[(a)]	5'-0"
Unit 1 Radiation Exhaust Vent Equipment Room 225[(a)]	5'-0"
Unit 1 Service Water Pump Room 226[(a)]	5'-0"
Unit 1 BAST and Pump Room 217[(a)]	5'-0"
Unit 1 Reactor Coolant Makeup Pump Room 216[(a)]	5'-0"
Unit 1 Charging Pump Room 115[(a)]	(-)10'-0"
Unit 1 Miscellaneous Waste Monitoring Room 113[(a)]	(-)10'-0"
Cask and Equipment Loading Area Rooms 419, 420, 425, and 426[(a)]	45'-0"
Solid Waste Processing[(a)]	45'-0"
Corridors 200, 202, 212, and 219[(a)]	5'-0"
Corridors 100, 103, and 116[(a)]	(-)10'-0"
Cable Chase 1A[(a)]	45'-0"
Cable Chase 1B[a(a)]	45'-0"
Unit 1 ECCS Pump Room 119[(a)]	(-)15'-0"
Hot Instrument Shop Room 222[(a)]	5'-0"
Hot Machine Shop Room 223[(a)]	5'-0"
1A Diesel Generator Building - Preaction Systems 1, 2, and 3	45'-6"

(a) Sprinklers required to ensure the operability of redundant safe shutdown equipment.

TABLE 15.7.6-2

SPRINKLER LOCATIONS -- UNIT 2

SPRINKLER LOCATION	CONTROL VALVE ELEVATION
Unit 2 Auxiliary Feedwater Pump Room 605[a]	12'-0"
Unit 2 East Piping Area Room 408[a]	45'-0"
Unit 2 East Electrical Penetration Room 409[a]	45'-0"
Unit 2 West Electrical Penetration Room 414[a]	45'-0"
Cable Chase 2A[a]	45'-0"
Cable Chase 2B[a]	45'-0"
Unit 2 Main Steam Piping Room 309[a]	45'-0"
Unit 2 Component Cooling Pump Room 201[a]	5'-0"
Unit 2 East Piping Area 203[a]	5'-0"
Unit 2 Radiation Exhaust Vent Equipment Room 204[a]	5'-0"
Unit 2 Service Water Pump Room 205[a]	5'-0"
Unit 2 BAST and Pump Room 215[a]	5'-0"
Unit 2 Reactor Coolant Makeup Pump Room 216A[a]	5'-0"
Unit 2 Charging Pump Room 105[a]	(-)10'-0"
Unit 2 Miscellaneous Waste Monitoring Room 106[a]	(-)10'-0"
Unit 2 ECCS Pump Room 101[a]	(-)15'-0"
2A Diesel Generator	45'-0"
2B Diesel Generator	45'-0"
Unit 2 East Pipe Penetration Room 206/310[a]	5'-0"

[a] Sprinklers required to ensure the operability of redundant safe shutdown equipment.

15.7.7 HALON SYSTEM

NORMAL CONDITION

TNC 15.7.7 The Halon Systems located in the following locations, shall be operable:

a. Cable spreading room total flood system, and associated vertical cable chase 1C, and

b. 4160 volt switchgear room 27 foot and 45 foot elevation.

The Halon System storage tanks must have at least 95% of full charge weight (or level) and 90% of full charge pressure.

APPLICABILITY Whenever equipment protected by the Halon System is required to be operable.

CONTINGENCY MEASURES

Nonconformance	Contingency Measures	Restoration Time
A. Both primary and backup Halon Systems protecting the area are inoperable.	A.1 Establish an hourly fire watch with backup fire suppression equipment. AND A.2 Restore system to operable status.	1 hour 14 days
B. Contingency measure A.2 and associated restoration time is not met.	B.1 See Section 15.0.3.	

VERIFICATION REQUIREMENTS

TVR	Verification	Frequency
15.7.7.1	Verify each manual, power-operated, and automatic valve in the flow path is in its correct position.	31 days
15.7.7.2	Verify halon storage tank weight (level) is ≥ 95% of full charge weight.	6 months
15.7.7.3	Verify halon storage tank pressure is ≥ 90% of full charge pressure.	6 months
15.7.7.4	Verify by visual inspection the nozzle(s) and visible flow paths are clear of obstructions.	12 months

15.7.7 HALON SYSTEM - Continued

VERIFICATION REQUIREMENTS - Continued

TVR	Verification	Frequency
15.7.7.5	Verify the system actuates manually and automatically upon receipt of a simulated actuation signal.	18 months
	This includes associated ventilation dampers and fire door release mechanisms.	
15.7.7.6	Perform a system flow test on the Halon System.	Within 72 hours following completion of major maintenance or modifications
	The test will require flow through the headers and nozzles to assure no blockage.	

15.7.8 FIRE HOSE STATIONS

NORMAL CONDITION

TNC 15.7.8 The required fire hose stations identified in Tables 15.7.8-1 and 15.7.8-2 shall be operable.

Shutting fire hose station containment isolation valves for the purpose of performing an ILRT does not make the fire hose station inoperable.

Shutting fire hose system isolation valve(s), listed per Table 15.7.8-3, for the purpose of performing surveillance testing (e.g., TVR 15.7.5.10) does not make the fire hose station inoperable provided an operator is dedicated to immediately open the valve and return it to its normal position (e.g., locked open) and is in direct communication with the Control Room.

APPLICABILITY

Whenever equipment in the areas protected by the fire hose stations is required to be operable.

CONTINGENCY MEASURES

Nonconformance	Contingency Measures	Restoration Time
A. One or more required fire hose stations inoperable.	A.1 Route a fire hose from any operable plant fire hose station or fire hydrant to the unprotected area(s) with the inoperable fire hose station. AND	1 hour
	A.2 Restore fire hose stations to operable status.	14 days
B. Contingency measure A.2 and associated restoration time are not met.	B.1 See Section 15.0.3.	

VERIFICATION REQUIREMENTS

TVR	Verification	Frequency
15.7.8.1	Verify by visual inspection that the required equipment is at each fire hose station located outside Containment.	31 days
15.7.8.2	Verify by visual inspection that the required equipment is at each fire hose station located inside Containment.	During each scheduled reactor shutdown, but not required more frequently than every 31 days
15.7.8.3	Deleted.	
15.7.8.4	Deleted.	

15.7.8 FIRE HOSE STATIONS - Continued

VERIFICATION REQUIREMENTS - Continued

TVR	Verification	Frequency
15.7.8.5	Verify valve operability and no flow blockage for hose station valves inside Containment by partially opening each hose station valve.	24 months
15.7.8.6	Deleted.	
15.7.8.7	Verify valve operability and no flow blockage for hose station valves outside Containment by partially opening each hose station valve.	36 months
15.7.8.8	Deleted.	

TABLE 15.7.8-1

FIRE HOSE STATIONS -- UNIT 1

	LOCATION	ELEVATION	NUMBER OF HOSE STATIONS
1.	Containment	10'	2
		45'	2
		69'	2
2.	Auxiliary Building	-15'[(a)]	1[(b)]
		-10'[(a)]	2[(b)]
		5'	6
		27'	3
		45'	5
		69'[(a)]	4
3.	Turbine Building, Heater Bay Outside Service Water Pump Rooms and Auxiliary Feedwater Pump Rooms	12'	3
	Outside Switchgear Room	27'	2
	Outside Switchgear Room	45'	3
4.	Intake Structure	10'[(a)]	1
5.	Diesel Generator Building	35'	1
		45'	1
		66'	1
		80'	1

(a) Fire Hose Stations required for primary protection to ensure the operability of safety-related equipment.

(b) Hose Stations that serve both Units 1 and 2.

TABLE 15.7.8-2

FIRE HOSE STATIONS -- UNIT 2

	LOCATION	ELEVATION	NUMBER OF HOSE STATIONS
1.	Containment	10'	2
		45'	2
		69'	2
2.	Auxiliary building	-15'[a]	1[b]
		-10'[a]	2[b]
		5'	3
		27'	2
		45'	4
		69'[a]	3
3.	Turbine Building, Heater Bay Outside Service Water Pump Rooms and Auxiliary Feedwater Pump Rooms	12'	2
	Outside Switchgear Room	27'	1
	Outside Switchgear Room	45'	2
4.	Intake Structure	10'[a]	1

[a] Fire Hose Stations required for primary protection to ensure the operability of safety-related equipment.

[b] Hose Stations that serve both Units 1 and 2.

TABLE 15.7.8-3

FIRE HOSE STATION ISOLATION VALVES -- UNIT 1 AND UNIT 2

AUX BUILDING

VALVE	HOSE STATION
0-FP-419	69-6
0-FP-420	45-28
0-FP-421	45-29
0-FP-422	45-30
0-FP-423	45-21
0-FP-450	27-15
0-FP-496	5-20
0-FP-497	5-21
0-FP-521	-10-2
0-FP-653	69-6 & 69-9

1A DG BUILDING

VALVE	HOSE STATION
0-FP-835	DG-1A-1
	DG-1A-2
	DG-1A-3
	DG-1A-4

15.7.9 YARD FIRE HYDRANTS AND HYDRANT HOSE HOUSES

NORMAL CONDITION

TNC 15.7.9 The following yard fire hydrants and associated hydrant hose houses shall be operable.

a. No. 6 yard hydrant and associated hydrant hose house, which provides primary protection for Unit 2 RWT blockhouse, and

b. No. 7 yard hydrant and associated hydrant hose house, which provides primary protection for Unit 1 RWT blockhouse.

APPLICABILITY

Whenever equipment in the areas protected by the yard fire hydrants is required to be operable.

CONTINGENCY MEASURES

Nonconformance	Contingency Measures	Restoration Time
A. One or more yard fire hydrants or associated hydrant hose houses inoperable.	A.1 Have sufficient additional lengths of 2.5 inch diameter hose located in an adjacent operable hydrant hose house to provide service to the unprotected area(s) if the inoperable fire hydrant or associated hydrant hose house is the primary means of fire suppression.	1 hour
	AND	
	A.2 Restore yard fire hydrant(s) or hydrant hose house(s) to operable status.	14 days
B. Contingency Measure A.2 and associated restoration time are not met.	B.1 See Section 15.0.3.	

VERIFICATION REQUIREMENTS

TVR	Verification	Frequency
15.7.9.1	Verify by visual inspection of the hydrant hose house that the required equipment is at hydrant hose house.	31 days
15.7.9.2	Verify that the hydrant barrel is dry and that the hydrant is not damaged.	6 months
	The verification is required to be performed every 6 months (once during March, April, or May; and once during September, October, or November) by visually inspecting each yard fire hydrant.	

15.7.9 YARD FIRE HYDRANTS AND HYDRANT HOSE HOUSES - Continued

VERIFICATION REQUIREMENTS - Continued

TVR	Verification	Frequency
15.7.9.3	Perform a hose hydrostatic test at a pressure at least 50 psig above the maximum pressure available at any yard fire hydrant.	12 months
15.7.9.4	Perform an inspection of gaskets.	12 months
	This requires the replacement of any degraded gaskets in the couplings.	
15.7.9.5	Perform a flow check of each hydrant.	12 months

15.7.10 FIRE BARRIER PENETRATIONS

<u>NORMAL CONDITION</u>	<u>TNC 15.7.10</u>	Fire barrier penetrations (i.e., cable penetration barriers, fire doors, and fire dampers), in the fire zone boundaries, protecting safe shutdown areas, shall be operable.
<u>APPLICABILITY</u>	At all times.	

CONTINGENCY MEASURES

Nonconformance	Contingency Measures	Restoration Time
A. One or more fire barrier penetrations inoperable.	A.1 Establish a continuous fire watch on at least one side of the affected penetration.	1 hour
	<u>OR</u>	
	A.1.2.1 Verify operability of fire detectors on at least one side of the inoperable fire barrier.	1 hour
	<u>AND</u>	
	A.1.2.2 Establish an hourly fire watch patrol.	1 hour
	<u>OR</u>	
	A.1.3 Verify operability of automatic sprinkler systems on both sides of the inoperable fire barrier (including the water flow alarm and supervisory system).	1 hour
	<u>AND</u>	
	A.2 Restore penetration fire barrier(s) to operable status.	7 days
B. Contingency measure A.2 and associated restoration time are not met.	B.1 See Section 15.0.3.	

15.7.10 FIRE BARRIER PENETRATIONS - Continued

VERIFICATION REQUIREMENTS

TVR	Verification	Frequency
15.7.10.1	Perform a visual inspection of fire barrier penetrations.	18 months AND Prior to returning a fire barrier penetration to functional status following repairs or maintenance

15.8 NOT USED

15.9 REFUELING OPERATIONS

15.9.1 DECAY TIME

NORMAL CONDITION — TNC 15.9.1 Reactor shall be subcritical for at least 100 hours.

APPLICABILITY — During movement of irradiated fuel in the reactor vessel.

CONTINGENCY MEASURES

Nonconformance	Contingency Measures	Restoration Time
A. Reactor subcritical < 100 hours.	A.1 Suspend operations involving movement of irradiated fuel in the reactor vessel.	Immediately

VERIFICATION REQUIREMENTS

TVR	Verification	Frequency
15.9.1.1	Verify reactor has been subcritical for at least 100 hours.	Prior to movement of irradiated fuel in the reactor vessel
	This verification shall consist of verifying the date and time of the subcriticality prior to movement of irradiated fuel in the reactor vessel.	

15.9.2 COMMUNICATIONS

NORMAL CONDITION	TNC 15.9.2	Direct communications shall be maintained between the Control Room and personnel at the refueling station.

APPLICABILITY During the movement of irradiated fuel in Containment

CONTINGENCY MEASURES

Nonconformance	Contingency Measures	Restoration Time
A. Direct communications between the Control Room and personnel at the refueling station cannot be maintained.	A.1 Suspend the movement of irradiated fuel in Containment.	Immediately

VERIFICATION REQUIREMENTS

TVR	Verification	Frequency
15.9.2.1	Demonstrate direct communications between the Control Room and personnel at the refueling station.	Within 1 hour prior to movement of irradiated fuel in Containment AND Once per 12 hours thereafter

15.9.3 REFUELING MACHINE

NORMAL CONDITION

TNC 15.9.3 The refueling machine shall be operable.

a. The main hoist shall be used for the movement of fuel assemblies and shall be operable with:
   1. A minimum capacity of at least 1610 pounds with the refueling pool dry and at least 1437 pounds with the refueling pool flooded.
   2. An overload cutoff limits of ≤ 3500 pounds.

b. Auxiliary hoists shall be used for the movement of CEAs that are being removed from or inserted into fuel assemblies in the core and shall be operable with:
   1. A minimum capacity of 1000 pounds, and
   2. A load indicator that shall be used to prevent lifting loads in excess of 1000 pounds.

APPLICABILITY During movement of CEAs or fuel assemblies within the reactor pressure vessel.

CONTINGENCY MEASURES

Nonconformance	Contingency Measures	Restoration Time
A. Refueling machine main hoist inoperable.	A.1 Suspend movement of fuel assemblies within the reactor pressure vessel.	Immediately
B. All refueling machine auxiliary hoists inoperable.	B.1 Suspend movement of CEAs within the reactor pressure vessel.	Immediately

VERIFICATION REQUIREMENTS

TVR	Verification	Frequency
15.9.3.1	Perform a load test on the refueling machine main hoist.	Within 72 hours prior to the initial start of movement of fuel assemblies within the reactor pressure vessel for a refueling operation which consists of either a fuel offload and onload OR a fuel shuffle.
	A load test of at least 1610 pounds with the refueling pool dry or at least 1437 pounds with the refueling pool flooded will be performed. The load cutoff must also be tested by demonstrating an automatic load cut off when the crane load exceeds ≤ 3500 pounds.	
15.9.3.2	Perform a load test on all the refueling machine auxiliary hoists and associated load indicators to be used.	Within 72 hours prior to the initial start of movement of CEAs within the reactor pressure vessel for a refueling operation which consists of either a fuel offload and onload OR a fuel shuffle.
	The load test requires a load of at least 1000 pounds.	

15.9.4 CRANE TRAVEL - SPENT FUEL POOL

NORMAL CONDITION — TNC 15.9.4 Loads in excess of 1,600 pounds shall be prohibited from travel over fuel assemblies in the SFP.

This TNC does not apply to loads handled by the single-failure-proof Spent Fuel Cask Handling Crane.

APPLICABILITY — Whenever fuel assemblies are stored in the SFP.

CONTINGENCY MEASURES

Nonconformance	Contingency Measures	Restoration Time
A. Loads in excess of 1,600 pounds over the fuel assemblies in the SFP.	A.1 Place the crane load in a safe condition.	Immediately

VERIFICATION REQUIREMENTS

TVR	Verification	Frequency
15.9.4.1	Verify the weight of each load over fuel assemblies in the SFP is ≤ 1,600 pounds. This is not required to be performed when moving fuel assemblies and CEAs over fuel assemblies in the SFP. This is not required to be performed if the single-failure-proof Spent Fuel Cask Handling Crane is used.	Prior to movement over the fuel assemblies in the SFP
15.9.4.2	Visually inspect slings and special lifting devices and verify they are operable. This is only required to be performed during Spent Fuel Cask Handling Crane operation over the SFP.	7 days prior to use of the Spent Fuel Cask Handling Crane AND every 7 days thereafter
15.9.4.3	In addition to the requirements of TVR 15.9.4.2, pre-operational and periodic tests and preventive maintenance shall be performed.	Per plant procedures

15.10 NOT USED

15.11 RADIOACTIVE EFFLUENTS

15.11.1 EXPLOSIVE GAS MIXTURE

NORMAL CONDITION — TNC 15.11.1 The concentration of oxygen in the Waste Gas Holdup System shall be limited to less than or equal to 4% by volume.

APPLICABILITY At all times.

CONTINGENCY MEASURES

Nonconformance	Contingency Measures	Restoration Time
A. The concentration of oxygen in a waste gas decay tank is greater than 4% by volume.	A.1 Suspend all additions of waste gases to the waste gas decay tank. AND	Immediately
	A.2 Reduce the concentration of oxygen in the waste gas decay tank holdup system to within limits.	Immediately

VERIFICATION REQUIREMENTS

TVR	Verification	Frequency
15.11.1.1	Collect and analyze a sample from the waste gas decay tanks to determine if the concentration of oxygen is ≤ 4% by volume.	Following waste gas decay tank isolation AND 7 days
	This frequency is only required for the inservice tank.	
15.11.1.2	Collect and analyze a sample from the waste gas surge tank.	7 days AND 24 hours during power escalation from Mode 6 through Mode 3

15.11.2 GAS STORAGE TANKS

NORMAL CONDITION — TNC 15.11.2 — The quantity of radioactivity contained in each gas storage tank shall be limited to less than or equal to 58,500 curies noble gasses (considered as Xe-133).

APPLICABILITY — At all times.

CONTINGENCY MEASURES

Nonconformance	Contingency Measures	Restoration Time
A. The quantity of radioactive material in a gas storage tank greater than 58,500 curies noble gas.	A.1 Suspend all additions of radioactive material to affected tank.	Immediately
	AND	
	A.2 Reduce the quantity of radioactive material in the affected tank to within the limits.	48 hours

VERIFICATION REQUIREMENTS

TVR	Verification	Frequency
15.11.2.1	Determine quantity of radioactive material in the in-service gas storage tank is within limits. This is not required if the RCS specific activity of Xe-133 is less than or equal to 150 μCi/ml.	24 hours

EAL 99-01 Rev 5 Upgrade

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Type	Name	Project Class	Modified	Modified By	Approval
	BGE Fuel Degradation EALs Calculation Worksheet JSB Associates February 18 1993	A,B,C	1/4/2011 1:50 PM	Jones, Michele	
	Letter dated March 6 1997 from Charles H Cruise to USNRC Revision to EALs Technical Basis Document	A,B,C	1/4/2011 1:53 PM	Jones, Michele	
	Letter LB Russell (BGE) to James H Joyner EAL Review Meeting June 6 1991	A,B,C	1/4/2011 1:53 PM	Jones, Michele	
	Radioactivity Release Emergency Action Levels September 1990 JSB Associates	A,B,C	1/4/2011 1:53 PM	Jones, Michele	

ATTACHMENT (3)

EAL COMPARISON MATRIX

Calvert Cliffs Nuclear Power Plant, LLC
February 1, 2011

BALTIMORE GAS AND ELECTRIC

MEMORANDUM

Chemistry Section

February 24, 1993

TO: E.H. Roach

FROM: R.L. Conatser

SUBJECT: FUEL DEGRADATION EAL

The attached document calculates the fuel degradation EAL corresponding to 5% failed fuel. Page 6 of this report contains the dose rates at one foot for the shielded sample bomb.

Please note that these values have changed from the values we discussed previously, because some of the assumptions in the calculation have changed.

If you have any questions, please contact me by calling extension 2086.

Richard L. Conatser

R.L. Conatser
Chemist

Attachments: Calculation Worksheet For BG&E Fuel Degradation EAL

RLC:rlc

c: J.M. Bills
Chemistry File: RLC194

EPU Received 2/26/93
Sent

FOLLOW UP
FILE NO. 4.14 EAL

Project: BG&E Fuel Degradation EAL

Purpose: Determine the RCS sample dose rate to use as an Emergency Action Level representative of "significant fuel degradation."

Inputs: ALERT EALs from NUREG-0654, Rev. 1:
1.b. Very high coolant activity sample (e.g., 300 μCi/cc equivalent of I-131)
1.c. Failed fuel monitor (PWR) indicates increase greater than 1% fuel failures within 30 minutes or 5% total fuel failures.

Fuel Clad Barrier Example EAL 2. Primary Coolant Activity Level, from Basic Information for Table 4 of NUMARC/NESP-007 indicates that 300 μCi/cc I-131 equivalent corresponds to about 5% to 10% fuel clad damage. "This amount of clad damage indicates ... the Fuel Clad Barrier is considered lost."

Sample container geometry from BG&E Calc. 200-DA-9201, page 16: 8 in. long, of 3/8 in. iron tube, containing 12.5 ml of coolant, shielded by 1/2 in. Pb cylindrical shield.

DCFs for:	I-131	I-132	I-133	I-134	I-135
are:	1.48E6	5.35E4	4E5	2.5E4	1.24E5

in Rem per Curie inhaled. (From TID 14844)

Assumptions:

1. The source term is based on a 24 month fuel cycle as contained in BG&E calculation 200-DA-9201 for the core inventory.

2. The gas gap inventory as a fraction of the core inventory is taken from CE Core Damage Assessment guidance and ANSI/ANS-5.4-1982, both of which provide similar values.

3. The concentrations used assume a gas gap release diluted into the RCS volume, without additional dilution by safety injection. Concentrations and therefore dose rates will be reduced by additional dilution. RCS Vol. = 2.7 E08 ml

4. Assume 1 hour elapses between fuel damage and RCS sampling.

Approach:

A. Compute the source term resulting from a 5% gap release into the RCS volume, both pressurized and degassed.

B. Compute the dose rate per ml of sample at 30 cm (1 ft.) for the source terms developed in A. above. Assume both the normal sample container and the Pb shielded sample container.

C. Present results in simple to use format.

Analyses:

A. Source Term

The source term applicable for determining fuel clad failure is a gap release. That is, the fission products which have migrated from the fuel matrix into the fuel/clad gap will be released first, and these should be used as the appropriate source term.

A fuel gap source term does not appear to be available from BG&E. NUREG-1465, Accident Source Terms for Light-Water Nuclear Power Plants, Draft Report for Comment, states the assumption that 5% of the core inventory of iodine fission products are available in the gap. This assumption may be conservative as a source term for calculation of off-site dose impacts of potential accidents, but it is non -conservative for the calculation of estimated fuel damage as a function of RCS sample dose rate. The bases for the previous statement follow.

Shorter-lived I-132, I-134, and I-135 have much higher gamma emissions energies and intensities than the longer-lived I-131 and I-133. Therefore they contribute significantly more to sample dose rate per unit concentration, but significantly less to off-site dose rate, since off-site dose conversion factors are based on the inhaled thyroid dose which is also a function of half-life. The fuel/clad gap concentration of radioiodines is also a function of half-life. This is clearly shown in the equations in ANSI/ANS-5.4-1982, Method for Calculating the Fractional Release of Volatile Fission Products from Oxide Fuel. Logically, the time required for a fission product to migrate from the fuel matrix to the gap will cause longer-lived fission products to exist in the gap in higher concentrations than shorter-lived isotopes of the same element.

It follows that assuming a constant 5% of core inventory in the gap for all iodine fission products significantly over-estimates the shorter lived I-132, I-134, and I-135 compared to the longer-lived I-131 and I-133. This would cause one to calculate a much higher sample dose rate for a given level of fuel clad damage than should actually be expected. Subsequently, using measured dose rates and comparing them to the higher calculated dose rate to estimate the amount of fuel clad damage could lead to under-estimating the fuel damage. (Note that this is but one of several indicators of fuel clad damage. It is not likely that the best assessment of the amount of damage will be based on an RCS sample dose rate measurement.)

Estimates of the fraction of core inventory available in the gap may be obtained from the CE Owners Group Core Damage Assessment guidance. Also, a distribution of the iodine isotopes in the fuel/clad gap may be obtained using Equation 5 of ANSI/ANS-5.4-1982 to calculate the fractional release F. Note that the fractional release F computed using the equations in ANSI/ANS-5.4-1982 are functions of burnup and temperature and half-life of the isotope being considered. Neither the burnup nor the temperature can be known before the fact, but they do not vary for analysis of the relative iodine fission products distribution. Burnup over a range of 0,000 to 60,000 MWd/t were computed for a temperatures of 1°K 1135°K = 1583°F and 1265°K = 1817°F. These temperatures were selected to obtain agreement between the CE Owners Group core damage assessment values and the results calculated using ANS-5.4-1982.

The C-E Owners Group core damage assessment guidance does not include either cesiums or a particulate contribution for the gap release.

Core inventory from BG&E calculation 200-DA-9201:

Nuclide	Half-life	CCNPP Core Inventory(Ci)	CE Damage Assessment Gap Fraction	ANS-5.4-'82 Computed Gap Fraction	Gap Fraction Assumed
Kr-85m	4.48 h	1.97E7		0.005	0.005
Kr-87	76.3 m	3.87E7	2E-7	0.003	0.003
Kr-88	2.84 h	5.54E7		0.004	0.004
Xe-133	5.245 d	1.39E8	9.3E-2	0.03	0.09
Xe-135	9.09 h	2.81E7		0.01	0.01
I-131	8.04 d	6.90E7	9.2E-2	0.093	0.09
I-132	2.3 h	9.78E7	7E-5	0.0103	0.01
I-133	20.8 h	1.39E8	4.5E-2	0.031	0.04
I-134	52.6 m	1.55E8		0.006	0.006
I-135	6.8 h	1.22E8	7.9E-3	0.018	0.02

The CE Computed Gap Fractions are from CE Owners Group Core Damage Assessment guidance, by dividing Table 3-5 data by Table 3-4 data for a 2700 MWT Plant Class.

See Appendix A for computation of ANSI/ANS-5.4-1982 Gap Fractions.

CCNPP Core Inventory * Release Fraction = Estimated Gap Activity

Estimated Gap Activity * 5% release * 10^6 μCi/Ci /2.7E8 ml = Sample Conc.

Isotope	Core Inventory(Ci)	Assumed Release Fraction	5% Gap Release RCS Conc.(μCi/cc) (1 hour decay)	Dose Equivalent I-131
Kr-85m	1.97E7	0.005	1.6E+01	
Kr-87	3.87E7	0.003	1.2E+01	
Kr-88	5.54E7	0.004	3.2E+01	
Xe-133	1.39E8	0.09	2.3E+03	
Xe-135	2.81E7	0.01	4.8E+01	
I-131	6.90E7	0.09	1.1E+03	1.1E+03
I-132	9.78E7	0.01	1.3E+02	4.8E+00
I-133	1.39E8	0.04	1.0E+03	2.7E+02
I-134	1.55E8	0.006	7.8E+01	1.3E+00
I-135	1.22E8	0.02	4.1E+02	3.4E+01
			Total =	1.5E+03

Note that this calculated source term is a factor of 5 higher than the 300 μCi/ml dose equivalent I-131 guidance from the NRC. The primary reason for this difference is the higher fraction of I-131 assumed in the gap.

For comparison, a 5% gap release assuming a constant 5% of the core inventory of iodines is available in the gap results in the following iodine concentrations (μCi/cc) after 1 hour:

I-131	I-132	I-133	I-134	I-135
640	670	1200	650	1000

This equals 1100 μCi/cc dose equivalent I-131.

B. Dose Rate at 1 foot from 12.5 ml sample

The dose rate at 1 foot from the center of the 12.5 ml sample bomb was calculated for the radioiodines and the noble gases, each separately. Data for the sample bomb was obtained from BG&E Calculation 200-DA-9201 Rev. A, Case B4X.04, after correcting for the sample bomb wall thickness. The following dimensions for the Reactor Coolant sample bomb were used:

Length = 20.32 cm
Source radius = 0.443 cm H_2O, 1 g/cc
Tube wall = 0.234 cm Fe, 7.84 g/cc
Shield = 1.27 cm Pb, 11.3 g/cc
Source volume = 12.528 cm^3

PC-SHIELD was used to calculate the dose rate at 1 foot from the sample bomb. The output of the PC-SHIELD calculations are contained in Appendix B. The results are:

Shielded Sample Bomb
42 mrem/hr at 1 ft. due to radioiodines (Depressurized sample in shielded sample bomb)
2.2 mrem/hr at 1 ft. due to noble gases in pressurized sample
44 mrem/hr at 1 ft. for pressurized sample in shielded sample bomb

Normal Sample Bomb
160 mrem/hr at 1 ft. due to radioiodines (Depressurized, unshielded sample bomb)
8.2 mrem/hr at 1 ft. due to noble gases in pressurized sample
168 mrem/hr at 1 ft. for pressurized sample in unshielded sample bomb

The calculated dose rates are a direct multiple of the assumed level of cladding failure. Thus, if an EAL equivalent to 2% clad failure rather than 5% clad failure is desired, the above results may be multiplied by 2/5.

C. Present Results in simple to use format

A 12.5 ml RCS sample in the shielded sample bomb with a dose rate greater than 40 mrem per hour at 1 foot is an indication of significant fuel damage, that is, a release on the order of 5% of the gap activity. Whether the sample is pressurized or not makes little difference, as the largest contribution to the dose rate after 1 hour decay is from the radioiodines expected following a gap release.

Similarly, a dose rate of 160 mrem/hour at 1 foot from an unshielded RCS sample bomb indicates a gap release on the order of 5%. Again, noble gases contribute relatively little to the total dose rate in these circumstances, so whether the sample is pressurized or not makes little difference.

References

Development of the Comprehensive Procedure Guideline for Core Damage Assessment, C-E Owners Group Task 467, May 1983 from BG&E CCNPP Chemistry Department files.

ANSI/ANS-5.4-1982, American National Standard Method for Calculating the Fractional Release of Volatile Fission Products from Oxide Fuel, Approved November 10, 1982.

NUREG-1228, Source Term Estimation During Incident Response to Severe Nuclear Power Plant Accidents, October 1988.

NUMARC/NESP-007, Methodology for Development of Emergency Action Levels, April 1990.

TID 14844, Calculation of Distance Factors for Power and Test Reactor Sites, AEC, March 1962.

NUREG-0654, Rev. 1, Criteria for Preparation and Evaluation of Radiological Emergency Response Plans and Preparedness in Support of Nuclear Power Plants, USNRC/FEMA, November 1980.

NUREG-1465, Accident Source Terms for Light-Water Nuclear Power Plants, Draft Report for Comment, USNRC, June 1992.

BG&E Calculation 200-DA-9201, Re-evaluation of Radiation Doses from Post-Accident Sampling, October 1992.

APPENDIX A

SOURCE TERM CALCULATIONS

Isotope	Half-Life	Lambda (1/sec)	Inventory (Ci)	Dose * Conversion Factor	Hi Temp Release Fraction	5% Gap Release RCS Conc. (1 hr. decayed)	Dose Equivalent I-131
Kr-85m	4.48h	4.30E-05	1.97E+07		0.005	1.6E+01	
Kr-87	76.3m	1.51E-04	3.87E+07		0.003	1.2E+01	
Kr-88	2.84h	6.78E-05	5.54E+07		0.004	3.2E+01	
Xe-133	5.245d	1.53E-06	1.39E+08		0.09	2.3E+03	
Xe-135	9.09h	2.12E-05	2.81E+07		0.01	4.8E+01	
I-131	8.04d	9.98E-07	6.90E+07	1.48E+06	0.09	1.1E+03	1.1E+03
I-132	2.3h	8.37E-05	9.78E+07	5.35E+04	0.01	1.3E+02	4.8E+00
I-133	20.8h	9.25E-06	1.39E+08	4.00E+05	0.04	1.0E+03	2.7E+02
I-134	52.6m	2.20E-04	1.55E+08	2.50E+04	0.006	7.8E+01	1.3E+00
I-135	6.8h	2.83E-05	1.22E+08	1.24E+05	0.02	4.1E+02	3.4E+01
							1.5E+03

% Gap Release = 5

Kr-85m	4.48h	4.30E-05	1.97E+07		0.05	1.6E+02	
Kr-87	76.3m	1.51E-04	3.87E+07		0.05	2.1E+02	
Kr-88	2.84h	6.78E-05	5.54E+07		0.05	4.0E+02	
Xe-133	5.245d	1.53E-06	1.39E+08		0.05	1.3E+03	
Xe-135	9.09h	2.12E-05	2.81E+07		0.05	2.4E+02	
I-131	8.04d	9.98E-07	6.90E+07	1.48E+06	0.05	6.4E+02	6.4E+02
I-132	2.3h	8.37E-05	9.78E+07	5.35E+04	0.05	6.7E+02	2.4E+01
I-133	20.8h	9.25E-06	1.39E+08	4.00E+05	0.05	1.2E+03	3.4E+02
I-134	52.6m	2.20E-04	1.55E+08	2.50E+04	0.05	6.5E+02	1.1E+01
I-135	6.8h	2.83E-05	1.22E+08	1.24E+05	0.05	1.0E+03	8.5E+01
							1.1E+03

* Rem/Ci inhaled, from TID 14844

PROBLEM: Develop estimates of the gap radioiodine activities available for release following fuel cladding failure.

APPROACH: Fractional (gap) radioiodine releases from the fuel to the gap are determined using the equilibrium equation (equation 5) from ANSI/ANS-5.4-1982, which is attached for reference.

From the referenced standard:
"For these conditions (varying temperature and power operations) the release fraction may be conservatively calculated using the equilibrium equation (Equation 5) <u>using the peak temperature reached during the last two halflives of operation</u>."

The temperatures used were selected to obtain results close to those presented in the C-E Owners Group core damage assessment guidance. These temperatures are consistent with those which would be expected for cladding failure, as indicated in NUREG-1228, Table 4.1.

CALCULATIONS: (Using MathCad)

$Bu = 20000, 30000 .. 60000$

$T := 1265$ Kelvin equivalent of 1817 F.

$$D(Bu, T) := 0.61 \cdot \exp\left[-\left[\frac{72300}{(1.987\ T)}\right]\right] \cdot 100^{\left[\frac{Bu}{28000}\right]} \cdot 7$$

See ANS-5.4 for explanation of terms. The factor of 7 at the end of the equation comes from section 4.1 High_Temperature Release Calculations of the standard.

For I-131 $\lambda := 9.98\ 10^{-7}$ Decay constant in 1/seconds.

$$m[\lambda, D] := \frac{\lambda}{D(Bu, T)}$$

D (Bu ,T)
$3.689\ 10^{-11}$
$1.911\ 10^{-10}$
$9.897\ 10^{-10}$
$5.126\ 10^{-9}$
$2.655\ 10^{-8}$

m[λ, D]
$2.705\ 10^{4}$
$5.223\ 10^{3}$
$1.008\ 10^{3}$
194.688
37.588

$$F[m] := 3\ \frac{1}{\sqrt{m}}\ \coth\left[\sqrt{m}\right] - \frac{1}{m}$$

$F(1008) = 0.093$

Evaluated at 40K MWd/t

For I-132:] := 8.37 10^{-5} Decay constant in 1/seconds.

m[ ],D]

2.269 10^{6}
4.38 10^{5}
8.457 10^{4}
1.633 10^{4}
3.152 10^{3}

F[$8.457\cdot10^{4}$] = 0.0103 Evaluated at 40K MWd/t

For I-133:] := 9.25 10^{-6} Decay constant in 1/seconds.

m[ ],D]

2.507 10^{5}
4.841 10^{4}
9.346 10^{3}
1.804 10^{3}
348.389

F (9346) = 0.031 Evaluated at 40K MWd/t

For I-134:] := 2.20 10^{-4} Decay constant in 1/seconds.

m[ ],D]

5.963 10^{6}
1.151 10^{6}
2.223 10^{5}
4.292 10^{4}
8.286 10^{3}

F[$2.223\cdot10^{5}$] = 0.006 Evaluated at 40K MWd/t

For I-135:] := 2.83 10^{-5} Decay constant in 1/seconds.

m[ ],D]

7.671 10^{5}
$1.481\cdot10^{5}$
$2.859\cdot10^{4}$
5.521 10^{3}
$1.066\cdot10^{3}$

F[$2.859\cdot10^{4}$] = 0.018 Evaluated at 40K MWd/t

Summary of Results at 1265K, 40K MWd/t:

Iodine isotope	I131	I132	I133	I134	I135
Release fraction	0.093	0.0103	0.031	0.006	0.018
C-E Guidance	0.092	0.007	0.045		0.0079

$T := 1135$

For I-131 $\lambda := 9.98 \cdot 10^{-7}$ Decay constant in 1/seconds.

$$\mu[\lambda, D] := \frac{\lambda}{D(Bu, T)}$$

$D(Bu,T)$	$\mu[\lambda,D]$
$1.368 \cdot 10^{-12}$	$7.295 \cdot 10^{5}$
$7.086 \cdot 10^{-12}$	$1.408 \cdot 10^{5}$
$3.67 \cdot 10^{-11}$	$2.719 \cdot 10^{4}$
$1.901 \cdot 10^{-10}$	$5.25 \cdot 10^{3}$
$9.846 \cdot 10^{-10}$	$1.014 \cdot 10^{3}$

$F(1014) = 0.093$

Evaluated at 60K MWd/t

For I-132: $\lambda := 8.37 \cdot 10^{-5}$ Decay constant in 1/seconds.

$\mu[\lambda,D]$
$6.118 \cdot 10^{7}$
$1.181 \cdot 10^{7}$
$2.28 \cdot 10^{6}$
$4.403 \cdot 10^{5}$
$8.501 \cdot 10^{4}$

$F[8.501 \cdot 10^{4}] = 0.0103$ Evaluated at 60K MWd/t

For I-133: $\lambda := 9.25 \cdot 10^{-6}$ Decay constant in 1/seconds.

$\mu[\lambda,D]$
$6.761 \cdot 10^{6}$
$1.305 \cdot 10^{6}$
$2.52 \cdot 10^{5}$
$4.866 \cdot 10^{4}$
$9.394 \cdot 10^{3}$

$F(9394) = 0.031$ Evaluated at 60K MWd/t

For I-134: $\lambda := 2.20 \cdot 10^{-4}$ Decay constant in 1/seconds.

$\mu[\lambda,D]$
$1.608 \cdot 10^{8}$
$3.105 \cdot 10^{7}$
$5.994 \cdot 10^{6}$
$1.157 \cdot 10^{6}$
$2.234 \cdot 10^{5}$

$F[2.234 \cdot 10^{5}] = 0.006$ Evaluated at 60K MWd/t

For I-`135: $\rfloor = 2.83\ 10^{-5}$ Decay constant in 1/seconds.

m[ ],D]
$2.069\ 10^{7}$
$3.994 \cdot 10^{6}$
$7.711\ 10^{5}$
$1.489 \cdot 10^{5}$
$2.874\ 10^{4}$

$F[2.874\ 10^{4}] = 0.018$ Evaluated at 60K MWd/t

Summary of Results at 1135 K, 60K MWd/t:

Iodine Isotope	I131	I132	I133	I134	I135
Release Fraction	0.093	0.0103	0.031	0.006	0.018

Conclusion:
As is evident from both the C-E Owners Group core damage assessment guidance and the above calculations using the methods of ASNI/ANS-5.4-1982, the distribution of the release fractions of the radioiodines to the gap is not constant.

PROBLEM: Develop estimates of the gap noble gases activities available for release following fuel cladding failure.

APPROACH: Fractional (Gap) noble gases released from the fuel to the gap are determined using equation 5 from ANSI/ANS-5.4-1982, which is attached for reference.

From the referenced standard:
"For those conditions (varying temperature and power operation), the release fraction may be conservatively calculated using the equilibrium equation (equation 5 in the standard), using the peak temperature reached during the last two halflives of operation."

These calculations are consistent with the previous ones for iodines, and are performed to obtain release fractions for a wider range of noble gases than is provided in the C-E Owners Group guidance on core damage assessment.

CALCULATIONS: (Using MathCad)

$Bu := 20000, 30000 .. 60000$

$T := 1135$ Kelvin equivalent of 1583 F.

$$D(Bu,T) := 0.61 \cdot \exp\left[-\left[\frac{72300}{(1.987\ T)}\right]\right] \cdot 100^{\left[\frac{Bu}{28000}\right]}$$

See ANS-5.4 for explanation of terms

For Kr-85m: $\lambda := 4.30\ 10^{-5}$ Decay constant in 1/seconds

$D(Bu,T)$
$1.954\ 10^{-13}$
$1.012\ 10^{-12}$
$5.243\ 10^{-12}$
$2.716\ 10^{-11}$
$1.407\ 10^{-10}$

$\mu[\lambda,D]$
$2.2\ 10^{8}$
$4.248 \cdot 10^{7}$
$8.201 \cdot 10^{6}$
$1.583\ 10^{6}$
$3.057 \cdot 10^{5}$

$$\mu[\lambda,D] = \frac{\lambda}{D(Bu,T)}$$

$$F[\mu] := 3 \cdot \frac{1}{\sqrt{\mu}} \cdot \coth\left[\sqrt{\mu}\right] - \frac{1}{\mu}$$

Equation 5 from ANS-5.4

$$F\left[3.06 \cdot 10^{5}\right] = 0.005$$

Evaluated at 60K MWd/t

For Kr-87: $\lambda = 1.51 \cdot 10^{-4}$ Decay constant in 1/seconds

$m[\lambda, D]$

$7.726 \cdot 10^{8}$
$1.492 \cdot 10^{8}$
$2.88 \cdot 10^{7}$
$5.56 \cdot 10^{6}$
$1.074 \cdot 10^{6}$

$F[1.07 \cdot 10^{6}] = 0.003$ Evaluated at 60K MWd/t

For Kr-88: $\lambda = 6.78 \cdot 10^{-5}$ Decay constant in 1/seconds

$m[\lambda, D]$

$3.469 \cdot 10^{8}$
$6.698 \cdot 10^{7}$
$1.293 \cdot 10^{7}$
$2.497 \cdot 10^{6}$
$4.82 \cdot 10^{5}$

$F[4.82 \cdot 10^{5}] = 0.004$ Evaluated at 60K MWd/t

For Xe-133: $\lambda := 1.53 \cdot 10^{-6}$ Decay constant in 1/seconds

$m[\lambda, D]$

$7.828 \cdot 10^{6}$
$1.511 \cdot 10^{6}$
$2.918 \cdot 10^{5}$
$5.634 \cdot 10^{4}$
$1.088 \cdot 10^{4}$

$F[1.09 \cdot 10^{4}] = 0.029$ Evaluated at 60K MWd/t

For Xe-135: $\lambda := 2.12 \cdot 10^{-5}$ Decay constant in 1/seconds

$m[\lambda, D]$

$1.085 \cdot 10^{8}$
$2.094 \cdot 10^{7}$
$4.043 \cdot 10^{6}$
$7.806 \cdot 10^{5}$
$1.507 \cdot 10^{5}$

$F[1.51 \cdot 10^{5}] = 0.008$ Evaluated at 60K MWd/t

Summary of Results at 1135K, 60K MWd/t:

Noble gas	Kr-85m	Kr-87	Kr-88	Xe-133	Xe-135
Release Fraction	0.005	0.003	0.004	0.03	0.008
C-E Guidance		<1E-6		0.0867	

$T := 1265$

For Kr-85m: $\lambda := 4.30 \cdot 10^{-5}$ Decay constant in 1/seconds

$$m[\lambda, D] := \frac{\lambda}{D(Bu,T)}$$

$D(Bu,T)$	$m[\lambda,D]$
$5.27 \cdot 10^{-12}$	$8.159 \cdot 10^{6}$
$2.73 \cdot 10^{-11}$	$1.575 \cdot 10^{6}$
$1.414 \cdot 10^{-10}$	$3.041 \cdot 10^{5}$
$7.323 \cdot 10^{-10}$	$5.872 \cdot 10^{4}$
$3.793 \cdot 10^{-9}$	$1.134 \cdot 10^{4}$

$F[3.04 \cdot 10^{5}] = 0.005$

Evaluated at 40K MWd/t

For Kr-87: $\lambda := 1.51 \cdot 10^{-4}$ Decay constant in 1/seconds

$m[\lambda,D]$
$2.865 \cdot 10^{7}$
$5.532 \cdot 10^{6}$
$1.068 \cdot 10^{6}$
$2.062 \cdot 10^{5}$
$3.981 \cdot 10^{4}$

$F[1.07 \cdot 10^{6}] = 0.003$ Evaluated at 40K MWd/t

For Kr-88: $\lambda := 6.78 \cdot 10^{-5}$ Decay constant in 1/seconds

$m[\lambda,D]$
$1.286 \cdot 10^{7}$
$2.484 \cdot 10^{6}$
$4.795 \cdot 10^{5}$
$9.258 \cdot 10^{4}$
$1.788 \cdot 10^{4}$

$F[4.8 \cdot 10^{5}] = 0.004$ Evaluated at 40K MWd/t

For Xe-133: $\lambda := 1.53 \cdot 10^{-6}$ Decay constant in 1/seconds

$m[\lambda,D]$
$2.903 \cdot 10^{5}$
$5.605 \cdot 10^{4}$
$1.082 \cdot 10^{4}$
$2.089 \cdot 10^{3}$
403.378

$F[1.082 \cdot 10^{4}] = 0.029$ Evaluated at 40K MWd/t

For Xe-135: $\lambda := 2.12 \cdot 10^{-5}$ Decay constant in 1/seconds

$m[\lambda,D]$
$4.023 \cdot 10^{6}$
$7.766 \cdot 10^{5}$
$1.499 \cdot 10^{5}$
$2.895 \cdot 10^{4}$
$5.589 \cdot 10^{3}$

$F[1.499 \cdot 10^{5}] = 0.008$ Evaluated at 40K MWd/t

Summary of Results for 1265K at 40K MWd/t:

Noble Gas	Kr-85m	Kr-87	Kr-88	Xe-133	Xe-135
Release Fraction	0.005	0.003	0.004	0.029	0.008

Conclusions:
Noble gases are not contained in the gap at a constant fraction of the core inventory.
3%, rather than 5%, is a more realistic assumption regarding the fraction of long-lived noble gases in the gap.

APPENDIX B

DOSE RATE CALCULATIONS

J. Stewart Bland Consultants, Inc.

PC - SHIELD REPORT
for
BG&E EAL for RCS sample dose rate, iodine contribution

Using only the photon contribution to dose

Prepared on 2/17/93

Press RETURN

3.050E+01 Cm X-Radial distance to the dose point
1.0[illegible] 1 Cm Y-Vertical distance from the end of the source to the dose point

2.032E+01 Cm SL-Source length
4.430E-01 Cm SR-Source radius and first shield
1.253E+01 cc Calculated source volume

5.000E+00 NT-Number of horizontal angle intervals for numerical integratio
n
1.100E+01 NP-Number of vertical angle intervals for numerical integration
5.000E+00 DR-Length of radial intervals for numerical integration

A total of 4 cylindrical shields are used
Shield 1 thickness is 4.430E-01 Cm
Shield 2 thickness is 2.340E-01 Cm
Shield 3 thickness is 1.270E+00 Cm
Shield 4 thickness is 2.855E+01 Cm
(the last shield is an air shield added by PC-SHIELD

Press RETURN

PC - SHIELD REPORT
for
BG&E EAL for RCS sample dose rate
SHIELD SPECIFICATIONS

A total of 4 shields are used
Shield 3 is used for calculating buildup

	Density in Shield gm/cc			
Shield Material	Shield 1	Shield 2	Shield 3	Shield 4
H20	1.00E+00	.00E+00	.00E+00	
Iron	.00E+00	7.84E+00	.00E+00	
Lead	.00E+00	.00E+00	1.13E+01	
AIR				0.001293

Press RETURN

SOURCE SPECIFICATIONS

Values shown are in curies. The source volume is 1.253E+01 cc.

Name	Amount
I 131	1.3781E-02
I 132	1.6286E-03
I 133	1.2528E-02
I 134	9.7718E-04
I 135	5.1365E-03

Press RETURN

PC - SHIELD REPORT
for--
BG&E EAL for RCS sample dose rate
DOSE RESULTS

GROUP	GROUP PRODUCTION RATE PHOTONS/SEC	GROUP AVERAGE ENERGY MEV	ENERGY FLUX AT DOSE POINT MEV/CM2/SEC	DOSE RATE AT DOSE POINT REM /HR
001	.000E+00	1.500E-02	.000E+00	.000E+00
002	.000E+00	2.500E-02	.000E+00	.000E+00
003	.000E+00	3.500E-02	.000E+00	.000E+00
004	.000E+00	4.500E-02	.000E+00	.000E+00
005	.000E+00	5.500E-02	.000E+00	.000E+00
006	.000E+00	6.500E-02	.000E+00	.000E+00
007	.000E+00	7.500E-02	.000E+00	.000E+00
008	1.326E+07	8.500E-02	.000E+00	.000E+00
009	.000E+00	9.500E-02	.000E+00	.000E+00
010	.000E+00	1.500E-01	.000E+00	.000E+00

Press RETURN

GROUP	GROUP PRODUCTION RATE PHOTONS/SEC	GROUP AVERAGE ENERGY MEV	ENERGY FLUX AT DOSE POINT MEV/CM2/SEC	DOSE RATE AT DOSE POINT REM /HR
011	3.110E+07	2.500E-01	6.398E-02	1.612E-07
012	4.171E+08	3.500E-01	1.307E+02	3.280E-04
013	4.404E+08	4.750E-01	1.352E+03	3.217E-03
014	1.441E+08	6.500E-01	1.516E+03	3.366E-03
015	1.706E+08	8.250E-01	3.938E+03	8.269E-03
016	4.292E+07	1.000E+00	1.521E+03	3.011E-03
017	1.578E+08	1.225E+00	7.562E+03	1.407E-02
018	2.847E+07	1.475E+00	1.785E+03	3.141E-03
019	4.728E+07	1.700E+00	3.556E+03	6.046E-03
020	2.892E+06	1.900E+00	2.496E+02	4.043E-04
021	.000E+00	2.100E+00	.000E+00	.000E+00
022	.000E+00	2.300E+00	.000E+00	.000E+00
023	.000E+00	2.500E+00	.000E+00	.000E+00
024	.000E+00	2.700E+00	.000E+00	.000E+00
025	.000E+00	3.000E+00	.000E+00	.000E+00
TOTAL	1.496E+09		2.161E+04	4.185E-02

Press RETURN

J. Stewart Bland Consultants, Inc.

PC - SHIELD REPORT
for
BG&E EAL for RCS sample dose rate, noble gases contribution

Using only the photon contribution to dose

Prepared on 2/17/93

Press RETURN

3.050E+01 Cm X-Radial distance to the dose point
1.01 Cm Y-Vertical distance from the end of the source to the dose point

2.032E+01 Cm SL-Source length
4.430E-01 Cm SR-Source radius and first shield
1.253E+01 cc Calculated source volume

5.000E+00 NT-Number of horizontal angle intervals for numerical integratio
n
1.100E+01 NP-Number of vertical angle intervals for numerical integration
5.000E+00 DR-Length of radial intervals for numerical integration

A total of 4 cylindrical shields are used
Shield 1 thickness is 4.430E-01 Cm
Shield 2 thickness is 2.340E-01 Cm
Shield 3 thickness is 1.270E+00 Cm
Shield 4 thickness is 2.855E+01 Cm
(the last shield is an air shield added by PC-SHIELD

Press RETURN

PC - SHIELD REPORT
for
BG&E EAL for RCS sample dose rate, noble gases contribution
SHIELD SPECIFICATIONS

A total of 4 shields are used
Shield 3 is used for calculating buildup

	Density in Shield gm/cc			
Shield Material	Shield 1	Shield 2	Shield 3	Shield 4
H20	1.00E+00	.00E+00	.00E+00	
Iron	.00E+00	7.84E+00	.00E+00	
Lead	.00E+00	.00E+00	1.13E+01	
AIR				0.001293

Press RETURN

SOURCE SPECIFICATIONS

Values shown are in curies. The source volume is 1.253E+01 cc.

Name	Amount
KRm85	2.0045E-04
KR 87	1.5034E-04
KR 88	4.0090E-04
XE 133	2.8814E-02
XE 135	6.0134E-04

Press RETURN

PC - SHIELD REPORT
for
BG&E EAL for RCS sample dose rate, noble gases contribution
DOSE RESULTS

GROUP	GROUP PRODUCTION RATE PHOTONS/SEC	GROUP AVERAGE ENERGY MEV	ENERGY FLUX AT DOSE POINT MEV/CM2/SEC	DOSE RATE AT DOSE POINT REM /HR
001	.000E+00	1.500E-02	.000E+00	.000E+00
002	.000E+00	2.500E-02	.000E+00	.000E+00
003	.000E+00	3.500E-02	.000E+00	.000E+00
004	.000E+00	4.500E-02	.000E+00	.000E+00
005	.000E+00	5.500E-02	.000E+00	.000E+00
006	.000E+00	6.500E-02	.000E+00	.000E+00
007	.000E+00	7.500E-02	.000E+00	.000E+00
008	6.024E+08	8.500E-02	.000E+00	.000E+00
009	.000E+00	9.500E-02	.000E+00	.000E+00
010	7.799E+06	1.500E-01	9.787E-12	2.476E-17

Press RETURN

GROUP	GROUP PRODUCTION RATE PHOTONS/SEC	GROUP AVERAGE ENERGY MEV	ENERGY FLUX AT DOSE POINT MEV/CM2/SEC	DOSE RATE AT DOSE POINT REM /HR
011	2.014E+07	2.500E-01	4.142E-02	1.044E-07
012	1.083E+06	3.500E-01	3.393E-01	8.515E-07
013	3.226E+06	4.750E-01	9.902E+00	2.357E-05
014	6.452E+05	6.500E-01	6.789E+00	1.507E-05
015	2.857E+06	8.250E-01	6.597E+01	1.385E-04
016	.000E+00	1.000E+00	.000E+00	.000E+00
017	.000E+00	1.225E+00	.000E+00	.000E+00
018	1.988E+06	1.475E+00	1.246E+02	2.193E-04
019	1.335E+05	1.700E+00	1.004E+01	1.707E-05
020	.000E+00	1.900E+00	.000E+00	.000E+00
021	3.402E+06	2.100E+00	3.038E+02	4.830E-04
022	6.942E+06	2.300E+00	7.453E+02	1.140E-03
023	8.344E+05	2.500E+00	9.542E+01	1.431E-04
024	.000E+00	2.700E+00	.000E+00	.000E+00
025	.000E+00	3.000E+00	.000E+00	.000E+00
TOTAL	6.514E+08		1.362E+03	2.181E-03

Press RETURN

J. Stewart Bland Consultants, Inc.

PC - SHIELD REPORT
for
BG&E EAL for RCS sample dose rate, iodine contribution, unshielded

Using only the photon contribution to dose

Prepared on 2/17/93

Press RETURN

3.050E+01 Cm X-Radial distance to the dose point
1.01 1 Cm Y-Vertical distance from the end of the source to the dose point

2.032E+01 Cm SL-Source length
4.430E-01 Cm SR-Source radius and first shield
1.253E+01 cc Calculated source volume

5.000E+00 NT-Number of horizontal angle intervals for numerical integratio
n
1.100E+01 NP-Number of vertical angle intervals for numerical integration
5.000E+00 DR-Length of radial intervals for numerical integration

A total of 3 cylindrical shields are used
Shield 1 thickness is 4.430E-01 Cm
Shield 2 thickness is 2.340E-01 Cm
Shield 3 thickness is 2.982E+01 Cm
(the last shield is an air shield added by PC-SHIELD

Press RETURN

PC - SHIELD REPORT
for
.E EAL for RCS sample dose rate, iodine contribution, unshielded
SHIELD SPECIFICATIONS

A total of 3 shields are used
Shield 2 is used for calculating buildup

Density in Shield gm/cc

Shield Material	Shield 1	Shield 2	Shield 3
H20	1.00E+00	.00E+00	
Iron	.00E+00	7.84E+00	
AIR			0.001293

Press RETURN

SOURCE SPECIFICATIONS

Values shown are in curies. The source volume is 1.253E+01 cc.

Name	Amount
I 131	1.3781E-02
I 132	1.6286E-03
I 133	1.2528E-02
I 134	9.7718E-04
I 135	5.1365E-03

Press RETURN

PC - SHIELD REPORT

for--

BG&E EAL for RCS sample dose rate, iodine contribution, unshielded

DOSE RESULTS

GROUP	GROUP PRODUCTION RATE PHOTONS/SEC	GROUP AVERAGE ENERGY MEV	ENERGY FLUX AT DOSE POINT MEV/CM2/SEC	DOSE RATE AT DOSE POINT REM /HR
001	.000E+00	1.500E-02	.000E+00	.000E+00
002	.000E+00	2.500E-02	.000E+00	.000E+00
003	.000E+00	3.500E-02	.000E+00	.000E+00
004	.000E+00	4.500E-02	.000E+00	.000E+00
005	.000E+00	5.500E-02	.000E+00	.000E+00
006	.000E+00	6.500E-02	.000E+00	.000E+00
007	.000E+00	7.500E-02	.000E+00	.000E+00
008	1.326E+07	8.500E-02	1.937E+01	6.101E-05
009	.000E+00	9.500E-02	.000E+00	.000E+00
010	.000E+00	1.500E-01	.000E+00	.000E+00

Press RETURN

GROUP	GROUP PRODUCTION RATE PHOTONS/SEC	GROUP AVERAGE ENERGY MEV	ENERGY FLUX AT DOSE POINT MEV/CM2/SEC	DOSE RATE AT DOSE POINT REM /HR
011	3.110E+07	2.500E-01	6.033E+02	1.520E-03
012	4.171E+08	3.500E-01	1.146E+04	2.877E-02
013	4.404E+08	4.750E-01	1.642E+04	3.908E-02
014	1.441E+08	6.500E-01	7.327E+03	1.627E-02
015	1.706E+08	8.250E-01	1.102E+04	2.314E-02
016	4.292E+07	1.000E+00	3.355E+03	6.642E-03
017	1.578E+08	1.225E+00	1.499E+04	2.788E-02
018	2.847E+07	1.475E+00	3.252E+03	5.724E-03
019	4.728E+07	1.700E+00	6.210E+03	1.056E-02
020	2.892E+06	1.900E+00	4.235E+02	6.860E-04
021	.000E+00	2.100E+00	.000E+00	.000E+00
022	.000E+00	2.300E+00	.000E+00	.000E+00
023	.000E+00	2.500E+00	.000E+00	.000E+00
024	.000E+00	2.700E+00	.000E+00	.000E+00
025	.000E+00	3.000E+00	.000E+00	.000E+00
TOTAL	1.496E+09		7.508E+04	1.603E-01

Press RETURN

J. Stewart Bland Consultants, Inc.

PC - SHIELD REPORT
for
BG&E EAL for RCS sample dose rate, noble gases contribution, unshielded

Using only the photon contribution to dose

Prepared on 2/17/93

Press RETURN

3.050F+01 Cm X-Radial distance to the dose point
1.0[illegible]1 Cm Y-Vertical distance from the end of the source to the dose point

2.032E+01 Cm SL-Source length
4.430E-01 Cm SR-Source radius and first shield
1.253E+01 cc Calculated source volume

5.000E+00 NT-Number of horizontal angle intervals for numerical integratio
n
1.100E+01 NP-Number of vertical angle intervals for numerical integration
5.000E+00 DR-Length of radial intervals for numerical integration

A total of 3 cylindrical shields are used
Shield 1 thickness is 4.430E-01 Cm
Shield 2 thickness is 2.340E-01 Cm
Shield 3 thickness is 2.982E+01 Cm
(the last shield is an air shield added by PC-SHIELD

Press RETURN

PC - SHIELD REPORT
for
BG&E EAL for RCS sample dose rate, noble gases contribution
SHIELD SPECIFICATIONS

A total of 3 shields are used
Shield 2 is used for calculating buildup

	Density in Shield gm/cc		
Shield Material	Shield 1	Shield 2	Shield 3
H20	1.00E+00	.00E+00	
Iron	.00E+00	7.84E+00	
AIR			0.001293

Press RETURN

SOURCE SPECIFICATIONS

Values shown are in curies. The source volume is 1.253E+01 cc.

Name	Amount
KRm85	2.0045E-04
KR 87	1.5034E-04
KR 88	4.0090E-04
XE 133	2.8814E-02
XE 135	6.0134E-04

Press RETURN

PC - SHIELD REPORT
for
BG&E EAL for RCS sample dose rate, noble gases contribution
DOSE RESULTS

GROUP	GROUP PRODUCTION RATE PHOTONS/SEC	GROUP AVERAGE ENERGY MEV	ENERGY FLUX AT DOSE POINT MEV/CM2/SEC	DOSE RATE AT DOSE POINT REM /HR
001	.000E+00	1.500E-02	.000E+00	.000E+00
002	.000E+00	2.500E-02	.000E+00	.000E+00
003	.000E+00	3.500E-02	.000E+00	.000E+00
004	.000E+00	4.500E-02	.000E+00	.000E+00
005	.000E+00	5.500E-02	.000E+00	.000E+00
006	.000E+00	6.500E-02	.000E+00	.000E+00
007	.000E+00	7.500E-02	.000E+00	.000E+00
008	6.024E+08	8.500E-02	8.800E+02	2.772E-03
009	.000E+00	9.500E-02	.000E+00	.000E+00
010	7.799E+06	1.500E-01	8.890E+01	2.249E-04

Press RETURN

GROUP	GROUP PRODUCTION RATE PHOTONS/SEC	GROUP AVERAGE ENERGY MEV	ENERGY FLUX AT DOSE POINT MEV/CM2/SEC	DOSE RATE AT DOSE POINT REM /HR
011	2.014E+07	2.500E-01	3.906E+02	9.842E-04
012	1.083E+06	3.500E-01	2.975E+01	7.468E-05
013	3.226E+06	4.750E-01	1.203E+02	2.863E-04
014	6.452E+05	6.500E-01	3.280E+01	7.283E-05
015	2.857E+06	8.250E-01	1.846E+02	3.876E-04
016	.000E+00	1.000E+00	.000E+00	.000E+00
017	.000E+00	1.225E+00	.000E+00	.000E+00
018	1.988E+06	1.475E+00	2.271E+02	3.996E-04
019	1.335E+05	1.700E+00	1.753E+01	2.981E-05
020	.000E+00	1.900E+00	.000E+00	.000E+00
021	3.402E+06	2.100E+00	5.499E+02	8.743E-04
022	6.942E+06	2.300E+00	1.226E+03	1.876E-03
023	8.344E+05	2.500E+00	1.600E+02	2.400E-04
024	.000E+00	2.700E+00	.000E+00	.000E+00
025	.000E+00	3.000E+00	.000E+00	.000E+00
TOTAL	6.514E+08		3.908E+03	8.222E-03

Press RETURN

3. Noble Gas Release Calculations

3.1 High-Temperature Release Calculations. For calculating high-temperature release of noble gases from fuel pellets, the column of fuel pellets is divided into radial and axial nodes, and the model is applied to each node for the appropriate local values of temperature and burnup.

Six or more radial nodes of equal volume or equal radial increment, and ten or more axial nodes of equal length shall be used, unless otherwise justified. The irradiation period shall be divided into a series of burnup (time) increments such that the temperature and power in each increment can be assumed constant. These increments shall not exceed 2,000 MWd/t provided that the burnup values used in the analysis correspond to the midpoint of the burnup increments. Otherwise, the burnup increments shall not exceed 1,000 MWd/t.

In the following equations, approximations have been made so that only finite number of terms remains. The error introduced by making these approximations is about one part in 10^5, which is considered insignificant in gas release calculations.

3.1.1 Long-Lived Nuclides (halflife greater than one year). For nuclides with halflife greater than one year (e.g., Kr-85), equations for stable nuclides are assumed to apply. The fractional release F at the end of a single burnup (time) increment is:

$$F = 1 - g(\tau), \qquad \text{(Eq. 1)}$$

where $\tau = D't$, t is time (sec), and the other definitions can be obtained by omitting the subscripts from the appropriate expressions following Equation 2.

The cumulative fractional release F_k at the end of k burnup (time) increments ($k \geq 2$) is:

$$F_k = 1 - \left\{\sum_{i=1}^{k-1} [B_i(\tau_i g_i - \tau_{i+1} g_{i+1})/D_i'] + B_k \Delta t_k g_k\right\} \Big/ \sum_{i=1}^{k} B_i \Delta t_i. \qquad \text{(Eq. 2)}$$

The following definitions apply:

B_i is the fission product production rate (birth rate) during the i^{th} step,

Δt_i is the length of the i^{th} time step (sec),

$$\tau_1 = \sum_{i=1}^{k} D_i' \Delta t_i, \ \tau_2 = \sum_{i=2}^{k} D_i' \Delta t_i, \ldots, \tau_k = D_k' \Delta t_k,$$

$$g_i = g(\tau_i) = 1 - 4\sqrt{\tau_i/\pi} + 3\tau_i/2 \quad \text{for } \tau \leq 0.1,$$

$$g_i = g(\tau_i) = \frac{1}{15\tau_i} - \frac{6}{\tau_i} \sum_{n=1}^{3} \frac{\exp(-n^2\pi^2\tau_i)}{n^4\pi^4} \quad \text{for } \tau > 0.1,$$

$$D_i' = \left[(D_o/a^2)\exp(-Q/RT_i)\right] \times 100^{Bu_i/28,000},$$

$D_o/a^2 = 0.61\ \mathrm{sec}^{-1}$ where the notation D_o/a^2 is retained for consistency with published references,

$Q = 72,300$ cal/mol,

$R = 1.987$ cal/mol K,

T_i is the temperature (K) during the i^{th} time increment,

and

Bu_i is the accumulated burnup (MWd/t) at the midpoint of the i^{th} time increment.

3.1.2 Short-Lived Nuclides (halflife less than one year). The release fraction F for an irradiation period at constant temperature and power, for $\tau < 0.1$, is:

$$F = \frac{3}{1 - \exp(-\mu\tau)} \left[\frac{1}{\sqrt{\mu}}[\mathrm{erf}(\sqrt{\mu\tau}) - 2\sqrt{\mu\tau/\pi}\exp(-\mu\tau)] - \frac{1 - (1 + \mu\tau)\exp(-\mu\tau)}{\mu}\right], \qquad \text{(Eq. 3)}$$

and, for $\tau > 0.1$, the release fraction is:

$$F = 3\left[\frac{1}{\sqrt{\mu}}\coth(\sqrt{\mu}) - \frac{1}{\mu}\right] - \frac{6\mu}{\exp(\mu\tau) - 1} \times \sum_{n=1}^{3} \frac{1 - \exp(-n^2\pi^2\tau)}{n^2\pi^2(n^2\pi^2 + \mu)}. \qquad \text{(Eq. 4)}$$

The following definitions apply:

$\mu = \lambda/D'$,

$\tau = D't$,

λ is the decay constant (sec^{-1}),

t is the time (sec) during the constant-temperature and constant-power irradiation period,

$D' = [D_o/a^2) \exp(-Q/RT)] \times 100^{Bu/28,000}$,

T is the temperature (K),

Bu is the total accumulated burnup (MWd/t) including all prior operating periods,

and

D_o/a^2, Q, and R are defined in 3.1.1.

These equations are valid when no concentration of the nuclides of interest exists from prior operating periods. For most practical applications, a shutdown period of four halflives satisfies this criterion.

Formulations for release fractions during varying temperature and power operation cannot be easily obtained in closed form.

For those conditions, the release fraction may be conservatively calculated using the equilibrium equation:

$$\left[F = 3 \quad \frac{1}{\sqrt{\mu}} \coth(\sqrt{\mu}) - \frac{1}{\mu}\right], \qquad \text{(Eq. 5)}$$

using the peak temperature reached during the last two halflives of operation.

3.2 Low-Temperature Release Calculations.

3.2.1 Long-Lived Nuclides (halflife greater than one year). The cumulative fractional release F is independent of temperature and given by:

$$F = 7 \times 10^{-8} Bu, \qquad \text{(Eq. 6)}$$

where Bu is the rod-average accumulated burnup (MWd/t).

3.2.2 Short-Lived Nuclides (halflife less than one year). The release fraction F is:

$$F = (1/\lambda)[10^{-7}\sqrt{\lambda} + 1.6 \times 10^{-12}P], \qquad \text{(Eq. 7)}$$

where P is the specific power (megawatts per metric ton of heavy metal) and λ is the decay constant (sec^{-1}). For conservatism, the specific power P should correspond to the maximum power level during the last two halflives of operation.

3.3 Precursor Effects. The effect of precursors is believed to be negligibly small except for Xe-133 and Xe-135. When calculating the total release fraction F_{total} for these nuclides of xenon, the fractional release of these nuclides, calculated in 3.1 and 3.2, should be corrected for the effect of precursors ($F^{I\text{-}133}$ and $F^{I\text{-}135}$, respectively) as follows:

$$F^{Xe\text{-}133}_{total} = F^{I\text{-}133} + F^{Xe\text{-}133} - \left(F^{I\text{-}133} \times F^{Xe\text{-}133}\right) \qquad \text{(Eq. 8)}$$

$$F^{Xe\text{-}135}_{total} = F^{I\text{-}135} + F^{Xe\text{-}135} - \left(F^{I\text{-}135} \times F^{Xe\text{-}135}\right) \qquad \text{(Eq. 9)}$$

4. Iodine, Cesium, and Tellurium Release Calculations.

4.1 High-Temperature Release Calculations. The noble gas model described in 3.1 is used except that the diffusion parameters are altered as follows:

$D'_{iodine}/D'_{noble} = 7$,

$D'_{cesium}/D'_{noble} = 2$,

$D'_{tellurium}/D'_{noble} = 30$.

4.2 Low-Temperature Release Calculations. Release fractions are assumed to be the same as those for the noble gases as described in 3.2.

JSB

J. STEWART BLAND ASSOCIATES, INC

RADIOACTIVITY RELEASE EMERGENCY ACTION LEVELS

SEPTEMBER 1990

by

Jim B. McIlvaine, P. E.

JSB Associates, Inc.

133 Defense Highway, Suite 208

Annapolis, MD 21401

FINAL September 1990

RADIOACTIVITY RELEASE
EMERGENCY ACTION LEVELS

EXECUTIVE SUMMARY

A revised set of Radioactivity Release Emergency Action Levels (EALs) is proposed based on the most recent evaluation of source term distributions for various accidents and the Dose Rate Conversion Factors (DCFs) which are used consistently for emergency preparedness off-site dose assessment. Use of EP methods and values, rather than ODCM methods and values, assures consistency within the Emergency Plan.

Proposed Radioactivity Release EALs follow:

Monitor	Unusual Event	ALERT
WRNGM (U1+U2)-RI-5415	1.8 E 5 μCi/sec	1.8 E 6 μCi/sec
MAIN VENT (U1+U2)-RE-5415	5.8 E 4 cpm	5.8 E 5 cpm
WASTE PROC (U1+U2)-RE-5410	2.0 E 5 cpm	Off Scale, High
FUEL HANDLING 0-RE-5420	1.7 E 5 cpm	Off Scale, High
ACCESS CONTROL 0-RE-5425	Off-Scale, High	NA
ECCS Pp. Rm. 1/2-RE-5406	Off-Scale, High	NA

These Radioactivity Release EALs are based on:
- A conservative assessment of the expected source term radionuclide distribution for accidents as presented in BG&E-EP9 and considering release pathways;
- The dose rate conversion factors published by Kocher and used consistently in the ERPIPs, but which differ from the DCFs used in the ODCM; and,
- Guidance in Appendix 1 of NUREG-0654.

The Radioactivity Release EALs developed herein differ from the current values primarily because of differences in the assumed source term radionuclide distribution. Differences between ERPIP and ODCM DCFs also contribute to the difference in EALs.

INTRODUCTION AND BACKGROUND

The purpose of this document is to explain the bases and reasoning leading to the proposed Calvert Cliffs Radioactivity Release Emergency Action Levels (EALs) for Unusual Event and Alert classifications.

NUREG-0654 (Ref. 1) Appendix 1 guidelines call for Notification of Unusual Event if radiological effluent technical specifications are exceeded, and further recommend that an ALERT be declared for radiological effluents greater than 10 times technical specification limits. The Calvert Cliffs Technical Specifications (T.S. 3.11.2) related to gaseous effluents restrict both the offsite dose rate and the total offsite air dose. The offsite dose rate is logically the limiting Technical Specification for Emergency Planning purposes.

Offsite dose rate limits exist for the total body, skin, and the "maximum organ", which for CCNPP is the child thyroid, inhalation pathway. Different nuclides have different dose rate conversion factors for each of the three different dose rate limits considered. The value selected for the Radioactivity Release EAL for each monitor should therefore be established for the most restrictive Technical Specification dose rate limit applicable for the radionuclide distribution existent at the monitor. Each accident scenario applicable to the monitored pathway should be evaluated.

The radionuclide distribution impacts both the effective Dose Conversion Factor which is used to assess offsite dose rates and the effluent monitor response which is used to determine when the EAL is reached. Considerable efforts have already been devoted to assessing potential source terms. The accident source terms in use for accident dose assessment are presented in BG&E-EP9, Calvert Cliffs Nuclear Power Plant Accident Source Terms, Rev. 2, April 1990. (Ref. 2) A comparison with other source terms used at Calvert Cliffs is presented in this report.

The values of the Dose Conversion Factors used in the ERPIPs come from an article by Kocher in the September 1983 Health Physics.

(Ref. 3) These values differ from the dose conversion factors used in the Offsite Dose Calculation Manual (ODCM), the Kocher values for noble gases being generally lower. The ODCM is used to evaluate the environmental impacts of normal releases over long time periods. The Kocher values are newer dose rate conversion factors for immersion in contaminated air, and are used consistently throughout the Calvert Cliffs Emergency Response Plan Implementing Procedures (ERPIPs) and in the RADDOSE dose assessment model. The Kocher values are therefore used in this study.

This study resulted from discovery of an inconsistency in the Radioactivity Release EALs which was identified during the development of responses to NRC comments regarding the EALs. The NRC did not comment specifically regarding the value of the Radioactivity Release EAL for the Main Vent monitors. However, it was determined that the EAL was established using ODCM methods and values. When the EAL was reached, calculations of off-site dose rates using ERPIP methods and values result in values greater than the Technical Specification limit, which is the basis for the EAL. It is this inconsistency which is being addressed and corrected.

APPROACH

Once the appropriate limit has been established, determination of an appropriate EAL for radioactivity releases is straightforward, although several variable factors must be considered. Two variables, meteorology and the Dose Conversion Factors (DCFs), impact the calculation of offsite dose rate, but choosing the most conservative values for each variable can lead to an EAL value which is too low. Unnecessarily low EAL values could lead to declaration of unwarranted UNUSUAL EVENTs or ALERTs. Other factors to consider are the Technical Specifications limits on Whole Body, Skin, and Maximum Organ dose rates. Which dose rate is most restrictive will depend on the radionuclide distribution assumed, which in turn will vary as a function of accident type.

The approach used to establish appropriate Radioactivity Release EALs follows:

- BG&E-EP9 was used as the source term reference for all accident types;
- DCFs were calculated for the expected radionuclide distribution for each accident type;
- The DCFs were evaluated with respect to all accident types and the Technical Specifications and an appropriate DCF was selected;
- The annual average X/Q (2.2E-6 sec/m^3) was assumed as appropriate meteorology; and,
- In accordance with NUREG-0654 guidance, EALs were set for expected monitor response at Technical Specification limits for Notification of Unusual Event and at 10 times Technical Specification limits for an Alert.

The Whole Body dose-rate limit was determined to be the controlling Technical Specification limit for the Waste Gas Decay Tank Rupture (WGDTR) and the Loss of Coolant Accident with release of Reactor Coolant (LOCAR). The Whole Body dose-rate limit also controls for the Steam Generator Tube Rupture (SGTR) accidents as long as the release exits through the condenser to the Main Vent. (This pathway achieves additional iodine and particulate washout in the condenser.) For the Fuel Handling Incident (FHI), determination of the most limiting dose rate depends upon which values are used to compute the dose rate. The ODCM dose rate conversion values result in the Whole Body dose rate being more limiting; the Kocher values and the values used in RADDOSE for the Child Thyroid dose rate result in the Maximum Organ dose rate being more limiting for this accident. The comparison of Whole Body limits with Skin and Maximum Organ dose rate limits is discussed in the **CALCULATIONS AND RESULTS** section under Dose Rate Conversion Factors. The specific dose rate limit used to calculate each EAL value is stated in the EAL Computation

Use of BG&E-EP9 as the source term basis document was determined to be appropriate for several reasons. It is the most recent and current study; it is comprehensive, and thoroughly researched and reviewed; and in one very important aspect - the noble gases distribution calculated for the Calvert Cliffs core - it is consistent with other bases documents which were used previously in establishing the emergency procedures. That is, the noble gases distribution found in BG&E-EP9 is identical, to 2 significant figures, to that derived from NUREG-0771, which was previously used to develop source terms for the ERPIPs.

CALCULATIONS AND RESULTS

Source Terms

For purposes of establishing the Radioactivity Release EALs, the distribution of radionuclides is as important as the absolute magnitude of the release. The dose rate varies as a function of the individual isotopes; i.e., 1 curie per cubic meter of Xe 135 produces a greater dose rate than 1 curie per cubic meter of Xe 133. Therefore, the distribution of radionuclides has as much bearing on the dose rate as does the release rate.

The original ERPIPs were developed using NUREG-0771, Table 3.2 source term values normalized to 2700 MWt as the basis. (Ref.4) BG&E-EP9 is based on the later NUREG-1228. Comparison of the distribution of the noble gases between these two base documents, shown in Table 1, shows agreement to two significant digits in the relative contributions of the noble gases. Therefore, the difference in base document - NUREG-0771 or NUREG-1228 - makes no appreciable difference for these purposes.

TABLE 1
ACCIDENT SOURCE TERM COMPARISON
Derived Core Inventories in MegaCuries

Isotope	Kr-85	Kr-85m	Kr-87	Kr-88	Xe-133	Xe-135
$t_{½}$ (hrs)	93910	4.48	1.27	2.84	125.9	9.09
BG&E-EP9	0.645	27.7	54.2	78.4	196	39.2
% Total	0.16%	6.99%	13.68%	19.79%	49.48%	9.90%
NUREG0771	0.445	19.7	38.7	55.4	139	28.1
% Total	0.16%	7.00%	13.76%	19.69%	49.41%	9.99%
ORIGEN	1.49	13.7	24.8	34.6	149	32.2
% Total	0.58%	5.36%	9.70%	13.50%	58.3%	12.6%

A comparison was also made to the Calvert Cliffs Nuclear Power Plant specific source term analysis done by Combustion Engineering using the ORIGEN computer code and assuming a 24 month fuel cycle. (Ref. 5) Agreement between BG&E-EP9 and the ORIGEN results is good, although not as good as the agreement between results based on NUREG-0771 and NUREG-1228. (See Table 1.) The ORIGEN results, when the same six significant noble gas radionuclides are considered, indicate more relative Xenon isotopes and Krypton 85 and relatively less of the other Krypton radionuclides. This results in a lower calculated effective dose-rate conversion factor when using the ORIGEN results. This further demonstrates the conservatism in use of BG&E-EP9 as the basis for accident source terms.

The noble gas radionuclide distribution for both gap and core releases are nearly identical, so only the RCS and gap release need be considered with respect to noble gases distribution. See Table 2.

TABLE 2
NOBLE GASES DISTRIBUTION
RCS, Gap, and Core
(Curies)

Isotope	Kr-85	Kr-85m	Kr-87	Kr-88	Xe-133	Xe-135
RCS Inventory	91.7	34.1	32.0	59.7	554	181
% of Total	9.63%	3.58%	3.36%	6.27%	58.2%	19.0%
Gap Inventory	1.94E4	8.30E5	1.62E6	2.35E6	5.88E6	1.18E6
% of Total	0.16%	6.99%	13.6%	19.8%	49.5%	9.93%
Core Inventory	6.45E5	2.77E7	5.42E7	7.84E7	1.96E8	3.92E7
% of Total	0.16%	6.99%	13.7%	19.8%	49.5%	9.90%

Above values are from BG&E-EP9, Table 1, page T2.

Dose Rate Conversion Factors

The effective Dose-Rate Conversion Factor for a specific mix of airborne radioactive material may be determined using the method presented in the CCNPP ODCM, Section 3.4.5. The method is defined by the equation:

$$DCF_{EFF} = \Sigma_i \; DCF_i \times f_i$$

where: DCF_{eff} = the effective whole body dose-rate conversion factor from all radionuclides released;

DCF_i = the whole body dose-rate conversion factor for radionuclide i from Kocher; and,

f_i = the fractional abundance of radionuclide i relative to the total noble gas activity.

Use of this equation with the radionuclide distributions presented in BG&E-EP9, Table 11, and the Kocher dose-rate conversion factors yields the following effective Whole Body DCFs, in mrem/hour per $\mu Ci/m^3$:

Waste Gas Decay Tank rupture: DCF_{eff} = 0.0455;

Fuel Handling Incident: DCF_{eff} = 0.0208;

Steam Line Break: DCF_{eff} = 0.465;

Steam Generator Tube Rupture, without fuel damage: DCF_{eff} = 0.148;

Steam Generator Tube Rupture, with gap release: DCF_{eff} = 1.08;

LOCA, with RCS release: DCF_{eff} = 0.0962;

LOCA, with Gap release: DCF_{eff} = 0.243;

LOCA, with core melt: DCF_{eff} = 0.243;

RCS distribution of noble gases: DCF_{eff} = 0.144.

The spreadsheet used in the calculation of these values is presented in Appendix A.

For all the accident types except the Loss of Coolant Accidents (LOCAs), the duration of the release is assumed to be one hour or less. For these cases the change in the radionuclide mix due to decay during the release is not too great and a time variable DCF was not calculated. Rather, the DCF is based on the total curies released during the postulated accident.

For the LOCA scenarios, with 6 hour release periods, decay during the release does materially effect the DCF calculation. The DCFs used herein for a 6 hour LOCA release are also based on the total activity estimated to be released. (BG&E-EP9, Table 9 or 11) An evaluation was done to compare this value for the DCF with the DCF as a function of time after reactor shutdown. (The evaluation and results are contained in Appendix B.) The DCFs decrease rather quickly as a function of time. The mid-range value of the DCF over 6 hours is always greater than the average, and is also greater than the value at 3 hours. The DCF based on the total activity released during a LOCA is always between the mid-range value and the average value of the DCF as a function of time.

The Kocher dose rate conversion factors result from calculations which assume a semi-infinite cloud of uniformly distributed radioactive material. Therefore the above numbers do not include a finite cloud correction factor. For the purpose of this determination of appropriate Emergency Action Levels, the Kocher values are judged to be adequate, without a finite cloud correction factor.

The Whole Body dose rate is the more limiting Technical Specification compared to the Skin dose rate limit. This was determined by evaluating the effective whole body and skin dose rate conversion factors from the noble gases for each accident type. Values from the ODCM were used in this evaluation for consistency and simplicity. The results, presented in Table 3, show that the effective skin DCF is less than 6 times the effective whole body DCF. Since the Technical Specification skin dose rate limit is 6 times the whole body dose rate limit, the whole body dose rate limit will control for each of the evaluated accidents. (See Appendix C for supporting data.)

TABLE 3
COMPARISON OF WHOLE BODY AND SKIN
DOSE-RATE CONVERSION FACTORS
(mrem/hour per μCurie/m^3)

ACCIDENT TYPE	WHOLE BODY DCF	SKIN DCF
WGDTR	6.57E-2	0.132
FHI	3.37E-2	8.02E-2
LOCAR	0.131	0.256

Likewise, an evaluation was also done to determine for which accidents the maximum organ dose-rate limit of Technical Specification 3.11.2.1 b. was more limiting than the Total Body dose-rate limit. (Evaluation and results are shown in Appendix

D.) The RADDOSE values of the dose conversion factors were used with the source terms in BG&E-EP9 to calculate both Total Body and Maximum Organ DCFs for each accident. The ratio of the DCFs were compared to the fraction of the release comprised of radioiodines to determine if the maximum organ dose-rate was most limiting. When all iodine reduction factors are considered, the Total Body dose-rate limit is limiting for each accident except the Steam Line Break (SLB) and possibly the Fuel Handling Incident (FHI).

The Steam Line Break (SLB) has a larger effective Dose-Rate Conversion Factor than any other potential accident analyzed except for the Steam Generator Tube Rupture with a Gap release (SGTRG). However, this accident is not used to establish any Radioactivity Release EAL, for two reasons. First, although the DCF_{eff} is high, the total curies available within the secondary system is small. Noble gases are continuously removed in the Condenser, and iodines and particulates are continuously removed by the Condensate Cleanup System. Also, a Technical Specification limits the allowable level of dose equivalent I-131. Even with a high DCF_{eff}, with a very small source quantity the off-site doses will be small. Second, the Steam Line Break accident outside of containment results in a unmonitored release from the Turbine Building. The escaping steam is not released via a normal, monitored vent pathway, so no monitor exists to provide a Radioactivity Release EAL for the SLB.

EAL Computation

Determination of the appropriate Emergency Action Level for radioactivity releases is simply the calculation of the release rate which equates to the Technical Specification limit and determining the expected monitor response at that release rate. Computations for each monitor(s) used for EALs follow.

Main Vent Release Pathway

For this EAL computation only those accidents where releases via the main vent are probable are considered. These accidents include: Waste Gas Decay Tank Rupture (WGDTR); Fuel Handling Incident (FHI); Steam Generator Tube Rupture without fuel damage (SGTRR); Steam Generator Tube Rupture with Fuel damage (SGTRG); and all the Loss of Coolant Accidents (LOCAR, LOCAG, and LOCAM). Steam Generator Tube Rupture releases will be via the main vent only until the Main Steam Isolation Valves close. And as long as the releases are via the Main Vent, radioiodine releases should be lower than projected in BG&E-EP9 due to additional iodine washout in the condenser, leaving only the noble gases as a significant source. Likewise for the LOCA scenarios, Containment leakage that exits via the Main Vent has probably leaked into Penetration Room(s) and ventilation exhaust from these rooms is filtered in charcoal filters to remove radioiodines. Therefore, the RCS distribution of noble gases is used to establish DCF_{eff}.

There is a significant difference between the effective DCF for RCS releases vis-a-vis fuel rod gap releases. The less conservative RCS noble gas distribution is used to determine the allowable release rate instead of the more conservative fuel gap distribution. This assures that the EAL is above the Main Vent Monitor's alarm setpoint, which is only reasonable since the alarm setpoint has a safety margin in it. Use of this less conservative distribution is justified by the observation that it is highly unlikey that a noble gas distribution equivalent to a gap release could exist at the Main Vent monitors without some other indicator which would have already caused an appropriate emergency notification. It is also most likely that - during progression of an accident - an RCS distribution will exist in a release pathway prior to fuel damage.

The Wide Range Noble Gas Monitor has a more predictable response to a range of noble gas distributions. Consequently, if there is significant disagreement between the WRNGM and the MVGM, the WRNGM should be used for EAL classification.

Wide Range Noble Gas Monitor (U1+U2)-RI-5415

The basis for the WRNGM EAL should be consistent with the basis for the MVGM EAL, as both monitors are in the same release path. Therefore, the EAL should be set based on the RCS noble gas distribution.

NOUE at Tech. Spec. 3.11.2.1 limit:

Tech Spec Whole Body limit = 500 mrem/yr = 0.057 mrem/hr

0.057 mrem/hr / 0.144 mrem/hr per μCi/m^3 = 0.40 μCi/m^3 at site boundary

0.40 μCi/m^3 / 2.2 E-6 sec/m^3 = 1.8 E+5 μCi/sec release from site

NOUE EAL: (U1+U2)-RI-5415 at 1.8 E 5 μCi/sec

ALERT EAL: (U1+U2)-RI-5415 at 1.8 E 6 μCi/sec

Main Vent Gaseous Monitors (U1+U2)-RE-5415

The basis for the MVGM EAL should be consistent with the basis for the WRNGM EAL, as both monitors are in the same release path. Therefore, the EAL should be set based on the RCS noble gas distribution.

NOUE at Tech. Spec. 3.11.2.1 limit:

0.40 μCi/m^3 / 2.2 E-6 sec/m^3 = 1.8 E+5 μCi/sec release from site

Total main vent flow rate = 63.7 + 57.5 = 121.2 m^3/sec

1.8 E+5 μCi/sec / 121.2 m^3/sec = 1.5 E-3 μCi/cc in both vents

The minimum Notification of Unusual Event concentration is 1.5 E-3 μCi/cc in both main vents simultaneously. Table 4 presents the development of the expected monitor response for the Main Vent Gaseous Monitor.

TABLE 4
MAIN VENT MONITOR RESPONSE
AT RADIOACTIVITY RELEASE EAL

ISOTOPE	RCS CONC.	% TOTAL	EAL CONC.	RESPONSE cpm/10^{-6}	EAL cpm
Kr-85	0.43	9.62	1.4 E-4	35	4.9 E3
Kr-85m	0.16	3.58	5.4 E-5	55	3.0 E3
Kr-87	0.15	3.36	5.0 E-5	218	1.1 E4
Kr-88	0.28	6.26	9.4 E-5	189	1.8 E4
Xe-133	2.6	58.17	8.7 E-4	1.87	1.6 E3
Xe-135	0.85	19.01	2.9 E-4	70	2.0 E4
Totals	4.47	100%	1.5 E-3		5.8 E4

The computed EAL value of a total of 5.8 E4 cpm from both Main Vent Gaseous Monitors is comfortably higher than the current ALARM setpoint of 1200 cpm on each monitor.

The ALERT EAL should be 10 times the Unusual Event EAL, or 5.8 E 5 cpm for (U1+U2)-RE-5415.

Waste Processing Monitor (U1+U2)-RE-5410

For the Waste Processing Monitor the noble gas distribution associated with a Waste Gas Decay Tank Rupture is the most likely source for an accident sufficient to cause implementation of the Emergency Plan.

NOUE at Tech Spec 3.11.2.1 limit:

0.057 mrem/hr / 0.0455 mrem/hr per $\mu Ci/m^3$ = 1.3 $\mu Ci/m^3$

1.3 $\mu Ci/m^3$ / 2.2 E-6 sec/m^3 = 5.7 E 5 $\mu Ci/sec$ release from site

Waste Processing Ventilation Flow = 49,500 cfm = 23.4 m^3/sec

5.7 E 5 $\mu Ci/sec$ / 23.4 m^3/sec = 2.4 E-2 $\mu Ci/cc$

Monitor Response at EAL Concentration

Isotope	% Total	EAL Conc.	Response cpm/10^{-6}	EAL Response
Kr-85	0.5	1.3 E-4	35	4.2 E 3
Kr-85m	0.8	1.9 E-4	55	1.1 E 4
Kr-87	0.4	9.6 E-5	218	2.1 E 4
Kr-88	1.3	3.1 E-4	189	5.9 E 4
Xe-133	93.1	2.2 E-2	1.87	4.2 E 4
Xe-135	3.9	9.4 E-4	70	6.6 E 4
		Total cpm at EAL Conc. =		2.0 E 5

ALERT EAL at ten times NOUE EAL = 2.0 E 6 cpm for (U1+U2)-RE-5410 or either monitor off-scale, high.

Fuel Handling Monitor 0-RE-5420

For the Fuel Handling Monitor the EAL should be established based on a Fuel Handling Incident (FHI). However, whether to set the EAL based on the Total Body dose-rate limit or the Child Thyroid (Maximum Organ) dose-rate limit is not easily determined.

Starting with BG&E-EP9 FHI source terms and using ODCM values for dose conversion factors results in the Total Body dose-rate limit being the limiting Technical Specification. Using Kocher and RADDOSE values results in the Maximum Organ (Child Thyroid) dose-rate limit being more restrictive. This anomaly is due solely to different values for dose-rate conversion factors, but it does suggest that either limit is justifiable.

The Maximum Organ (Child Thyroid) dose rate is related to the amount of radioiodines released, which in turn is dependent upon iodine washout in the Spent Fuel Pool. Washout may result in a reduction factor of from 100 to 10,000. At the low end, the Maximum Organ dose-rate limit could be limiting; at the high end the Total Body dose-rate limit is most restrictive. Analysis indicates that if iodine washout in the Spent Fuel Pool results in a DF > 252, then the Total Body dose-rate limit controls. Since this DF is towards the low end of the expected range, this EAL will be established based on a Total Body dose-rate limit.

Only noble gases will be considered in determining the monitor response at the EAL. This will result in a conservative EAL. The monitor is located upstream of the charcoal filters. It will respond to both noble gases and radioiodines released from the Spent Fuel Pool, after iodine washout in the pool but before reduction through the charcoal filters. The Fuel Handling Monitor does not distinguish between noble gases and radioiodines, so radioiodines which are released and sensed at the monitor will cause earlier attainment of the EAL.

NOUE at Tech Spec 3.11.2.1 limit:

0.057 mrem/hr / 0.0208 mrem/hr per $\mu Ci/m^3$ = 2.7 $\mu Ci/m^3$

2.7 μCi/m^3 / 2.2 E-6 sec/m^3 = 1.2 E 6 μCi/sec release from site

Fuel Handling Area Ventilation Flow = 32,000 cfm = 15.1 m^3/sec

1.2 E 6 μCi/sec / 15.1 m^3/sec = 8.0 E-2 μCi/cc

Monitor Response at EAL Concentration

Isotope	% Total	EAL Conc.	Response cpm/10^{-6}	EAL Response
Kr-85	0.5	4.0 E-4	35	1.4 E 4
Kr-85m	0	0	55	0
Kr-87	0	0	218	0
Kr-88	0	0	189	0
Xe-133	99.4	8.0 E-2	1.87	1.5 E 5
Xe-135	0.1	8.0 E-5	70	5.6 E 3

Total cpm at EAL Conc. = 1.7 E 5 cpm

ALERT EAL at ten times NOUE EAL = Off-scale, High. Use 9 E5 cpm.

Access Control Monitor 0-RE-5425

Although significant airborne releases via the Access Control Point are unlikely, the monitor is placed in a potential release pathway. Therefore determination of an EAL is prudent. This determination must assume an airborne radionuclide distribution, and the distribution should reflect a feasible accident so that the EAL may be realistic. With complete failure of the Auxiliary Building ventilation systems and a reactor coolant leak into the Aux. Building, a RCS noble gas distribution could be present. Such a distribution would be conservative vis-a-vis other potential accident distributions such as the WGDTR or FHI. Therefore, a RCS noble gases distribution will be assumed to determine the Access Control Point EAL.

NOUE at Tech. Spec. 3.11.2.1 limit:

Tech Spec Whole Body limit = 500 mrem/yr = 0.057 mrem/hr

0.057 mrem/hr / 0.144 mrem/hr per $\mu Ci/m^3$ = 0.40 $\mu Ci/m^3$ at site boundary

0.40 $\mu Ci/m^3$ / 2.2 E-6 sec/m^3 = 1.8 E+5 $\mu Ci/sec$ release from site

Access Control Point vent flow rate = 13,900 cfm = 6.56 m^3/sec

1.8 E+5 $\mu Ci/sec$ / 6.56 m^3/sec = 2.7 E-2 $\mu Ci/cc$

Access Control Point
Monitor Response at EAL Concentration

Isotope	% Total	EAL Conc.	Response cpm/10^{-6}	EAL Response
Kr-85	9.62	2.6 E-3	35	9.1 E 4
Kr-85m	3.58	9.8 E-4	55	5.4 E 4
Kr-87	3.36	9.2 E-4	218	2.0 E 5
Kr-88	6.26	1.7 E-3	189	3.2 E 5
Xe-133	58.17	1.6 E-2	1.87	3.0 E 4
Xe-135	19.01	5.2 E-3	70	3.6 E 5

Total cpm at EAL Conc. = Off-scale, High

ECCS Pump Room U1/U2-RE-5406

Following a Loss of Coolant Accident, equipment leakage into the ECCS Pump Room could release the RCS distribution of noble gases into the ECCS Pump Room. Initially flow is from the Refueling Water Storage Tank, but recirculation can begin in about 30 minutes, therefore the RCS distribution without decay may be assumed without significant additional conservatism.

NOUE at Tech. Spec. 3.11.2.1 limit:

Tech Spec Whole Body limit = 500 mrem/yr = 0.057 mrem/hr

0.057 mrem/hr / 0.144 mrem/hr per μCi/m^3 = 0.40 μCi/m^3 at site boundary

0.40 μCi/m^3 / 2.2 E-6 sec/m^3 = 1.8 E+5 μCi/sec release from site

ECCS Pump Room vent flow rate = 3,000 cfm = 1.42 m^3/sec

1.8 E+5 μCi/sec / 1.42 m^3/sec = 0.13 μCi/cc

ECCS Pump Room
Monitor Response at EAL Concentration

Isotope	% Total	EAL Conc.	Response cpm/10^{-6}	EAL Response
Kr-85	9.62	1.2 E-2	35	4.2 E 5
Kr-85m	3.58	4.6 E-3	55	2.5 E 5
Kr-87	3.36	4.3 E-3	218	9.4 E 5
Kr-88	6.26	8.0 E-3	189	1.5 E 6
Xe-133	58.17	7.4 E-2	1.87	1.4 E 5
Xe-135	19.01	2.4 E-2	70	1.7 E 6

Total cpm at EAL Conc. = Off-scale, High

CONCLUSIONS

Revisions to ERPIP 3.0 Emergency Action Levels are suggested to reflect the accident source term assumptions of BG&E-EP9.

Default dose-rate conversion factors presented in the ERPIPs should be revised to be consistent with those used in the emergency dose assessment code RADDOSE and with the expected sources for the different types of accidents as presented in BG&E-EP9. Other portions of the ERPIPs, such as graphs of monitor response, will likewise need revision to maintain consistency. A comprehensive ERPIP review is recommended upon acceptance of RADDOSE and BG&E-EP9.

REFERENCES

1. NUREG-0654 / FEMA-REP-1, Rev. 1, Criteria for Preparation and Evaluation of Radiological Emergency Response Plans and Preparedness in Support of Nuclear Power Plants, Nov. 1980

2. BG&E-EP9, Calvert Cliffs Nuclear Power Plant Accident Source Terms, Rev. 2, April 1990

3. D.C. Kocher, "Dose-Rate Conversion Factors for External Exposure to Photons and Electrons", p. 665-686, in Health Physics, Vol. 45, No. 3, September 1983

4. E.T. Reimer memorandum to G.C. Rudigier, Oct. 6,1987, Subject: CCNPP Accident Source Terms, BG&E-EP9

5. Combustion Engineering (J.E. Baum) letter to BG&E (J.A. Mihalcik), 9/15/87, Subject: BG&E Fuel Fission Products Isotopic Inventories from ORIGEN

6. Westinghouse (C.H. Griesacker) letter to BG&E (Boyd Wylie), 6/26/81, Subject: Request by Pete Crinigan for various isotope response sensitivity for the monitors at Calvert Cliffs Nuclear Power Plant

APPENDIX A

EFFECTIVE TOTAL BODY DOSE CONVERSION FACTORS

FISSION PRODUCT	AIR IMMERSION DCF	WGDTR	WGDTR DCF	FHI	FHI DCF	SLB	SLB DCF	SGTRR	SGTRR DCF	SGTRG	SGTRG DCF	LOCAR	LOCAR DCF	LOCAG	LOCAG DCF	LOCAM	LOCAM DCF
Kr-85	3.03	2.62e2	1.382e-2	1.1e1	1.472e-2	2.32e-1	1.430e-2	2.75e1	2.902e-1	5.8e0	4.799e-3	2.29e-1	3.287e-1	4.83e0	6.518e-3	1.61e2	6.496e-3
Kr-85m	96.3	4.41e2	7.393e-1	5.49e-3	2.335e-4	8.64e-2	1.692e-1	1.02e1	3.4210e0	2.49e2	6.5481e0	5.51e-2	2.5134e0	1.34e2	5.7469e0	4.46e3	5.7192e0
Kr-87	527.99	2.4e2	2.2060e0	7.33e-15	1.71e-15	8.1e-2	8.699e-1	9.59e0	1.7635e1	4.87e2	7.0218e1	2.34e-2	5.8522e0	1.19e2	2.7982e1	3.97e3	2.7912e1
Kr-88	1313.63	7.69e2	1.7586e1	2.56e-5	1.485e-5	1.51e-1	4.0349e0	1.79e1	8.1893e1	7.04e2	2.5254e2	7.79e-2	4.8472e1	3.06e2	1.7902e2	1.02e4	1.7842e2
Xe-133	20.7	5.35e4	1.9279e1	2.25e3	2.0571e1	1.4e0	5.895e-1	1.66e2	1.1967e1	1.76e3	9.9489e0	1.36e0	1.3335e1	1.44e3	1.3275e1	4.81e4	1.3258e1
Xe-135	146.99	2.23e3	5.7064e0	2.97e0	1.928e-1	4.59e-1	1.3724e0	5.44e1	2.7849e1	3.52e2	1.4129e1	3.65e-1	2.5413e1	2.36e2	1.5449e1	7.87e3	1.5404e1
		5.7442e4	100.00%	2.2640e3	99.87%	2.4094e0	1.52%	2.8559e2	96.39%	3.5578e3	32.61%	2.1104e0	99.71%	2.2398e3	99.26%	7.4761e4	98.88%
I-131	222.18			8.6e-2	8.439e-3	5.15e0	2.3275e1	5.76e-2	4.457e-2	1.17e1	7.099e-1	4.75e-5	4.999e-3	9.68e-1	9.578e-2	4.84e1	1.432e-1
I-132	1381.22			5.91e-11	3.61e-11	2.44e0	6.8554e1	2.69e-1	1.2940e0	1.66e1	6.2613e0	1.03e-4	6.739e-2	6.38e-1	3.925e-1	3.19e1	5.867e-1
I-133	358.61			2.07e-2	3.279e-3	8.55e0	6.2369e1	1.79e-1	2.236e-1	2.35e1	2.3014e0	1.35e-4	2.293e-2	1.77e0	2.827e-1	8.87e1	4.236e-1
I-134	1600.86			5.33e-26	3.77e-26	1.55e0	5.0474e1	4.35e-1	2.4253e0	2.62e1	1.1454e1	7.59e-5	5.755e-2	4.58e-1	3.265e-1	2.29e1	4.882e-1
I-135	967.27			1.17e-4	4.998e-5	7.49e0	1.4737e2	3.33e-1	1.1218e0	2.07e1	5.4678e0	2.07e-4	9.484e-2	1.29e0	5.557e-1	6.43e1	8.282e-1
		0		1.068e-1	.06%	2.518e1	75.70%	1.2736e0	3.44%	9.87e1	2.42%	5.684e-4	.26%	5.124e0	.68%	2.562e2	1.01%
Cs-134	929.26			2.45e-2	1.006e-2	1.03e0	1.9470e1	9.08e-3	2.939e-2	2.59e0	4.4507e2	7.57e-6	3.332e-3	2.16e-1	8.939e-2	4.31e0	5.333e-2
Cs-136	1305.19			8.37e-3	4.825e-3	1.18e-1	3.1328e0	1.11e-3	5.046e-3	1.04e0	2.5101e2	9.21e-7	5.694e-4	8.57e-2	4.982e-2	1.71e0	2.972e-2
Cs-137	.95			1.54e-2	6.462e-6	1.38e0	2.667e-2	1.2e-2	3.970e-5	1.62e0	2.846e-1	1e-5	4.500e-6	1.35e-1	5.712e-5	2.7e0	3.416e-5
Rb-86	62.94			7.62e-5	2.118e-6					8.97e-3	1.044e-1			7.44e-4	2.085e-5	1.49e-2	1.249e-5
Te-127	4.22			1.9e-7	3.54e-10					4.07e-3	3.176e-3			2.74e-4	5.150e-7	8.23e-1	4.625e-5
Te-127m	1.87			7.07e-6	5.839e-9					7.59e-4	2.625e-4			6.32e-5	5.263e-8	1.9e-1	4.731e-6
Te-129	35.82			3.14e-23	4.97e-25	1.49e-1	1.086e-1	3.07e-2	3.830e-3	2.14e-2	1.418e-1	6.9e-6	1.171e-4	4.81e-4	7.673e-6	1.44e0	6.868e-4
Te-129m	21.84			3.26e-5	3.145e-7	2.39e-2	1.062e-2	2.43e-4	1.848e-5	3.66e-3	1.478e-2	2.02e-7	2.090e-6	3.04e-4	2.957e-6	9.12e-1	2.652e-4
Te-131m	861.68			1.61e-5	6.127e-6	1.09e-1	1.9105e0	1.92e-3	5.762e-3	8.97e-3	1.4293e0	1.49e-6	6.081e-4	6.98e-4	2.679e-4	2.1e0	2.410e-2
Te-132	126.72			4.14e-4	2.317e-5	1.68e-1	4.330e-1	2.17e-3	9.577e-4	8.28e-2	1.9403e0	1.76e-6	1.056e-4	6.72e-3	3.792e-4	2.02e1	3.409e-2
Sb-127	393.67			2.34e-5	4.069e-6					4.21e-3	3.065e-1			3.43e-4	6.014e-5	1.03e0	5.399e-3
Sb-129	870.12			1.95e-9	7.49e-10					2.28e-2	3.6687e0			1.22e-3	4.728e-4	3.65e0	4.229e-2
Sr-89	5.03					1.76e-2	1.801e-3	1.79e-4	3.136e-6			1.49e-7	3.550e-7			3.78e0	2.532e-4
Sr-90	1.22					1.54e-3	3.822e-5	1.53e-5	6.501e-8			1.28e-8	7.397e-9			1.49e-1	2.421e-6
Sr-91	420.28					3.68e-2	3.146e-1	1.23e-3	1.800e-3			8.32e-7	1.656e-4			3.6e0	2.015e-2
Ba-140	110.67					1.54e0	3.4668e0	1.66e-2	6.398e-3			1.38e-5	7.234e-4			1.83e1	2.697e-2
Co-58	582.9					5.89e-1	6.9838e0	5.88e-3	1.194e-2			4.9e-6	1.353e-3			4.48e-2	3.477e-4
Co-60	1499.48					6.89e-2	2.1016e0	6.78e-4	3.541e-3			5.65e-7	4.013e-4			1.67e-2	3.334e-4
Mo-99	95.46					6.18e-1	1.2000e0	8.19e-3	2.723e-3			6.62e-6	2.993e-4			8.93e0	1.135e-2
Tc-99m	76.87					1.19e-1	1.861e-1	6.01e-3	1.609e-3			3.61e-6	1.314e-4			5.81e0	5.947e-3
Ru-103	280.47					9.53e-1	5.4370e0	9.59e-3	9.368e-3			7.98e-6	1.060e-3			4.42e-1	1.651e-3
Ru-105	468.85															1.88e-1	1.174e-3
Ru-106						1.16e1		1.15e-1				9.59e-5				1.01e-1	0
Rh-105	46.04															1.86e-1	1.140e-4
Y-90	8.36															2.17e-4	2.416e-8
Y-91	7.31					6.49e-4	9.650e-5	6.65e-6	1.693e-7			5.53e-9	1.915e-8			6.89e-3	6.707e-7
Zr-95	443.51					4.95e-2	4.466e-1	4.99e-4	7.708e-4			4.15e-7	8.718e-5			8.62e-3	5.091e-5
Zr-97	114.89															7.66e-3	1.172e-5
Nb-95	460.41					3.37e-2	3.156e-1	3.58e-4	5.741e-4			2.98e-7	6.499e-5			8.61e-3	5.279e-5
La-140	1427.68					2.04e0	5.9244e1	3.2e-2	1.591e-1			2.53e-5	1.711e-2			8.74e-3	1.662e-4
Ce-141	45.2					1.87e-2	1.719e-2	1.92e-4	3.022e-5			1.59e-7	3.404e-6			8.61e-3	5.182e-6
Ce-143	155.44					2.07e-1	6.545e-1	3.58e-3	1.938e-3			2.8e-6	2.062e-4			7.03e-3	1.455e-5
Ce-144	10.77					5e-1	1.095e-1	5.12e-3	1.920e-4			4.26e-6	2.173e-5			4.89e-3	7.013e-7
Pr-143	2.31															7.43e-3	2.285e-7
Nd-147	78.56															3.42e-3	3.578e-6
Np-239	97.99					2e-1	3.987e-1	2.81e-3	9.590e-4			2.26e-6	1.049e-4			9.1e-2	1.187e-4
Pu-238	.05															3.28e-6	2.18e-12
Pu-239	.05															1.21e-6	8.06e-13
Pu-240	.05															1.21e-6	8.06e-13
Pu-241																1.96e-4	0
Am-241	11.02															9.78e-8	1.44e-11
Cm-242	.06															2.87e-5	2.29e-11
Cm-244	.05															1.32e-6	8.79e-13
		0		4.884e-2	.07%	2.1571e1	22.79%	2.652e-1	.17%	5.4076e0	64.97%	1.987e-4	.03%	4.475e-1	.06%	8.0791e1	.11%
		5.7442e4	4.5531e1	2.2641e3	2.0805e1	4.9161e1	4.6506e2	2.8713e2	1.4841e2	3.6619e3	1.0836e3	2.1112e0	9.6188e1	2.2454e3	2.4327e2	7.5098e4	2.4345e2

APPENDIX B

TIME DEPENDENCY OF TOTAL BODY DOSE-RATE CONVERSION FACTORS

Time After Shutdown (hours)	Dose-Rate Conversion Factor in mrem/hr per $\mu Ci/m^3$ LOCAR	LOCAG	LOCAM
0	0.144	0.365	0.366
1	0.122	0.313	0.314
2	0.105	0.266	0.267
3	0.090	0.225	0.225
4	0.079	0.190	0.190
5	0.070	0.160	0.161
6	0.062	0.136	0.136
Mid-Range Value	0.103	0.251	0.251
Average Value	0.096	0.236	0.237
DCF Based on Total Curies Released in 6 Hours	0.096	0.243	0.243

	AIR IMMERSION DCF (Rem/hr / Ci/m³)	HALF-LIFE (days)	LOCAR[1] PRIMARY SYSTEM INVENTORY (Ci)	t = 0 DCF	t = 1 hr INVENTORY (Ci)	t = 1 hr DCF	t = 2 hr INVENTORY (Ci)	t = 2 hr DCF	t = 3 hr INVENTORY	t = 3 hr DCF	t = 4 hr INVENTORY	t = 4 hr DCF	t = 5 hr INVENTORY	t = 5 hr DCF	t = 6 hr INVENTORY	t = 6 hr DCF
Kr-85	3.03e0	3.95e3	9.17e2	2.916e-1	9.1699e2	3.069e-1	9.1699e2	3.202e-1	9.1698e2	3.318e-1	9.1697e2	3.422e-1	9.1697e2	3.516e-1	9.1696e2	3.602e-1
Kr-85m	96.3	.183	3.41e2	3.4458e0	2.9122e2	3.0981e0	2.4872e2	2.7602e0	2.1241e2	2.4429e0	1.8141e2	2.1516e0	1.5493e2	1.8880e0	1.3231e2	1.6519e0
Kr-87	527.99	.0528	3.2e2	1.7729e1	1.8520e2	1.0802e1	1.0719e2	6.5219e0	6.2035e1	3.9117e0	3.5903e1	2.3348e0	2.0779e1	1.3884e0	1.2026e1	8.232e-1
Kr-88	1313.63	.117	5.97e2	8.2292e1	4.6644e2	6.7687e1	3.6443e2	5.5169e1	2.8473e2	4.4670e1	2.2246e2	3.5992e1	1.7381e2	2.8893e1	1.3580e2	2.3127e1
Xe-133	20.7	5.28	5.54e3	1.2033e1	5.5098e3	1.2599e1	5.4797e3	1.3072e1	5.4499e3	1.3473e1	5.4201e3	1.3819e1	5.3906e3	1.4121e1	5.3612e3	1.4388e1
Xe-135	146.99	.384	1.81e3	2.7918e1	1.6789e3	2.7262e1	1.5573e3	2.6379e1	1.4445e3	2.5357e1	1.3398e3	2.4257e1	1.2428e3	2.3117e1	1.1528e3	2.1968e1
I-131	222.18	8.05	1.92e-1	4.476e-3	1.913e-1	4.696e-3	1.906e-1	4.881e-3	1.899e-1	5.040e-3	1.893e-1	5.179e-3	1.886e-1	5.302e-3	1.879e-1	5.413e-3
I-132	1381.22	.0958	8.95e-1	1.297e-1	6.621e-1	1.010e-1	4.898e-1	7.796e-2	3.623e-1	5.977e-2	2.681e-1	4.560e-2	1.983e-1	3.466e-2	1.467e-1	2.627e-2
I-133	358.61	.875	5.97e-1	2.247e-2	5.776e-1	2.288e-2	5.589e-1	2.310e-2	5.407e-1	2.316e-2	5.232e-1	2.311e-2	5.062e-1	2.297e-2	4.898e-1	2.277e-2
I-134	1600.86	.0366	1.45e0	2.436e-1	6.588e-1	1.165e-1	2.993e-1	5.522e-2	1.360e-1	2.600e-2	6.178e-2	1.218e-2	2.807e-2	5.686e-3	1.275e-2	2.647e-3
I-135	967.27	.28	1.11e0	1.127e-1	1.0012e0	1.070e-1	9.031e-1	1.007e-1	8.146e-1	9.411e-2	7.348e-1	8.754e-2	6.628e-1	8.113e-2	5.979e-1	7.498e-2
Cs-134	929.26	750	3.03e-2	2.955e-3	3.030e-2	3.110e-3	3.030e-2	3.245e-3	3.030e-2	3.362e-3	3.030e-2	3.467e-3	3.029e-2	3.563e-3	3.029e-2	3.650e-3
Cs-136	1305.19	13	3.71e-3	5.081e-4	3.702e-3	5.337e-4	3.694e-3	5.556e-4	3.685e-3	5.745e-4	3.677e-3	5.911e-4	3.669e-3	6.060e-4	3.661e-3	6.195e-4
Cs-137	.95	11000	4.01e-2	3.997e-6	4.010e-2	4.208e-6	4.010e-2	4.390e-6	4.010e-2	4.550e-6	4.010e-2	4.692e-6	4.010e-2	4.821e-6	4.010e-2	4.939e-6
Te-132	126.72	3.25	7.25e-3	9.640e-5	7.186e-3	1.006e-4	7.122e-3	1.040e-4	7.059e-3	1.068e-4	6.997e-3	1.092e-4	6.935e-3	1.112e-4	6.874e-3	1.129e-4
Sr-89	5.03	50.5	5.97e-4	3.151e-7	5.967e-4	3.315e-7	5.963e-4	3.457e-7	5.960e-4	3.580e-7	5.956e-4	3.690e-7	5.953e-4	3.789e-7	5.950e-4	3.880e-7
Sr-90	1.22	11000	5.12e-5	6.555e-9	5.120e-5	6.900e-9	5.120e-5	7.198e-9	5.120e-5	7.460e-9	5.120e-5	7.693e-9	5.120e-5	7.905e-9	5.120e-5	8.098e-9
Ba-140	110.67	12.8	5.54e-2	6.434e-4	5.528e-2	6.758e-4	5.515e-2	7.034e-4	5.503e-2	7.273e-4	5.490e-2	7.484e-4	5.478e-2	7.672e-4	5.466e-2	7.842e-4
Mo-99	95.46	2.8	2.73e-2	2.735e-4	2.702e-2	2.849e-4	2.674e-2	2.942e-4	2.647e-2	3.018e-4	2.620e-2	3.080e-4	2.593e-2	3.132e-4	2.566e-2	3.176e-4
Ru-103	280.47	39.5	3.84e-1	1.130e-2	3.837e-1	1.189e-2	3.834e-1	1.239e-2	3.832e-1	1.283e-2	3.829e-1	1.323e-2	3.826e-1	1.358e-2	3.823e-1	1.390e-2
La-140	1427.68	1.67	1.07e-1	1.603e-2	1.052e-1	1.659e-2	1.034e-1	1.701e-2	1.016e-1	1.732e-2	9.985e-2	1.756e-2	9.814e-2	1.773e-2	9.646e-2	1.785e-2
Ce-144	10.77	284	1.71e-2	1.933e-5	1.710e-2	2.034e-5	1.710e-2	2.122e-5	1.709e-2	2.199e-5	1.709e-2	2.267e-5	1.709e-2	2.329e-5	1.709e-2	2.386e-5
Np-239	97.99	2.35	9.38e-3	9.645e-5	9.265e-3	1.003e-4	9.152e-3	1.034e-4	9.041e-3	1.058e-4	8.930e-3	1.078e-4	8.821e-3	1.094e-4	8.713e-3	1.107e-4
			9.5299e3	1.4425e2	9.0523e3	1.2214e2	8.6774e3	1.0452e2	8.3732e3	9.0430e1	8.1192e3	7.9106e1	7.9021e3	6.9946e1	7.7131e3	6.2488e1

1. LOCAR PRIMARY SYSTEM ACTIVITY from BG&E-EP9, Table 6, column entitled LOCARO (Coolant) 1% Failed Fuel Released Activity (Ci).

	AIR IMMERSION DCF Rem/hr / Ci/m3	HALF-LIFE (days)	LOCAG RELEASED ACTIVITY INVENTORY (Ci)	t = 0 DCF	t = 1 hr INVENTORY (Ci)	t = 1 hr DCF	t = 2 hr INVENTORY (Ci)	t = 2 hr DCF	t = 3 hr INVENTORY	t = 3 hr DCF	t = 4 hr INVENTORY	t = 4 hr DCF	t = 5 hr INVENTORY	t = 5 hr DCF	t = 6 hr INVENTORY	t = 6 hr DCF
Kr-85	3.03	3950	1.94e4	4.934e-3	1.9400e4	5.613e-3	1.9400e4	6.216e-3	1.9400e4	6.742e-3	1.9399e4	7.200e-3	1.9399e4	7.599e-3	1.9399e4	7.948e-3
Kr-85m	96.3	.183	8.3e5	6.7088e0	7.0885e5	6.5186e0	6.0538e5	6.1647e0	5.1701e5	5.7109e0	4.4154e5	5.2085e0	3.7709e5	4.6946e0	3.2205e5	4.1936e0
Kr-87	527.99	.0528	1.62e6	7.1792e1	9.3758e5	4.7273e1	5.4263e5	3.0297e1	3.1405e5	1.9020e1	1.8176e5	1.1755e1	1.0519e5	7.1803e0	6.0882e4	4.3466e0
Kr-88	1313.63	.117	2.35e6	2.5911e2	1.8361e6	2.3032e2	1.4345e6	1.9927e2	1.1208e6	1.6888e2	8.7567e5	1.4090e2	6.8416e5	1.1619e2	5.3454e5	9.4950e1
Xe-133	20.7	5.28	5.88e6	1.0216e1	5.8479e6	1.1560e1	5.8160e6	1.2731e1	5.7843e6	1.3734e1	5.7528e6	1.4587e1	5.7214e6	1.5311e1	5.6902e6	1.5927e1
Xe-135	146.99	.384	1.18e6	1.4558e1	1.0945e6	1.5363e1	1.0152e6	1.5780e1	9.4170e5	1.5877e1	8.7348e5	1.5727e1	8.1021e5	1.5396e1	7.5152e5	1.4937e1
I-131	222.18	8.05	3.92e3	7.310e-2	3.9060e3	8.287e-2	3.8920e3	9.144e-2	3.8780e3	9.883e-2	3.8642e3	1.052e-1	3.8503e3	1.106e-1	3.8365e3	1.153e-1
I-132	1381.22	.0958	5.53e3	6.411e-1	4.0910e3	5.396e-1	3.0264e3	4.420e-1	2.2388e3	3.547e-1	1.6562e3	2.802e-1	1.2252e3	2.188e-1	9.0641e2	1.693e-1
I-133	358.61	.875	7.84e3	2.360e-1	7.5855e3	2.598e-1	7.3393e3	2.783e-1	7.1010e3	2.921e-1	6.8705e3	3.018e-1	6.6475e3	3.082e-1	6.4317e3	3.119e-1
I-134	1600.86	.0366	8.76e3	1.1770e0	3.9799e3	6.084e-1	1.8082e3	3.061e-1	8.2151e2	1.508e-1	3.7324e2	7.319e-2	1.6957e2	3.509e-2	7.7042e1	1.668e-2
I-135	967.27	.28	6.91e3	5.610e-1	6.2329e3	5.757e-1	5.6222e3	5.751e-1	5.0713e3	5.627e-1	4.5744e3	5.420e-1	4.1261e3	5.160e-1	3.7218e3	4.868e-1
Cs-134	929.26	750	8.64e2	6.739e-2	8.6397e2	7.667e-2	8.6393e2	8.489e-2	8.6390e2	9.208e-2	8.6387e2	9.833e-2	8.6383e2	1.038e-1	8.6380e2	1.085e-1
Cs-136	1305.19	13	3.46e2	3.790e-2	3.4523e2	4.303e-2	3.4447e2	4.754e-2	3.4370e2	5.146e-2	3.4294e2	5.483e-2	3.4218e2	5.774e-2	3.4142e2	6.026e-2
Cs-137	.95	11000	5.42e2	4.322e-5	5.4200e2	4.917e-5	5.4200e2	5.445e-5	5.4200e2	5.906e-5	5.4199e2	6.307e-5	5.4199e2	6.656e-5	5.4199e2	6.962e-5
Te-132	126.72	3.25	2.77e1	2.946e-4	2.7455e1	3.322e-4	2.7212e1	3.646e-4	2.6971e1	3.920e-4	2.6733e1	4.150e-4	2.6496e1	4.341e-4	2.6262e1	4.500e-4
Sr-89	5.03	50.5	0	0	0	0	0	0	0	0	0	0	0	0	0	0
Sr-90	1.22	11000	0	0	0	0	0	0	0	0	0	0	0	0	0	0
Ba-140	110.67	12.8	0	0	0	0	0	0	0	0	0	0	0	0	0	0
Mo-99	95.46	2.8	0	0	0	0	0	0	0	0	0	0	0	0	0	0
Ru-103	280.47	39.5	0	0	0	0	0	0	0	0	0	0	0	0	0	0
La-140	1427.68	1.67	0	0	0	0	0	0	0	0	0	0	0	0	0	0
Ce-144	10.77	284	0	0	0	0	0	0	0	0	0	0	0	0	0	0
Np-239	97.99	2.35	0	0	0	0	0	0	0	0	0	0	0	0	0	0
			1.1914e7	3.6518e2	1.0472e7	3.1323e2	9.4567e6	2.6607e2	8.7182e6	2.2483e2	8.1637e6	1.8965e2	7.7352e6	1.6013e2	7.3953e6	1.3563e2

LOCAG RELEASED ACTIVITY INVENTORY from BG&E-EP9, Table 6, column entitled LOCAGO (Gap) Released Activity (Ci).

	AIR IMMERSION DCF (Rem/hr / Ci/m³)	HALF-LIFE (days)	LOCAM RELEASED ACTIVITY INVENTORY (Ci)	t = 0 DCF	t = 1 hr INVENTORY (Ci)	t = 1 hr DCF	t = 2 hr INVENTORY (Ci)	t = 2 hr DCF	t = 3 hr INVENTORY	t = 3 hr DCF	t = 4 hr INVENTORY	t = 4 hr DCF	t = 5 hr INVENTORY	t = 5 hr DCF	t = 6 hr INVENTORY	t = 6 hr DCF
Kr-85	3.03	3950	6.45e5	4.910e-3	6.4500e5	5.588e-3	6.4499e5	6.189e-3	6.4499e5	6.714e-3	6.4498e5	7.170e-3	6.4498e5	7.568e-3	6.4497e5	7.916e-3
Kr-85m	96.3	.183	2.77e7	6.7016e0	2.3657e7	6.5136e0	2.0204e7	6.1612e0	1.7254e7	5.7082e0	1.4736e7	5.2064e0	1.2585e7	4.6929e0	1.0748e7	4.1923e0
Kr-87	527.99	.0528	5.42e7	7.1895e1	3.1369e7	4.7354e1	1.8155e7	3.0355e1	1.0507e7	1.9058e1	6.0811e6	1.1780e1	3.5195e6	7.1957e0	2.0369e6	4.3561e0
Kr-88	1313.63	.117	7.84e7	2.5874e2	6.1254e7	2.3006e2	4.7858e7	1.9908e2	3.7391e7	1.6874e2	2.9214e7	1.4080e2	2.2825e7	1.1610e2	1.7833e7	9.4885e1
Xe-133	20.7	5.28	1.96e8	1.0193e1	1.9493e8	1.1537e1	1.9387e8	1.2708e1	1.9281e8	1.3711e1	1.9176e8	1.4563e1	1.9071e8	1.5287e1	1.8967e8	1.5903e1
Xe-135	146.99	.384	3.92e7	1.4476e1	3.6360e7	1.5281e1	3.3727e7	1.5699e1	3.1284e7	1.5797e1	2.9017e7	1.5649e1	2.6915e7	1.5320e1	2.4966e7	1.4864e1
I-131	222.18	8.05	1.96e5	1.094e-1	1.9530e5	1.241e-1	1.9460e5	1.369e-1	1.9390e5	1.480e-1	1.9321e5	1.575e-1	1.9252e5	1.656e-1	1.9183e5	1.726e-1
I-132	1381.22	.0958	2.77e5	9.612e-1	2.0492e5	8.092e-1	1.5159e5	6.631e-1	1.1214e5	5.321e-1	8.2962e4	4.204e-1	6.1373e4	3.283e-1	4.5402e4	2.540e-1
I-133	358.61	.875	3.92e5	3.532e-1	3.7928e5	3.889e-1	3.6696e5	4.167e-1	3.5505e5	4.374e-1	3.4353e5	4.520e-1	3.3237e5	4.616e-1	3.2158e5	4.671e-1
I-134	1600.86	.0366	4.38e5	1.7616e0	1.9900e5	9.108e-1	9.0410e4	4.583e-1	4.1076e4	2.259e-1	1.8662e4	1.096e-1	8.4786e3	5.256e-2	3.8521e3	2.498e-2
I-135	967.27	.28	3.46e5	8.408e-1	3.1210e5	8.631e-1	2.8152e5	8.623e-1	2.5393e5	8.438e-1	2.2905e5	8.129e-1	2.0661e5	7.739e-1	1.8636e5	7.301e-1
Cs-134	929.26	750	1.73e4	4.039e-2	1.7299e4	4.596e-2	1.7299e4	5.090e-2	1.7298e4	5.522e-2	1.7297e4	5.897e-2	1.7297e4	6.224e-2	1.7296e4	6.510e-2
Cs-136	1305.19	13	6.91e3	2.266e-2	6.8947e3	2.573e-2	6.8794e3	2.843e-2	6.8641e3	3.078e-2	6.8489e3	3.280e-2	6.8337e3	3.454e-2	6.8185e3	3.605e-2
Cs-137	.95	11000	1.08e4	2.578e-5	1.0800e4	2.934e-5	1.0800e4	3.249e-5	1.0800e4	3.525e-5	1.0800e4	3.764e-5	1.0800e4	3.973e-5	1.0800e4	4.156e-5
Te-132	126.72	3.25	8.3e4	2.642e-2	8.2266e4	2.981e-2	8.1538e4	3.272e-2	8.0817e4	3.518e-2	8.0102e4	3.724e-2	7.9394e4	3.896e-2	7.8691e4	4.039e-2
Sr-89	5.03	50.5	1.52e4	1.921e-4	1.5191e4	2.185e-4	1.5183e4	2.418e-4	1.5174e4	2.622e-4	1.5165e4	2.799e-4	1.5157e4	2.952e-4	1.5148e4	3.086e-4
Sr-90	1.22	11000	5.97e2	1.830e-6	5.9700e2	2.082e-6	5.9700e2	2.306e-6	5.9700e2	2.502e-6	5.9699e2	2.672e-6	5.9699e2	2.820e-6	5.9699e2	2.950e-6
Ba-140	110.67	12.8	7.37e4	2.049e-2	7.3534e4	2.327e-2	7.3368e4	2.571e-2	7.3203e4	2.783e-2	7.3038e4	2.966e-2	7.2873e4	3.123e-2	7.2709e4	3.259e-2
Mo-99	95.46	2.8	3.69e4	8.850e-3	3.6521e4	9.968e-3	3.6147e4	1.093e-2	3.5776e4	1.173e-2	3.5409e4	1.240e-2	3.5046e4	1.295e-2	3.4686e4	1.341e-2
Ru-103	280.47	39.5	1.77e3	1.247e-3	1.7687e3	1.418e-3	1.7674e3	1.570e-3	1.7661e3	1.702e-3	1.7648e3	1.816e-3	1.7635e3	1.915e-3	1.7623e3	2.002e-3
La-140	1427.68	1.67	3.69e1	1.324e-4	3.6267e1	1.480e-4	3.5646e1	1.612e-4	3.5035e1	1.718e-4	3.4434e1	1.804e-4	3.3844e1	1.871e-4	3.3264e1	1.924e-4
Ce-144	10.77	284	1.96e1	5.303e-7	1.9598e1	6.035e-7	1.9596e1	6.683e-7	1.9594e1	7.250e-7	1.9592e1	7.742e-7	1.9590e1	8.170e-7	1.9588e1	8.545e-7
Np-239	97.99	2.35	3.69e2	9.084e-5	3.6449e2	1.021e-4	3.6004e2	1.117e-4	3.5565e2	1.197e-4	3.5130e2	1.263e-4	3.4701e2	1.317e-4	3.4277e2	1.360e-4
			3.9804e8	3.6616e2	3.4975e8	3.1399e2	3.1578e8	2.6670e2	2.9109e8	2.2537e2	2.7256e8	1.9013e2	2.5824e8	1.6057e2	2.4689e8	1.3605e2

LOCAM RELEASED ACTIVITY INVENTORY from BG&E-EP9, Table 6, column entitled LOCAMO (Melt) Released Activity (Ci).

APPENDIX C

EVALUATION OF LIMITING DOSE-RATE

TOTAL BODY VERSUS SKIN

To establish appropriate Radioactivity Release EALs, one must determine which dose-rate limit of Technical Specification 3.11.2.1.a. is more restrictive, the Total Body or Skin dose-rate limit. To do this, it is sufficient to compute the Total Body DCF and the Skin DCF for each accident type and to compare the ratio of these values to the ratio of the Total Body and Skin dose-rate Technical Specification limits.

For simplicity, and to assure consistency of inputs, these calculations are made using the ODCM values for dose-rate conversion factors. The ODCM contains all the required data in consistent units and in a convenient format. Skin dose factors are not readily available in the ERPIPs or RADDOSE. Since the intent is to determine the limiting dose rate, it is preferrable to use one document which contains all the required inputs.

The following Attachment 1 from the ODCM presents values related to the Total Body and Skin dose factors for the noble gases. The ODCM does not contain skin dose factors for iodines or particulates via either airborne or liquid pathways. Page 24 of the ODCM also presents formulae for determining the effective total body dose factor due to gamma emissions and the effective skin dose factor due to beta and gamma emissions. Using these values and equations, effective Total Body and Skin DCFs may be computed for each noble gas and for the noble gas distributions of each postulated accident.

Table C-1 presents the Total Body and Skin DCFs and the Skin to Total Body DCF ratio for all the noble gases. By inspection of Table C-1 it is apparent that the noble gases will not cause a Skin dose rate that is 6 times the Total Body dose rate except when Kr-85 is dominant. Because of its long half-life, Kr-85 is most significant in those accidents which include some decay of the source term prior to release, e.g., the WGDTR and the FHI. Table C-2 shows the calculated effective Total Body and Skin DCFs for the source terms associated with WGDTR, FHI, and LOCAR. It is clear from this Table that the Skin DCF is not 6 times the Total Body DCF for any of these accidents. It is therefore concluded that the Total Body dose-rate limit is more restrictive than the Skin dose-rate limit of Technical Specification 3.11.2.1.

TABLE C-1
NOBLE GAS DOSE-RATE CONVERSION FACTORS

Isotope	mrem/hr per $\mu Ci/m^3$ Total Body DCF	Skin DCF	DCF(Skin) / DCF(TB)
Kr-85	1.84E-3	1.55E-1	84.24
Kr-85m	1.34E-1	3.21E-1	2.40
Kr-87	6.76E-1	1.89E 0	2.80
Kr-88	1.68E 0	2.18E 0	1.30
Xe-133	3.36E-2	7.93E-2	2.36
Xe-135	2.07E-1	4.53E-1	2.19

TABLE C-2
EFFECTIVE TOTAL BODY AND SKIN DCFs
FOR SPECIFIC ACCIDENTS

Isotope	TB DCF mrem/hr	SKIN DCF per $\mu Ci/m^3$	WGDTR (Ci)	FHI (Ci)	LOCAR (Ci)
Kr-85	.00184	.155	262	11	.229
Kr-85m	.134	.321	441	5.49E-3	.0551
Kr-87	.676	1.89	240	7.33E-15	.0234
Kr-88	1.68	2.18	769	2.56E-5	.0779
Xe-133	.0336	.0793	53500	2250	1.36
Xe-135	.207	.453	2230	2.97	.365
Total			57442	2264	2.1104
Effective DCF (Skin)			.132	.080	.256
Effective DCF (Total Body)			.0657	.0337	.131
DCF(Skin) / DCF(TB)			2.01	2.38	1.96

APPENDIX D

EVALUATION OF MAXIMUM ORGAN DOSE RATES

Technical Specification 3.11.2.1.b. limits the dose rate to any organ, from I-131 and all radionuclides in particulate form with half-life longer than 8 days, to not more than 1500 mrem per year (0.17 mrem/hour). The following evaluation was done to determine whether any Radioactivity Release EAL should be established with respect to this Technical Specification limit. Based on this evaluation and arguments presented herein, all the Radioactivity Release EALs are set based on the Total Body dose-rate limit.

For the Calvert Cliffs Nuclear Power Plant, the Child Thyroid, Inhalation pathway, is the limiting maximum organ dose. (Ref: ODCM, p.16) Radioiodines are the predominant contributors to the Child Thyroid, Inhalation dose rate, and only the radioiodines are considered herein. (Te-132 may be neglected with no loss of accuracy.) However, all the radioiodines - not just I-131 - are included in the evaluation. This is considered appropriate for accident evaluations, whereas the Technical Specification is meant to limit normal releases.

A way to evaluate whether the maximum organ dose rate is limiting is to compare Maximum Organ DCF / Total Body DCF ratio to the fraction of the release which contributes to the maximum organ dose rate. For example, if the Maximum Organ DCF is 4 times the Total Body DCF, and the nuclides contributing to the Maximum Organ dose rate comprise 1/4 of the total release, then the Maximum Organ dose rate and the Total Body dose rate would be identical. (A radionuclide may contribute to both the Total Body and the Maximum Organ dose rates simultaneously.) The Maximum Organ dose-rate limit is three times the Total Body dose-rate limit. Therefore, nuclides contributing to the Maximum Organ dose rate would have to exceed 3/4 of the release before the Maximum Organ dose-rate limit would be more restrictive than the Total Body dose-rate limit. Stated another way, the Maximum Organ dose-rate limit will not control unless the fraction of the release which contributes to the Maximum Organ dose exceeds 3 times the Total Body DCF / Maximum Organ DCF ratio.

Table D-1 presents the data used to calculate effective Child Thyroid DCFs for each accident type contained in BG&E-EP9, along with the calculated result.

Table D-2 presents the comparison of the parameters used to determine when the Child Thyroid, Inhalation dose-rate limit is more restrictive than the Total Body dose-rate limit. Some explanation is required for proper interpretation of the data.

The Fuel Handling Incident results indicate that the Child Thyroid should be limiting. Due to the uncertainty in iodine washout in the Spent Fuel Pool, the EAL for this incident is based on the Total Body dose-rate limit. This is discussed in detail in the section presenting the calculation of the Fuel Handling Monitor EAL.

The Steam Line Break (SLB) consists predominantly of an iodine release, since noble gases are continuously removed from the steam cycle through the condenser off-gas system. However, the Technical Specification limiting dose equivalent I-131 in the steam system effectively limits the consequences of a SLB. Also, a SLB in the Turbine Building results in essentially an unmonitored release. Therefore, no Radioactivity Release EAL exists for this event.

The Steam Generator Tube Rupture (SGTRR or SGTRG) source terms presented in BG&E-EP9 shows that the Child Thyroid dose-rate limit controls in each accident. However, Radioactivity Release EAL for these accidents must consider both system design and reduction phenomena. Until such time as the Main Steam Isolation Valves close, releases from a SGTR will be via the Condenser air removal system, which exhausts to the Main Vent. There will be very effective iodine removal in the condenser. In fact, NUREG-1228 assumes complete iodine and particulate removal in the Condenser. Reduction of the radioiodines will leave only the noble gases in the release, so the Total Body dose-rate limit should control for these accidents also.

The Total Body dose-rate limit is most restrictive for the LOCAR. For the LOCAG release the Child Thyroid is indicated as the controlling dose-rate limit, but consideration should be given to the most likely leakage pathways. For this accident, the most likely leakage is through containment penetrations into the Penetration Rooms. In an accident situation, the ventilation of these rooms draws air out of the rooms and through charcoal filters before discharging the exhaust to the Main Vent. This additional iodine reduction would result in the Total Body dose-rate limit being most restrictive.

The LOCAM is purposely analyzed to be the worst case scenario. It is presumed that this accident leads to failure of the containment, so leakage from the containment would not necessaril y be filtered. In such circumstance, the Child Thyroid dose-rate limit would be controlling. However, there is no monitor which can be used for a Radioactivity Release EAL in this situation. Such an EAL would be superfluous in any case, as a General Emergency will have been indicated for other reasons already.

Table D-1
EFFECTIVE CHILD THYROID DOSE-RATE CONVERSION FACTORS
(Rem/hour per Curie/cu.meter)

Isotope	Child Thyroid DCF	Curies released from BG&E-EP9						
		FHI	SLB	SGTRR	SGTRG	LOCAR	LOCAG	LOCAM
I-131	1.96e6	8.61e-2	5.15e0	5.76e-2	1.18e1	4.75e-5	9.69e-1	4.84e1
I-132	1.15e4	5.9e-11	2.44e0	2.69e-1	1.66e1	1.03e-4	6.39e-1	3.2e1
I-133	3.31e5	2.07e-2	8.55e0	1.79e-1	2.35e1	1.35e-4	1.78e0	8.89e1
I-134	3.11e3	5.3e-26	1.55e0	4.35e-1	2.63e1	7.59e-5	4.58e-1	2.29e1
I-135	5.74e4	1.17e-4	7.49e0	3.33e-1	2.07e1	2.07e-4	1.29e0	6.44e1
Total		1.069e-1	2.518e1	1.2736e0	9.89e1	5.684e-4	5.136e0	2.566e2
Child Thyroid DCF		1.6425e6	5.3165e5	1.5366e5	3.2727e5	2.6581e5	5.0063e5	5.0049e5

TABLE D-2
EVALUATION OF LIMITING DOSE-RATE

PARAMETER	Accident Type WGDTR	FHI	SLB	SGTRR	SGTRG	LOCAR	LOCAG	LOCAM
Total Body DCF	4.55e1	2.08e1	4.65e2	1.48e2	1.08e3	9.62e1	2.43e2	2.43e2
Child Thyroid DCF	0	1.64e6	5.32e5	1.54e5	3.27e5	2.66e5	5.01e5	5e5
3*(TB DCF/CT DCF)	Indeter	3.805e-5	2.622e-3	2.883e-3	9.908e-3	1.085e-3	1.455e-3	1.458e-3
Total Release	5.74e4	2.27e3	4.77e1	2.87e2	3.67e3	2.11e0	2.25e3	7.52e4
Total Iodines	0	1.07e-1	2.52e1	1.27e0	9.89e1	5.69e-4	5.13e0	2.57e2
Fraction Iodines	0	4.714e-5	5.283e-1	4.425e-3	2.695e-2	2.697e-4	2.28e-3	3.418e-3
Limiting Dose Rate	TB	CT	CT	CT	CT	TB	CT	CT

DOSE FACTORS FOR NOBLE GASES

Nuclide	Dose to People +		Dose to Air #	
	Gamma-Body K(i)	Beta-Skin L(i)	Gamma M(i)	Beta N(i)
AR-41	8.840E+03	2.690E+03	9.300E+03	3.280E+03
KR-83M	8.000E-02	0.000E+00	1.930E+01	2.880E+02
KR-85	1.610E+01	1.340E+03	1.720E+01	1.950E+03
KR-85M	1.170E+03	1.460E+03	1.230E+03	1.970E+03
KR-87	5.920E+03	9.730E+03	6.170E+03	1.030E+04
KR-88	1.470E+04	2.370E+03	1.520E+04	2.930E+03
KR-89	1.660E+04	1.010E+04	1.730E+04	1.060E+04
KR-90	1.560E+04	7.290E+03	1.630E+04	7.830E+03
XE-131M	9.150E+01	4.760E+02	1.560E+02	1.110E+03
XE-133	2.940E+02	3.060E+02	3.530E+02	1.050E+03
XE-133M	2.510E+02	9.940E+02	3.270E+02	1.480E+03
XE-135	1.810E+03	1.860E+03	1.920E+03	2.460E+03
XE-135M	3.120E+03	7.110E+02	3.360E+03	7.390E+02
XE-137	1.420E+03	1.220E+04	1.510E+03	1.270E+04
XE-138	8.830E+03	4.130E+03	9.210E+03	4.750E+03

+ -- mrem/yr per uCi/cu.m

-- mrad/yr per uCi/cu.m

6 October 1987

To: G. C. Rudigier
From: E. T. Reimer

Subj: Calvert Cliffs Nuclear Power Plant Accident Source Terms, BGE-EP9

A comparison was performed between subject core inventories of radionuclides and those determined by use of NUREG-0771 Table 3.2 values normalized to 2700 MWt full power. Table 1 illustrates this comparison and the percent error that the subject values vary from the NUREG-0771 values. It is also obvious that the subject study did not include all the radiologically important isotopes and the inventory for reactors that NRC, etc., expect to be potentially released to the atmosphere during an reactor accident. If the subject core inventories are to be accepted for use in emergency response assessment for offsite releases, the inventories should contain all the radiologically important isotope inventories mentioned in NUREG-0771, and these latter inventories placed in ERPIP Appendix C.1 for potential use in offsite dose assessment.

Should you have any questions or comments concerning this matter, please contact me.

Eugene T. Reimer
Plant Health Physicist

Attachment

Copy to RSS staff routing

EPU Received ~~10/6/88~~ 10/6/89 GK
Sent
FOLLOW UP
FILE NO. 10.3.1 DOSE ASST. SYS.

TABLE 1. COMPARISON BETWEEN NUREG-0771 AND NUREG-1226 DERIVED CORE INVENTORIES (MCi)

Group/Radionuclide	Full Power Core Inventory NUREG-0771	NUREG-1226	NUREG-1226 % ERROR
NOBLE GASES			
Krypton-85	0.455	0.645	41.76
Krypton-85m	19.719	27.700	40.47
Krypton-97	38.680	54.200	40.12
Krypton-88	55.365	78.400	41.61
Xenon-133	138.792	196.000	41.22
Xenon-135	28.062	39.200	39.69
Iodine-131	69.017	97.900	41.85
Iodine-132	97.837	138.000	41.05
Iodine-133	138.792	196.000	41.22
Iodine-134	154.719	219.000	41.55
Iodine-135	122.107	173.000	41.68
ALKALI METALS			
Rubidium-86	0.021		
Cesium-134	6.143	8.640	40.65
Cesium-136	2.427	3.460	42.56
Cesium-137	3.868	5.420	40.12
TELLURIUM-ANTIMONY			
Tellurium-127	4.778		
Tellurium-127m	0.910		
Tellurium-129	25.028		
Tellurium-129m	4.323		
Tellurium-131m	10.618		
Tellurium-132	97.837	138.000	41.05
Antimony-127	5.006		
Antimony-129	26.545		
ALKALINE EARTHS			
Strontium-89	76.601	108.000	40.99
Strontium-90	3.034	4.260	40.41
Strontium-91	89.494		
Barium-140	130.449	184.000	41.05
NOBLE METALS & COBAL			
Cobalt-58	0.637		
Cobalt-60	0.235		
Molybdenum-99	130.449	184.000	41.05
Technetium-99m	114.522		
Ruthenium-103	89.494	127.000	41.91
Ruthenium-105	58.399		
Ruthenium-106	20.478	28.800	40.64
Rhodium-105	40.197		
RARE EARTHS, REFACTORY OXIDES & TRANSURANICS			
Yttrium-90	3.185		

Yttrium-91	97.837		
Zirconium-95	122.107		
Zirconium-97	122.107		
Niobium-95	122.107		
Lanthanum-140	130.449	184.000	41.05
Cerium-141	122.107		
Cerium-143	106.180		
Cerium-144	69.017	97.900	41.85
Praseodymium-143	106.180		
Neodymium-147	49.298		
Neptunium-239	1365.169	1840.000	34.78
Plutonium-238	0.046		
Plutonium-239	0.017		
Plutonium-240	0.017		
Plutonium-241 (Feeds	2.806		
Americium-241	0.001		
Curium-242	0.410		
Curium-244	0.019		

Westinghouse
Electric Corporation

Nuclear Instrumentation and
Control Department

1111 Schilling Road
Hunt Valley Maryland 21031
(301) 667 1000

June 26, 1981

Baltimore Gas & Electric Co.
Gas & Electric Building
Baltimore, Maryland 21203

ATTN: Mr. Boyd Wylie - Engineering Dept.

Subject: Request by Pete Crinigan for various isotope response sensitivity for the monitors at Calvert Cliffs Nuclear Plant.

Dear Boyd:

As requested by Mr. Pete Crinigan of your office several weeks ago, we have accumulated various reference documents which should give you the information you need. We have included several detailed sensitivity curves similar to the originals in your RMS manual, and in addition, we have included three tables giving the sensitivities of the process monitors to various isotopes. The tables include:

1. Noble Gas Monitors using 912NB3 G-M tubes and 6 inches of lead shielding.
2. Liquid Monitors using 6S4/2 Scintillation tubes and 5.5 or 7.5 inches of lead shielding. According to our records, your liquid has 7.5 inches of shielding. Thus, you should use information under "Note 1".
3. Airborne Monitors using 6S4/2 Scintillation tubes and 4.5 inches of lead shielding.

We regret the delay in forwarding this information but will be glad to assist you in the future if possible.

Sincerely,

C. H. Griesacker
RMS Proj. Engr.
301-667-5115 x3108

car

CC: P. Crinigan

RESPONSE OF THE WESTINGHOUSE NICD'S GAS MONITOR S

TO VARIOUS ISOTOPES

NON-OPERATING CO
FOR INFORMATION C

Isotope	Vents/Purges Note 1 $CPM/10^{-6}$ u Ci/Cc	Note 2 WGDT $CPM/10^{-6}$ u Ci/Cc
Br^{84}	236	441
Rb^{88}	242	453
Rb^{89}	150	279
Sr^{89}	129	241
Sr^{90}	17.3	32
Sr^{91}	159	297
Sr^{92}	32	59
Y^{90}	170	316
Y^{91}	133	249
Y^{92}	213	399
Zr^{95}	5	8.5
Nb^{95}	1	~ 1
Mo^{99}	106	198
I^{131}	26	48
I^{132}	167	309
I^{133}	112	209
I^{134}	180	335
I^{135}	125	233
Te^{132}	1	~1
Te^{134}	< 1	~ 1
Cs^{134}	34	62
Cs^{136}	6	8.6
Cs^{137}	20	36
Cs^{138}	207	386
Ba^{140}	83	154
La^{140}	131	243
Ce^{144}	1	~ 1
Pr^{144}	195	365

...ope	CPM/10^{-6} u Ci/Cc	... CPM / u Ci/Cc
Kr^{85}	35	8000
Kr^{85m}	55	
Kr^{87}	218	23000
Kr^{88}	189	15000
Xe^{133}	1.87	300 (includes XE-13[illegible] and XE-133m)
Xe^{133m}	<1	
Xe^{135}	70	5000
Xe^{135m}	1	
Xe^{138}	173	
Mn^{54}	1	
Mn^{56}	191	
Co^{58}	1	
Co^{60}	200	
Fe^{59}	11	
Tc^{99m}	1	
Cr^{51}	<<1	

Used on: Note 1 - 1-RE-1752, 2RE-1752, 1-RE-5406, 2RE-5406
0-RE-5420, 0-RE5425, 0-RE-5350, 1-RE-541-, 2RE-54
Note 2 - 0-RE-2191

Theoretical response calculated from the known responses of the calibratio[illegible] isotopes Kr^{85} and Xe^{133}.

TABLE 1

THEORETICAL RESPONSE OF WESTINGHOUSE OFF-LINE LIQUID MONITORS TO VARIOUS ISOTOPES.

Isotope	Response 10^6 CPM/uCi/cc
Br-84	3.03
Rb-88	.94
Rb-89	4.46
Sr-89	0
Sr-90	0
Sr-91	2.06
Sr-92	2.50
Y-90	4.86
Y-91	.08
Y-92	.61
Zr-95	2.53
Nb-95	2.58
Mo-99	.52
I-132	10.2
I-133	2.32
I-134	5.74
I-135	3.61
Te-132	2.32
Te-134	1.03
Cs-134	5.83
Cs-136	7.04
Cs-137	2.20
Cs-138	4.02
Ba-140	1.16
La-140	4.86
Ce-144	0
Pr-144	.066
Kr-85	.01
Kr-85m	.36

TABLE 1 (Continued)

Isotope	Response 10^6 CPM/uCi/cc
Kr-87	3.48
Kr-88	3.87
Xe-133	0
Xe-133m	.36
Xe-135	3.30
Xe-135m	2.06
Xe-1[illegible]	?
Mn-54	2.58
Mn-56	3.69
Co-58	3.38
Co-60	5.16
Fe-59	2.65
Tc-99m	0
Cr-51	.23

ATTACHMENT 9.2.2

RMS RESPONSE FACTORS

(cpm/μCi/cc)

ISOTOPE	COLUMN 1 MV NOBLE GAS (RIC-5415)	COLUMN 2 MV GASEOUS (RI-5415)	COLUMN 3 WASTE GAS DISCHARGE (RI-2191)
Ba-140	1.8	8.3E+07	1.5E+08
Ce-144	.95	1.0E+06	1.0E+06
Co-58	1.6	1.0E+06	1.0
Co-60	2.2	2.0E+08	1.0
Cs-134	1.3	3.4E+07	6.2E+07
Cs-137	1.4	2.0E+07	3.6E+07
Cs-138	2.2	2.1E+08	3.9E+08
I-131	1.5	2.6E+07	4.8E+07
I-132	2.1	1.7E+08	3.1E+08
I-133	2.0	1.1E+08	2.1E+08
I-135	2.0	1.3E+08	2.3E+08
Kr-85	1.7	3.5E+07	8.0E+03
Kr-85m	2.4E-8	5.5E+07	1.0
Kr-87	2.2	2.2E+08	2.3E+04
Kr-88	2.0	1.9E+08	1.5E+04
La-140	2.1	1.3E+08	2.4E+08
Rb-88	2.2	2.4E+08	4.5E+08
Xe-133	1.0	1.9E+06	3.2E+02
Xe-133m	2.4E-8	1.0E+06	1.0
Xe-135	1.9	7.0E+07	5.0E+03
Xe-135m	2.4E-8	1.0E+06	1.0

NOTE: Responses for isotopes not listed above may be obtained from the GSC.

ED

CHARLES H. CRUSE
Vice President
Nuclear Energy

Baltimore Gas and Electric Company
Calvert Cliffs Nuclear Power Plant
1650 Calvert Cliffs Parkway
Lusby, Maryland 20657
410 495-4455

BGE

March 6, 1997

U. S. Nuclear Regulatory Commission
Washington, DC 20555

ATTENTION: Document Control Desk

SUBJECT: Calvert Cliffs Nuclear Power Plant
Unit Nos. 1 & 2; Docket Nos. 50-317 & 50-318
Independent Spent Fuel Storage Facility Docket No. 72-8
Revision to Emergency Action Levels Technical Basis Document

REFERENCE: (a) Letter from Mr. D. G. McDonald, Jr. (NRC) to Mr. R. E. Denton (BGE), dated April 12, 1994, Emergency Action Levels, Calvert Cliffs Nuclear Power Plant Units 1 and 2 (TAC Nos. M87080 and M87081)

Enclosed for your information is Revision 5 to the Calvert Cliffs Nuclear Power Plant Emergency Action Levels Technical Basis Document. The Technical Basis Document meets Regulatory Guide 1.101 criteria (i.e., NUMARC/NESP-007, Methodology for Development of Emergency Action Levels), and has been approved by the NRC (Reference a).

The Emergency Action Level changes that will be effected by Revision 5 of the Technical Basis Document will be incorporated in Revision 18, Change 9, of Calvert Cliffs Emergency Response Plan Implementation Procedure 3.0, "Immediate Actions." Emergency Response Plan Implementation Procedure 3.0, Revision 18, Change 9, will be implemented approximately 90 days from the date of this letter, but not earlier than 45 days. This implementation date allows for staff training and for other internal document processing requirements. The Emergency Action Level Technical Basis Document, Revision 5, changes have been reviewed with appropriate State and local agencies.

Should you have questions regarding this matter, please contact Mr. T. E. Forgette, Director-Emergency Planning, at (410) 495-4996.

Very truly yours,

CHC/GT/dlm

Enclosure: Revision 5 to the Calvert Cliffs Nuclear Power Plant Emergency Action Levels Technical Basis Document

cc: **(Without Enclosure)**
D. A. Brune, Esquire
J. E. Silberg, Esquire
Director, Project Directorate I-1, NRC
A. W. Dromerick, NRC
H. J. Miller, NRC
Resident Inspector, NRC
R. I. McLean, DNR
J. H. Walter, PSC

CN600002

bcc: **(Without Enclosure)**
G. C. Creel
R. E. Denton
P. E. Katz
P. G. Chabot
J. R. Lemons
R. P. Heibel
T. J. Ćamilleri
K. R. Eser
S. B. Haggerty
L. A. Larragoite
K. R. Neddenien
OSSRC Secretary
M. G. Polak
Correspondence Evaluator

(With Enclosure)
G. Tesfaye
T. E. Forgette
File 35.03
Electronic Docket File

CHC/GT/gt/dlm

NRC 97-014

CN600003

ENCLOSURE

CALVERT CLIFFS NUCLEAR POWER PLANT

EMERGENCY ACTION LEVELS TECHNICAL BASIS DOCUMENT

REVISION 5

CN600004

Baltimore Gas & Electric Company
Docket Nos. 50-317 and 50-318
Independent Spent Fuel Storage Facility Docket No. 72-8
March 6, 1997

ENCLOSURE

CALVERT CLIFFS NUCLEAR POWER PLANT
EMERGENCY ACTION LEVELS TECHNICAL BASIS DOCUMENT
REVISION 5

Only the revised pages of the Technical Basis Document are included in this transmittal. New words/information are identified by bold letters and revision bars. Deleted words/information are identified by line-out and revision bars. Typographical and reference corrections are identified by revision bars only.

SPECIFIC REVISION ITEMS

PAGE		ITEM
iii		Table of Contents
		Inserted missing page number.
A:2		Administrative Control of the Emergency Action Level (EAL) Technical Basis
		Updated procedure reference from CCI-154 to RM-1-100.
R:11	RA1	Unplanned Radioactive Release Exceeding 200 X Technical Specification Limits for At Least 15 Minutes
		Added a question and answer explaining the use of an alarm rather than a value.
R:12	RA1	Unplanned Radioactive Release Exceeding 200 X Technical Specification Limits for At Least 15 Minutes
		Page overflow.
R:14	RA2	Damage or Uncovery of Single Irradiated Fuel Assembly Outside the Reactor Vessel
		Added a question and answer explaining the use of an alarm rather than a value.
R:16	RA3	Radiation Increases that Impede Safe Plant Operation
		Revised EAL to refer to "Areas of Concern for Safe Shutdown" table.
		Removed definition of "required."
B:3	BU2	Reactor Coolant System Leakage
		Added reference to "cool down."
		Corrected EAL to apply to leakage within the capacity of a charging pump. Added reference to cooldown.
B:4	BU3	Fuel Clad Degradation
		Updated discussion of AOP-6A vis-à-vis Chemistry Action Level.
		Added a reference to Improved Technical Specifications.

CN600005

<u>ENCLOSURE</u>

CALVERT CLIFFS NUCLEAR POWER PLANT
EMERGENCY ACTION LEVELS TECHNICAL BASIS DOCUMENT
REVISION 5

PAGE		ITEM
B:5	BU3	Fuel Clad Degradation
		Added a reference to Improved Technical Specifications.
		Revised EAL to refer to Improved Technical Specifications.
		Added a reference to Improved Technical Specifications.
B:25	RCB4	Coolant Leakage
		Page overflow
B:26	RCB4	Coolant Leakage
		Added a reference to EOP-6.
		Added a new EAL for EOP-6 implementation.
		Subdivided EAL for EOP-5 and EOP-8 implementation.
		Added a reference to EOP-6.
Q:18	QS3	Loss of Water Level that Can Uncover Fuel in the Reactor Vessel
		Corrected units of Reactor Vessel Level Monitor Sytem reading to water level.
E:1	EU1	Loss of Offsite Power
		Added a Nuclear Management and Resources Council (NUMARC) question and answer explaining guidance on cross-tied units.
		Corrected AOP-3F Mode reference to shutdown.
		Deleted discussion on EOP-2 as an entry condition of the EAL.
		Added a reference to credit taken for ability of Calvert Cliffs to cross-tie units.
E:2	EU1	Loss of Offsite Power
		Revised EAL. Combined two EALs into one. Eliminated EOP-2 as an entry condition.
E:4	EU2	Loss of Vital 125 Volt DC Power for Greater Than 15 Minutes
		Added reference to applicability of 1A DG Bus 14 to EAL entry condition.
		Added question and answer explaining why reference is included.
E:5	EU2	Loss of Vital 125 Volt DC Power for Greater Than 15 Minutes
		Administrative change. Changed bold to standard, deleted strike outs.
E:6	EU2	Loss of Vital 125 Volt DC Power for Greater Than 15 Minutes
		Administrative change. Changed bold to standard, deleted strike outs.

CN600006

ENCLOSURE

CALVERT CLIFFS NUCLEAR POWER PLANT
EMERGENCY ACTION LEVELS TECHNICAL BASIS DOCUMENT
REVISION 5

PAGE		ITEM
E:8	EA2	Only One AC Power Source Available to Supply 4 kV Emergency Buses
		Added reference explaining acceptability of back up power sources.
E:15	EG1	Prolonged Station Blackout
		Revised reference to Calvert Cliffs status as a one-hour station blackout coping category plant.
		Added reference explaining the basis of station blackout core uncovery time of four hours.
E:16	EG1	Prolonged Station Blackout
		Page overflow.
T:1	TU1	Confirmed Security Event With Potential Degradation in the Level of Safety of the Plant
		Added a reference to explain that Security defines intrusion and sabotage.
		Deleted reference to "hostile manner."
T:2	TA1	Security Event in the Plant Protected Area
		Added a reference to explain that Security defines intrusion and sabotage.
		Deleted reference to "hostile manner."
T:3	TS1	Security Event in a Plant Vital Area
		Added a reference to explain that Security defines intrusion and sabotage.
		Deleted reference to "hostile manner."
		Revised EAL to refer to "Areas of Concern for Safe Shutdown."
		Rearranged two EALs into one integrated statement.
		Deleted EAL to incorporate it into a single integrated statement.
		Added a reference to NRC Information Notice 96-71.
I:3	IA1	Fire or Explosion Affecting Safe Shutdown
		Added a new EAL for AOP-9 implementation.
		Renumbered EAL.
I:4	IA1	Fire or Explosion Affecting Safe Shutdown
		Renumbered EALs.

CN600007

PAGE		ITEM
N:4	NA1	Natural Phenomena Affecting Safe Shutdown
		Revised EAL to refer to "observable damage" and "Area of Concern for Safe Shutdown."
O:1	OU1	Site Emergency Coordinator Judgment
		Typographical errors corrected: flammable.
		Renumbered EAL
O:2	OU1	Site Emergency Coordinator Judgment
		Added new EAL for EOP-4 implementation and its basis.
O:5	OU3	Destructive Phenomena
		Revised EAL to refer to "observable damage" rather than security report.
		Revised EAL to refer to "observable" rather than "visible," and adds plant protected area.
		Corrected EAL to refer to "protected area" rather than "safe shutdown."
		Added a reference to "potential damage."
		Removed reference to "maintaining safe shutdown."
		Added clarification explaining that effect of "safe shutdown" is escalation to Alert.
		Removed reference to table of "Areas of Concern for Safe Shutdown."
		Revised EAL to refer to "observable damage" rather than safe shutdown.".
O:8	OA2	Toxic of Flammable Gases Affecting Safe Shutdown
		Revised EAL to refer to "Areas of Concern for Safe Shutdown."
O:10	OA3	Destructive Phenomena Affecting Safe Shutdown
		Revised EAL to refer to "Areas of Concern for Safe Shutdown" and "observable damage."
O:11	OA3	Destructive Phenomena Affecting Safe Shutdown
		Revised EALs to refer to "Areas of Concern for Safe Shutdown" and "observable damage."

CN600008

CALVERT CLIFFS NUCLEAR POWER PLANT UNITS 1 & 2

EMERGENCY ACTION LEVELS
TECHNICAL BASIS DOCUMENT

REVISION 5

PREPARED:	Emergency Planning - G. C. Rudigier	DATE:	2/28/97
REVIEWED:	J. V. Grooms Plant Operations - J. V. Grooms	DATE:	2/28/97
REVIEWED:	Operations Training - J. T. Huber	DATE:	3/3/97
REVIEWED:	Chemistry Programs - G. K. Barley	DATE:	3/3/97
REVIEWED:	Edward H Roach Radiation Safety - E. H. Roach	DATE:	3/3/97
REVIEWED:	Design Engineering-Mechanical -C. J. Ludlow	DATE:	3/3/97
REVIEWED:	Design Engineering-Electrical -R. B. Sydnor	DATE:	3/3/97
REVIEWED:	Nuclear Engineering - S. A. Henry	DATE:	3/3/97
REVIEWED:	Security - D. M. Dean	DATE:	3/3/97
REVIEWED :	John M. Osborne Licensing - J. M. Osborne	DATE:	3/3/97
APPROVED:	Director-Emergency Planning Unit - T. E. Forgette	DATE:	3.4.97
REVIEWED:	POSRC 97-020 Meeting No.	DATE:	3-5-97
APPROVED	Plant General Manager	DATE:	3/5/97

CN600009

Effective Date: with ERPIP 3.0, Revision 18, Change 9

LIST OF EFFECTIVE PAGES

PAGE	REVISION
i	3
ii	1
iii	5
A:1	1
A:2	5
A:3	1
G:1	0
G:2	0
G:3	0
G:4	0
G:5	0
G:6	0
G:7	0
G:8	3
G:9	0
G:10	0
R:1	0
R:2	0
R:3	1
R:4	0
R:5	1
R:6	0
R:7	0
R:8	0
R:9	0
R:10	0
R:11	5
R:12	5
R:13	0
R:14	5
R:15	0
R:16	5
R:17	0
R:18	0
R:19	0
R:20	0
R:21	0
R:22	0
R:23	0
R:24	0
B:1	0
B:2	0
B:3	5
B:4	5
B:5	5
B:6	0
B:7	0

LIST OF EFFECTIVE PAGES

PAGE	REVISION
B:8	0
B:9	0
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B:11	0
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B:16	0
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Q:10	0
Q:11	0
Q:12	0
Q:13	0
Q:14	3
Q:15	3
Q:16	3
Q:17	3
Q:18	5
Q:19	3
Q:20	3
Q:21	3

LIST OF EFFECTIVE PAGES

PAGE	REVISION
E:1	5
E:2	5
E:3	4
E:4	5
E:5	5
E:6	5
E:7	3
E:8	5
E:9	1
E:10	0
E:11	1
E:12	0
E:13	0
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O:7	0
O:8	5
O:9	0
O:10	5
O:11	5
O:12	0
O:13	0
O:14	0
O:15	0
O:16	0

TABLE OF CONTENTS

TABLES

CN600013

Administrative revisions shall not change the intent of the Basis AND shall not cause a wording difference with ERPIP 3.0, Attachment 1.

IV.B.1.b. Administrative revisions shall be approved by the Director-Emergency Planning.

c. Administrative revisions approved by the Director-Emergency Planning will be distributed in accordance with PR-2-100, Document and Drawing Control.

2. Technical revision.

a. Technical revisions shall be reviewed by:

(1) Emergency Planning
(2) Nuclear Operations
(3) Operations Training
(4) Chemistry Programs
(5) Radiation Safety
(6) Nuclear Engineering
(7) Design Engineering
(8) Nuclear Security
(9) Licensing

b. The Emergency Planning reviewer will collect and reconcile review comments. Reviews will be documented on the Basis review/approval sheet.

c. Technical revisions shall be approved by the Director-Emergency Planning. The Director will consider review comments and their reconciliation.

d. Technical revisions shall be submitted to POSRC and the Plant General Manager in accordance with NS-2-101, Conduct of the Plant Operations and Safety Review Committee/Procedure Review Committee/Qualified Reviewer.

e. Technical revisions approved by the Plant General Manager shall be submitted to the NRC for information in accordance with ~~CCI-154~~ **RM-1-100**, Preparation of NRC Correspondence. This submittal shall specify that a revision to ERPIP 3.0, Attachment 1, Emergency Action Levels, to implement the Basis document change, will be processed in forty-five (45) days.

f. After action IV.B.2.e. is complete (i.e., the correspondence is mailed) then a revision to ERPIP 3.0, Immediate Actions, Attachment 1, Emergency Action Levels may be initiated in accordance with ERPIP 900, Preparation of Emergency Response Plan and Emergency Response Plan Implementation Procedures. The effective date

CN600014

RA1 Threshold for RI-5421, RI-5422

Release Rate = 3.2 E+7 µCi/sec (see above)

Release Coefficient (for SG Tube Rupture) = 6.1 E+2 $\frac{\mu Ci/cm^3}{rem/h}$

Atmospheric Dump Valve Flow Rate = 1.4 E+6 cm^3/sec

Safety Valve Flow Rate = 2.4 E+6 cm^3/sec

Main Steam Monitor Reading (rem/h) = $\frac{\text{Release Rate}}{\text{Release Coefficient x Flow Rate}}$

For safety valve rem/h = $\frac{3.2\text{ E+7}}{6.1\text{ E+2 x 2.4 E+6}}$

= .022 rem/h (read as .02) (0.2 mSv/h)

For atmospheric dump valve rem/h = $\frac{3.2\text{ E+7}}{6.1\text{ E+2 x 1.4 E+6}}$

= .038 rem/h (read as .04) (0.4 mSv/h)

The minimum reading for RI-5421/5422 is 10 mrem/h due to the "keep alive" source. Twenty mrem/h would be difficult to read accurately. The high alarm setpoint for these monitors is set at 47 mrem/h ± 5 mrem/h. Therefore, for human factors reasons, the existence of the high alarm setpoint is used as the threshold for this EAL.

Thus, EAL 2 uses the lower value and is written as:

Valid Main Steam Effluent Monitor (RI-5421, RI-5422) High Alarm for GREATER THAN 15 Minutes

CCNPP Questions and Answers (Radioactivity Release)

- **Why isn't a value given for the EAL 2 monitor reading?**

  It is difficult to read this monitor at the response level that is required. It is more appropriate to use the alarm set point in consideration of human factors.

Valid means that the indication is from instrumentation determined to be operable in accordance with the Technical Specifications or has been verified by other independent methods such as indications displayed on the control panels, reports from plant personnel, or radiological survey results. Based on the March 14, 1993 SG tube rupture event at Palo Verde Unit 2, the main steam effluent monitors (RI-5421, RI-5422) may read N^{16} immediately following SG tube rupture and prior to reactor trip. However, given the short half-life of N^{16}, this should clear within the first minute following reactor trip.

Although Calvert Cliffs does not have a perimeter monitoring system, field monitoring could reliably detect radioactive releases.

Thus EAL 3 is written as:

Field Survey Dose Rate Reading of 10 mrem/h or greater at Site Boundary

CN600015

RADIOACTIVITY RELEASE

Source Documents/References/Calculations:

1. System Descriptions
   - No. 15, Radiation Monitoring System

2. Off-Site Dose Calculation Manual (ODCM) for the Baltimore Gas & Electric Company Calvert Cliffs Nuclear Power Plant

3. Radioactivity Release Emergency Action Levels, J.B. McIlvaine, JSB Associates, Inc., September 1990

4. Emergency Response Plan Implementation Procedures
   - ERPIP 821, Accidental Radioactivity Release Monitoring and Sampling Methods

5. BG&E Internal Memorandum, J. R. Hill (Nuclear Plant Operations) to R. L. Wenderlich, CE Operations Subcommittee Meeting - Trip Report, April 16, 1993

6. 10 CFR Part 20, Standards for Protection Against Radiation; Final Rule, 56 FR 23360, May 21, 1991

7. Calvert Cliffs Instructions
   •CCI-302, Calvert Cliffs Alarm Manual, Main Steam Effl Rad Monitor 2C24B

Thus, EAL 1 is written as:

AOP-6E, Loss of Refueling Pool Level, is Implemented AND Valid Containment Radiation Alarm (RI-5316A/B/C/D)

CCNPP Questions and Answers (Radioactivity Release)

- **Why isn't a value given for the EAL 1 monitor reading?**

  Since the alarm set point and the action level are the same, human factor consideration is to reference the alarm to eliminate the need to visually follow a monitor output during a containment event.

Thus, EAL 2 is written as:

AOP-6D, Fuel Handling Incident, is Implemented AND ANY of the Following: • Valid Containment Radiation Alarm (RI-5316A/B/C/D) • Valid Fuel Handling Area Ventilation Exhaust Radiation Monitor (RI-5420) Reading of AT LEAST 2E+4 CPM • Valid Spent Fuel Service Platform Monitor (RI-7025) Reading of AT LEAST 100 mrem/h

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

The containment radiation alarm corresponds to a dose rate of 200 mrem/h.

The value for RI-5420 was determined based on a fuel handling accident damaging one fuel rod in an average (unpeaked) fuel assembly. The results of the calculation, showing RI-5420 response versus age of the assembly (time after shutdown), is shown as Figure R1. The value of 2E4 CPM corresponds to the minimum expected response and is significantly higher than the alarm setpoint of 600 CPM.

One hundred mrem/h is used for the Service Platform Monitor (RI-7025) because it corresponds to the administrative limit for a high radiation area and is significantly higher than the dose rates expected for fuel handling activities.

Expected increases in monitor readings due to controlled evolutions (such as lifting the reactor vessel head during refueling) should not result in emergency declaration. Nor should momentary increases due to events such as resin transfers or controlled movement of radioactive sources result in emergency declaration. In-plant radiation level increases that would result in emergency declaration are also *unplanned, e.g.*, outside the limits established by an existing radioactive discharge permit.

Source Documents/References/Calculations:

1. System Descriptions
   - No. 15, Radiation Monitoring System

2. Abnormal Operating Procedures
   - AOP-6D, Fuel Handling Incident
   - AOP-6E, Loss of Refueling Pool Level

3. Ogden Calculation #RA-1, 0-RI-5420 Detector Response to Fuel Handling Accident

CN600017

For areas requiring infrequent access, the (Site-Specific) value(s) should be based on radiation levels which result in exposure control measures intended to maintain doses within normal occupational exposure guidelines and limits (*i.e.*, 10 CFR 20), and in doing so, will impede necessary access. For many areas, it may be possible to establish a single <Generic> EAL that represents a multiple of the normal radiation levels (*e.g.*, 1000 times normal). However, areas that have normally high dose rates may require a lower multiple (*e.g.*, 10 times normal).

Plant-Specific Information

The control room is required to be continuously occupied following design basis accidents. All actions required to achieve and maintain cold shutdown can be accomplished from the control room. Post-accident doses have been evaluated and shown to be less than limits based on GDC 19. On a control room high radiation signal, the control room emergency ventilation system automatically switches into a recirculation mode of operation with flow through the HEPA filters and charcoal absorber banks. EAL 1 is based on the GDC 19 limit recommended by NUMARC.

Thus, EAL 1 is written as:

Valid Control Room Radiation Monitor (RI-5350) Reading GREATER THAN 15 mrem/h

This corresponds to a dose rate of 0.15 mSv/h. *Valid* means that the indication is from instrumentation determined to be operable in accordance with the Technical Specifications or has been verified by other independent methods such as indications displayed on the control panels, reports from plant personnel, or radiological survey results.

EAL 2 addresses event sequences outside the plant design basis. Entry into any area with exposure rates of at least 10 Rem/h (100 m Sv/h) could result in an individual exceeding 10 CFR 20 limits (5 REM/yr) with approximately 30 minutes.

Thus, EAL 2 is written as:

Exposure Rate of 10 rem/h or greater in ~~Areas Required to Achieve or Maintain Safe Shutdown~~ **an Area of Concern for Safe Shutdown**

~~*Required* means that entry into the area is not optional and is imperative based on existing conditions.~~
Areas of concern for Safe Shutdown are listed below.

Areas of Concern for Safe Shutdown	
• Control Room	• Electrical Penetration Rooms
• Control Room HVAC Room	• Auxiliary Feedwater Pump Room
• Cable Spreading Room	• Charging Pump Rooms
• Cable Chases	• Diesel Generator Rooms
• Switchgear Room	• Diesel Generator Building (0C/1A)
• ECCS Pump Room	• Refueling Water Tank (RWT) 11(21)
• Service Water Pump Room	• Condensate Storage Tank (CST) 12
• Component Cooling Pump Room	• Pretreated Water Storage Tank (PWST) 11(21)
• Main Steam Penetration Room	• Fuel Oil Storage Tank (FOST) 12
This list of Safe Shutdown areas is displayed on the EAL Tables to assure that all areas related to Safe Shutdown are considered by the SEC.	

Expected increases in monitor readings due to controlled evolutions (such as lifting the reactor vessel head during refueling) do not result in emergency declaration. Nor should momentary increases due to events such as resin transfers or controlled movement of radioactive sources result in emergency declaration. In-plant radiation level increases that would result in emergency declaration are also *unplanned*, *e.g.*, outside the limits established by an existing radioactive discharge permit. The containment radiation monitor readings should only apply to this IC when personnel are in containment for normal maintenance, inspection, surveillance, testing, or refueling activities.

Source Documents/References/Calculations:

1. System Descriptions

CN600018

Calvert Cliffs Units 1 and 2 are Combustion Engineering designed reactors. These reactors use a programmed pressurizer water level that varies as a function of T_{avg} and load. The Chemical Volume Control System includes three fixed flow positive displacement charging pumps and a variable letdown system. Each charging pump has a capacity of 44 GPM. The letdown system valves regulate letdown flow from 28 GPM to 128 GPM. The nominal configuration is one charging pump with ~40 GPM letdown flow. The letdown flow is varied as necessary to maintain programmed pressurizer level. Additional charging pumps are automatically started when necessary to maintain pressurizer level.

AOP-2A, Excessive Reactor Coolant Leakage, is implemented if any entry conditions are met; this includes the results of STP-0-27-1/2, Reactor Coolant Leakage Evaluation. STP-0-27-1/2 will indicate leakage in excess of Technical Specification 3.4.6.2 allowable limits. Control room personnel require approximately 5 to 15 minutes to implement AOP-2A if RCS leakage exceeds the capacity of one charging pump. In general, Calvert Cliffs does not distinguish between identified or unidentified leakage when AOP-2A is implemented. Per AOP-2A, if leakage exceeding the capacity of one charging pump (11 GPM leakage with minimum letdown flow or greater than 39 GPM with letdown isolated) could not be isolated, then the reactor must be shutdown (tripped) **and cooled down.**

If RCS leakage is less than the capacity of one charging pump, STP-0-27-1/2 would be performed to determine the leak rate and the reactor would be maintained at power. It requires approximately 3 to 6 hours to perform STP-0-27-1/2 to determine the amount of unidentified leakage.

Calvert Cliffs EALs have been written to be consistent with procedural requirements. These leakage rates are very similar to the NUMARC generic leakage. AOP-2A specifies certain flow paths that can be isolated to terminate RCS leakage. If isolation of the leakage path is successful (*e.g.*, isolating a leaking pressurizer power operated relief valve), reactor operation can continue and this EAL does not apply. However, if RCS leakage could not be isolated, then under these conditions the reactor would have to be shut down in accordance with technical specifications. The EAL language was picked to assure that: (1) leakage is greater than net RCS make-up flow threshold of 11 GPM, and (2) Such leakage could not be isolated in accordance with procedural requirements.

Thus, the Calvert Cliffs EAL is written as:

AOP-2A, Excessive Reactor Coolant Leakage, Is Implemented For RCS Leakage ~~Exceeding~~ **within** the Capacity of One Charging Pump AND Reactor Shutdown **OR Cooldown** is Required

NUREG 1449 raises concerns regarding events involving leakage through RCS temporary boundaries. RCS leakage EALs apply to all operational modes at Calvert Cliffs. This will assure that leakage is appropriately addressed for cold shutdown and refueling modes and address NRC concerns about leakage through temporary RCS boundaries as they apply to EALs.

<u>Source Documents/References/Calculations</u>:

1. Technical Specifications
   • TS 3.4.6.2, Reactor Coolant System Leakage

2. Abnormal Operating Procedures
   • AOP-2A, Excessive Reactor Coolant Leakage

3. Surveillance Test Procedure (STP) O-27-1/2, RCS Leakage Evaluation

4. NUREG 1449, Shutdown and Low-Power Operation at Commercial Nuclear Power Plants in the United States, Draft for Comment, February 1992

CN600019

Emergency Classification Level: UNUSUAL EVENT

Applicable Operational Modes: ALL

Calvert Cliffs Initiating Condition:

BU3 Fuel Clad Degradation

NUMARC Recognition Category: System Malfunction

NUMARC Initiating Condition:

SU4 Fuel Clad Degradation

Barrier: Fuel Clad

NUMARC Generic Basis:

This IC is included as an Unusual Event because it is considered to be a potential degradation in the level of safety of the plant and a potential precursor of more serious problems. <Generic> EAL 1 addresses (Site-Specific) radiation monitor readings such as failed fuel monitors, etc., that provide indication of fuel clad integrity. <Generic> EAL 2 addresses coolant samples exceeding coolant technical specifications for iodine spike. Escalation of this IC to the Alert level is via the <Fission Product Barrier Degradation EALs>.

Plant-Specific Information:

A significant rise in the count rate on the Activity Monitor or valid actuation of the "RADIATION MONITOR LEVEL HI" alarm can be due to either fuel clad failure or to crud burst. In accordance with AOP-6A, the response to high RCS activity level is to notify Plant Chemistry to perform a sample analysis to determine what radionuclides caused the radiation alarm. This means that the monitor indications are not sufficient alone to determine whether fuel clad damage has occurred at Calvert Cliffs. Thus, <Generic> EAL 1 is not appropriate for use at Calvert Cliffs.

Clad damage is determined from specific activity levels contained in reactor coolant samples. Per AOP-6A, **when RCS activity is less than Chemistry** ~~Fuel Reliability Plan~~ **Action Level 1 values, the operator may return to the appropriate operating procedure. Per CP-204, Chemistry Action Level 1 is for specific activity levels greater than 0.5 μCi/gram I^{131} DEQ *or greater than 50/Ebar μCi/gram of gross radioactivity.*** ~~4 is defined as Dose Equivalent I^{131} (I^{131} DEQ) of at least the~~ Technical Specification Section 3.4.8. **(Improved Technical Specifications Section 3.4.15) requires the specific activity of the reactor coolant to be within** ~~limits. These are~~:

a. ~~Not more than~~ 1 μCi/gram I^{131} DEQ.

b. ~~Not more than~~ 100/Ebar μCi/gram of gross radioactivity.

The specific activity of the reactor coolant may be as high as the limits defined by Technical Specification Figure 3.4.8-1 **(Improved Technical Specification Section 3.4.15-1)** for up to 48 hours. The lowest limit for this figure corresponds to 60 μCi/gram I^{131} DEQ. Scaling down from the value shown for FCB3, Radiation, corresponding 1500 μCi/gram I^{131} DEQ an RCS sample dose rate at one foot is computed as shown in the equation below.

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RCS Sample Reading For 60 µCi/gram I^{131} DEQ

Refer to EAL FCB3 , Radiation

$$\text{BU3 Value} = \frac{60\ \mu\text{Ci/gram}}{1500\ \mu\text{Ci/gram}} \times 168\ \text{mrem/h} = 6.7\ \text{mrem/h}$$

Read as 6 mrem/h (.06 mSv/h)

Thus, the EAL 1 is written as:

Dose Rate at One Foot from RCS Sample of AT LEAST 6 mrem/h

This corresponds to a dose rate of 0.06 mSv/h.

Technical Specification 3.4.8 (**Improved Technical Specification Section 3.4.15**) Reactor Coolant System - Specific Activity is addressed by EAL 2:

Thus EAL 2 is written:

Fuel Clad Degradation Indicated by RCS Sample Activity GREATER THAN Tech Spec 3.4.8 ***(Improved Technical Specification Section 3.4.15)*** *Allowable Limits*

Source Documents/References/Calculations:

1. Technical Specifications
   - TS 3.4.8, Reactor Coolant System - Specific Activity

   Improved Technical Specification
   - **TS 3.4.15, RCS Specific Activity**
2. Abnormal Operating Procedures
   - AOP-6A, Response to High RCS Activity
3. BG&E Fuel Degradation EALs Calculation Worksheet, JSB Associates, February 18, 1993

CN600021

Calvert Cliffs Emergency Action Level:

RCB4 Coolant Leakage

NUMARC Emergency Action Level:

RCS 2 RCS Leak Rate

- *Potential Loss* - Unisolable Leak Exceeding the Capacity of One Charging Pump in the Normal Charging Mode

RCS 3 SG Tube Rupture

- *Potential Loss* - (Site-Specific) Indication that a SG is Ruptured and Has a Non-Isolable Secondary Line Break OR (Site-Specific) Indication that a SG is Ruptured and a Prolonged Release of Secondary Coolant is Occurring From the Affected SG to the Environment

RCS 5 Other (Site-Specific) Indications

NUMARC Generic Basis:

[RCS 2, RCS 3]
<Loss EALs are addressed under IC RCB2, Temperature.> <>

The Potential Loss EAL is based on the inability to maintain normal liquid inventory within the Reactor Coolant System (RCS) by normal operation of the Chemical and Volume Control System which is considered as one centrifugal charging pump discharging to the charging header. <This indication, applying to any RCS leakage including primary-to-secondary leakage> assures that any event that results in significant RCS inventory shrinkage or loss (*e.g.*, events leading to reactor scram and ECCS actuation) will result in no lower than an "Alert" emergency classification.

[RCS 5]
This EAL is to cover other (site-specific) indications that may indicate loss or potential loss of the RCS barrier, including indications from containment air monitors or any other (site-specific) instrumentation.

Plant-Specific Information:

The Calvert Cliffs Chemical and Volume Control System (CVCS) uses three positive displacement horizontal pumps with a capacity of 44 GPM each. The pressurizer level control program regulates letdown purification subsystem flow by adjusting the letdown flow control valve so that the reactor coolant pump (RCP) controlled leak-off plus the letdown flow matches the input from the operating charging pump. Equilibrium pressurizer level conditions may be disturbed due to RCS temperature changes, power changes, or RCS inventory loss due to leakage. A decrease in pressurizer water level below the programmed level will result in a control signal to start one or both standby charging pumps to restore water level. An increase in pressurizer water level above the programmed level will result in a control signal to increase letdown purification flow rate and initiate a backup signal to stop the two standby charging pumps.

A start signal is sent to all three charging pumps on a Safety Injection Actuation Signal (SIAS), aligning the charging pumps suction to the Boric Acid Storage Tanks (BASTs) via the boric acid pumps. All three charging pumps will then inject highly concentrated boric acid into the RCS to ensure that the reactor is shutdown. Potential Loss of the RCS corresponds to conditions where the CVCS can not maintain pressurizer water level within normal limits requiring transition into the EOPs when the reactor is initially critical.

Thus, Potential Loss EAL 1 is written as:

RCS Leakage Exceeds Available CVCS Capacity

However, review showed that an appropriate site-specific Potential Loss EAL could be developed based on entry into EOP-5, Loss of Coolant Accident, **EOP-6, Steam Generator Tube Rupture**, or EOP-8, Functional Recovery Procedure, for an RCS leak.

Thus, Potential Loss EAL 2 ~~is~~ **, 3 and 4 are** written as:

EOP-5, Loss of Coolant Accident, ~~Or EOP-8, Functional Recovery Procedure,~~ is Implemented for RCS Leakage

EOP-6, Steam Generator Tube Rupture, is implemented for RCS leakage.

EOP-8, Functional Recovery Procedure, is implemented for RCS leakage.

<u>Source Documents/References/Calculations</u>:

1. Abnormal Operating Procedures
   - AOP-2A, Excessive Reactor Coolant Leakage

2. Emergency Operating Procedures
   - EOP-5, Loss of Coolant Accident
   - **EOP-6, Steam Generator Tube Rupture**
   - EOP-8, Functional Recovery Procedure

3. Surveillance Test Procedure (STP) 0-27-1/2, RCS Leakage Evaluation

4. Updated Final Safety Analysis Report
   - Section 9.1, Chemical and Volume Control System

CN600023

Emergency Classification Level: SITE EMERGENCY

Applicable Operational Modes: 5, 6

Calvert Cliffs Initiating Condition:

QS3 Loss of Water Level That Can Uncover Fuel in the Reactor Vessel

NUMARC Initiating Condition:

SS5Loss of Water Level in the Reactor Vessel That Has or Will Uncover Fuel in the Reactor Vessel

Barrier: FUEL CLAD

NUMARC Generic Basis:

Under the conditions specified by this IC, severe core damage can occur and reactor coolant system pressure boundary integrity may not be assured. <> For PWRs, this IC covers sequences such as prolonged boiling following loss of decay heat removal.

Thus, declaration of a Site <E>mergency is warranted under the conditions specified by the IC. Escalation to a General Emergency is via <Radioactivity Release IC RG1, Off-Site Dose of AT LEAST 1 REM (EDE+CEDE) Whole Body or 5 REM (CDE) Thyroid>.

Plant-Specific Information:

Sequences that can result in uncovery of fuel in the reactor vessel (indirectly by prolonged boiling) include leakage through SG nozzle dams, pipe breaks in the Shutdown Cooling (SDC) System or Chemical & Volume Control System (CVCS), or loss of the SDC function. These leakage sources are outside the reactor vessel and at most could only result in water level decreases to the bottom of the hot leg elevation. This water level decrease would cause loss of SDC suction. In-core instrumentation (ICI) penetrations for Calvert Cliffs are through the vessel head. Thus, these do not have to be considered for this IC.

A review of attachments to AOP-3B, Abnormal Shutdown Cooling Conditions, shows that depending on previous power history and assuming an initial RCS temperature of 140°F, boiling in the core can begin in as little as 7 minutes following loss of SDC during mid-loop operation. AOP-3B also shows that under these conditions, without any operator action, core uncovery can begin within about 80 minutes after loss of SDC.

Available methods to restore RCS inventory and to remove core heat include restoring the SDCS, injecting into the RCS from the Refueling Water Tank (RWT) using the HPSI, LPSI, CS or charging pumps, using the steam generators as a heat sink, using the Refueling Pool as a heat sink, aligning a LPSI pump to take suction from the RWT, or even injecting into the RCS using Safety Injection Tanks (SITs). *Given the number of methods to restore inventory, and the amount of time available, it is highly unlikely that this IC will be entered.*

Thus, the EAL is written as:

AOP-3B, Abnormal Shutdown Cooling Conditions, Is Implemented AND ANY of the Following Conditions Exist:

- Alternate Methods for Restoring RCS Inventory Are NOT Effective
- Valid RVLMS Reading Indicating ~~0% Level~~ **water level above core is ten inches or less.**
- Valid CET Reading Indicating Superheat Conditions

CN600024

Emergency Classification Level: UNUSUAL EVENT

Applicable Operational Modes: ALL

Calvert Cliffs Initiating Condition:

EU1 Loss of Off-Site Power

NUMARC Recognition Category: System Malfunction

NUMARC Initiating Condition:

SU1 Loss of All Off-Site Power to Essential Busses for Greater Than 15 Minutes

Barrier: Not Applicable

NUMARC Generic Basis:

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

NUMARC Questions and Answers, June 30, 1993 (System Malfunction)

1. **Does the EAL of SU1 apply to one unit whose essential busses can be energized from another (unaffected) unit at a multi-unit site?**

   SU1 does apply to this situation. Plants that have the capability to cross-tie power from a companion unit may take credit for the redunant power source in the associated EAL for this IC. Inability to effect that cross-tie within 15 minutes is grounds for declaring the Unusual Event.

Multi-unit stations with shared safety functions should further consider how this IC may affect more than one unit and how this may be a factor in escalating the emergency class.

Plant-Specific Information:

Procedure EOP-2, Loss of Off-Site Power, would be implemented under the conditions of concern. AOP-3F applies to the other operational modes when the plant is ~~critical~~ **shutdown**. Per EOP-2, the following are symptoms of a loss of off-site power:

- Momentary loss of Control Room lighting on both Units.
- 500KV Red Bus and Black Bus power available lights are de-energized.
- Diesel Generators automatically start.
- 13KV Service Buses 12 and 22 power available lights are de-energized.
- No RCPs are running on either Unit.
- Reactor Trip occurs due to RCS low flow.

~~For consistency with procedural requirements and to reflect potential severity, separate EALs have been developed for hot and cold conditions. With the plant initially operating in Mode 1 or 2, EOP-2 would be entered on a loss of off-site power. Under these conditions, restoring off-site power is expected to take no less than 15 minutes based on procedure implementation. Therefore, EAL 1 does not use the generic 15 minute threshold. EOP-2 may also be implemented if single phase natural circulation is to be used for RCS heat removal although at least one 13KV Service Bus is energized. Unusual Event declaration is not appropriate for this use of the procedure.~~ **Being a two unit site with the ability to cross-tie power from the other (unaffected) unit, credit is taken for the redundant power source.**

Thus, the EAL ~~1~~ is written as:

~~EOP-2,~~ Loss of All Off-site 500 kV Power to both 4 kV Safety Related Buses on ~~, Implemented On~~ Either Unit for ~~loss of off-site power.~~ ***Greater than 15 minutes.***

~~EAL 2 addresses loss of off-site power when EOP-2 does not apply.~~

~~Thus, EAL 2 is written as:~~

~~Loss of Off-Site Power for GREATER THAN 15 Minutes~~

Source Documents/References/Calculations:

1. Technical Specifications
   - TS 3.8.1, A.C. Sources

2. Emergency Operating Procedures
   - EOP-2, Loss of Off-Site Power

3. Abnormal Operating Procedures
   - AOP-3F, Loss of Off-Site Power While in Modes 3, 4, 5, or 6

CN600026

There is one battery charger fed from Unit 1 and another battery charger fed from Unit 2 connected to each 125 volt dc bus. The ac power for both battery chargers per bus is obtained from the same load group. The reserve battery is connected to its own charger when it is not connected to a safety related 125 volt dc bus.

Each of the four 125 volt dc power sources is equipped with the following instrumentation in the control room to enable continual operator assessment of 125 volt dc power source condition:

- DC bus undervoltage alarm
- Battery current indication
- Charger current indication
- Charger malfunction alarm (including input ac undervoltage, output dc undervoltage, and output dc overvoltage)
- DC bus voltage indication, and
- DC ground indication

Components affected by the loss of 125 volt dc buses 11, 12, 21, or 22 are listed in table EU2-1. **Loss of the new Diesel Generator 1A 125 volt DC bus 14 does not constitute an entry condition for this EAL.**

CCNPP Questions and Answers (Electrical)

- **Why does the 125 volt DC bus 14 need to be addressed in the basis if it has no impact on the EAL?**

  Site Emergency Coordinators asked for documentation in the basis, that the new 125 volt DC bus 14 was considered for the electrical EAL's. AOP-7J lists the equipment that is lost if bus 14 is lost.

Thus, the EAL is written as:

AOP-7J, Loss of 120 Volt Vital AC or 125 Volt Vital DC Power, is Implemented AND 125 Volt DC Power Lost for GREATER THAN 15 Minutes

Source Documents/References/Calculations:

1. Abnormal Operating Procedures
   - AOP-7J, Loss of 120 Volt Vital AC or 125 Volt Vital DC Power
2. Updated Final Safety Analysis Report
3. BG&E Drawing 61-030-E, Single Line Diagram, Vital 120V AC & 125V DC - Emergency 250V DC
4. BG&E Drawing 61-057-E, Block Diagram - Auxiliary System Load Groups - Units 1 & 2

CN600027

Table E-1: Effects of Lost 125 Volt DC Buses 11, 21, 12, and 22			
Loss of 11 125 volt dc Bus	Loss of 21 125 volt dc Bus	Loss of 12 125 volt dc Bus	Loss of 22 125 volt dc Bus
Channel ZD ESFAS and AFAS Sensor Cabinets de-energized	Channel ZE ESFAS and AFAS Sensor Cabinets de-energized	Channel ZF ESFAS and AFAS Sensor Cabinets de-energized	Channel ZG ESFAS and AFAS Sensor Cabinets de-energized
CNTMT Area Rad Monitor out of service	CNTMT Area Rad Monitor out of service	CNTMT Area Rad Monitor out of service	CNTMT Area Rad Monitor out of service
Channel A RPS Cabinet de-energized	Channel B RPS Cabinet de-energized	Channel C RPS Cabinet de-energized	Channel D RPS Cabinet de-energized
Loss of 2A EDG field flash and control power; the start solenoids fail shut (Unit 2 only)	Loss of 2B EDG field flash and control power; the start solenoids fail shut (Unit 2 only)	Loss of 1B EDG field flash and control power; the start solenoids fail shut (Unit 1 only)	
Loss of breaker position indication: ·Normal power supply to the 11A/21A and 12A/22A RCPs ·11/21, 12/22, 15/25, and 16/26 4 KV buses ·11A/21A, 11B/21B, 12A/22A, and 12B/22B 480 Volt Buses ·11 and 12 13 KV buses (Unit 1 only)	Loss of breaker position indication: ·Normal power supply to the 11B/21B and 12B/22B RCPs ·13/23 and 14/24 4 KV buses ·13A/23A, 13B/23B, 14A/24A, and 14B/24B 480 Volt Buses		
Loss of Unit 2 Annunciation	All Unit 1 Annunciator lights de-energized (Status Panels remain energized)		
CC CNTMT SUPPLY fails shut	CC CNTMT RETURN fails shut		
12 SG AFW STM SUPP & BYPASS valves fail shut	11 SG AFW STM SUPP & BYPASS valves fail shut		
Loss of SRW to the Turbine Building	Loss of SRW to the Turbine Building		
IA and PA may be lost due to loss of SRW to the Turbine Building	IA and PA may be lost due to loss of SRW to the Turbine Building		
Channel A ESFAS and AFAS Actuation Cabinets de-energized	Channel B ESFAS and AFAS Actuation Cabinets de-energized		
11/21 SRW, 11/21 CC, and 11/21 ECCS Pump Room HX SW outlet valves fail open	12/22 SRW, 12/22 CC, and 12/22 ECCS Pump Room HX SW outlet valves fail open	—	
11/21 Main Steam Effluent Rad Monitor out of service	12/22 Main Steam Effluent Rad Monitor out of service		
11 and 12 SFP Heat Exchangers lose cooling flow due to SRW outlet CVs failing shut (Unit 1 only)	11 and 12 SFP Heat Exchangers lose cooling flow due to SRW outlet CVs failing shut (Unit 1 only)		
11/21 MSIV loses position indication, but can still be closed from 1CO3/2CO3	12/22 MSIV loses position indication, but can still be closed from 1CO3/2CO3		
CNTMT High Range Monitor Channel A out of service		CNTMT High Range Monitor Channel B out of service	

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Table E-1: Effects of Lost 125 Volt DC Buses 11, 21, 12, and 22 (Continued)			
Loss of 11 125 volt dc Bus	Loss of 21 125 volt dc Bus	Loss of 12 125 volt dc Bus	Loss of 22 125 volt dc Bus
Loss of open signal to the Turbine Bypass Valves and loss of quick open signal to the ADVs (Unit 1 only)			
Aux Spray Valve fails shut			
IA downstream of the CNTMT IA Control Valve is isolated ("CNTMT IA ISOLATED IA-2085-CV CLOSED" alarm does NOT actuate)			
CNTMT Gaseous Monitor out of service			
Gaseous and Liquid Waste release control valves fail shut (Unit 1 only)			
	11B/21B and 12B/22B RCPs are untrippable from 1CO6/2CO6		
	Loss of letdown due to 1/2-CVC-516-CV failing shut		
	AFW Turbine Driven Train Flow Control Valves 11 SG and 12 SG fail open (Unit 1 only)		
	PORV-404 inoperable in MPT ENABLE (Unit 1 only)		
TCBs 1 and 5 trip	TCBs 2, 6, and 9 trip	TCBs 3 and 7 trip	TCBs 4 and 8 trip
			Loss of plant oscillograph (Unit 1 only)

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Emergency Classification Level: ALERT

Applicable Operational Modes: 1, 2, 3, 4

Calvert Cliffs Initiating Condition:

EA2 Only One AC Power Source Available to Supply 4 kV Emergency Busses

NUMARC Recognition Category: System Malfunction

NUMARC Initiating Condition:

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

Barrier: Not Applicable

NUMARC Generic Basis:

This IC and its associated <Generic> EAL are intended to provide an escalation from IC <EU1, Loss of Off-Site Power>. The condition indicated by this IC is the degradation of the off-site and on-site power systems such that any additional single failure would result in a station blackout. This condition could occur due to a loss of off-site power with a concurrent failure of one diesel generator to supply power to its emergency busses. Another related condition could be the loss of all off-site power and loss of on-site emergency diesels with only one train of emergency busses being backfed from the unit main generator, or the loss of on-site emergency diesels with only one train of emergency busses being backfed from off-site power. The subsequent loss of this single power source would escalate the event to a Site <E>mergency in accordance with IC <ES1, Station Blackout>.

<Generic> EAL 1b should be expanded to identify the control room indications of the status of Site-specific power sources and distribution busses that, if unavailable, establish single failure vulnerability.

At multi-unit stations, the EALs should allow credit for operation of installed design features, such as cross-ties or swing diesels, provided that abnormal or emergency operating procedures address their use. However, these stations must also consider the impact of this condition on other shared safety functions in developing the site specific EAL.

Plant-Specific Information:

The EAL addresses conditions while operating in Modes 1, 2, 3, or 4 under which only one method is available to supply the emergency buses and loss of that method will result in a Station Blackout. **Acceptable back up power sources with respect to this EAL include the non-safety related 0C diesel generator and the 13 kV SMECO tie line. The 13 kV SMECO tie line can back up both units. When one or more of these sources are available to back up the Unit experiencing a loss of offsite power or loss of a safety related diesel generator the entry condition for the EAL is not being met and the EAL does not apply.**

Thus, the EAL is written as:

Only One Power Source (Off-site or Diesel) is Available to Supply Unit 1 (Unit 2) Safety Related 4 kV busses for GREATER THAN 15 Minutes AND the Unit is Not on Shutdown Cooling (this is a condition where any additional single failure will result in Station Blackout).

Source Documents/References/Calculations:
1. Updated Final Safety Analysis Report
   - Section 8, Electric Power Systems

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Under conditions where a diesel generator is supplying power to one Unit, it should not be considered available as a power supply for the other Unit.

The first part of this EAL corresponds to the threshold conditions for IC ES1, Station Blackout for GREATER THAN 15 Minutes. The second part of the EAL addresses the conditions that will escalate the SBO to General Emergency. Occurrence of any one of these conditions following SBO is sufficient for escalation to General Emergency. These conditions are: (1) SBO coping capability, or (2) indications of inadequate core cooling. Each of these conditions is discussed below:

1. SBO Coping Capability

Calvert Cliffs ~~falls within the~~ **is licensed both for a** four hour SBO coping category **AND a one hour SBO coping category**. The ability of each unit to cope with a four hour SBO duration was based on an assessment of condensate inventory required for decay heat removal, Class 1E battery capacity, compressed air availability or manual operation of certain valves, effects of loss of ventilation, containment isolation valve operability, and reactor coolant inventory loss. A plant-specific analysis indicates that the expected rates of reactor coolant inventory loss under SBO conditions do not result in core uncovery in a SBO of four hours. Therefore, makeup systems in addition to those currently available under SBO conditions are not required to maintain core cooling under natural circulation (including reflux boiling). *Thus, conditions in which restoration of AC power within 4 hours is NOT likely are included in the EAL.*

Installation of a SBO diesel also allows Calvert Cliffs to operate as a plant having a one hour coping capability. This allowance is in recognition that sufficient diesel generator back-up reduces the likelihood of station black-out. The analysis for the four hour coping category however, provides the source of an appropriate estimate of the time to core uncovery following a station black out from which the plant can not recover. This time (four hours) is used as the basis for determining when to declare a general emergency subsequent to a prolonged station black out.

2. Indications of Inadequate Core Cooling

Calvert Cliffs does not use Critical Safety Function Status Trees. Calvert Cliffs uses Safety Function Status Checks developed by the Combustion Engineering Owners' Group (C-E OG) which are based on logic similar to that used for CSFSTs developed for Westinghouse PWRs. The applicable acceptance criteria for Core and RCS Heat Removal are shown on the Safety Function Status Checks. They are:

Steam Generators Available for RCS Heat Removal

1. Adequate secondary side liquid inventory in at least one steam generator as indicated by level between -170 and +30 inches, and

2. Adequate source of feedwater available to assure continued liquid inventory available as indicated by Condensate Storage Tank level greater than 5 feet, and

3. Steam Generators acting as effective heat sink as indicated by Cold Leg Temperatures (T_{COLD}) constant or lowering.

Primary Side Conditions for Core and RCS Heat Removal

1. Adequate core heat removal as indicated by Core Exit Thermocouple readings less than superheated, and

2. Either of the following:
   - Natural circulation established as indicated by the temperature difference between Hot Leg Temperature (T_{HOT}) and T_{COLD} of between 10 °F and 50 °F, or
   - Forced circulation effective as indicated by T_{HOT} - T_{COLD} less than 10 °F.

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Per CEN-152, superheated conditions indicate core uncovery and inadequate core cooling.

Thus, the EAL is written as:

EOP-7, Station Blackout, is Implemented AND ANY of the Following: • Restoration of Power to ANY Vital 4kV Bus Is NOT Likely Within 4 Hours • Valid CET Readings Indicate Superheat Temperatures • Core and RCS Heat Removal Using Steam Generators Can NOT Meet Acceptance Criteria

Valid means that the indication is from instrumentation determined to be operable in accordance with the Technical Specifications or has been verified by other indications displayed on the control panels.

Can NOT is used because the ability to meet the final acceptance criteria is the appropriate concern, not whether intermediate acceptance criteria are not being achieved at any given moment.

Source Documents/References/Calculations:

1. Emergency Operating Procedures
   - EOP-7, Station Blackout
   - EOP-8, Functional Recovery Procedure

2. CEN-152, Emergency Procedure Guidelines

3. Letter, Daniel G. MacDonald (US Nuclear Regulatory Commission) to G.C. Creel (BG&E), Response to Station Blackout Rule - Calvert Cliffs Nuclear Power Plant, Units 1 and 2, TAC Numbers 68525 (Unit 1) and 68256 (Unit 2), October 10, 1990

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SECURITY

Emergency Classification Level: UNUSUAL EVENT

Applicable Operational Modes: ALL

Calvert Cliffs Initiating Condition:

TU1 **Confirmed Security Event With Potential Degradation in the Level of Safety of the Plant**

NUMARC Recognition Category: Hazards and Other Conditions Affecting Plant Safety

NUMARC Initiating Condition:

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

Barrier: Not Applicable

NUMARC Generic Basis:

This EAL is based on (Site-specific) Site Security Plan. Security events which do not represent at least a potential degradation in the level of safety of the plant are reported under 10 CFR 73.71 or in some cases under 10 CFR 50.72. The plant Protected Area Boundary is typically that part within the security isolation zone and is defined in the (Site-specific) security plan. Bomb devices discovered within the plant Vital Area would result in <> escalation <to a higher emergency classification level via other Security Event ICs>.

Plant-Specific Information:

The Calvert Cliffs EALs address the generic areas of concern and include the ISFSI. **Nuclear Security will determine whether or not intrusion or sabotage exists in accordance with the Safeguards Contingency Plan.** Attempted intrusion means that intruders are not successful in getting past the innermost fence of the double fence that surrounds the plant protected area. Sabotage within the ISFSI includes discovery of a bomb device. Intruders are armed or unarmed personnel that are attempting to or have gained unauthorized access ~~in a hostile manner~~.
Sabotage (including discovery of a bomb device) inside the Plant Protected Area warrants escalation to an Alert level emergency. A Site Emergency is warranted if sabotage occurs in an area of concern for safe shutdown of either reactor.

Thus, EAL 1 is written as:

"Security Emergency" or "Security Alert" Declared for Attempted Intrusion into the Plant Protected Area

EAL 2 is written as:

"Security Event" Declared for:
- Sabotage Within or to ISFSI Protected Area
- Intrusion Into ISFSI Protected Area

Source Documents/References/Calculations:

None

CN600033

SECURITY

Emergency Classification Level: ALERT

Applicable Operational Modes: ALL

Calvert Cliffs Initiating Condition:

TA1 Security Event in the Plant Protected Area

NUMARC Recognition Category: Hazards and Other Conditions Affecting Plant Safety

NUMARC Initiating Condition:

HA4 Security Event in a Plant Protected Area

Barrier: Not Applicable

NUMARC Generic Basis:

This class of security events represents an escalated threat to plant safety above that contained in the Unusual Event. For the purposes of this IC, a civil disturbance which penetrates the protected area boundary can be considered a hostile force. Intrusion into a vital area by a hostile force will escalate this event to a Site <Emergency>.

Plant-Specific Information:

The Calvert Cliffs EALs address the generic areas of concern. **Nuclear Security will determine whether or not intrusion or sabotage exists in accordance with the Safeguards Contingency Plan.** Sabotage includes discovery of a bomb device. Intruders are armed or unarmed personnel that have gained unauthorized access ~~in a hostile manner~~.

Thus, EAL 1 is written as:

"Security Emergency" or "Security Alert" Declared For: • Intrusion into the Plant Protected Area • Sabotage inside the Plant Protected Area

Source Documents/References/Calculations:

None

CN600034

SECURITY

Emergency Classification Level: SITE EMERGENCY

Applicable Operational Modes: ALL

Calvert Cliffs Initiating Condition:

TS1 Security Event in a Plant Vital Area

NUMARC Recognition Category: Hazards and Other Conditions Affecting Plant Safety

NUMARC Initiating Condition:

HS1 Security Event in Plant Vital Area

Barrier: Not Applicable

NUMARC Generic Basis:

This class of security events represents an escalated threat to plant safety above that contained in the Alert IC in that a hostile force has progressed from the Protected Area to the Vital Area. < >

Plant-Specific Information:
The Calvert Cliffs EALs address the generic areas of concern. **Nuclear Security will determine whether or not intrusion or sabotage exists in accordance with the Safeguards Contingency Plan.** Sabotage includes discovery of a bomb device. Intruders are armed or unarmed personnel that have gained unauthorized access ~~in a hostile manner~~.

Thus, the EAL ~~1~~ is written as:

"Security Emergency " or "Security Alert" Declared For: • Intrusion into an Aarea of ~~the plant that is a concern for safe shutdown of either reactor~~ **Concern for Safe Shutdown.** • Sabotage within an Aarea of ~~the plant that is a concern for safe shutdown of either reactor~~ **Concern for Safe Shutdown.**

The list of areas of concern for Safe Shutdown are shown below and are prominently displayed on the EAL Table.

Areas of Concern for Safe Shutdown	
• Control Room	• Electrical Penetration Rooms
• Control Room HVAC Room	• Auxiliary Feedwater Pump Room
• Cable Spreading Room	• Charging Pump Rooms
• Cable Chases	• Diesel Generator Rooms
• Switchgear Room	• Diesel Generator Building (0C/1A)
• ECCS Pump Room	• Refueling Water Tank (RWT) 11(21)
• Service Water Pump Room	• Condensate Storage Tank (CST) 12
• Component Cooling Pump Room	• Pretreated Water Storage Tank (PWST) 11(21)
• Main Steam Penetration Room	• Fuel Oil Storage Tank (FOST) 12
This list of Safe Shutdown areas is displayed on the EAL Tables to assure that all areas related to Safe Shutdown are considered by the SEC.	

~~EAL 2 is written as:~~

~~Sabotage within an area of the plant that is a concern for safe shutdown of either reactor.~~

Source Documents/References/Calculations:

1. **NRC Information Notice No. 96-71: Licensee Response to Indications of Tampering, Vandalism, or Malicious Mischief.**

Emergency Classification Level: ALERT

Applicable Operational Modes: ALL

Calvert Cliffs Initiating Condition:

IA1 Fire or Explosion Affecting Safe Shutdown

NUMARC Recognition Category: **Hazards and Other Conditions Affecting Plant Safety**

NUMARC Initiating Condition:

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

Barrier: Not Applicable

NUMARC Generic Basis:

(Site-specific) Areas containing functions and systems required for the safe shutdown of the plant should be specified. (Site-Specific) Safe Shutdown Analysis should be consulted for equipment and plant areas required for the applicable mode. This will make it easier to determine if the fire or explosion is potentially affecting one or more trains of safety systems. Escalation to a higher emergency class, if appropriate, will be based on <Equipment Failure, Electrical, Fission Product Barrier Degradation, Radioactivity Release, or SEC Judgement ICs>. <>

With regard to explosions, only those explosions of sufficient force to damage permanent structures or equipment required for safe operation within the identified plant area should be considered. As used here, an explosion is a rapid, violent, unconfined combustion, or a catastrophic failure of pressurized equipment, that potentially imparts significant energy to near-by structures and materials. The inclusion of a "report of visible damage" should not be interpreted as mandating a lengthy damage assessment before classification. No attempt is made in this <Generic> EAL to assess the actual magnitude of the damage. The occurrence of the explosion with reports of evidence of damage (*e.g.*, deformation, scorching) is sufficient for the declaration. The declaration of an Alert and the activation of the TSC will provide the <SEC> with the resources needed to perform these damage assessments. The <SEC> also needs to consider any security aspects of the explosions, if applicable.

Plant-Specific Information:

Each Calvert Cliffs unit uses the Abnormal Operating Procedures (AOP) 9A through 9S to address fires within the plant protected and vital areas that are of particular concern because they contain equipment required for safe shutdown.

Thus, EAL 1 is written as:

AOP-9 series Implemented for Fire..

There are two independent clocks for determining the magnitude of a fire based on time. One clock starts when a fire is detected. For practical purposes a fire is detected when the report of the fire is received in the Control Room. Report of a fire may be by Control Room fire alarm or by voice message. A fire alarm refers to 1C24B, Fire System Control Panel, for fire detection and fire suppression system actuation. Fire pump running and trouble alarms by themselves do not constitute a report of a fire. This clock includes: the time it takes to confirm or verify the fire report, plus the response team assembly time, plus the time it takes the responders to establish a fire fighting strategy, plus the time it takes to actually extinguish the fire.

Thus, EAL ~~1~~2 is written as:

Fire in an Area of Concern for Safe Shutdown that is not extinguished within 30 minutes of its detection.

Visible smoke is sufficient to conclude that a fire exists. Flames do not have to exist. Odor by itself does not constitute a fire.
A fire is extinguished when the Fire Brigade Leader determines that active combustion has ceased and there is no immediate danger of the fire spreading.

The other clock for determining the magnitude of a fire is the time it takes to extinguish the fire. This clock begins when the first extinguishing agent is applied to the fire.

Thus, EAL ~~2~~ **3** is written as:

Fire in an Area of Concern for Safe Shutdown that is not extinguished within 15 minutes of the first extinguishing agent being applied.

This EAL accounts for situations where the time to validate and respond to the fire is short.

EAL ~~3~~ **4** is written as:

Explosion in an Area of Concern for Safe Shutdown.

An explosion is a rapid, violent, unconfined combustion, a catastrophic failure of pressurized equipment, or a violent electric arc, of sufficient force to potentially damage equipment, structures or components.

Fire and/or explosion in the Control Room HVAC Room may lead to power being lost to the alternate shutdown panels. Thus, the Control Room HVAC Room (Room 512) has been added to the areas of concern for safe shutdown. The list of areas of concern for Safe Shutdown are shown below and are prominently displayed on the EAL Table.

Areas of Concern for Safe Shutdown	
• Control Room	• Electrical Penetration Rooms
• Control Room HVAC Room	• Auxiliary Feedwater Pump Room
• Cable Spreading Room	• Charging Pump Rooms
• Cable Chases	• Diesel Generator Rooms
• Switchgear Room	• Diesel Generator Buildings (0C/1A)
• ECCS Pump Room	• Refueling Water Tank (RWT) 11(21)
• Service Water Pump Room	• Condensate Storage Tank (CST) 12
• Component Cooling Pump Room	• Pretreated Water Storage Tank (PWST) 11(21)
• Main Steam Penetration Room	• Fuel Oil Storage Tank (FOST) 21
This list of Safe Shutdown areas is displayed on the EAL Tables to assure that all areas related to Safe Shutdown are considered by the SEC.	

The significance of these EALs is not that safety systems have been degraded. What is significant is that a fire of such magnitude that it can not be extinguished in the times specified exists in an area of concern for safe shutdown. Likewise, an explosion is significant because it occurred in an area of concern for safe shutdown, not because it degraded safety systems.

EAL 2 is written as:

Verified Report to Control Room of ~~Visible Damage to Safe Shutdown Equipment~~ **Observable Damage in an Area of Concern for Safe Shutdown.**

Verification of damage can be by physical observation, or by indications of degraded equipment performance in the Control Room or at local control stations.

EAL 3 uses a sustained wind speed of 90 MPH to address high winds striking the Plant Vital Area as recommended by NUMARC. This speed is chosen to assure that the wind speed is within the design capability of the meteorological tower.

Thus, EAL 3 is written as:

Sustained Wind Speed GREATER THAN 90 MPH (40 meters/second) for AT LEAST 15 Minutes

The duration of 15 minutes is selected to indicate sustained winds and to preclude wind gusts. Wind speeds are also provided here in meters/second for dose assessment input. The conversion equation is as follows:

90 miles/hour x 5280 feet/mile x (1 hour/3600 seconds) x 1 meter/3.2808 feet) = 40 meters/second

Per UFSAR Section 2.8.3.6, the still water level used for Intake Structure analysis is 17.6 feet MSL. This is above the top of the range of the Tide Level Recorder (0-LR-5195). The top of the Traveling Screen cover housings is about 18 feet MSL. EAL 4 indicates achieving the design water level.

Thus, EAL 4 is written as:

Bay Water Level At Or Above the Top of the Traveling Screen Cover Housing

Per UFSAR Section 2.8.3.7, the predicted extreme low tide is -3.6 feet MSL and the plant is designed to safely operate at an extreme low water level of -6.0 feet MSL. EAL 5 is based on the lower elevation.

Thus, EAL 5 is written as:

Bay Water Level Is AT LEAST 6 Feet Below Mean Sea Level

Surveillance Test Procedures provide a way to determine Bay level.

Source Documents/References/Calculations:

1. Updated Final Safety Analysis Report
2. Operating Instruction (OI) 46, Seismic Measurement Equipment
3. BG&E Drawing 60-220-E (M-31), Equipment Location Service Building, Water Treatment Area & Intake Structure Section "J-J"
4. BG&E Internal Memorandum, J.E. Thorp to R.E. Denton, Emergency Action Level Review Criteria, June 1, 1990

Emergency Classification Level: UNUSUAL EVENT

Applicable Operational Modes: ALL

Calvert Cliffs Initiating Condition:

OU1 SEC Judgement

NUMARC Recognition Category: **Hazards and Other Conditions Affecting Plant Safety**

NUMARC Initiating Condition:

HU5 Other Conditions Existing Which in the Judgement of the Emergency Director Warrant Declaration of an Unusual Event

Barrier: Not Applicable

NUMARC Generic Basis:

This <Generic> EAL is intended to address unanticipated conditions not addressed explicitly elsewhere but that warrant declaration of an emergency because conditions exist which are believed by the <SEC> to fall under the Unusual Event emergency class.

From a broad perspective, one area that may warrant <SEC> judgement is related to likely or actual breakdown of site specific event mitigating actions. Examples to consider include inadequate emergency response procedures, transient response either unexpected or not understood, failure or unavailability of emergency systems during an accident in excess of that assumed in accident analysis, or insufficient availability of equipment and/or support personnel.

Specific examples of actual events that may require <SEC> judgement for Unusual Event declaration are listed here for consideration. However, this list is by no means all inclusive and is not intended to limit the discretion of the site to provide further examples.

- Aircraft crash on-site
- Train derailment on-site
- Near-site explosion which may adversely affect normal site activities.
- Near-site release of toxic or ~~flammagle~~ **flammable** gas which may adversely affect normal site activities
- Uncontrolled RCS cooldown due to Secondary Depressurization

It is also intended that the <SEC's> judgement not be limited by any list of events as defined here or as augmented by the site. This list is provided solely as examples for consideration and it is recognized that actual events may not always follow a pre-conceived description.

Plant-Specific Information:

Site Emergency Coordinator (SEC) is the title for the emergency director function at Calvert Cliffs.

Thus, ~~the~~ EAL 1 is written as:

Any Condition Which in the SEC's Judgement Indicates Potential Degradation in the Level of Safety of the Plant

In this manner, the EAL addresses conditions that fall under the Notification of Unusual Event emergency classification description contained in NUREG-0654, Appendix 1 that is retained under the NUMARC methodology.

Uncontrolled RCS cooldown due to secondary depressurization is given as an example under this initiating condition. In order to reduce the need for judgment in recognizing this condition, a separate EAL is written for EOP-4 implementation. EOP-4 is implemented for this condition at Calvert Cliffs. Other examples given in the generic basis are addressed as specific EAL's and under OU2 and OU3.

Thus EAL 2 is written as:

EOP-4, Excess Steam Demand Event, is Implemented.

Source Documents/References/Calculations:

1. Emergency Response Plan
2. NUREG-0654/FEMA-REP-1, Criteria for Preparation and Evaluation of Radiological Emergency Response Plans and Preparedness in Support of Nuclear Power Plants, Revision 1, October 1980, Appendix 1

EAL 1 is written as:

~~Nuclear Security Report of an~~ **Explosion Causing Observable Damage to Permanent Equipment or Structures** Within the Plant Protected Area Or Within the ISFSI Protected Area

EAL 2 is written as:

~~Visible~~ **Observable** Damage ~~to Safe Shutdown Equipment Or~~ to Permanent Equipment or Structures Within the **Plant Protected** Area **Or Within the** ISFSI Protected Area

EAL 3 is written as:

Turbine Failure Causing Observable Casing Damage

Observable is used to indicate that such damage can be readily seen and does not require special equipment or techniques to see or measure.

EAL 4 is written as:

Vessel or Vehicle Collision Causing Observable Damage to ~~Safe Shutdown~~ **Permanent Equipment or Structures Within the Plant Protected Area.**

EAL 5 is written as:

Vessel or Vehicle Collision Causing Observable Damage to Structures Containing Dry Stored Spent Fuel

EALs 4 and 5 address airplane, helicopter, barge, boat, train, car, or truck collisions ~~into~~ **that may potentially damage** equipment required to achieve or maintain safe shutdown or with the Horizontal Storage Modules and their associated structural supports. These EALs do not include vehicle crashes with each other, damage to office structures, damage to equipment ~~not required to achieve or maintain safe shutdown~~ **that does not affect plant safety,** or damage to structures that are not required to maintain the integrity of the dry spent fuel stored in the ISFSI~~. Safe Shutdown areas and equipment of concern are identified below~~. **Actual damage in areas of Concern for Safe Shutdown are escalated to Alert whereas damage in the plant protected area is recognized as only having the potential for affected safe shutdown.**

~~Areas of Concern for Safe Shutdown~~	
~~• Control Room~~	~~• Electrical Penetration Rooms~~
~~• Control Room HVAC Room~~	~~• Auxiliary Feedwater Pump Room~~
~~• Cable Spreading Room~~	~~• Charging Pump Rooms~~
~~• Cable Chases~~	~~• Diesel Generator Rooms~~
~~• Switchgear Room~~	~~• Diesel Generator Building (0C/1A)~~
~~• ECCS Pump Room~~	~~• Refueling Water Tank (RWT) 11(21)~~
~~• Service Water Pump Room~~	~~• Condensate Storage Tank (CST) 12~~
~~• Component Cooling Pump Room~~	~~• Pretreated Water Storage Tank (PWST) 11(21)~~
~~• Main Steam Penetration Room~~	~~• Fuel Oil Storage Tank (FOST) 12~~
~~This list of Safe Shutdown areas is displayed on the EAL Tables to assure that all areas related to Safe Shutdown are considered by the SEC.~~	

EAL 6 is written as:

Flooding ~~of Rooms Containing Safe Shutdown Equipment~~ **Causing Observable Damage to Permanent Equipment or Structures Within the Plant Protected Area**

Flooding indicates that the net water flow into the room results in elevated water levels, may be more than available drain capacity, and if continued, can prevent operation of equipment in the room. Thus, minor water level increases that may result in wet floors and do not pose a challenge to equipment operation are not included in this EAL. Areas containing equipment required for Safe Shutdown are listed above. The rooms located below MSL include the ECCS Pump Rooms and the Charging Pump Rooms. The Shutdown Cooling Heat Exchangers are also located in the ECCS Pump Rooms. Such flooding can result in a potential degradation in the level of safety of the Calvert Cliffs plant and is therefore included in this EAL.

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Emergency Classification Level: ALERT

Applicable Operational Modes: ALL

Calvert Cliffs Initiating Condition:

OA2 Toxic or Flammable Gases Affecting Safe Shutdown

NUMARC Recognition Category: **Hazards and Other Conditions Affecting Plant Safety**

NUMARC Initiating Condition:

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

Barrier: Not Applicable

NUMARC Generic Basis:

This IC is based on gases that have entered a plant structure affecting the safe operation of the plant. This IC applies to buildings and areas contiguous to plant Vital Areas or other significant buildings or areas (*i.e.*, Service Water Pumphouse). The intent of this IC is not to include buildings (*i.e.*, warehouses) or other areas that are not contiguous or immediately adjacent to plant Vital Areas. It is appropriate that increased monitoring be done to ascertain whether consequential damage has occurred. Escalation to a higher emergency class, if appropriate, will be based on <Electrical, Equipment Failure, Radioactivity Release, Fission Product Barrier Degradation, or SEC Judgement ICs.> <>

Plant-Specific Information:

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

Thus, the EAL is written as:

Toxic or Flammable Gas Making ~~Safe Shutdown Areas Inaccessible~~ **an Area of Concern for Safe Shutdown Inaccessible.**

This EAL also addresses releases that could originate from the Cove Point Liquid Natural Gas Plant.

The areas of concern for safe shutdown are identified below.

Areas of Concern for Safe Shutdown	
• Control Room	• Electrical Penetration Rooms
• Control Room HVAC Room	• Auxiliary Feedwater Pump Room
• Cable Spreading Room	• Charging Pump Rooms
• Cable Chases	• Diesel Generator Rooms
• Switchgear Room	• Diesel Generator Building (0C/1A)
• ECCS Pump Room	• Refueling Water Tank (RWT) 11(21)
• Service Water Pump Room	• Condensate Storage Tank (CST) 12
• Component Cooling Pump Room	• Pretreated Water Storage Tank (PWST) 11(21)
• Main Steam Penetration Room	• Fuel Oil Storage Tank (FOST) 12
This list of Safe Shutdown areas is displayed on the EAL Tables to assure that all areas related to Safe Shutdown are considered by the SEC.	

Emergency Classification Level: ALERT

Applicable Operational Modes: ALL

Calvert Cliffs Initiating Condition:

OA3 Destructive Phenomena Affecting Safe Shutdown

NUMARC Recognition Category: Hazards and Other Conditions Affecting Plant Safety

NUMARC Initiating Condition:

HA1 Natural and Destructive Phenomena Affecting the Plant Vital Area

Barrier: Not Applicable

NUMARC Generic Basis:

Generic EALs 1, 2, and 3 are addressed under IC NA1, Natural Phenomena Affecting Safe Shutdown.

<Generic> EAL 4 should specify the types of instrumentation or indications including judgement which are to be used to assess occurrence.

<Generic> EAL 5 is intended to address such items as plane or helicopter crash, or on some sites, train crash, or barge crash into a plant vital area.

<Generic> EAL 6 is intended to address the threat to safety-related equipment imposed by missiles generated by main turbine rotating component failures. This (site-specific) list of areas should include all safety-related equipment, their controls, and their power supplies. This EAL is, therefore, consistent with the definition of an ALERT in that if missiles have damaged or penetrated areas containing safety-related equipment the potential exists for substantial degradation of the level of safety of the plant.

<Generic> EAL 7 covers other (Site-Specific) phenomena such as flood.

Each of these <generic> EALs is intended to address events that may have resulted in a plant vital area being subjected to forces beyond design limits, and thus damage may be assumed to have occurred to plant safety systems. The initial "report" should not be interpreted as mandating a lengthy damage assessment prior to classification. No attempt is made in th<ese> EAL<s> to assess the actual magnitude of the damage. Escalation to a higher emergency class, if appropriate, will be based on <Equipment Failure, Electrical, Fission Product Barrier Degradation, Radioactivity Release, or SEC> Judgement ICs.

Plant-Specific Information:

The Calvert Cliffs EALs are based on report to the control room of damage affecting safe shutdown functions.

EAL 1 addresses airplane, helicopter, barge, boat, train, car, or truck collisions. This EAL does not include vehicle crashes with each other, damage to office structures, or damage to structures that are not safety-related.

Thus, EAL 1 is written as:

Vessel or Vehicle Collision ~~Affecting the Ability to Achieve Or Maintain Safe Shutdown~~ Causing Observable Damage in an Area of Concern for Safe Shutdown

CN600043

EAL 2 is written as:

Missiles ~~Affecting the Ability to Achieve Or Maintain Safe Shutdown~~ **Causing Observable Damage in an *Area of Concern for Safe Shutdown***

EAL 3 is written as:

Flooding ~~Affecting the Ability to Achieve Or Maintain Safe Shutdown~~ **Causing Observable Damage in an Area of Concern for Safe Shutdown**

Determination of whether the collision, missiles, or flooding are *affecting* ability to achieve or maintain safe shutdown is determined by physical observation, or by Control Room/local control station indications. Observation of damage to systems should be used to discriminate between major flooding and minor flooding or flooding in areas having a low probability of affecting safe shutdown. Operability determinations are not expected prior to declaration of this event-based EAL.

The list of areas of concern for Safe Shutdown are shown below and are prominently displayed on the EAL Table.

Areas of Concern for Safe Shutdown	
• Control Room	• Electrical Penetration Rooms
• Control Room HVAC Room	• Auxiliary Feedwater Pump Room
• Cable Spreading Room	• Charging Pump Rooms
• Cable Chases	• Diesel Generator Rooms
• Switchgear Room	• Diesel Generator Building (0C/1A)
• ECCS Pump Room	• Refueling Water Tank (RWT) 11(21)
• Service Water Pump Room	• Condensate Storage Tank (CST) 12
• Component Cooling Pump Room	• Pretreated Water Storage Tank (PWST) 11(21)
• Main Steam Penetration Room	• Fuel Oil Storage Tank (FOST) 12
This list of Safe Shutdown areas is displayed on the EAL Tables to assure that all areas related to Safe Shutdown are considered by the SEC.	

Source Documents/References/Calculations:

1. Updated Final Safety Analysis Report

BALTIMORE
GAS AND
ELECTRIC

CHARLES CENTER • P.O. BOX 1475 • BALTIMORE, MARYLAND 21203-1475

LEON B. RUSSELL
MANAGER
NUCLEAR SAFETY &
PLANNING DEPARTMENT

June 6, 1991

Mr. James H. Joyner, Chief
Facilities Radiological Safety and Safeguards Branch
Division of Radiation Safety and Safeguards
U.S. Nuclear Regulatory Commission
Region I
475 Allendale Road
King of Prussia, PA 19406

SUBJECT: Calvert Cliffs Nuclear Power Plant
Unit Nos. 1 & 2; Docket Nos. 50-317 & 50-318
Emergency Action Level Review Meeting

REFERENCES: (a) Letter from Mr. J. H. Joyner (NRC) to Mr. G. C. Creel (BG&E), dated February 12, 1991

(b) Letter from Mr. G. C. Creel (BG&E) to Mr. J. H. Joyner (NRC), dated March 29, 1991

Dear Mr. Joyner:

This is to confirm arrangements for Baltimore Gas and Electric Company (BG&E) to meet with your staff on Thursday, June 20, 1991, 10:00 a.m., in the Nuclear Regulatory Commission (NRC) Region I Offices, King of Prussia, Pennsylvania. The purpose of this meeting is to present a proposed draft revision to Emergency Action Levels (EALs).

As you are aware, Reference (a) discusses the results of NRC staff review of Revision 14, Change 3 to the Emergency Classification and Action Level Scheme for Calvert Cliffs Nuclear Power Plant. This review concluded that the EAL scheme has been improved, but that additional changes are needed to meet the guidance of NUREG-0654. Reference (b) provides BG&E comment on the Staff's review and a schedule for implementing changes to the EAL scheme. In keeping with Reference (b), BG&E will present an EAL proposed revision in draft on June 20, 1991. The specifics on the revision will be discussed during this meeting.

I want to thank you for making your staff available for this meeting. It is our endeavor to resolve NRC staff concerns in a timely manner. Getting together to review this proposal in draft should facilitate an understanding of the issues and expedite NRC's formal review (Region and NRR).

EPU Received 6/10/91
Sent

FOLLOW UP
FILE NO. 4.3 ERPP Gen Corres

BALTIMORE GAS AND ELECTRIC

CHARLES CENTER • P.O. BOX 1475 • BALTIMORE, MARYLAND 21203-1475

LEON B. RUSSELL
MANAGER
NUCLEAR SAFETY &
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EPII Received 6/10/91
Date

FOLLOW UP
FILE NO. 4.3 ERAP Gen Corres

I envision this to be a working meeting of one to two hour duration. An agenda in attached. Please don't hesitate to contact me at (301) 260-6680 if questions arise.

Sincerely,

L B Russell

L. B. Russell
Manager
Nuclear Safety & Planning Department

LBR/TEF/GLD/bjd

Attachment

cc: D. A. Brune, Esquire
J. E. Silberg, Esquire
R. A. Capra, NRC
D. G. McDonald, Jr., NRC
T. T. Martin, NRC
L. E. Nicholson, NRC
R. I. McLean, DNR
J. H. Walter, PSC

ATTACHMENT (1)

BG&E / NRC MEETING

Proposed EAL Revision

Thursday, June 20, 1991

King of Prussia, Pennsylvania

10:00 a.m.	Personnel introduction and introductory remarks.	L. B. Russell
10:10 a.m.	Overview: Background Meeting process	T. E. Forgette
10:20 a.m.	EAL proposed revision presentation/ discussion Unusual Event Alert Site Emergency General Emergency	All
11:20 a.m. (+)	Meeting recount and adjournment	T. E. Forgette

CALVERT CLIFFS NUCLEAR POWER PLANT

BGE

PROBABILISTIC RISK ASSESSMENT

Individual Plant Examination of External Events
Summary Report
August 1997

97-031
8/26/97

CALVERT CLIFFS NUCLEAR POWER PLANT
PROBABILISTIC RISK ASSESSMENT
PROJECT TEAM

PROJECT MANAGER

Bruce B. Mrowca

BGE TEAM MEMBERS

Gayle A. Blizzard

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Timothy A. Rogers

Eric A. Schade

Jeff L. Stone

DOCUMENT PREPARATION SUPPORT

Helen C. Buck, Dawn M. Cox

LIST OF EFFECTIVE PAGES

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B	A226	Unit 1 Service Water Pump Room
C	A227/A316	East Piping Penetration Rooms
D	A228	Unit 1 Component Cooling Water Pump Room
E	CSR	Unit 1 & 2 Cable Spreading Room
E1	CSR	Passageway
F	SWGR	Switchgear Rooms
G	A315	Unit 1 Main Steam Isolation Valve Room
H	A318	Unit 1 Purge Air Supply Fan Room
I	MCR	Main Control Room
J	A419	Adjoining areas of the 45' Elevation of the Auxiliary Building
K	A423	Unit 1 West Electrical Penetration Room
L	A429	Unit 1 East Electrical Penetration Room
M	A512	Control Room HVAC Room, Spent Fuel Vent Room, Unit 1 Main Vent Fan Room, Unit 1 Containment Access Area
N	A529	Unit 1 69' West Electrical Room
O	AB	Auxiliary Building Stairtowers
P	CC-AB	Cable Chases 1A, 1B, 2A, 2B, A518, and A517
Q	CC-C	Cable Chases 1C & 2C
R	INTAKE	Intake Structure
S	T603	Unit 1 AFW Pump Room
T	TB	Unit 1 and 2 Turbine Building
U	YARD	Yard Areas

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List of Acronyms

AC	Air Conditioning
ACA	Access Control Area
ADV	Atmospheric Dump Valve
AFAS	Auxiliary Feedwater Actuation System
AFW	Auxiliary Feedwater
AIB	Auxiliary Improvement Bulletins
AOPs	Abnormal Operating Procedures
APCSB	Auxiliary and Power Conversion System Branch
ASA	All Support Available
ASCE	American Society of Civil Engineers
ATWS	Automated Transient Without Scram
BAST	Boric Acid Storage Tank
BGE	Baltimore Gas & Electric
BTP	Branch Technical Position
CAC	Containment Air Coolers
CCDP	Conditional Core Damage Probability
CCFPRA	Calvert Cliffs Fire Probabilistic Risk Assessment
CCNPP	Calvert Cliffs Nuclear Power Plant
CCPRA	Calvert Cliffs Probabilistic Risk Assessment
CCSPRA	Calvert Cliffs Seismic Probabilistic Risk Assessment
CCW	Component Cooling Water
CDF	Core Damage Frequency
CEA-MG	Controlled Element Assemblies - Motor Generator
CEDM	Control Element Drive Mechanism
CEDS	Control Element Drive System
CH	Channel
CKV	Check Valve
CNTMT	Containment
CR	Control Room
CRS	Circuit & Raceway System
CSAS	Containment Spray Actuation System
CSR	Cable Spreading Room
CST	Condensate Storage Tank
CV	Control Valve
DAS	Data Acquisition System
DBE	Design Basis Earthquake
DHR	Decay Heat Removal
DWT	Demineralized Water Storage Tank
ECCS	Emergency Core Cooling System
EDGs	Emergency Diesel Generators
EHC	Electro-Hydraulic Control
EMD	Emergency Management Division
EOPs	Emergency Operating Procedures
EPRI	Electrical Power Research Institute
ERPIP	Emergency Response Plan Implementation Procedures
ESFAS	Emergency Safety Feature Actuation System
FAA	Federal Aviation Agency

List of Acronyms (cont'd)

FCIA	Fire Compartment Interaction
FCR	Facility Change Request
FEDB	Fire Events Database
FEMA	Federal Emergency Management Agency
FIVE	Fire-Induced Vulnerability Evaluation
FOST	Fuel Oil Storage Tank
FPRA	Fire Probabilistic Risk Analysis
FRSS	Fire Risk Scoping Study
HCLPF	High Confidence of Low Probability of Failure
HCR	Human Cognitive Reliability
HDR	Header
HELB	High Energy Line Break
HEPA	High Efficiency Particulate Air
HGL	Hot Gas Layer
HMR	Human Action Methodology Report
HPSI	High Pressure Safety Injection
HRA	Human Reliability Analysis
HRR	Heat Release Rate
HVAC	Heating Ventilation Air Conditioning
HX	Heat Exchanger
IA/PA	Instrument Air/Plant Air
IE	Initiating Event
IEEE	Institute of Electrical and Electronic Engineers
ILRT	Individual Leak Rate Test
IPE	Individual Plant Examination
IPEEE	Individual Plant Examination of External Events
ISFSI	Independent Spent Fuel Storage Installation
ISFSI	Independent Spent Fuel Storage Installation
LLNL	Lawrence Livermore National Laboratory
LLOCA	Large LOCA
LNG	Liquified Natural Gas
LNG	Liquified Natural gas
LOCA	Loss of Coolant Accident
LOOP	Loss Of Offsite Power
LPSI	Low Pressure Safety Injection
LTM	Low Trajectory Missile
MCCs	Motor Control Center
MCR	Main Control Room
MFW	Main Feedwater
MOV	Motor Operated Valve
MSDS	Material Safety Data Sheet
MSIV	Main Steam Isolation Valve
MSL	Mean Sea Level
MTC	Moderate Temperature Coefficient
MU	Make Up
NAS	Naval Air Station
NFPA	National Fire Protection Association

List of Acronyms (cont'd)

NM	Nautical Mile
NRC	Nuclear Regulatory Commission
NSAC	Nuclear Safety Analysis Center
NSR	Non-Safety Related
NSSS	Nuclear Steam Supply System
OBE	Operational Basis Earthquake
OBE	Operational Basis Earthquake
OI	Operating Instructions
OL	Operating License
OP	Operator
ORE	Operator Reliability Experiments
OTCC	Once Through Core Cooling
PCB	Polychlorinated Biphenyl
PDS	Plant Damage State
PE	Performance Evaluation
PEG	Plant Engineering Guideline
PGA	Peak Ground Acceleration
PMH	Probable Maximum Hurricane
PMP	Probable Maximum Precipitation
PNS	Probability of Non-Suppression
PORV	Power Operated Relief Valve
POSRC	Plant Operations Safety Review Committee
PP	Pump
PRA	Probabilistic Risk Assessment
PSF	Pounds Per Square Foot
PSF	Performance Shaping Factors
PTS	Pressure Thermal Shock
PWST	Pre-Treated Water Storage Tank
QA	Quality Assurance
RAS	Recirculation Actuation System
RC	Reactor Coolant
RCP	Reactor Coolant Pump
RCS	Reactor Coolant System
RCW	Reactor Coolant Waste
RLE	Review Level Earthquake
RMIEP	Risk Methods Integration and Evaluation Program Methods Development
RMS	Radiation Monitoring System
RPS	Reactor Protection System
RRS	Reactor Regulating System
RWT	Refueling Water Storage Tank
S&A	Stevenson & Associates
S/D	Shutdown
S/G	Steam generator
SBO	Station Blackout
SCBA	Self-Contained Breathing Apparatus
SF	Split Fraction
SFP	Spent Fuel Pool

List of Acronyms (cont'd)

SGIs	Steam Generator Isolation Signal
SIAS	Safety Injection Actuation System
SLB	Steam Line Break
SLI	Success Likelihood Index
SLIM-MAUD	Success Likelihood Index Methodology - Multi-Attribute Utility Decomposition
SNL	Sandia National Laboratory
SPT	Standard Penetration Test
SQUG GIP	Seismic Qualification Utility Group Generic Implementation Procedure
SR	Safety Related
SRP	Standard Review Plan
SRT	Seismic Review Team
SRV	Safety Relief Valve
SRW	Service Water
SSE	Safety Shutdown Earthquake
SSI	Soil Structural Interaction
SSSA	Spurious Safety System Actuation
STP	Surveillance Test Procedure
SW	Saltwater
SWAC	Salt Water Air Compressor
SWGR	Switchgear
T/G	Turbine Generator
TBV	Turbine Bypass Valve
TNT	Trinitrotoluene
TSC	Technical Support Center
TURB	Turbine
UFSAR	Updated Final Safety Analysis Report
USDOT	United States Department of Transportation
USI	Unresolved Safety Issue
USNRC	United States Nuclear Regulatory Commission
UV	Undervoltage
VCT	Volume Control Tank
VMT	Vessel Melt Through
WANO	World Association of Nuclear Operators
XFMER	Transformer
ZPA	Zero Period Acceleration

SECTION 2

EXAMINATION DESCRIPTION

2.1 Introduction

The requirements for CCNPP's IPEEE are satisfied with this submittal. CCNPP's IPEEE was started shortly after the completion of the IPE. As in the IPE, one of CCNPP's objectives was to utilize the in-house staff to the maximum extent possible to retain the knowledge within the utility. The IPEEE process was initiated consistent with the requirements of Generic Letter 88-20, Supplement 4 and using NUREG-1407 as guidance. The examination process, methodology and information assembly are described in the following sections.

2.2 Conformance with Generic Letter and Supporting Material

The major objectives of CCNPP's IPEEE are to meet the purposes of the IPEEE stated in the Generic Letter 88-20 (i.e., to develop an overall appreciation of severe accident behavior due to external events, to understand the most likely severe accident sequences that could occur under full-power operating conditions for these events, to gain a qualitative understanding of the overall likelihood of core damage and radioactive material release, and if necessary, to reduce the overall probability of core damage and radioactive material release by appropriate modifications to plant operating procedures and hardware that would help prevent or mitigate severe accidents).

As evidenced by the results documented in the various sections of this report, the active and direct involvement in the development and performance of the various IPEEE tasks has provided the means for BGE: (1) to understand severe accidents at the broad and detailed levels, (2) to understand the most likely severe accident sequences, dominant contributors to these sequences, and the resulting consequences, (3) to have an overview of the significance and magnitude of the most probable core damage sequences and release states, (4) to identify the vulnerabilities and recommend corrective actions in terms of design modification and/or procedure changes in order to reduce the overall core damage frequency.

2.3 General Methodology

The IPEEE for CCNPP is performed using the approaches identified in NUREG-1407. The general methodology being used specifically for each task is summarized below: The report is written from a Unit 1 perspective. Key differences between the Units are assessed.

<u>Seismic</u>

A seismic PRA is performed for CCNPP using the approaches described in NUREG-1407 and EPRI NP-6041-SL. Section 3 of this report provides a detailed description of the methods used. A very comprehensive component walkdown list is first developed followed by a screening process using the results from extensive plant walkdowns. The results from the A-46 project are also utilized for screening. HCLPF (high-confidence-of-low-probability of failure) calculations are performed for the second screening followed by detailed fragility calculations for all the non-screened components. A HCLPF of 0.3g is used for screening. Soil liquefaction, soil-structural interaction and structural fragility analyses are also performed and the results are used as input for the component HCLPF and fragility calculations.

The seismic initiating events are developed using the results from the fragility calculations. The annual probability of exceedance for peak ground acceleration from the Revised Livermore Seismic Hazard Estimate (NUREG -1488) for CCNPP is used for the initiating events binning. All the non-screened components are grouped as a surrogate component which is assumed to lead directly to core damage when failed. The seismic impact in terms of core damage frequency and containment performance are quantified using a modified version of CCNPP's IPE submittal model.

Contractors with specific seismic expertise were utilized, for example, EQE International performed the component walkdown, HCLPF screening and fragility calculations. Stevenson & Associates performed all the soil structural related analyses.

Fire

A fire PRA is performed for CCNPP using the approaches described in NUREG-1407 and the EPRI Fire PRA Implementation Guide. Section 4 of this report provides a detailed description of the methods used. The Fire Induced Vulnerability Evaluation (FIVE) methodology is used as guidance for the evaluation of specific scenarios within a compartment when the failure of the entire compartment is proven to be significant. The quantification of the fire induced core damage frequency is performed using a modified version of CCNPP's IPE submittal model. A containment performance review is also performed to identify any sequences that would lead to containment failure.

Other External Events

For the risk assessment of the other external events, the approaches provided in NUREG 1407 are followed, i.e., an initial screening analysis is performed followed by bounding or detailed analyses as necessary. Thus, the analysis process basically follows the steps outlined in Figure 5.1 of NUREG-1407. Section 5 of this report provides the detailed description of the methods used.

2.4 Information Assembly

The IPEEE process includes a considerable effort to assemble information relevant to all of the external events analyzed. A variety of internal and external information sources are utilized to support the IPEEE. Examples of internal documents are: UFSAR, OIs, AOPs, EOPs, ERPIP, various design calculations, etc. Examples of external documents are: NUREGs, Regulatory Guides, vendor's design calculations including Bechtel and turbine vendors, federal and local governments' reports on related subjects, etc.

The reference sections provided in Sections 3, 4 and 5 of this report provides more details on the information obtained for the seismic, fire and other events analyses.

SECTION 3

SEISMIC ANALYSIS

3.0 Methodology Selection

CCNPP's PRA was performed to satisfy the IPE requirements for internal initiating events on CCNPP. To assess the risk contribution and significance of seismic initiated events to the total plant risk, Baltimore Gas and Electric Company selected the PRA method for the seismic analysis to meet the requirements of the IPEEE.

The PRA method is selected over the seismic margins method because it allows component risk ranking and provides the ability to relate the seismic risk to the risk from internal events and other external events providing insight to the total risk profile.

The Seismic PRA is performed using the methodology outlined in NUREG-1407 (Ref. 3-13). The general transient linked event tree model is used to quantify seismic induced core damage frequency. A seismic event tree is added in front of the model to assess the probability of system failures directly due to the seismic event. The impact of the seismic event tree top event failures are then cascaded into the general transient event trees using split fraction assignment rules to reflect the dependencies of the seismic failures on the downstream top events.

Thirty seismic initiating events are used to generate accident sequences. These sequences are then analyzed to identify any potential vulnerabilities.

3.0.1 Coordination with USI A-46

Since the USI A-46 and Seismic IPEEE programs are similar in many respects, the programs where possible were coordinated to gain efficiency. EQE International, Inc. was contracted to perform equipment walkdowns and evaluations for both the USI-A46 and Seismic IPEEE programs.

The primary area of overlap between the two programs occurred in the component walkdown evaluation. The USI A-46 walkdowns followed the guidelines in the Seismic Qualification Utility Group (SQUG) Generic Implementation Procedure (GIP). These walkdowns are documented with detailed seismic evaluation worksheets, calculations and photographs. The seismic PRA takes into account the results of the A-46 walkdowns and anchorage calculations for equipment that appears in both programs.

3.1 Seismic PRA

3.1.1 Seismic Hazard Evaluation

3.1.1.1 Seismic Hazard Curves

Seismic hazard curves were developed for the CCNPP site by EPRI and LLNL. NUREG-1407 states that if only one set of curves are used for the analysis, the higher of the two should be used. For the CCNPP site, the LLNL curves are higher. The updated LLNL curves, provided in NUREG-1488 (Ref. 3-5), are used for the CCNPP Seismic PRA (CCSPRA). These curves are used as the basis for the seismic initiating event frequencies used for quantification and for the soil-structure interaction analysis and structure fragility calculations performed by S&A (Ref. 3-1).

Table 3-1 shows the Updated LLNL Seismic Hazard Curve Values for CCNPP. This table gives estimates for the annual probability of exceeding ten peak ground acceleration levels ranging from 50 to 1000 cm/sec/sec. The CCSPRA uses these values to generate thirty bins ranging from 12.5 to 1500 cm/sec/sec. These bins are shown in Table 3-2. A relatively high number of bins is used to attempt to minimize the conservatism in the calculation of the core damage frequency. Acceleration levels below the given range are logarithmically extrapolated and those above the given range are exponentially extrapolated to obtain the corresponding frequencies.

3.1.1.2 Liquefaction Analysis

The liquefaction analysis was performed, along with the soil-structure interaction analysis and structure fragility calculations, by S&A. The results of the analysis are contained in Ref. 3-1 and are summarized here.

CCNPP is located on a deep-soil site. The stratigraphy consists of a sequence of Quaternary, Tertiary, and Cretaceous sand, silt, clay, and gravel deposits which are about 2,500 feet thick at the site.

Existing available soil boring data includes that obtained from borings made in the 1967 & 1968 time frame prior to the initial excavation work as well as more recent borings made for the new Diesel Generator Project.

The new Diesel Generator Project added two new diesel generators, one self-cooling safety-related diesel in a concrete building and one augmented quality diesel enclosed in a steel frame building. The safety-related Diesel Generator Building was evaluated for liquefaction analysis. This building is referred to here and in Ref. 3-1 as the New Diesel Generator Building.

The results of the liquefaction evaluation conducted for the Containments, Auxiliary Building, Turbine Building, and the Intake Structure, indicate that the supporting soils are not subject to liquefaction for the Review Level Earthquake (RLE) with a peak ground acceleration of 0.4g. For the new Diesel Generator Building, initial liquefaction is expected to occur at a median peak ground acceleration of 0.27g with a range of 0.2g to 0.36g.

The stratum considered in determining the Zero Period Acceleration (ZPA) levels at which liquefaction could be imminent below the New Diesel Generator Building is located 16 feet below the foundation of the building and is (on average) nine feet thick. Due to this, plus the fact that the differential contact pressure is small compared to the existing overburden pressure prior to excavation (an increase of 1.2ksf), stability failure is not considered a realistic hazard. Hence, the only remaining issue for the new Diesel Generator Building is liquefaction-induced settlement, which is addressed below.

3.1.1.3 Seismically Induced Settlements of Structures

Seismic-induced settlements are estimated using empirical correlations. For the Containments, Auxiliary Building, and Intake Structure, essentially no seismic-induced settlements are indicated. Total seismic-induced settlement for the ground without any structures (i.e., considering the full soil column and blow counts from test borings made in areas not directly below any structure) is estimated to be 0.25 inches. Total seismic-induced settlements for the Turbine Building are estimated to be 0.5 inches and a total settlement ranging from 0.3 inches to 0.75 inches for the new Diesel Generator Building, again due to the range in Standard Penetration Test (SPT) blow counts for the six test borings. This amount of settlement is not considered significant.

A conservative review of the differential displacements between buildings (performed for the Containments and Auxiliary Buildings) indicates a probability of 0.1 that the buildings will impact if subjected to the RLE and assuming out-of-phase displacement. The effects of this interaction are not modeled.

3.1.2 Review of Plant Information and Walkdowns

The walkdowns were conducted by a Seismic Review Team (SRT) consisting of personnel from EQE and CCNPP.

The purposes of the walkdown were to: 1) visually inspect and screen-out inherently rugged components from further review, 2) define the failure modes (such as anchorage failure) and elements which are not screened, and 3) add to the analysis any seismic interaction items judged to be a potentially serious problem. The scope of the walkdowns and screening covered only equipment and components. Structure and soil analyses were performed separately by S&A and is discussed in Sections 3.1.1.2 and 3.1.3.1.

The component walkdown list is compiled from the components in the PRA database and the Q-list (Ref. 3-22). Initial screening is performed on these components as described in IPEEE Seismic Component Walkdown List, Ref. 3-3. For example, check valves are inherently rugged and are screened from the walkdown list. In addition, systems such as Main Feedwater, Condensate, and Circulating Water are not walked down because of their low importance in the PRA. Components that passed this initial screening comprise the walkdown list and were walked down.

The Seismic IPEEE is closely coordinated with USI A-46. Most of the components included in the Seismic IPEEE walkdown are also on the USI A-46 Safe Shutdown Equipment List. The scope of the Seismic IPEEE walkdown includes the additional work necessary to supplement the work done during the USI A-46 walkdown.

The scope of the walkdown also includes the identification of higher capacity equipment and components which may be screened out from explicit consideration in the CCSPRA.

NUREG-1407 directs that the walkdown follow the guidelines of EPRI NP-6041-SL, Rev. 1 (Ref. 3-4). The guidelines contained therein are used for both the walkdown and the screening out of higher capacity components.

3.1.2.1 Component Screening

For screening, the guidelines of Table 2-4 of EPRI NP-6041 are used. The guidelines are set to screen earthquakes of about 0.3g and 0.5g peak ground acceleration (0.8g and 1.2g peak spectral acceleration). For USI A-46, the equipment is evaluated for the plant's safe shutdown during an earthquake with a peak ground acceleration of 0.15g. Therefore, equipment and components are screened as follows:

- At 0.3g peak ground acceleration, if caveats of column 1 of EPRI NP-6041 are met and the A-46 anchorage calculation showed a factor of safety greater than two.

- At 0.5g peak ground acceleration, if caveats of column 2 of EPRI NP-6041 are met and the A-46 anchorage calculation showed a factor of safety greater than four.

One exception to the above has to do with seismic interactions, particularly block walls. For seismic interactions, the walkdown must note potential seismic interactions for earthquake levels above the Safe Shutdown Earthquake (SSE). These are not documented in the A-46 walkdowns. For block walls, nearby equipment and components could be screened at 0.3g if the wall is qualified by elastic analysis for IEB 80-11 (Ref. 3-9). It is assumed that all walls are reinforced and anchored in accordance with BGE Drawing 62-128-E, "Masonry Details." Equipment near walls qualified by inelastic analysis could be screened at 0.3g if the SRT judged that there is sufficient margin based on the wall height and thickness. In no case is equipment near masonry walls screened at the 0.5g level.

For seismic-fire and seismic-flood interaction, potential fire and flood sources are identified by BGE and then evaluated per the EPRI NP-6041 guidelines at a review level of 0.3g. Those sources not screened out are identified for further review and possible fragility analysis.

The walkdown lists and screening results are contained in Attachment A to Ref. 3-3. Attachment A is a complete walkdown and screening report prepared by EQE. It contains sections on methodology, the seismic review team, screening, seismic-fire interaction and seismic-flood interaction.

3.1.3 Analysis of Plant System and Structure Response

3.1.3.1 Structure Fragility Analysis

The Soil-Structure Interaction (SSI) Analysis and structure fragility analysis were performed by S&A and are documented in Ref. 3-1. The information presented here is from that report.

To estimate realistic fragility values for structures and components, it is necessary to consider an earthquake level where a number of components would be expected to fail or near their failure limit. The RLE was defined by the median shape Uniform Hazard Spectrum for CCNPP from NUREG-1488 for a 10,000-year return period at a peak ground acceleration (pga) of 0.4g set at 50 Hz. The pga is four times the associated median pga of 0.1g from NUREG-1488 and 2.67 times the Design Basis Earthquake of 0.15g.

Probabilistic seismic SSI analysis for five of the CCNPP major site buildings were performed. The buildings considered for probabilistic seismic SSI analysis were:

- Containment
- Auxiliary Building
- Intake Structure
- Turbine Building
- New Emergency Diesel Generator Building

Additionally, seismic response analyses are performed for the Fire Pump House and the Condensate Storage Tank and Fuel Oil Storage Tank enclosures for the purpose of calculating fragilities for these structures.

The results of the SSI analyses includes development of probabilistic in-structure response spectra for use in calculating equipment fragilities, and probabilistic nodal forces and moments in the structure models for use in calculating structure fragilities. The structure floor response spectra are contained in Ref. 3-1. Table 3-5 shows the fragility values for the structures.

3.1.3.2 Seismic Relay Chatter Analysis

An extensive relay evaluation (Ref. 3-12) was performed at CCNPP for the USI A-46 program. The evaluation found no 'bad actor' relays and therefore, for focused scope plants such as CCNPP, completion of the USI A-46 review satisfies the IPEEE intent for review of relays.

3.1.3.3 Seismic-Fire Interaction Analysis

NUREG-1407 and NUREG/CR-5088 explain that the IPEEE seismic walkdown and screening should address the following for seismic-fire interaction: Seismically-induced fires, seismic actuation of fire suppression systems and seismic degradation of fire suppression systems.

Seismic-Induced Fires

Seismic-induced fires are addressed by examining sources of flammable liquids or combustible gases which, in the presence of an ignition source, could damage equipment on the IPEEE component list. Materials may be screened from fire source consideration if they meet one of the following criteria:

a. The fire source materials are not explosive nor combustible and they have a high flash point.

b. The system containing the fire source material is not pressurized and operates in a low temperature environment.

c. The fire source material is not near vital equipment.

Based on the fire screening criteria above, the screened and not screened fire sources are listed in Ref. 3-2. The non-screened fire sources are then walked down. Generally, the non-screened fire sources includes the storage tanks and piping that contain flammable liquids or combustible gases. These sources include components such as the Volume Control Tanks, EDG Fuel Oil Day Tanks, hydrogen seal oil skid, hydrogen piping to the main generators, lube-oil piping to the main turbine, etc.

In addition to evaluating components specifically identified as potential fire sources, the Component Table (walkdown list) (Attachment A, Ref. 3-3) includes a column indicating whether or not the Seismic Review Team identified a seismic-induced fire concern for each component listed.

In the safety-related buildings, most of the tanks and piping are screened at either 0.3 or 0.5g. The non-safety-related buildings with flammables are the Fire Pump House, Turbine Building and the North Service Building.

CCNPP has two Fuel Oil Storage Tanks (FOST) which have fragilities of 0.22g; This is less than the 0.3g screening criteria. However, in the case of tank failure, the contents and possible resulting fire should be confined to the area immediately around the tank. The No. 21 FOST sits inside a tornado proof concrete building. Inside the building, but surrounding the lower portion of the tank, are concrete walls designed to contain the contents of the tank if it ruptures. The concrete structure, which has a HCLPF of 3.70g, is quite rugged and is not expected to fail in the earthquake range of interest.

The No. 11 FOST is surrounded by an earthen berm designed to contain the fuel oil if the tank fails. A fire here would probably affect 500 KV Unit 1 output to the Switchyard by burning the overhead power lines leading to the Switchyard. However, this impact is bounded by switchyard failure, with its much lower HCLPF of 0.09g. Therefore, fires resulting from FOST failure are not expected to have a significant direct impact on other equipment. The only remaining issue for a FOST fire is the smoke generated by the fire.

Smoke generated by a large outside fire could potentially be drawn into the Control Room and Cable Spreading Room Ventilation system and adversely affect the Control Room Operators. The likelihood of smoke from fires in various outside areas is addressed in this report in Section 4, Fire Analysis. Protection from smoke being drawn into the Control Room and Cable Spreading Room is addressed in this report in Section 7.0, Plant Improvement and Unique Safety Features.

There are various outside transformers located around the main plant buildings. Most of these are included in the generic fragility used for 500KV Switchyard and loss of off-site power. Assuming that the transformers have a similar capacity, their HCLPF, which corresponds to about a 1% failure probability, is 0.09g. The probability of exceeding 0.09g is approximately 3.7E-04 (based on the LLNL seismic hazard curves for CCNPP). The base frequencies for transformer fires analyzed in Section 4 are on the order of E-02. Therefore, the impact of a seismic-induced transformer fire is bounded by the fire initiating event frequencies. Section 4, Fire Analysis, also includes the impact of smoke from each transformer on the Control Room Ventilation System.

The only other outside fire source is the hydrogen storage area located in the tank farm. This fire source is evaluated in Section 4, Fire Analysis. The area consists of nine hydrogen cylinders in an area surrounded by a fence and concrete barriers. The tanks are far enough away from any vital equipment so that a fire here is not expected to have any adverse impact.

All fire sources are either screened at a HCLPF of 0.3g or are not near any vital equipment, and therefore, are not a significant risk.

Seismic Actuation of Fire Suppression Systems

This is discussed below in Seismic-Flood Interaction.

Seismic Degradation of Fire Suppression Systems

During the walkdown, the smoke detectors, halon bottles and piping were found to be well secured and screened at 0.5g. The fire system pumps, motors, and piping were walked down and are screened at a HCLPF of at least 0.3 g.

Therefore, there appears to be no significant risk associated with Seismic-Fire interaction. More details concerning particular components and screening criteria may be found in Ref. 3-3.

3.1.3.4 Seismic-Flood Interaction Analysis

Flood analysis includes equipment damage from spray as well as high water level. These risks may be created by failure of fluid system piping, water storage tanks inside safety-related buildings and by spurious actuation of fire suppression system.

Based on the plant walkdowns, all the safety-related piping systems and water storage tanks inside the Containment and the Auxiliary Building are considered seismically rugged and screened at a minimum HCLPF of 0.3g. The non-safety-related piping, particularly the Fire Protection system in the safety-related buildings, is well supported and is also screened at a HCLPF of at least 0.3g.

In the Turbine Building, piping in the proximity of electrical components and the Fire Protection system are screened at 0.3g. Part of the Service Water (SRW) piping did not meet the HCLPF screening due to spatial interaction. Thus, SRW could be a potential flood source if the piping breaks during a seismic event. However, the results of the IPE Internal Flooding analysis indicate that the flooding from SRW source in the Turbine Building presents no significant risk to plant safety.

Equipment damage from spray could also be caused by seismic actuation of the Fire Protection system or from failure of Fire Protection piping. All the Fire Protection piping in the safety and non-safety-related buildings is screened at a minimum HCLPF of 0.3g, as a result of the equipment walkdowns. Actuation due to relay chatter is not considered a credible risk for CCNPP, based on the relay evaluation performed for USI-A46.

Based on the above, it is concluded that seismic-induced flooding is unlikely except for SRW in the Turbine Building. However, this seismic flooding event presents no significant risk to plant safety.

3.1.4 Evaluation of Component Fragilities and Failure modes

The methodology used to screen components and calculate component fragilities is described in Ref. 3-2. The information in this section is taken from that report.

Relatively strong components may be identified and screened out by verifying during the plant walkdown that these rugged elements comply with the caveats in Tables 2-3 and 2-4 of Ref. 3-6. Use of the first column of these tables coupled with anchorage calculations based on a peak ground acceleration of 0.3g implies a minimum HCLPF of 0.3g.

If the plant component screening is performed at a HCLPF level of 0.3g, then the plant HCLPF capacity cannot exceed this level even if all components exceed the screening criteria. This is because a component is screened out based on its strength being greater than the screening level, but it is not known how much greater. Ref. 3-5 gives a simple approach to obtain a generally conservative estimate of the mean CDF if the plant HCLPF is known. Assuming a plant HCLPF of 0.3g and using the median seismic hazard curve for CCNPP from Ref. 3-4, this procedure estimates a core damage frequency of 3E-6. It is concluded on this basis that a HCLPF screening level of 0.3g peak ground acceleration was appropriate.

The screening tables in Ref. 3-6 require anchorage calculations for certain types of equipment. For a 0.3g HCLPF screening, the anchorage calculations would be based on a peak ground acceleration of 0.3g. It was noted that the USI A-46 Equipment Review performed anchorage calculations based on a peak ground acceleration of 0.15g. Thus, if equipment reviewed under A-46 had a factor of safety (capacity to demand ratio) of 2.0 or greater, and also satisfied the caveats of the screening tables in Ref. 3-6, it would have a HCLPF capacity above 0.3g. (It should be noted that the SSE ground response spectrum used for A-46 has less amplification than the NUREG-0098 spectrum assumed in Ref. 3-6, however, it is judged that conservatism's in the A-46 criteria compensated for this difference.)

Ref. 3-3 tabulates the results of the walkdown and screening for CCSPRA. For items that are in the scope of the A-46 review, those which had a safety factor of 2.0 or greater are screened out from further consideration. A-46 items with a safety factor less than 2.0, as well as items which are CCSPRA only or A-46 outliers, remained for consideration in developing fragilities for IPEEE. Probabilistic floor response spectra have been provided in Ref. 3-1 for use in development of the equipment fragilities.

3.1.4.1 HCLPF Calculation for Screening A-46 Equipment

The initial activity is to determine the most vulnerable items for fragility development. For A-46 equipment, this is accomplished by estimating an approximate HCLPF using the results of the A-46 calculations and the probabilistic floor spectra. Components with an estimated HCLPF greater than 0.3g may be screened from further fragility analysis.

From Ref. 3-4, a HCLPF may be conservatively estimated as:

$$\text{HCLPF} = \frac{C}{D} \times \text{PGA}$$

where: C = component capacity

D = 84th percentile seismic demand on component from Ref. 3-1 ground response spectrum

PGA = Peak acceleration of reference ground response spectrum

The reference ground response spectrum of Ref. 3-1 is the 1E-4 median uniform hazard spectrum (UHS) shape for CCNPP from Ref. 3-5 with a PGA of 0.4g.

Therefore:

$$\text{HCLPF}_{UHS} = \frac{C}{D_{UHS}} \times 0.4g$$

The capacity may be conservatively determined from the A-46 review as:

$$C = FS \times D_{A-46}$$

where: FS = Ratio of capacity to demand in the A-46 analysis

From Ref. 3-6, the demand from the Ref. 3-1 earthquake may be estimated from the A-46 analysis as:

$$D_{UHS} = D_{A-46} \times \frac{Sa_{84}}{Sa_{A-46}}$$

where: Sa_{A-46} = Spectral acceleration used in the A-46 analysis

Sa_{84} = 84th percentile spectral acceleration at the component frequency due to the Ref. 3-1 earthquake (Ref. 3-1).

Combining the above yields:

$$HCLPF = FS \times \frac{Sa_{A-46}}{Sa_{84}} \times 0.4g$$

If the HCLPF is 0.3g or greater, the component is screened out from development of fragilities.

Results of the comparison are shown in Table 1 of Ref. 3-2. A-46 components not screened out are listed in Table 2 of Ref. 3-2. These components are then further screened. For example, several distribution panels had their anchorage modified for A-46 and are screened out. Several Unit 2 125VDC distribution panels had anchorage modifications for A-46 and are screened out. Similar anchorage modifications will be made for several Unit 1 125VDC panels in the Spring 1998 Refueling Outage and these are also screened out. The inverters have been replaced using new rugged anchorage so they also are of no further concern.

Fragility calculations are performed on the remaining components. The results of these calculations are shown in Reference 3-2 and Table 3-3 of this report.

3.1.5 Analysis of Plant Systems and Sequences

3.1.5.1 Seismic Event Trees

The Seismic PRA model consists of six event trees from the internal events model with a seismic event tree linked to the front of the model. The six internal event model event trees are modified for the seismic model. The event trees are listed below:

- Seismic Event Tree

  <u>Modified Internal Event trees</u>
  - Support 1
  - Support 2
  - General Transient 1
  - General Transient 2
  - Long Term
  - Plant Damage State

The seismic event tree contains eight seismic top events. Each seismic top event models an important system function that may be failed by a seismic event. Each top event in the seismic event tree is linked to the corresponding top event in one of the internal event trees. For example, Seismic Top Event LE models the 0C Emergency Diesel Generator (EDG). EDG 0C is modeled by Top Event GJ in the Support 1 Event Tree of the internal events model. If EDG 0C fails due to a seismic event, it is automatically failed in the Support 1 event tree because the 'rules' for Top Event GJ require Top Event LE success. In this way, each seismic top event is linked to the corresponding top event in the support or front-line event tree from the internal events model.

3.1.5.2 Seismic Top Events

Fragilities are calculated for components as shown in Table 3-3. A Seismic Top Event is created for each system that had component fragilities with HCLPF values less than 0.3g. In addition, two surrogate top events are created, as explained below.

Several components in Table 3-3 do not have top events. They were dispositioned as follows:

Line 1 lists a fragility for control cabinets for EDGs 1B, 2A and 2B. However, these EDGs are all SRW dependent. Since SRW has a significantly lower fragility, the function of these EDGs is bounded by the SRW Top Event LG.

The components listed in Line 3 of Table 3-3 are relief valves associated with Steam Generator Blowdown and Reactor Coolant System Waste Process Sample Systems. These components are screened because by themselves, they do not cause a LOCA. The relief valve failure must occur concurrently with another component failure or event and the combined probability was very small.

Line 4 of Table 3-3 is a block wall which could impact certain components. However, further analysis showed that its impact was bounded by other fragilities already considered.

The remaining components have top events developed for them.

Top Event LA Surrogate Systems sustain a seismic event

During the component walkdowns, most of the plant components reviewed are found to be rugged and are screened at either the 0.3g or 0.5g screening level. Although these components are screened out from further evaluation, that does not mean they will not fail. They still have some contribution to seismic risk, particularly at the higher g levels. Top Event LA was created to model this risk contribution.

Of the systems whose components were walked down but screened out because they are rugged, six are important enough that their failure (of any one of the six) has a high probability of causing core damage. These systems are 125VDC Distribution, 4KV Buses, 480V Buses & MCCs, Auxiliary Feedwater, Reactor Protection System and Condensate Storage Tanks.

Top Event LA consists of these six systems under an 'or' gate. The failure of any one system will fail the top event. In the seismic model, Top Event LA failure leads directly to core damage.

Each system under the 'or' gate is assigned a fragility. The first five systems named above are assigned a fragility based on the screening level and the guidance for assigning surrogate fragilities given in Ref. 3-2. Although these systems have capacities that exceed the screening criteria, we do not know by how much they exceed it. Therefore, we conservatively set the fragilities at the screening level. This resulted in a HCLPF of 0.3g and a median acceleration capacity of 1.2 g for the first five systems. The sixth system listed above, Condensate Storage Tanks, had a fragility analysis performed by S&A (Ref. 3-10). This analysis determined the median and HCLPF capacities to be 0.96 and 0.43 g respectively and these values are used for the CST fragility.

The failure probabilities for Top Event LA are determined as follows. For a given g level earthquake, the conditional failure probability of a single surrogate system is the failure fraction of its fragility, associated with that g level. These failure fractions are calculated as described in Section 3.1.5.3 below and whose values are shown in Table 3-4. Let the failure fraction of one of the 0.3g systems equal 'x' and the failure fraction of the CST system equal 'y'. The success likelihood of a single system is (1-x) or (1-y). The likelihood of all six systems succeeding (Top Event LA success) is $(1-x) * (1-x) * (1-x) * (1-x) * (1-x) * (1-y)$ which may be re-written as $(1-x)^5 * (1-y)$. The failure probability (Top Event LA fails) is simply 1 - this value or $1 - [(1-x)^5 * (1-y)]$. This calculation is performed for each of the 30 initiating event g levels. The values are shown in Table 3-4.

The seismic model is quantified both with and without Top Event LA. This is discussed more in Section 3.1.5.5.

Top Event LB Refueling Water Storage Tanks sustain a seismic event

This top event models the availability of a borated water supply to the suction of the HPSI, LPSI and Containment Spray pumps. This top event is linked to Top Event RT (RWT supplies flow during LOCAs) in the SMCGT2 internal events model event tree.

Top Event LE OC EDG sustains a seismic event

This top event models the 0C Emergency Diesel Generator. This EDG, the 'Station Blackout' EDG, may be aligned to any one of the four safety-related 4KV Buses. The limiting components are three air conditioning components whose fragilities are calculated and are shown on lines 13, 14, and 15 of Table 3-3. These three fragilities are combined to form one fragility. Top Event LE is linked to Top Event GJ (EDG 0C starts and runs) in the Support 1 (SMCSUP1) event tree of the internal events model.

Top Event LG SRW Headers 11, 12, 21 and 22 sustain a seismic event

This top event models the function of the SRW system to provide cooling to various components, including EDGs 1B, 2A and 2B. The components which limit the seismic capacity of this system are several SRW coolers and SRW piping located in the Turbine Building. These components are listed on lines 6 through 12 of Table 3-3. These fragilities are combined to a single fragility representing the SRW system. Top Event LG is linked to the following top events of the SMCSUP2 event tree.

- S3 SRW Header 11 provides sufficient flow
- S4 SRW Header 12 provides sufficient flow
- GW SW Header 22 & SRW Header 22 operate
- GZ SW Header 21 & SRW Header 21 operate

Top Event LH CR HVAC sustains a seismic event

This top event models the Control Room and Cable Spreading Room HVAC system. The seismically limiting component is a control cabinet whose fragility is shown on line two of Table 3-3. Top Event LH is linked to Top Event HH (Control Room/Cable Spreading Room HVAC provides adequate ventilation) of the SMCSUP1 event tree.

Top Event LJ 500KV Switchyard sustains a seismic event

This top event models the function of the 500KV switchyard to supply off-site power. The fragility is based on a generic value from Ref. 3-7. Top Event LJ is linked to Top Event OP (grid fails to remain energized following a plant trip) in the SMCSUP1 event tree.

Top Event LK Secondary Systems sustain a seismic event

This top event models some systems (mostly secondary) that were not walked down because of their low importance in the PRA and to reduce the scope of the walkdowns. These are assumed failed for all seismic initiating events except for those at or below the Operational Basis Earthquake (OBE = 0.08G) and where Top Event LK is successful. Top Event LK is a surrogate top event representing these systems. It has practically no impact on CDF but was created to speed the model. It is assigned a fragility based on the most limiting fragility (500KV Switchyard). We assumed it had median acceleration capacity of one-half and twice the logarithmic standard deviations for uncertainty (β_u) and randomness (β_r). The below-DBE initiating events are SMCP03, SMCP05, and SMCP08. The top events that are set to failure for all IEs, except where LK is successful are listed in the table below.

Top Event	Event Tree	Description
BS	SMCGT1	TBVs modulate
BV	SMCGT1	TBVs quick open
MC	SMCGT1	Condensate is available
MN	SMCGT1	MFW is adequate after Trip
MP	SMCGT1	MFW ramps back after Reactor Trip
VC	SMCGT1	Condenser vacuum is available
CV	SMCGT2	2 of 3 Charging Pumps Borate as required
NR	SMCSUP2	Non Safety-related Instrument Air/Plant Air is available

Top Event LK is only questioned if all of the preceding Seismic Top Events are successful. If any one of the preceding Seismic Top Events fail, Top Event LK is set to guaranteed failure.

Although the fragility assigned to Top Event LK is probably realistic, some of the systems modeled by LK could possibly have lower seismic capacities than assumed. Although this would make LK non-conservative, the impact of this possibility is not very significant. Most of the functions LK represents are already modeled in the internal events model by initiating events such as Loss of Main Feedwater, Loss of Condenser Vacuum, or Loss of Instrument Air. These initiating events have comparable or greater frequencies than the three lowest-g seismic initiating events used with LK.

Top Event LL Containment Isolation sustains a seismic event

This top event models the function of the Containment Isolation system and penetrations to contain a LOCA. All of the electrical penetrations, piping penetrations and containment isolation valves whose failure would lead to containment release that were walked down are screened at a 0.5g. Therefore, a surrogate fragility of 0.5g is assigned to the Containment Isolation Top Event. Top Event LG is linked to Top Event SI (Penetrations greater than 4 inches function) in the SMCLT event tree. Top Event LG failure leads to a large release.

3.1.5.3 Seismic Split Fraction Values

Each seismic top event modeled represents some system function. The probability of functional failure increases with increasing seismic level, measured in ground acceleration g. The conditional probability that the function fails is the seismic fragility. The fragility is assumed to be lognormally distributed. The fragility curve has failure fraction values between zero and one on the ordinate and peak ground acceleration values on the abscissa. The fragility curve may be completely defined by two parameters, such as the median acceleration capacity Am, and the composite logarithmic standard deviation of the underlying normal distribution, B_c.

The seismic split fraction values are calculated for each top event using the spreadsheet, shown in Table 3-4. The spreadsheet calculates a failure fraction for each top event at each of 30 g levels (there is one g level for each of the 30 initiating events).

The fragility curves are truncated at the low and high ends. This reduces the number of split fractions and speeds up the quantification.

The fragility curves are truncated at the low end at a failure fraction of 5.0E-05. Below this failure fraction, the function is assumed to survive the seismic event. This truncation is reasonable because the lognormal distribution tends to overstate the failure likelihood at the low probability end of the distribution. This is because most components, particularly ductile components, have some cut-off limit below which, there is essentially zero probability of failure.

The fragility curves are truncated at the high end at a failure fraction of 0.95, except for Top Event LB, which is truncated at .50 and Top Event LE, which was truncated at 0.90. Top events are set to guaranteed failure above the high-end truncation. LB and LE are truncated at a lower limit because the model runs more efficiently without a significant increase in core damage frequency.

3.1.5.4 Seismic Human Action Analysis

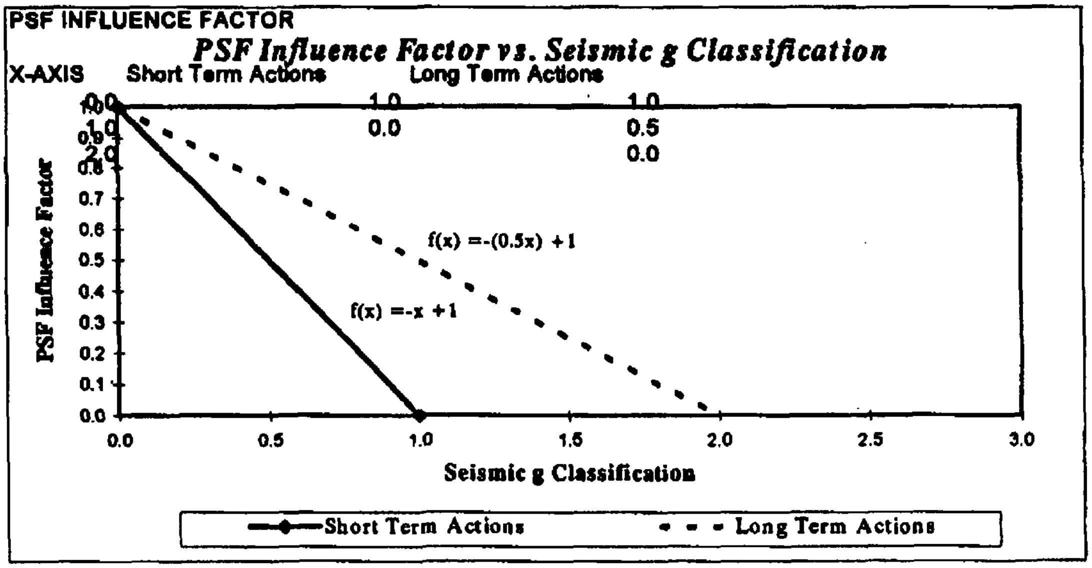
Since we expect a seismic event to adversely affect operator performance, the human action failure rates used in the Internal Events CCPRA are adjusted for seismic scenarios. The non-seismic human action failure rates are calculated using performance shaping factors (PSFs). Those PSFs that were judged to be affected by a seismic event are multiplied by influence factors to account for the lower success rate following a seismic event.

Different influence factors are used for human actions depending on whether it is a short term or long term action. Short term actions are those that must be completed within 15 minutes. All other human actions beyond the 15 minute time frame are considered long term. A seismic event is assumed to have a greater impact on short term actions because of the initial shock imposed by a seismic event and the more limited amount of time to complete the action.

The impact on human actions is also affected by the magnitude of the seismic event. Three influence factors are used for the short term actions and three different influence factors are used for the long term actions, depending on the magnitude of the seismic event. The seismic initiating events are grouped into three bins: seismic events from 0.01 to 0.281 g (bin 1), seismic events between .282 and 0.663 g (Bin 2), and seismic events between 0.664 and 0.918g (bin 3). For seismic events above 1.02g, all human actions are set to guaranteed failure (see assumptions below).

Values for the PSF influence factors are obtained from a straight line graph of PSF influence factor vs. seismic g level, shown below. There are two lines plotted, one line for the short-term actions and one line for the long-term actions. The endpoints for the long-term action line are established by assuming no impact (PSF influence factor = 1) at zero g and guaranteed failure (PSF influence factor= 0) at 2.0 g (see assumptions below). The short-term action line was anchored assuming no impact (PSF influence = 1) at zero g and at guaranteed failure at one-half the seismic value of the long-term action line, or 1.0 g. This makes a more severe impact on the short-term actions for any given seismic event.

Graph 1:



*** short term actions are those actions that must be completed in 15 minutes or less; all other human actions beyond the 15 minute timeframe will be considered long term actions.**

where: f is the influence factor of PSF due to a seismic condition
x is the seismic loading frame in g units

Assumption: while x = 0 (no earthquake) the PSFs are unchanged

while x = 2g (long term actions) or x = 1g (short term actions), PSFs drop to 0 (for direct PSFs) or 10 (for indirect PSFs). In other words, the specific action is set to guaranteed failure.

In analyzing the graph, the impact on the actual human action failure probability is established through the PSF influence factor. At the guaranteed failure point (X g for long term actions, and Y g for short term actions) the appropriate direct acting PSFs (those PSFs whose survey rankings are better for higher ranked numbers) are multiplied by an influence factor of zero while all indirect acting PSFs (those PSFs whose survey rankings are worse for higher ranked numbers) are set to a value of ten. For all other actions, a seismic g classification corresponds to a specific PSF influence factor that is applied to all relevant PSFs. The specific seismic bins are as follows:

SEISMIC BIN	RANGE OF VALUES	PSF INFLUENCE FACTOR	
1	0.013g - 0.281g	LT = 0.8595	ST = 0.719
2	0.282g - 0.663g	LT = 0.6685	ST = 0.337
3 (see note below)	0.664g - 1.02g	LT = 0.4900	ST = failure
	1.02g or greater	LT = failure	ST = failure

* ST = Short Term Actions
LT = Long Term Actions

Note: Since the Human Action Analysis was performed, the upper bound of Bin 3 was lowered to 0.918g. The frequencies of seismic levels above 0.918g are conservatively grouped with the 1.5g initiating event, which goes directly to core damage. This reduced the number of initiating events with no significant impact on core damage.

The PSF influence factor for each bin is obtained by finding the influence value associated with the highest seismic level in that bin. This results in all short-term actions being set to guaranteed failure for all Bin 3 initiating events. There is one initiating event that falls above Bin 3. Initiating Event SMC1P5, which has a 1.5g seismic level, falls in the category where all human actions are set to guaranteed failure. This initiating event goes directly to core damage.

As mentioned earlier, the human action failure rates for the internal events PRA are determined using PSFs. The methodology used to calculate the Internal Events CCPRA Human Action failure probabilities is a hybrid which incorporates the best features of several well-known Human Reliability Analysis (HRA) methodologies and uses concepts from more advanced HRA approaches and available empirical data. The SLIM-MAUD approach and the HCR (Human Cognitive Reliability) models form the basis of the CCPRA HRA methodology. Calvert Cliffs' human action methodology is documented in the Human Error Probability Methodology Report, Ref. 3-15.

The basic rationale underlying SLIM-MAUD is that the likelihood of an error occurring in a particular situation depends on the combined effects of a set of performance shaping factors. These PSFs include both human traits (e.g., operator competence, morale, motivation, etc.) and conditions of the work setting (e.g., time available to complete a task, procedures, indications, etc.) that are likely to influence an individual's performance.

Not every PSF is likely to be degraded by a seismic event. The areas that will most likely be degraded for a seismic situation are: time to complete a specific action, concurrent actions in progress, communication, training and experience, Control Room indication, and the ability to access certain areas of the plant.

Assumptions

The following assumptions are used to analyze the impact of a seismic event on various human actions.

- We assume that the impact on human actions will vary with the strength of the seismic event. We assume that operators will be unable to perform required actions at seismic levels high enough to fail the building structure, because they may be physically blocked by fallen debris. If some part of the building collapses, then the operators will probably be unable to perform actions, at least in that area. Most of the human actions take place in the Auxiliary Building (median capacities for the various buildings are shown in Table 3-5). The average median capacity is 2.0 g. This level is used to establish the guaranteed failure point for human actions.

- At a seismic g classification of zero there will be no degradation of the human action failure probability. The impact on the various PSFs from a seismic g classification of zero to the guaranteed failure point is assumed to be a linear relationship.

- The weighting factor associated with a specific PSF is assumed to remain the same for a seismic scenario. The weighting factor accounts for how important a PSF is in a transient. The PSF is assumed to have the same importance in a seismic situation although the ability associated with a specific PSF may be degraded. The way to account for this degradation is by adjusting the survey response for the specific PSF rather than adjusting the weighting factor linked to the PSF.

- The conversion constant ('a' constant) used to convert the Success Likelihood Index (SLI) to a failure probability is assumed to remain constant in a seismic scenario. The human action methodology for CCPRA currently uses seven 'a' constants to calculate failure probabilities for different cognitive binning of the specific action in question. These same 'a' constants will be utilized to calculate the seismic human action failure probabilities since the 'a' constant is just a conversion constant. The specific impact of the seismic event will be seen in the degradation of the specific PSFs as stated earlier.

Specific Adjustments made to the Performance Shaping factors

Specific adjustments made to the various types of performance shaping factors:

Time-dependent PSFs:

VT1 Operator Stress Level

This PSF is an indicator of the operator's time, stress level, and mental load. Just several minutes after the earthquake, the operators may not have recovered from the initial shock of an earthquake. But after the initial shock, the effect slowly diminishes. Therefore, VT1 is judged as a time-dependent factor.

VD3 Number of Concurrent Actions in Progress

Certainly, in the early stages of a seismic event, a variety of different activities can be commencing, therefore, producing some chaos in the Control Room. The plant begins to stabilize farther into the event, thus reducing the amount of concurrent actions in progress.

VA4, 5, 6 Communication

In the first several minutes, communication may be more difficult due to the initial shock of the seismic event as compared to later in the event when the plant starts to stabilize.

VA1, 3 Adequacy of Personnel

VA1 and VA3 refer to the adequacy of personnel inside/outside the Control Room to perform a desired task. In the first several minutes, some operators may not be mentally or physically available because they may not have recovered from the initial shock of the earthquake or they might be busy in some non-task-related jobs.

Time Independent PSFs:

VI1, VI2 Initial and Secondary Indications

A signal transmission line may be damaged by the earthquake, but the damage is assumed not recovered in the duty period. Therefore, the related PSFs will be degraded but not considered as time-dependent.

VL1, VL2 Equipment Location and Accessibility

It is assumed that an earthquake will degrade the ability of an operator to get to certain locations (if required) but will not be related to time. This PSF models the impact of fallen structures blocking access to equipment. We expect this at the higher g levels. This PSF, by itself, will severely degrade the human action failure probability at the highest g level.

Unchanged PSFs:

VP1, VP3 Procedure Quality and Adequacy

These PSFs rank the quality and adequacy of any procedure that the operators may utilize to mitigate a transient. Since the operators will still enter the same procedures during a seismic event combined with a plant trip, these PSFs are assumed to be unaffected by an earthquake.

VC1, VC2, VC3, and VC4 Consequences of Performing/Not Performing Action

These factors are only related to the accident scenario and will be unaffected by an earthquake.

VD1, VD2 Impact of Previous Related Successful/Unsuccessful Actions

We assume these PSFs will not be affected by an earthquake.

g-level Independent PSFs:

VE1, VE2, VE3 Operator Training and Experience in Identifying, Diagnosing, and Performing a Specific Action

Regardless of the earthquake's g-level, most of the operators have not experienced a seismic event. Therefore, a generic influence factor of 0.5 is applied.

The human action failure rates are multiplied by the adjusted performance shaping factors. The human action failure rates are incorporated into the appropriate split fraction failure probabilities in the master frequency file. A portion of the master frequency file is shown in Table 3-8.

Figure 3-1 shows the general trend of increasing human action failure probability versus peak ground acceleration (PGA) level for short-term actions. Note that all short-term human actions are set to guaranteed failure at a pga of 1.0g. Figure 3-2 shows the general trend of the long term human action failure probabilitys. Note that these are set to guaranteed failure at a pga of 2.0g.

3.1.5.5 Seismic Quantification

The event trees are quantified to determine point estimate frequencies for core damage sequences and plant damage states that result from each of the 30 ranges in peak ground acceleration. The ranges, shown in Table 3-2, are divided up into three large bins. The initiators in each bin are quantified as a separate group because of the impact of seismic level on human actions.

The three levels of impact dictated that there would be three different failure probabilities for each split fraction whose value is affected by a human action. Approximately 200 of the roughly 1,400 split fraction failure probabilities used in the seismic quantification are affected. Having multiple values for the same split fraction (depending on the PGA level) is managed by making three different master frequency files in RISKMAN. The initiators are run in three separate batches, each batch using a separate MFF.

The linked seismic event tree, support trees and front-line trees are quantified at each of the 30 PGA ranges using the mean hazard curve and mean seismic fragility curves. There is a seismic initiating event for each of the 30 seismic bins. An initiating event is an earthquake causing a peak ground acceleration at CCNPP corresponding to a PGA level within a bin. The PGA level of the initiating event determines the failure fraction to be used for each top event fragility. Table 3-4 shows the failure fraction (conditional failure probability) of each seismic top event for each of the 30 PGA ranges. For each seismic top event, the split fraction values are equal to the failure fraction calculated from the corresponding fragility curve at the PGA of the initiating event.

The very high (above 0.95) and very low (below 0.0005) failure fraction values correspond to the tails of the lognormal distribution representing the fragility curve. Split fractions were not assigned for these failure fractions. Rather, the fragility curve is truncated, and the failure probability is set to guaranteed failure for the high end or guaranteed success for the low end. This reduces the number of split fractions and sequences generated. Fragility curve truncation is explained in detail in Section 3.1.5.3.

Table 3-4 shows that the conditional failure probabilities for the seismic top events vary considerably over the acceleration range. At each g level, successes and failures in the seismic event tree determine which top events are failed seismically in the support and front-line trees, according to the way the trees are linked. If a top event in a support or front-line tree is not affected by seismic failure, its conditional failure probability is based on the updated IPE (except for those failure probabilities affected by the seismic impact on human actions). In this manner, the linked event trees account for the combined contributions from all possible seismic and non-seismic failures at each acceleration.

The seismic event trees are quantified both with and without the contribution of surrogate Top Event LA. When Top Event LA is included, it is the single largest contributor to CDF. The most useful information gained by including LA is a more accurate estimate of total seismic CDF. However, this estimate is somewhat conservative because of the way the surrogate is calculated. As explained in Section 3.1.5.2, the surrogate systems are assigned fragilities at the screening level, when in reality, these systems likely have some higher capacity. However, the results which include the surrogate are probably closer to the actual CDF, so these are presented as the CCNPP Seismic CDF.

The total seismic CDF for Unit 1, quantified with surrogate top event LA is 1.29E-05. Unit 1 CDF quantified without LA is 1.07E-05.

3.1.5.6 Seismic Sequence Analysis

For performing component importance calculations and identifying potential vulnerabilities, results quantified without Top Event LA are considered more meaningful. The top 100 sequences quantified without Top Event LA are shown in Table 3-7. Most of these sequences fall into one of two main categories. The first type is a Spurious Safety System Actuation (SSSA) with a loss of Auxiliary Feedwater (AFW). The second category is a seismic-induced Station Blackout (SBO) and loss of Auxiliary Feedwater.

An SSSA is the spurious actuation of the Engineered Safety Features Actuation System, Auxiliary Feed Actuation System (AFAS) and Reactor Protection System (RPS). SSSA occurs in the Internal Events PRA and is explained in detail in the Calvert Cliffs IPE Summary Report (Ref. 3-14). It occurs when two of four Vital 120VAC buses de-energize.

In most of the seismic sequences involving the SSSA, it is caused by a seismic-induced Loss of Offsite Power (LJ) and loss of all but one EDG. The three SRW-dependent EDGs fail whenever SRW fails (LG). Note that SRW and Off-site power are guaranteed to fail for all Bin 3 and most of the Bin 2 initiating events. If either of the two remaining self-cooled EDGs fail, an SSSA could occur (if both of the self-cooled EDGs failed at this point, a Station Blackout would result).

Of the two self-cooled EDGs, EDG 0C (LE) and EDG 1A (GE), EDG 0C has a lower seismic capacity and is the more likely to fail. LE is guaranteed to fail for all Bin 3 initiating events. In either case, failure of one of the self-cooled EDGs leaves only one EDG operating.

Each EDG is normally aligned to supply two of four Vital 120VAC buses. Each of the other two Vital 120VAC buses are supplied by a 125VDC battery through a vital inverter. The 125VDC batteries will last four hours with no SIAS (SIAS does not occur until the SSSA occurs at about the four-hour point). After about four hours, these two 125VDC batteries will deplete, resulting in a loss of their downstream buses, panels and loads.

Prior to battery depletion, the operators must align the battery-supplied Vital 120VAC buses to their back-up buses. Failure of this action (Top Event XW) results in de-energizing two of four Vital 120VAC buses. This results in failing two of four sensor channels and actuating the two out of four trip logic in virtually all protective actions of our Engineered Safety Features Actuation System, Reactor Protection System, and Auxiliary Feed Actuation System. This event is referred to a spurious systems actuation (SSSA). An SSSA causes these significant events to occur.

- UV Channels A and B activate and lock in. This opens the feeder breakers to safety-related 4KV buses and sheds all major 4KV loads, including SRW and Safety Injection and the motor-driven AFW pump. Lock-in of the UV signal prevents the 4KV loads from re-starting. The three SRW-dependent EDGs (1B, 2A, 2B) will fail in 10 to 20 minutes because they have no SRW cooling (in most of the seismic sequences SRW is lost anyway due to failure of one of the various SRW coolers/piping in the Turbine Building).

- Steam Generator Isolation Signal (SGIS) Channels A and B activate: This shuts both Main Steam Isolation Valves (MSIVs), which fails MFW.

- Reactor Protection System (RPS) activates: This sends an open signal to both Power Operated Relief Valves (PORVs). However, when the SSSA is caused by a loss of all power except EDG 1A - as in many of the seismic sequences - the PORVs (on both units) lose 125VDC power and remain closed. The HPSI pumps cannot provide make-up to the RCS due to the locked in UV signal.

- AFAS Block Channels A & B activate: This isolates AFW flow to both steam generators.

  At this point, operator action is required to recover from the spurious AFAS by opening the AFW block valves (Top Event QZ) and locally control AFW flow (Top Event HX). If the operator fails to locally control flow, the steam generators eventually over-fill and the resulting water in the steam supply line is assumed to fail the operating AFW pump turbine. This leaves only the standby AFW turbine, and failure of this turbine (Top Event F1R) will result in AFW failure.

Once-through Core Cooling is not an option because the High Pressure Safety Injection (HPSI) Pumps are locked out by the UV signal. Also, if the running EDG is EDG 1A, the PORVs are failed shut due to the loss of 125VDC power after 125VDC Battery 21 runs down.

When the 125VDC Bus 11 or 21 runs down, one of the AFW steam admission valves will shut on loss of power. This may require operator action to re-open the steam admission valve (Top Event MH).

Long term AFW cooling also requires operator action to shift the suction from Condensate Storage Tank (CST)12 to either CST 11 or the Demineralized Water Storage Tank. Failure of this action (Top Event F1) will result in a loss of core cooling.

Typically, automatic AFW flow control is lost and the operators must control AFW locally. If they underfeed the steam generators (Top Event UQ), the heat sink will be lost and core damage will result.

The second most common type of seismic sequence is a seismic-induced SBO and an associated AFW flow control failure, Top Event F1R. Top Event F1R models the short term availability of AFW flow paths, including the supports for AFW pumps. This top event includes human action BHEF1Q, which is required in the following scenario: A seismic-induced SBO and failure of all EDGs results in a loss of main feedwater on both units. The station batteries deplete after about four hours and steam generator (S/G) water level indication is lost. The operators are assumed to overfill the S/Gs. Consequently, the 11 AFW Turbine-Driven Pump fails due to the return of excess water carryover from the S/G. Operator action is required to drain the AFW steam supply header and start 12 AFW Turbine-Driven Pump within three hours of the overfill condition in accordance with the Emergency Operating Procedure.

3.1.5.7 CCNPP Unit 2 Assessment

A Unit 2 assessment performed for the internal events PRA determined that the differences between the two units are minor and do not warrant the completion of a Unit 2 PRA. For the seismic analysis, both units were walked down and the differences between the units are noted on the walkdown list (Ref. 3-3). Various components had different anchorage configurations and these differences are explained in the notes section of Ref. 3-3. None are judged to be significant.

The most significant difference between Unit 1 and Unit 2 is the lack of a SRW dependency on Emergency Diesel Generator 1A. EDG 1A is a newly-installed safety-related self-cooled diesel generator and backs up Unit 1's safety-related 4KV Bus 11. The Unit 2 equivalent bus is 4KV Bus 24. It is backed up by SRW-dependent EDG 2B. Since SRW has a relatively low fragility, the diesels dedicated to Unit 2 are more susceptible to failure from a seismic event.

The EDGs tie into the 4KV buses which in turn power the 480V, 125VDC, and vital inverters which power the 120VAC vital buses. At the 125VDC and 120VAC levels, multiple trains and cross ties between the units electrical buses eliminates the impact of the SRW dependency. That is, at the lower voltage levels, Unit 2 has half of its trains ultimately powered from the non-SRW-dependent EDG 1A, just as Unit 1 has. However, at levels above these (4KV and 480V), there are no Unit 2 buses backed by EDG 1A and the SRW dependency exists.

The SBO diesel, EDG 0C, is a newly-installed self-cooled diesel which may be aligned to a safety-related 4KV bus on either Unit. However, this diesel is not built to Seismic Class I specifications and it has a lower seismic capacity, due to the mounting of its Control Room air conditioning units and Switchgear Room air-handling unit. Since this diesel is intended to be aligned to either unit, it does not cause a difference between units for the seismic analysis. EDG 0C is modeled by Top Event LE in the Seismic Event Tree.

The most significant impact of the increased SRW dependency of Unit 2 is on the motor driven AFW pump. Unit 2's motor driven AFW pump (AFW Pump 23) is normally aligned to 4KV Bus 24, which is backed up by SRW dependent EDG 2B. Unit 1's motor driven pump is normally aligned to EDG 1A-backed 4KV Bus 11, and does not have the SRW dependency. However, this does not mean that Unit 2 AFW is entirely SRW dependent. Unit 1 AFW may be cross-connected to Unit 2, but this requires a human action. However, Unit 1 AFW would only be needed for Unit 2 if AFW Pump 13 is not available because of a LOOP and the Unit 2 turbine-driven AFW pumps failed or are not available.

The Unit 2 Seismic CDF is evaluated by quantifying a modified version of the Unit 1 Seismic Model. For the Unit 2 model, the diesel generators are 'rewired' so that the SRW-dependent diesels supply the 4KV buses on the Unit 1 side. This is accomplished by adding a SRW dependency to EDG 1A and removing it from EDG 2B in the seismic model.

The Unit 2 model quantification (with surrogate Top Event LA) shows that the Unit 2 seismic CDF is 1.52E-05. This is about 23% higher than the Unit 1 Seismic CDF (with Top Event LA). Even with this increase, the Unit 2 CDF is considered acceptable.

The sequences for Unit 1 and Unit 2 are very similar. As expected, Unit 2 has an increased dependency on SRW and on the EDG 0C. There are no other significant differences.

3.1.6 Analysis of Containment Performance

Containment penetrations and containment isolation valves are screened at 0.5g. Seismic-induced failure of containment penetrations or containment isolation valves result in a failure of the containment isolation function. This function is modeled by surrogate Top Event LL in the Seismic Event Tree. For CCNPP, a large release is defined as a break greater than a 4-inches diameter hole. Top Event LL is conservatively mapped to large release for all failures by making Top Event SI (containment penetrations greater than four-inch function) dependent on top event LL success.

The fragility assigned to the containment isolation function is based on the screening level used by the Seismic Review Team during the component walkdowns and the guidance provided by EQE in the fragility report (Ref. 3-2). This guidance states that for the purposes of defining surrogate element fragilities, components mounted in structures should conservatively have a median/HCLPF ratio of about four. Containment electrical penetrations and piping penetrations are screened at HCLPF of 0.5 g. Containment isolation valves are also walked down and all those whose failure could lead to containment bypass or release are also screened at 0.5g. Therefore, Top Event LL is assigned a fragility with a HCLPF of 0.5g. Using the recommended median/HCLPF ratio of four yields a median acceleration capacity of 2g.

The randomness, β_R, and uncertainty, β_U, are assigned as 0.40 and 0.44, respectively, based on the guidance given for establishing surrogate fragilities in Ref. 3-2. Top Event LL split fraction values are generated using a spreadsheet. The values are shown in Table 3-4.

The seismic structure fragility analysis (Ref. 3-1) determined a median acceleration capacity and HCLPF value for the containment shell, and the reinforced-concrete base slab. These values are shown in Table 3-5. The base slab is the most limiting and has a median acceleration capacity of 2.31g and HCLPF of 0.70g. This is bounded by the surrogate Top Event LA, which leads directly to core damage and the Containment Isolation Top Event LL, which is mapped to a large (greater than four-inch diameter) leak in containment. Therefore, containment structural failures are not modeled in this evaluation.

3.1.6.1 Plant Damage States

The Seismic Event Tree quantification binned the seismically induced core damage sequences into various plant damage states (PDS). The PDS define the entry condition into the Level 2 analysis, and define the thermodynamic conditions of the RCS at the time of core damage, the status of containment, and availability of both passive and active safety features that can terminate the accident or mitigate the release of radioactive materials to the environment. Table 3-6 contains the PDS with the three highest frequencies. The fractional contribution of each PDS to the different failure categories was obtained from Table 4.7.2.b of the IPE Summary Report, Reference 3-14.

The dominant seismic PDS, HRIF, contains 85% of the total seismic core damage (quantified without surrogate Top Event LA). HRIF is the second ranked PDS in the internal events model. HRIF represents a high pressure in the reactor coolant system at the time of vessel melt through (VMT), only the RCS volume of coolant is inside containment because safety injection failed, the containment is isolated, and containment cooling has failed. HRIF results from a seismic-induced SBO or spurious safety system actuation (SSSA) with a total loss of main and auxiliary feedwater. The containment is isolated but HPSI and Containment Spray are not available.

Most of the remaining seismic core damage (10% of total) is binned to PDS HRWF. HRWF results from a seismic-induced SBO with a total loss of feedwater, safety injection failure and failure of the containment isolation function. PDS HRWF represents a condition of high RCS pressure at vessel melt through, safety injection failure results in only the RCS volume of coolant on the containment floor, containment has a large (greater than four-inch diameter) leak area, and containment heat removal and fission product scrubbing are failed. PDS HRWF is a relatively severe PDS and did not appear in the Internal Events model and could not be 're-binned' to a higher-frequency, higher-consequence PDS from the Internal Events PRA because none existed.

PDS HRWF results partially due to linking Containment Isolation Top Event LL to the large-break-size containment failure. This modeling is conservative but was done because the actual response of containment isolation is not well understood and internal events PRA MAAP runs do not exist for PDS HRWF. After the seismic quantification was performed we realized that assuming all containment failures to be large may have been overly conservative. Therefore, HRWF is binned to the most likely containment failure categories. Since HRWF is a relatively severe PDS it will likely result in the more severe categories of early-small and early-large containment failures. The results shown in Table 3-6 show a range of values for PDS HRWF's contribution to these two containment failure categories. The values reflect the two extreme cases of HRWF resulting entirely in early-large release (conservative) to HRWF resulting entirely in early-small release (less conservative). The true value lies somewhere in this range, but it is not known where. In the worst case, the seismic contribution to early-large containment failure is 1.41E-06 per year.

About 5% of seismic core damage is binned to HGIP, which includes HHIP. HGIP represents a condition of high RCS pressure at VMT, HPSI and containment-spray pumps inject before VMT, the containment is isolated and not bypassed, and containment heat removal and fission product scrubbing are available.

3.2 USI A-45 and Other Seismic Safety Issues

The purpose of this section is to describe how the CCNPP IPEEE evaluates the Decay Heat Removal Safety Function based on results from the seismic analyses. The evaluation was performed to identify potential decay heat removal vulnerabilities for seismic events initiated from power operation, and to determine if the risk associated with the loss of decay heat removal can be reduced in a cost-effective manner.

The results of this evaluation support closure of USI A-45.

3.2.1 USI A-45 Background

The objective of the NRC's Shutdown Decay Heat Removal (DHR) Requirements Program (A-45) is:

> *"to evaluate the adequacy of current licensing requirements for ensuring that nuclear power plants do not pose an unacceptable risk to the public as a result of failure to remove decay heat" (Ref. 16).*

This issue was designated USI A-45 on December 24, 1980. At the time the A-45 program was started, the NRC's desire was to determine the possible advantages and disadvantages of backfitted, independent, bunkered DHR systems such as those being installed at plants in a few European countries.

In 1984, the A-45 program characterized the DHR systems and functional capabilities of U.S. plants and grouped them into eight groups (Ref. 17). A Shutdown DHR analysis of a Combustion Engineering two-loop pressurized water reactor was performed in August 1987 (Ref. 18).

The results of this and other NRC sponsored DHR case studies were summarized in a Shutdown Decay Heat Removal Analysis report (Ref. 19). The results of these studies prompted a policy statement concerning Shutdown Decay Heat Removal Requirements to be issued (Ref. 20). This SECY document informed utilities that the resolution of USI A-45 will be accomplished through plant-specific analyses under the IPE process. Due to the plant-specific nature of DHR vulnerabilities and corrective actions, the IPE and IPEEE were determined to be the most effective means for resolving this issue.

In November of 1988 the NRC issued Generic Letter 88-20 which requires the IPE process to resolve USI A-45 (Ref. 21). Resolution of USI A-45 requires:

> *"a determination of whether the DHR function is adequate and if cost-beneficial improvements could be identified."*

Appendix 5 of Generic Letter 88-20 identifies potential improvements to DHR systems.

3.2.2 Evaluation

The CCNPP IPE submittal addressed the issue of DHR. With the exception of external events, the CCNPP IPE concluded that USI A-45 was resolved. This section provides an evaluation of the CCNPP DHR functions with respect to the seismic analysis.

Individual sequence examination was performed to determine if any differences appeared in the seismic analysis that were not already identified in the internal events analysis. The major sequences from all seismic initiating events were evaluated and compared with all sequences from CCPRA internal events. Most of the seismic sequences appear in the internal events CCPRA. All additional sequences unique to seismic were the result of increased human action failure probabilities. In seismic, certain performance shaping factors for all human actions were increased proportional to the "g" level of the earthquake. This, in turn, increased the failure probabilities for these human actions (See section 3.1.5.4 for detailed discussion on seismic human actions).

CCNPP currently has effective operator training for these human actions, and it is not expected that additional training would benefit the failure probability of these human actions in the event of a seismic event.

Based on these results, the CCPRA has effectively evaluated the DHR function with respect to seismic events at CCNPP and USI A-45 is resolved.

3.2.3 Charleston Earthquake Issue

In accordance with NUREG 1407 and EPRI Report NP-6395-D, the Charleston issue has been closed for all but eight outlier plants identified through the Eastern U.S. Seismicity program. CCNPP is not one of those outlier plants. The Charleston issue is therefore considered closed.

3.3 References

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3-22 Calvert Cliffs Engineering Standard No. ES-011, System Structure, and Component (SSC) Evaluation

3-23 Calvert Cliffs Nuclear Power Plant, Units 1 and 2 Updated Final Safety Analysis Report (UFSAR), Volume II, Revision 20, February 5. 1997.

FIGURE 3-1

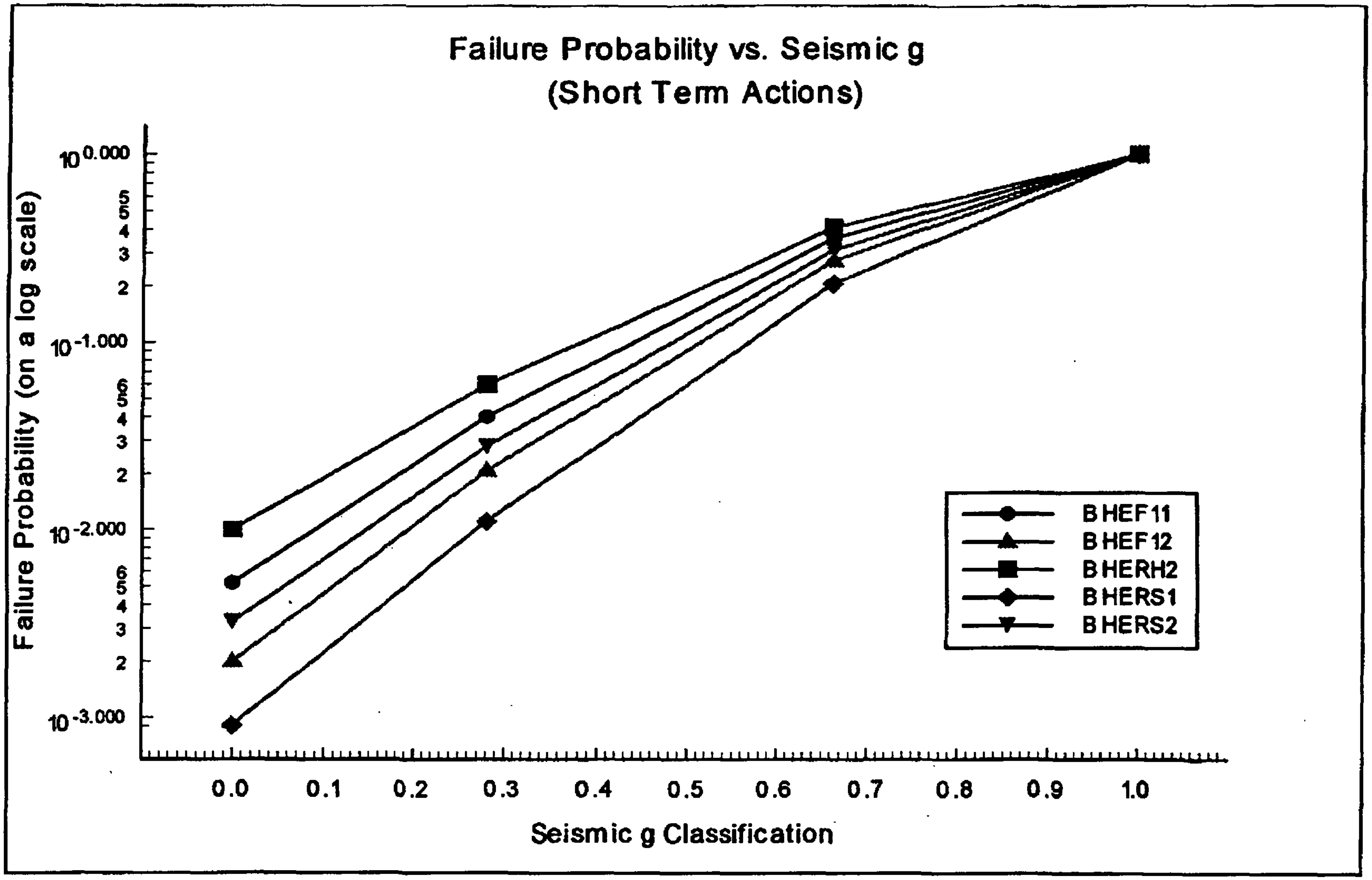

Failure Probability vs. Seismic g
(Short Term Actions)
Failure Probability (on a log scale)
$10^{0.000}$
$10^{-1.000}$
$10^{-2.000}$
$10^{-3.000}$
0.0
0.1
0.2
0.3
0.4
0.5
0.6
0.7
0.8
0.9
1.0
Seismic g Classification
BHEF11
BHEF12
BHERH2
BHERS1
BHERS2

FIGURE 3-2

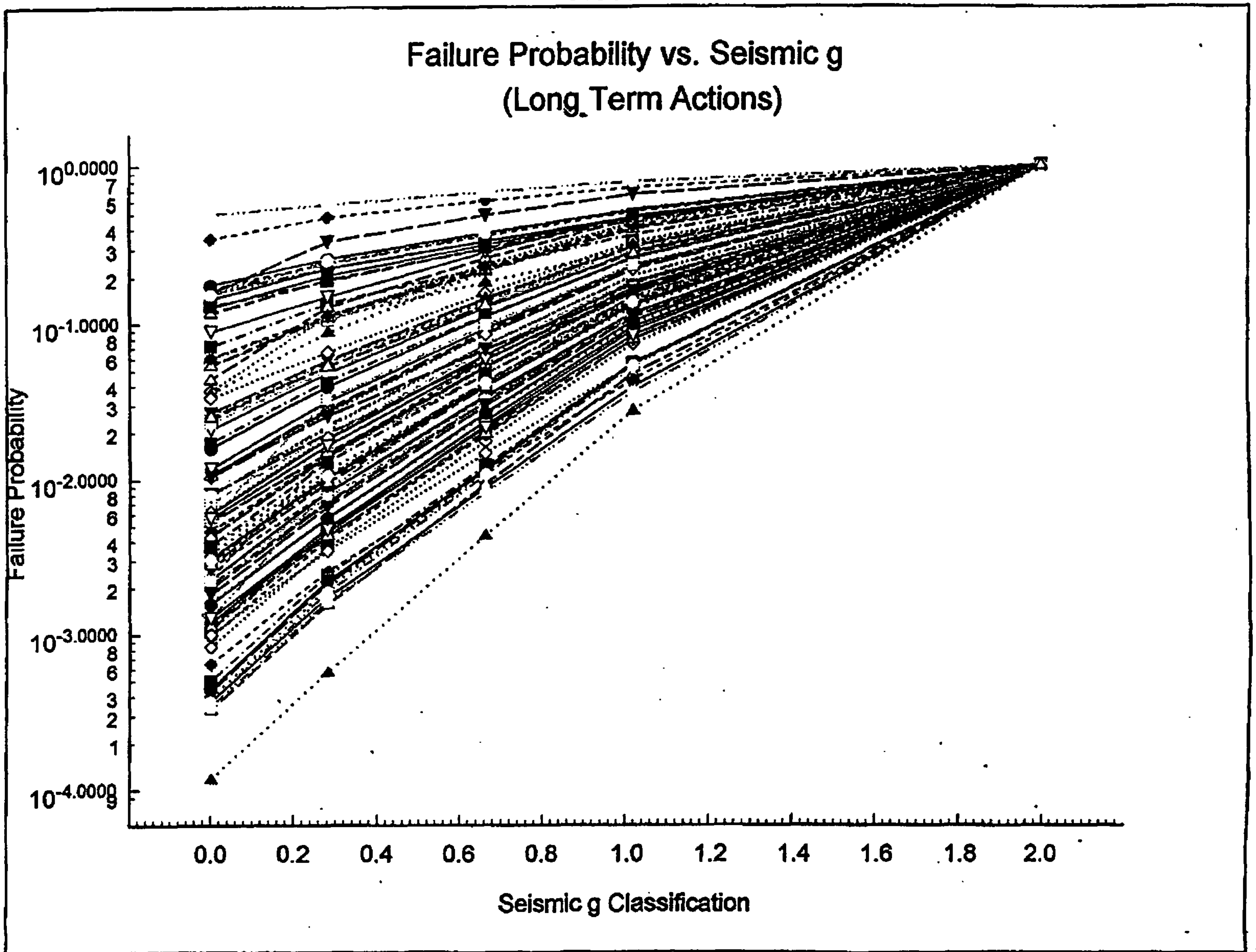

Failure Probability vs. Seismic g
(Long Term Actions)
Failure Probability
10^0.0000
7
5
4
3
2
10^-1.0000
8
6
4
3
2
10^-2.0000
8
6
4
3
2
10^-3.0000
8
6
3
2
1
10^-4.0000
9
0.0
0.2
0.4
0.6
0.8
1.0
1.2
1.4
1.6
1.8
2.0
Seismic g Classification

FIGURE 3-3

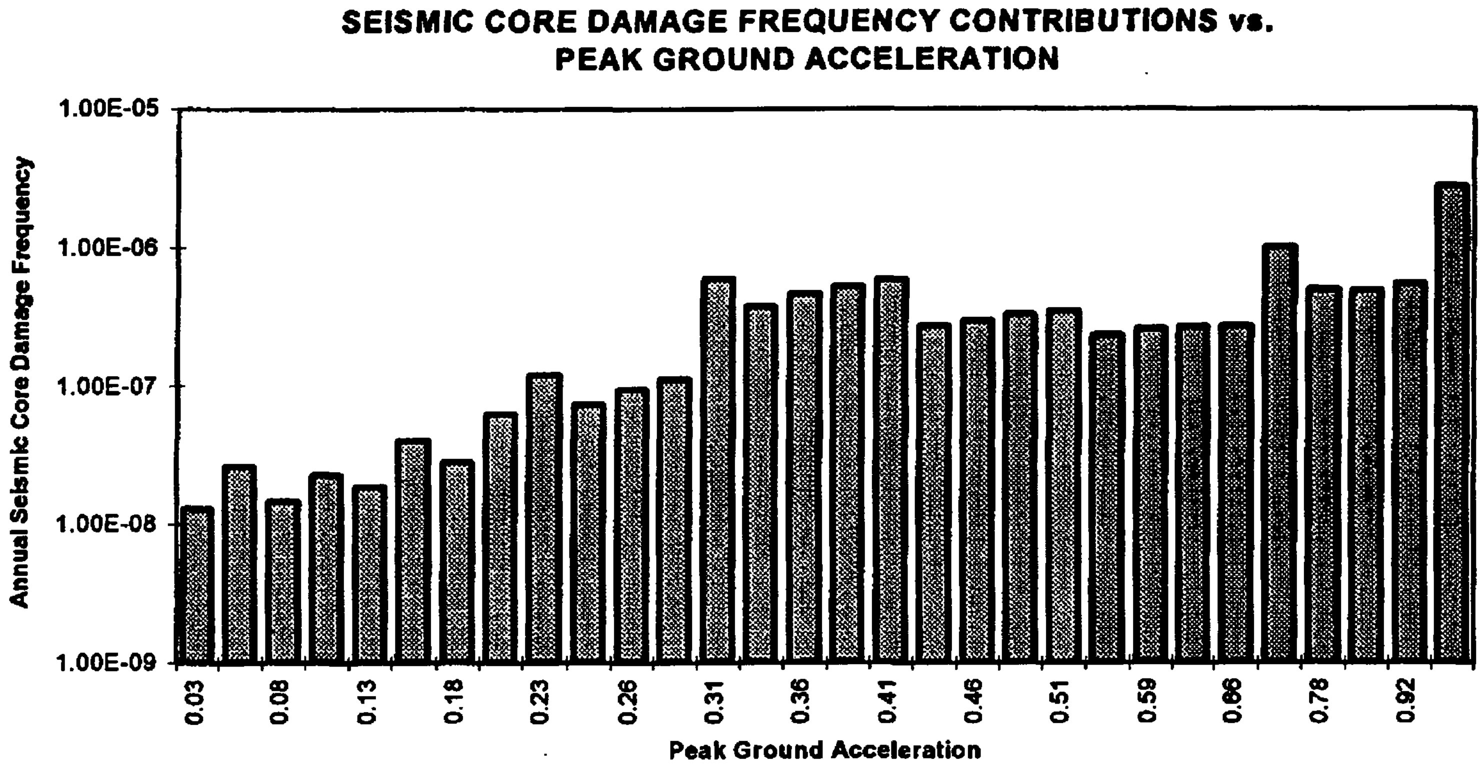

SEISMIC CORE DAMAGE FREQUENCY CONTRIBUTIONS vs.
PEAK GROUND ACCELERATION
Annual Seismic Core Damage Frequency
1.00E-05
1.00E-06
1.00E-07
1.00E-08
1.00E-09
0.03
0.08
0.13
0.18
0.23
0.26
0.31
0.36
0.41
0.46
0.51
0.59
0.66
0.78
0.92
Peak Ground Acceleration

TABLE 3-1

CCNPP SEISMIC HAZARD CURVE VALUES

Updated LLNL curves from NUREG 1488

Acceleration		Percentiles		
(cm/sec/sec)	Mean	15th	50th (median)	85th
50	.7674E-03	.6850E-04	.3090E-03	.1300E-02
75	.4321E-03	.3100E-04	.1620E-03	.7160E-03
150	.1459E-03	.6460E-05	.4310E-04	.2340E-03
250	.5891E-04	.1470E-05	.1350E-04	.8490E-04
300	.4141E-04	.8270E-06	.8380E-05	.5590E-04
400	.2292E-04	.2720E-06	.3780E-05	.2830E-04
500	.1402E-04	.1040E-06	.1900E-05	.1570E-04
650	.7565E-05	.2860E-07	.7710E-06	.7510E-05
800	.4498E-05	.9010E-08	.3370E-06	.4000E-05
1000	.2490E-05	.2220E-08	.1290E-06	.1900E-05

TABLE 3-2

SEISMIC INITIATING EVENTS

Batch Name	Initiators	Description	PGA	Frequency (per year)
SEISMIC1	SMCP03	SEISMIC EVENT	0.01G - 0.026G	6.7870E-03
	SMCP05	SEISMIC EVENT	0.027G - 0.051G	1.5350E-03
	SMCP08	SEISMIC EVENT	0.052G - 0.077G	3.3530E-04
	SMCP12	SEISMIC EVENT	0.078G - 0.115G	1.4310E-04
	SMCP13	SEISMIC EVENT	0.116G - 0.134G	7.1550E-05
	SMCP15	SEISMIC EVENT	0.135G - 0.153G	7.1550E-05
	SMCP18	SEISMIC EVENT	0.154G - 0.179G	2.1750E-05
	SMCP20	SEISMIC EVENT	0.180G - 0.204G	2.1750E-05
	SMCP23	SEISMIC EVENT	0.205G - 0.230G	2.1750E-05
	SMCP24	SEISMIC EVENT	0.231G - 0.242G	1.0870E-05
	SMCP26	SEISMIC EVENT	0.243G - 0.255G	1.0870E-05
	SMCP28	SEISMIC EVENT	0.256G - 0.281G	8.7500E-06
SEISMIC2	SMCP31	SEISMIC EVENT	0.282G - 0.306G	8.7500E-06
	SMCP33	SEISMIC EVENT	0.307G - 0.332G	4.6230E-06
	SMCP36	SEISMIC EVENT	0.333G - 0.357G	4.6230E-06
	SMCP38	SEISMIC EVENT	0.358G - 0.383G	4.6230E-06
	SMCP41	SEISMIC EVENT	0.384G - 0.408G	4.6230E-06
	SMCP43	SEISMIC EVENT	0.409G - 0.434G	2.2250E-06
	SMCP46	SEISMIC EVENT	0.435G - 0.459G	2.2250E-06
	SMCP49	SEISMIC EVENT	0.460G - 0.485G	2.2250E-06
	SMCP51	SEISMIC EVENT	0.486G - 0.510G	2.2250E-06
	SMCP55	SEISMIC EVENT	0.511G - 0.548G	1.5140E-06
	SMCP59	SEISMIC EVENT	0.549G - 0.587G	1.6140E-06
	SMCP63	SEISMIC EVENT	0.588G - 0.625G	1.6140E-06
	SMCP66	SEISMIC EVENT	0.626G - 0.663G	1.6140E-06
SEISMIC3	SMCP74	SEISMIC EVENT	0.664G - 0.740G	1.5330E-06
	SMCP78	SEISMIC EVENT	0.741G - 0.778G	7.6670E-07
	SMCP82	SEISMIC EVENT	0.779G - 0.816G	7.6670E-07
	SMCP92	SEISMIC EVENT	0.817G - 0.918G	1.0040E-06
	SMC1P5	SEISMIC EVENT	0.919G - 1.531G	3.4940E-06

TABLE 3-3

FRAGILITIES OF SELECTED EQUIPMENT FOR CCNPP

NO.	COMPONENT	Am	β_R	β_U	HCLPF	Impact
1	Emergency Diesel Generator Control Panels 1C61A,B,C, 2C61A,B,C	1.01	.39	.56	.21	bounded by SRW impact
2	Control Room HVAC Control Panels 1NB108, 2NB408	.66	.40	.52	.14	CR HVAC
3	SG Blowdown and Waste Sampling Hoods					determined to be negligible impact on CDF
	U1 B/D Waste	.46	.38	.56	.10	
	U2 B/D Waste*	.22	.38	.56	.05	
	U1 R/C Samples	.80	.38	.56	.17	
	U2 R/C Sample*	.38	.38	.56	.08	
4	Bounding Case Block Wall in Auxiliary Bldg. (A at Elev. 27' and C at Elev. 69')	1.06	.38	.44	.27	impact of this wall bounded by other components
5	Refueling Water Storage Tank	.48	.22	.44	.16	RWST
6	U2 Turbine Lube Oil Cooler	.54	.36	.40	.15	SRW
7	Generator Bus Duct Cooler	.61	.39	.41	.17	SRW
8	U2 EHC Cooler	.67	.31	.43	.20	SRW
9	U1 Turbine Lube Oil Cooler	.97	.31	.31	.35	SRW
10	Block Wall Near Waste Process Area Degassifier	.88	.34	.41	.26	SRW
11	TDAFP Room A/C Unit	.50	.33	.50	.13	SRW
12	Turbine Building SRW Piping	.44	.41	.30	.14	SRW
13	0CAHU1 DG Control Room A/C Unit	.67	.41	.37	.18	0C-EDG
14	0CAHU1C DG Control Room A/C Condenser Unit	.80	.29	.42	.25	0C-EDG
15	0CAHU2 DG Rm. Air Handler	.67	.41	.37	.18	0C-EDG

* Expansion anchors are only 1/4", thus very low capacity.

TABLE 3.4

SEISMIC SPLIT FRACTION VALUES

Fragility Parameters		Am	1.2		0.96		0.48	0.43	0.266	0.63	0.2	0.1	2
		Br	0.4		0.18		0.22			0.4	0.2	0.4	0.4
		Bu	0.44		0.3		0.44			0.52	0.25	0.5	0.44
		Bc	0.59		0.35		0.49	0.41	0.36	0.66	0.32	0.64	0.59
			Top Event			LA	LB	LE	LG	LH	LJ	LK	LL
IE No.	SUFFIX	pga	1 Surrog	5 Surrog's	CST	5 sur+CST	RWT	0C EDG	SRW	CR HVAC	LOOP	Secondary	Cont Isol
SMCP03	1	0.026	0.00%	0.00%	0.00%	0.00%	0.00%	0.00%	0.00%	0.00%	0.00%	1.77%	0.00%
SMCP05	2	0.051	0.00%	0.00%	0.00%	0.00%	0.00%	0.00%	0.00%	0.01%	0.00%	14.66%	0.00%
SMCP08	3	0.077	0.00%	0.00%	0.00%	0.00%	0.01%	0.00%	0.03%	0.07%	0.14%	34.18%	0.00%
SMCP12	4	0.115	0.00%	0.02%	0.00%	0.02%	0.18%	0.06%	0.59%	0.46%	4.20%	58.64%	0.00%
SMCP13	5	0.134	0.01%	0.06%	0.00%	0.06%	0.47%	0.22%	2.64%	0.92%	10.56%	67.62%	0.00%
SMCP15	6	0.153	0.03%	0.13%	0.00%	0.13%	1.01%	0.59%	6.22%	1.56%	20.14%	74.67%	0.00%
SMCP18	7	0.179	0.07%	0.34%	0.00%	0.34%	2.25%	1.63%	13.58%	2.76%	36.45%	81.84%	0.00%
SMCP20	8	0.204	0.14%	0.72%	0.00%	0.72%	4.10%	3.45%	23.05%	4.28%	52.47%	86.72%	0.01%
SMCP23	9	0.230	0.27%	1.36%	0.00%	1.36%	6.74%	6.36%	34.31%	6.23%	66.89%	90.33%	0.01%
SMCP24	A	0.242	0.35%	1.76%	0.00%	1.76%	8.19%	8.04%	39.64%	7.24%	72.42%	91.62%	0.02%
SMCP26	B	0.255	0.46%	2.28%	0.01%	2.29%	9.93%	10.13%	45.33%	8.40%	77.60%	92.81%	0.03%
SMCP28	C	0.281	0.73%	3.61%	0.02%	3.63%	13.62%	14.97%	56.06%	10.92%	85.39%	94.67%	0.05%
SMCP31	D	0.306	1.08%	5.28%	0.05%	5.33%	18.61%	20.33%	65.14%	13.56%	90.80%	95.97%	0.08%
SMCP33	E	0.332	1.54%	7.44%	0.12%	7.56%	22.68%	26.41%	73.09%	16.44%	94.33%	96.95%	0.13%
SMCP36	G	0.357	2.07%	9.95%	0.23%	10.16%	27.37%	32.60%	79.31%	19.33%	96.48%	97.66%	0.19%
SMCP38	H	0.383	2.74%	12.97%	0.43%	13.34%	32.32%	38.89%	84.44%	22.40%	97.88%	98.20%	0.27%
SMCP41	I	0.408	3.48%	16.24%	0.72%	16.86%	37.08%	44.90%	88.26%	25.39%	98.70%	98.60%	0.38%
SMCP43	J	0.434	4.36%	19.98%	1.16%	20.91%	41.89%	50.90%	91.31%	28.50%	99.22%	98.91%	0.51%
SMCP46	K	0.459	5.30%	23.85%	1.75%	25.16%	46.38%	56.32%	93.52%	31.47%	99.53%	99.13%	0.67%
SMCP49	L	0.485	6.38%	28.09%	2.55%	29.92%	50.84%	61.56%	95.24%	34.51%	99.72%	99.32%	0.86%
SMCP51	M	0.510	7.51%	32.31%	3.53%	34.70%	54.90%	66.14%	96.47%	37.37%	99.83%	99.45%	1.08%
SMCP55	N	0.548	9.37%	38.87%	5.45%	42.20%	60.62%	72.29%	97.77%	41.58%	99.92%	99.61%	1.47%
SMCP59	O	0.587	11.46%	45.58%	7.99%	49.93%	65.88%	77.61%	98.61%	45.71%	99.96%	99.71%	1.96%
SMCP63	P	0.625	13.63%	51.94%	11.00%	57.23%	70.42%	81.91%	99.12%	49.52%	99.98%	99.79%	2.52%
SMCP66	Q	0.663	15.92%	57.98%	14.50%	64.07%	74.43%	85.45%	99.44%	53.10%	99.99%	99.84%	3.17%
SMCP74	R	0.740	20.81%	68.86%	22.84%	75.97%	81.05%	90.73%	99.78%	59.69%	100.00%	99.91%	4.73%
SMCP78	T	0.778	23.31%	73.47%	27.40%	80.74%	83.69%	92.59%	99.86%	62.61%	100.00%	99.93%	5.62%
SMCP82	U	0.816	25.83%	77.56%	32.11%	84.76%	85.96%	94.09%	99.91%	65.33%	100.00%	99.95%	6.56%
SMCP92	V	0.918	32.62%	86.11%	44.91%	92.35%	90.63%	96.78%	99.97%	71.70%	100.00%	99.97%	8.82%
SMC1P5	Z	1.508	64.96%	99.47%	90.16%	99.95%	99.00%	99.89%	100.00%	90.83%	100.00%	100.00%	31.74%

TABLE 3-5

MEDIAN CAPACITIES and HCLPF VALUES FOR BUILDINGS and STRUCTURES

Building/Structure	Median Capacity	HCLPF
Containments		
Shell--With Prestress	7.78g	2.85g
Shell--Without Prestress	2.68g	0.98g
Reinforced Concrete Base Slab	2.31g	0.70g
Auxiliary Building--Main		
Reinforced Concrete Shear Wall	2.54g	0.95g
Steel Column	2.50g	0.90g
Auxiliary Building Penetration Area		
Reinforced Concrete Shear Wall	2.76g	1.05g
Reinforced Concrete Shear Wall	1.19g	0.45g
Intake Structure		
Reinforced Concrete Shear Wall	2.30g	0.93g
Reinforced Concrete Shear Wall	1.40g	0.55g
New EDG Building		
Reinforced Concrete Shear Wall	2.12g	0.80g
Reinforced Concrete Slab	1.24g	0.49g
Turbine Building		
Steel Columns	2.88g	1.02g
Fire Pump House		
Steel Beam	23.22g	8.51g
Condensate Storage Tank Tornado Enclosure--Reinforced Concrete Shear Wall	4.05g	1.84g
Fuel Oil Storage Tank Tornado Enclosure Reinforced Concrete Shear Wall	8.70g	3.70g

* values are referenced to peak ground acceleration in g's

TABLE 3-6

SEISMIC CONTRIBUTION TO FREQUENCIES OF MAJOR CONTAINMENT FAILURE CATEGORIES FROM THE KEY PLANT DAMAGE STATES

Containment Failure Category	Associated Seismic CDF	Percent of Seismic CDF	HRWF	HRIF	HGIP
			1.10E-06	9.07E-06	5.12E-07
I. Intact Containment	4.62E-07	4.33%			4.62E-07
II. Late Containment Failure	8.63E-06	80.9%		8.614E-06	2.05E-08
III. Early Small Containment Failure	1.70E-07 to 1.27E-06	1.3% to 11.9%	0 to 1.10E-06	1.632E-07	6.65E-09
IV. Early Large Containment Failure	3.13E-07 to 1.41E-06	2.9% to 13.2%	0 to 1.10E-06	2.902E-07	2.25E-08
V. Small Containment Bypass	0.00E+00	0.00%			
VI. Large Containment Bypass	0.00E+00	0.00%			
Total of all groups	1.07E-05	100%	100%	100%	100%

Note: The Seismic contribution to Containment failure categories IV and V is shown as a range of values. A range is shown because the contribution of PDS HRWF will be apportioned between the small and large early containment failures, but the ratio is unknown. Therefore, we show a range of values which reflect the contribution of PDS HRWF from being attributed entirely to early-large containment failures (conservative) to early-small containment failures. See section 3.1.6.1 for a more detailed explanation. These results reflect quantification without the surrogate Top Event LA.

TABLE 3-7

TOP 100 SEISMIC CORE DAMAGE SEQUENCES
(quantified without Surrogate Top Event LA)

Initiating Event	Bin	Frequency	PDS	Sequence	Cumulative CDF
SMC1P5	3	2.37E-06	HRIF	(1-(LLZ+SI4+SRA+SG4+SHH+RSL))	22.23%
SMC1P5	3	1.11E-06	HRWF	LLZ	32.60%
SMCP74	3	1.31E-07	HRIF	LER*XW2*FCH	33.83%
SMCP74	3	1.16E-07	HRIF	LER*F31	34.92%
SMCP92	3	9.47E-08	HRIF	XW2*FCH	35.81%
SMCP74	3	9.21E-08	HRIF	LER*H32*HSU*F1R	36.67%
SMCP92	3	8.40E-08	HRIF	F31	37.46%
SMCP74	3	7.94E-08	HRIF	LER*GE5*F1R	38.20%
SMCP74	3	7.39E-08	HRIF	LER*XW2*HX5*F1R	38.89%
SMCP82	3	6.81E-08	HRIF	LEU*XW2*FCH	39.53%
SMCP74	3	6.71E-08	HRIF	GJ5*XW2*FCH	40.16%
SMCP78	3	6.70E-08	HRIF	LET*XW2*FCH	40.79%
SMCP92	3	6.65E-08	HRIF	H32*HSU*F1R	41.41%
SMCP41	2	6.45E-08	HRIF	LEI*LGI*GE5*HX3*F1R	42.01%
SMCP82	3	6.04E-08	HRIF	LEU*F31	42.58%
SMCP74	3	5.95E-08	HRIF	GJ5*F31	43.13%
SMCP78	3	5.94E-08	HRIF	LET*F31	43.69%
SMCP92	3	5.74E-08	HRIF	GE5*F1R	44.23%
SMCP31	2	5.68E-08	HRIF	LGD*LJD*GE5*GJA*HX3*F1R	44.76%
SMCP38	2	5.34E-08	HRIF	LEH*LGH*GE5*HX3*F1R	45.26%
SMCP92	3	5.34E-08	HRIF	XW2*HX5*F1R	45.76%
SMCP74	3	5.21E-08	HRIF	LER*HX5*UQ5	46.25%
SMCP51	2	5.17E-08	HRIF	LEM*GE5*HX3*F1R	46.73%
SMCP66	2	4.85E-08	HRIF	LEQ*GE5*HX3*F1R	47.19%
SMCP49	2	4.81E-08	HRIF	LEL*GE5*HX3*F1R	47.64%
SMCP82	3	4.78E-08	HRIF	LEU*H32*HSU*F1R	48.08%
SMCP74	3	4.71E-08	HRIF	GJ5*H32*HSU*F1R	48.52%
SMCP78	3	4.70E-08	HRIF	LET*H32*HSU*F1R	48.96%
SMCP63	2	4.65E-08	HRIF	LEP*GE5*HX3*F1R	49.40%
SMCP74	3	4.65E-08	HRIF	LER*XW2*Q11*QZ4	49.83%
SMCP74	3	4.50E-08	HRIF	LER*H32*HSU*MH5	50.26%
SMCP41	2	4.48E-08	HRIF	LGI*GE5*GJA*HX3*F1R	50.68%
SMCP59	2	4.41E-08	HRIF	LEO*GE5*HX3*F1R	51.09%
SMCP38	2	4.28E-08	HRIF	LGH*GE5*GJA*HX3*F1R	51.49%
SMCP36	2	4.19E-08	HRIF	LEG*LGG*GE5*HX3*F1R	51.88%
SMCP74	3	4.19E-08	HRIF	LER*XW2*HX5*MH2	52.27%
SMCP74	3	4.15E-08	HRIF	GE5*GJA*F1R	52.66%
SMCP82	3	4.12E-08	HRIF	LEU*GE5*F1R	53.05%
SMCP46	2	4.12E-08	HRIF	LEK*LGK*GE5*HX3*F1R	53.43%
SMCP78	3	4.06E-08	HRIF	LET*GE5*F1R	53.81%
SMCP36	2	4.02E-08	HRIF	LGG*GE5*GJA*HX3*F1R	54.19%

TABLE 3-7

TOP 100 SEISMIC CORE DAMAGE SEQUENCES (Continued)

Initiating Event	Bin	Frequency	PDS	Sequence	Cumulative CDF
SMCP74	3	3.89E-08	HRIF	LER*GE5*MH5	54.55%
SMCP55	2	3.85E-08	HRIF	LEZ*GE5*HX3*F1R	54.91%
SMCP82	3	3.84E-08	HRIF	LEU*XW2*HX5*F1R	55.27%
SMCP74	3	3.79E-08	HRIF	LER*FCH*F33	55.63%
SMCP74	3	3.78E-08	HRIF	GJ5*XW2*HX5*F1R	55.98%
SMCP78	3	3.77E-08	HRIF	LET*XW2*HX5*F1R	56.34%
SMCP92	3	3.76E-08	HRIF	HX5*UQ5	56.69%
SMCP31	2	3.69E-08	HRIF	LED*LGD*LJD*GE5*HX3*F1R	57.03%
SMCP43	2	3.64E-08	HRIF	LEJ*LGJ*GE5*HX3*F1R	57.37%
SMCP33	2	3.50E-08	HRIF	LGE*LJE*GE5*GJA*HX3*F1R	57.70%
SMCP82	3	3.36E-08	HRIF	GJ5*XW2*FCH	58.02%
SMCP78	3	3.36E-08	HRIF	GJ5*XW2*FCH	58.33%
SMCP92	3	3.36E-08	HRIF	XW2*Q11*QZ4	58.64%
SMCP74	3	3.26E-08	HRIF	H32*HSU*FCH	58.95%
SMCP92	3	3.25E-08	HRIF	H32*HSU*MH5	59.25%
SMCP74	3	3.08E-08	HRIF	F3C	59.54%
SMCP92	3	3.02E-08	HRIF	XW2*HX5*MH2	59.83%
SMCP78	3	2.98E-08	HRIF	GJ5*F31	60.10%
SMCP33	2	2.96E-08	HRIF	LEE*LGE*LJE*GE5*HX3*F1R	60.38%
SMCP92	3	2.81E-08	HRIF	GE5*MH5	60.64%
SMCP92	3	2.74E-08	HRIF	FCH*F33	60.90%
SMCP82	3	2.70E-08	HRIF	LEU*HX5*UQ5	61.15%
SMCP74	3	2.67E-08	HRIF	GJ5*HX5*UQ5	61.40%
SMCP78	3	2.66E-08	HRIF	LET*HX5*UQ5	61.65%
SMCP23	1	2.60E-08	HRIF	LG9*LJ9*GE5*GJA*HX3*F1R	61.90%
SMCP41	2	2.59E-08	HRIF	LEI*LGI*H32*HSU*HX3*F1R	62.14%
SMCP74	3	2.55E-08	HRIF	LER*GE5*FCH	62.38%
SMCP51	2	2.44E-08	HRIF	GE5*GJA*HX3*F1R	62.61%
SMCP49	2	2.44E-08	HRIF	GE5*GJA*HX3*F1R	62.84%
SMCP82	3	2.41E-08	HRIF	LEU*XW2*Q11*QZ4	63.06%
SMCP74	3	2.38E-08	HRIF	GJ5*XW2*Q11*QZ4	63.28%
SMCP78	3	2.37E-08	HRIF	LET*XW2*Q11*QZ4	63.51%
SMCP74	3	2.36E-08	HRIF	LER*HSJ*HZ3*F1R	63.73%
SMCP82	3	2.36E-08	HRIF	GJ5*H32*HSU*F1R	63.95%
SMCP78	3	2.36E-08	HRIF	GJ5*H32*HSU*F1R	64.17%
SMCP82	3	2.34E-08	HRIF	LEU*H32*HSU*MH5	64.39%
SMCP74	3	2.30E-08	HRIF	GJ5*H32*HSU*MH5	64.60%
SMCP78	3	2.30E-08	HRIF	LET*H32*HSU*MH5	64.82%
SMCP46	2	2.28E-08	HRIF	LGK*GE5*GJA*HX3*F1R	65.03%
SMCP43	2	2.23E-08	HRIF	LGJ*GE5*GJA*HX3*F1R	65.24%
SMCP41	2	2.20E-08	HHIP	LEI*LGI*F31	65.45%

TABLE 3-7

TOP 100 SEISMIC CORE DAMAGE SEQUENCES (Continued)

Initiating Event	Bin	Frequency	PDS	Sequence	Cumulative CDF
SMCP31	2	2.19E-08	HRIF	LGD*LJD*GJ5*H32*HSU*HX3*F1R	65.65%
SMCP28	1	2.19E-08	HRIF	LGC*LJC*GE5*GJA*HX3*F1R*(1-(SRA))	65.86%
SMCP82	3	2.17E-08	HRIF	LEU*XW2*HX5*MH2	66.06%
SMCP38	2	2.15E-08	HRIF	LEH*LGH*H32*HSU*HX3*F1R	66.26%
SMCP74	3	2.14E-08	HRIF	GJ5*XW2*HX5*MH2	66.46%
SMCP78	3	2.14E-08	HRIF	LET*XW2*HX5*MH2	66.66%
SMCP92	3	2.14E-08	HRIF	H32*HSU*FCH	66.86%
SMCP51	2	2.08E-08	HRIF	LEM*H32*HSU*HX3*F1R	67.06%
SMCP74	3	2.08E-08	HRIF	LER*XW2*TF4	67.25%
SMCP74	3	2.03E-08	HRIF	GE5*GJA*MH5	67.44%
SMCP82	3	2.02E-08	HRIF	LEU*GE5*MH5	67.63%
SMCP41	2	2.01E-08	HRIF	LEI*LGI*GE5*HX3*MH5	67.82%
SMCP26	1	1.99E-08	HRIF	LGB*LJB*GE5*GJA*HX3*F1R*(1-(SRA))	68.00%
SMCP78	3	1.98E-08	HRIF	LET*GE5*MH5	68.19%
SMCP82	3	1.97E-08	HRIF	LEU*FCH*F33	68.37%
SMCP66	2	1.95E-08	HRIF	LEQ*H32*HSU*HX3*F1R	68.56%
SMCP82	3	1.94E-08	HRIF	LHU*GJ5*F31	68.74%
SMCP74	3	1.94E-08	HRIF	GJ5*FCH*F33	68.92%

TABLE 3-8

PARTIAL LIST OF SPLIT FRACTION DESCRIPTIONS AND VALUES

SPLIT FRACTION	DESCRIPTION	BIN 1 VALUE	BIN 2 VALUE	BIN 3 VALUE
F1R	AFW delivers adequate flow, given SBO with S/G overfill	4.85E-01	6.10E-01	7.38E-01
F31	AFW has adequate inventory (includes op action for tank switchover), no CNTRL room indication	1.22E-03	1.20E-02	8.37E-02
F33	AFW has adequate inventory (includes op action for tank switchover) given no indication	7.25E-03	3.20E-02	1.15E-01
FCH	AFW PP RM cooling operates, given a SBO condition, EOP8	7.83E-03	4.69E-02	2.37E-01
FF2	Operator aligns AFW Turbine Pump Room Emergency Cooling Fans, EOP-8	9.82E-02	3.34E-01	9.40E-01
FN4	Operator aligns N2 or starts SWACs to AFW CVs, given a dual unit SBO, EOP-8	6.16E-02	2.07E-01	6.60E-01
GB5	EDG 1A starts & provides power to 4KV Bus 11, LOOP > 11 hours, ASA	7.74E-02	7.74E-02	7.74E-02
GJA	EDG 0C starts & provides power to a 4KV Bus, LOOP > 11 hours, EDG 1A fails, ASA	2.34E-01	3.12E-01	4.74E-01
H32	OP starts stand-by Switchgear HVAC train within 2 hours, LOOP	1.87E-02	6.18E-02	1.78E-01
HSU	SWGR HVAC HDR OPERATES given HVAC train 12 fails due to loss of support; HVAC train 11 has support; 4KV XFMER 11 failure or LOOP causes momentary loss of 4KV Bus 11; Operator fails to start standby train; Instrument Air available or not.	5.04E-01	5.04E-01	5.04E-01
HX5	Operator controls AFW flow, locally due to CR AFW Flow control support unavailable for either flowpath where flow exists, S/G Level Ind avail, EOP-08	1.98E-02	6.08E-02	1.81E-01
HZ3	OP locally ventilates both Swgr Rms using temporary fans, LOOP, 4KV Bus 14 failed	1.65E-01	3.01E-01	4.96E-01
LED	0C EDG sustains a seismic event up to 0.306 g.	2.03E-01	2.03E-01	2.03E-01
LEG	0C EDG sustains a seismic event up to 0.357 g.	3.25E-01	3.25E-01	3.25E-01
LEH	0C EDG sustains a seismic event up to 0.383 g.	3.89E-01	3.89E-01	3.89E-01
LEI	0C EDG sustains a seismic event up to 0.408 g.	4.49E-01	4.49E-01	4.49E-01
LEJ	0C EDG sustains a seismic event up to 0.434 g.	5.09E-01	5.09E-01	5.09E-01
LEK	0C EDG sustains a seismic event up to 0.459 g.	5.63E-01	5.63E-01	5.63E-01
LEL	0C EDG sustains a seismic event up to 0.485 g.	6.15E-01	6.15E-01	6.15E-01
LEM	0C EDG sustains a seismic event up to 0.51 g.	6.61E-01	6.61E-01	6.61E-01
LEO	0C EDG sustains a seismic event up to 0.587 g.	7.76E-01	7.76E-01	7.76E-01
LEP	0C EDG sustains a seismic event up to 0.625 g.	8.19E-01	8.19E-01	8.19E-01
LEQ	0C EDG sustains a seismic event up to 0.663 g.	8.55E-01	8.55E-01	8.55E-01
LEZ	0C EDG sustains a seismic event up to 0.548 g.	7.23E-01	7.23E-01	7.23E-01
LG9	SRW Headers 11, 12, 21 and 22 sustain a seismic event up to 0.23 g.	3.43E-01	3.43E-01	3.43E-01
LGD	SRW Headers 11, 12, 21 and 22 sustain a seismic event up to 0.306 g.	6.51E-01	6.51E-01	6.51E-01
LGE	SRW Headers 11, 12, 21 and 22 sustain a seismic event up to 0.332 g.	7.31E-01	7.31E-01	7.31E-01
LGG	SRW Headers 11, 12, 21 and 22 sustain a seismic event up to 0.357 g.	7.93E-01	7.93E-01	7.93E-01
LGH	SRW Headers 11, 12, 21 and 22 sustain a seismic event up to 0.383 g.	8.44E-01	8.44E-01	8.44E-01

TABLE 3-8

PARTIAL LIST OF SPLIT FRACTION DESCRIPTIONS AND VALUES (Continued)

SPLIT FRACTION	DESCRIPTION	BIN 1 VALUE	BIN 2 VALUE	BIN 3 VALUE
LGI	SRW Headers 11, 12, 21 and 22 sustain a seismic event up to 0.408 g.	8.83E-01	8.83E-01	8.83E-01
LGJ	SRW Headers 11, 12, 21 and 22 sustain a seismic event up to 0.434 g.	9.13E-01	9.13E-01	9.13E-01
LGK	SRW Headers 11, 12, 21 and 22 sustain a seismic event up to 0.459 g.	9.35E-01	9.35E-01	9.35E-01
LHR	Control Room HVAC sustains a seismic event up to 0.74 g.	5.97E-01	5.97E-01	5.97E-01
LHT	Control Room HVAC sustains a seismic event up to 0.778 g.	6.26E-01	6.26E-01	6.26E-01
LHU	Control Room HVAC sustains a seismic event up to 0.816 g.	6.53E-01	6.53E-01	6.53E-01
LHV	Control Room HVAC sustains a seismic event up to 0.918 g.	7.17E-01	7.17E-01	7.17E-01
LJ9	500KV Switchyard sustains a seismic event up to 0.23 g.	6.69E-01	6.69E-01	6.69E-01
LJB	500KV Switchyard sustains a seismic event up to 0.255 g.	7.76E-01	7.76E-01	7.76E-01
LJC	500KV Switchyard sustains a seismic event up to 0.281 g.	8.56E-01	8.56E-01	8.56E-01
LJD	500KV Switchyard sustains a seismic event up to 0.306 g.	9.08E-01	9.08E-01	9.08E-01
LJE	500KV Switchyard sustains a seismic event up to 0.332 g.	9.43E-01	9.43E-01	9.43E-01
LLE	Containment Isolation sustains a seismic event up to 0.332 g.	1.26E-03	1.26E-03	1.26E-03
LLR	Containment Isolation sustains a seismic event up to 0.74 g.	4.73E-02	4.73E-02	4.73E-02
LLT	Containment Isolation sustains a seismic event up to 0.778 g.	5.62E-02	5.62E-02	5.62E-02
LLU	Containment Isolation sustains a seismic event up to 0.816 g.	6.58E-02	6.58E-02	6.58E-02
LLV	Containment Isolation sustains a seismic event up to 0.918 g.	9.52E-02	9.52E-02	9.52E-02
MH2	Operator recovers failed steam admission line to AFW turbine driven pumps, EOP-8	2.15E-02	1.03E-01	4.18E-01
QI1	OP recovers AFW at onset (w/in 10 minutes) of a spurious AFAS block (Cowboy)	2.55E-01	3.75E-01	5.28E-01
QZ4	OP recovers AFW following a spurious AFAS block, given PORVs open on SSSA	1.45E-02	5.26E-02	1.59E-01
RSL	Reactor Trip actuates, given a LOOP (4KV Buses12 & 13 de-energize)	3.26E-06	3.26E-06	3.26E-06
SRA	Cntmt Normal Sump Drain Line isolates on a LOCA, no support available	2.01E-03	2.01E-03	2.01E-03
TF4	AFW TURB PP 11 provides adequate flow given PP 13 successful or not?ed, both GBL and Turb Rec fails	3.75E-02	3.75E-02	3.75E-02
TG4	AFW Pump 12 works, AFW PP 11 successful & 13 fails, EOP8	2.02E-01	5.03E-01	1.00E+00
UQ5	OPs do not underfill S/Gs when AFW flow control is lost, given: S/G Level Ind avail, remote flow control failed, EOP-8	5.69E-02	1.15E-01	2.07E-01
XW2	OP supplies a 120VAC Vital Panel from 208/120 VAC Instrument Bus, given MCC 104R is available	1.06E-01	2.17E-01	3.98E-01

SECTION 4

INTERNAL FIRES ANALYSIS

4.0 Methodology Selection

The Nuclear Regulatory Commission (NRC) in Generic Letter (GL) No. 88-20 Supplement 4 (Ref. 4-1), "Individual Plant Examination of External Events (IPEEE) for Severe Accident Vulnerabilities" requested all utilities to perform an individual plant examination of external events to identify any severe accident vulnerabilities that may exist at their nuclear power plants and to submit the results of the examination to the NRC.

The Baltimore Gas and Electric Company selected the fire PRA method to assess the total core damage frequency due to fire events at CCNPP. This selection is based primarily on the expected future usefulness of a plant specific fire PRA.

4.1 Fire Hazard Analysis

A Level I PRA is used to evaluate the fire risk for CCNPP Unit 1. Unit 2 was reviewed and significant differences are evaluated. This summary report is written from the Unit 1 perspective. The Unit 2 assessment is contained in Section 4.6.7.

The general guidelines described in NUREG-1407 (Ref. 4-2) and the EPRI Fire PRA Implementation Guide (Ref. 4-3) are followed. The EPRI Fire Events Database (Ref. 4-4) is used for the quantification of fire ignition frequencies. The Fire-Induced Vulnerability Evaluation (FIVE) methodology (Ref. 4-5) is used as guidance for the evaluation of specific fire scenarios within a compartment when the failure of the entire compartment proved to be significant. The quantification of fire-induced core damage frequency (CDF) is obtained by propagating fire-induced failures through a modified version of CCPRA. This modified version is constructed from the General Transient module of CCNPP's updated internal events Level 1 PRA. This is referred to as the "base CCPRA."

A screening process is used to identify the compartments and compartment groups of concern. Compartments are grouped when barriers between them are not credited. See Section 4.3.1. A fire initiating event is assigned to each compartment, compartment group or for fire scenarios within a compartment. In addition, fire initiating events are developed for compartments which are grouped as a result from cross-zone fire spread (fire propagation across credited fire barriers). A CDF point estimate for each of these initiating events is calculated as follows:

$$F_{CDF} = F_I * P_e * P_{fs} * CCDP$$

F_{CDF} = fire-induced core damage initiating event frequency
F_I = fire ignition frequency
P_e = probability of induced equipment damage
P_{fs} = probability of fire suppression failure
CCDP = Conditional Core Damage Probability given the fire damage

The overall fire CDF is determined by summing all the compartment CDFs and scenario CDFs including the contribution due to cross-zone fire spread. These results are summarized in Section 4.6.6.

Cross-zone Propagation

The assessment of the likelihood of cross-zone propagation includes the determination of the fire frequency at which each barrier is challenged, the likelihood of suppression effectively stopping the fire growth and the likelihood that the barrier is effective in preventing the fire from spreading to an adjacent compartment. See Section 4.3.3 for details on the approach used in the Calvert Cliffs Fire PRA (CCFPRA).

Compartment Fire Modeling

Those compartments which exhibited unacceptable conditional CDFs when modeled as fully burned or those which have a large system functional loss potential are evaluated to determine compartment specific fire scenarios. See Section 4.3.4 for the detailed fire modeling process.

Fire scenarios are developed for over twenty compartments. These fire scenarios are then grouped, assigned an initiating event designator and then propagated through the PRA. See Attachments 4-A through 4-U for details on the development of fire scenarios for each compartment.

Cable Analysis

Cable analysis is perhaps the most important element of CCNPP's fire risk analysis. The cable routing used in CCFPRA addresses all functions considered in the base CCPRA. Therefore, assuming failure of those non Appendix R components where cable routing information is typically not available was not required. This approach required the routing and analysis of approximately 5,500 cables. It should be noted that cables for the containment mitigation and isolation functions, and those fire-induced initiating events that are in addition to a standard reactor trip are explicitly evaluated. See Section 4.4.1 for more details on the cable routing analysis.

Quantification of Fire-Risk

The method used to propagate the impact of fire induced fires through the PRA model is described in Section 4.6. Included in this discussion is a description of BGE's comprehensive treatment of human actions. In addition, the reactivity control functions which support the likelihood that the reactor trips on demand are explicitly modeled by the evaluation of impact of fires on the loss of the Reactor Protection System and the Diverse Scram System.

A containment performance review is also done to identify any sequences leading to containment failure that are different from those identified in the individual examination of internal events. See Section 4.7.

Fire Risk Scoping Study

Fire Risk Scoping Study Issues are addressed through specifically tailored walkdowns as defined in the FIVE methodology, including seismic fire interactions, effects of fire suppressant on safety repeated equipment, fire barrier effectiveness and control systems interactions. See Section 4.8.

4.1.1 Key Assumptions

The Calvert Cliffs Fire PRA (CCFPRA) uses an updated version of the IPE logic models. Consequently, fundamental assumptions associated with the Level 1 analysis are applicable. The following key assumptions are used in this analysis. Specific assumptions for each compartment are included in the analysis for those compartments.

1. The reactivity control functions which support the likelihood that the reactor trips on demand are explicitly modeled by the evaluation of impact of fires on the loss of the Reactor Protection System and the Diverse Scram System.
2. Internal events operator actions are assumed to be applicable. However, they are failed if a fire blocks the performance of the action in question and are degraded to reflect the impact of fire and smoke. See Section 4.6.1.
3. Fire induced spuriously opening PORVs and RCP Seal LOCAs due to the loss of the Component Cooling Water System are explicitly modeled.
4. For fires which impact off-site power, a 24 hour LOOP duration is assumed if off-site power is lost and the Control Room is abandoned, a four hour duration is assumed if a 4KV service transformer is lost with the Control Room manned and two hours is assumed for all other scenarios. Note that a fire in the switchyard has a similar impact as that of a standard loss of off-site power event and therefore its impact is considered to be included in the internal events loss of offsite power initiating event.
5. A 24 hour period is assumed as the base mission time for this analysis. This time is consistent with the internal events analysis and NUREG-1335 (Ref. 4-12).
6. Cross-zone propagation between Appendix R areas and between sub-divided areas within Appendix R areas with credited barriers is considered possible and is explicitly evaluated. Barriers which are not in an inspection and/or control program are assumed to be ineffective. See Section 4.3.3.
7. It is assumed that all automatic fire suppression systems are properly sized to effectively mitigate the growth and spread of fire within their zones of coverage.
8. It is assumed that the Halon 1301 extinguishing agent has no short term effects on equipment operation based on an evaluation of chemical characteristics, industry studies and operational experience.

4.2 Review of Plant Information and Walkdown

This section provides a description of the plant, particularly as it relates to the fire protection program and other aspects of the plant that bear on the Fire PRA for CCNPP. The IPE Summary Report (Ref. 4-6) for CCNPP provides descriptions and simplified diagrams of the plant systems that were modeled for that study, and therefore, they are not included here. This section also includes a discussion of the plant familiarization work and walkdowns that were done as part of the fire study.

4.2.1 Plant Fire Protection Program Description

The CCNPP Fire Protection Program (Ref. 4-11) is an integrated effort involving components, procedures and personnel used to carry out all activities of fire protection and prevention. The program provides the necessary controls to protect the health and safety of CCNPP workers and the general public, satisfy NRC and insurer requirements and safeguard Company assets with the goal to prevent fires and to minimize the consequences of any fire that does occur. To achieve this goal the program established a defense-in-depth approach by providing:

1. Design of systems and facilities that minimize the probability and consequences of fire; and that provide for the detection, annunciation, confinement and suppression of fire.
2. Controls to verify the operability and availability of equipment and systems through inspection and test programs.
3. Controls that prevent fires from starting and minimize potential fire hazards.
4. Controls that establish planned strategies in response to fires including trained fire fighting brigades.
5. Training programs that provide fire prevention and response training consistent with assigned duties and responsibilities.
6. Monitoring and continuous assessment of Fire Protection Program performance.
7. Periodic audits of the Fire Protection Program.

4.2.2 Plant Familiarization

This section describes the plant familiarization work performed to complete the CCFPRA. The CCFPRA is developed in a large part by BGE personnel who have familiarity with the plant and its systems. Their work is supported by the site's fire protection engineers. Key information used in support of the CCFPRA is described below. Information contained in the IPE (Ref. 4-6) and its update is also used, but is not included below.

4.2.2.1 Fire Hazards Analysis

The Calvert Cliffs Fire Hazards Analysis Summary Document (Ref. 4-7) provides a compartment by compartment summary of the fire protection features in plant compartments which contain safe shutdown components. This document identifies for each compartment:

- the major Appendix R safe shutdown equipment
- the Appendix R Abnormal Operating Procedure number
- the type of fire detection and suppression
- the Appendix R exemptions, if any
- a summary of the primary fixed combustibles
- the equivalent fire severity of the combustible loading

4.2.2.2 Interactive Cable Analysis

The Interactive Cable Analysis for Calvert Cliffs Nuclear Power Plant Unit 1 (Ref. 4-8) analyzes the consequences of a fire in any area associated with the safe shutdown of Unit 1. A similar document exist for Unit 2 (Ref. 4-9).

This document includes listing of equipment and cables in specific fire areas, previously defined in the Calvert Cliffs Nuclear Power Plant Fire Protection Evaluation Program (Ref. 4-10), required to reach hot standby and cold shutdown. It provides an analysis of the impact of losing each area as a result of a fire and identifies the available recovery actions.

4.2.2.3 Combustible Loading Analysis Report

The Combustible Loading Analysis identifies the assumptions, methodology and results of the July 1993, combustible loading re-analysis for Calvert Cliffs Nuclear Power Plant Units 1 and 2 (Ref. 4-14). This calculation:

- defined the specific plant area which required calculation
- calculated floor area for each specific plant area
- identified heat content (Btu) attributed to cable insulation
- identified heat content (Btu) attributed to fixed and transient combustibles
- summed the total heat content found in each area then calculated the heat load (Btu/ft^2) and fire severity (min.) for the plant area

This information was used as a prime input into the calculation for the fire ignition frequency for each area. See Section 4.3.2.

4.2.2.4 Abnormal Operating Procedures (AOPs)

These procedures are used by operators when responding to a particular abnormal situation in the plant, such as a fire. The AOPs which address mitigating a fire are listed below:

AOP-9A	Control Room Evacuation and Safe Shutdown due to a Severe Control Room Fire
AOP-9B	Safe Shutdown due to a Severe Cable Spreading Room Fire
AOP 9C	Safe Shutdown due to a Severe Fire in Room 100, 103, 104, 110 or 116 Auxiliary Building (-)10' and (-)15' Corridors
AOP-9D	Safe Shutdown due to a Severe Fire in Room 119 - Unit 1 No. 11 ECCS Pump Room
AOP-9E	Safe Shutdown due to a Severe Fire in Room 225 - Unit 1 5' Exhaust Fan Equipment Room
AOP-9F	Safe Shutdown due to a Severe Fire in Room 226 - Unit 1 Service Water Pump Room
AOP-9G	Safe Shutdown due to a Severe Fire in Room 227/316 - Unit 1 5' and 27' East Piping Penetration Rooms
AOP-9H	Safe Shutdown due to a Severe Fire in Room 228 - Unit 1 Component Cooling Pump Room
AOP-9I	Safe Shutdown due to a Severe Fire in Room 315 - Unit 1 Main Steam Penetration Room
AOP-9J	Safe Shutdown due to a Severe Fire in Room 317 - Unit 1 Switchgear Room 27'

AOP-9L	Safe Shutdown due to a Severe Fire in Cable Chase 1A - Unit 1 West Vertical Cable Chase
AOP-9M	Safe Shutdown due to a Severe Fire in Cable Chase 1B - Unit 1 East Vertical Cable Chase
AOP-9N	Safe Shutdown due to a Severe Fire in Room 408, 410, 413, 419, 424, 425, 426 or 428 Auxiliary Building - 45' Corridors and Sample Rooms
AOP-9P	Safe Shutdown due to a Severe Fire in Room 429 - Unit 1 Auxiliary Building 45' East Electrical Penetration Room
AOP-9Q	Safe Shutdown due to a Severe Fire in Room 430 - Unit 1 Switchgear Room 45'
AOP-9R	Safe Shutdown due to a Severe Fire in Room 603 - Unit 1 Auxiliary Feed Pump Room
AOP-9S	Safe Shutdown due to a Severe Fire in Room 423 - Unit 1 Auxiliary Building 45' West Electrical Penetration Room

4.2.2.5 Fire Fighting Strategies Manual

The Calvert Cliffs Fire Fighting Strategies Manual provides the operator and fire fighter with pertinent information in the case of a fire in vital areas, particularly for those areas containing safe shutdown equipment.

For each area there is an elevation floor plan with the appropriate compartment highlighted and access/egress routes clearly marked. Locations of fire extinguishers, hose stations and electrical outlets found on that elevation are also indicated.

Individual floor plans show a detailed area layout and indicate the location of fire fighting apparatus, important plant equipment, various hazards (fire, toxic, radiological, electrical, etc.) access/egress routes, smoke ejection routes and electrical power supply breaker numbers.

4.2.2.6 Fire Prevention Administrative Procedure

CCNPP Fire Prevention Administrative Procedure, SA-1-100, establishes administrative controls and requirements to prevent fires and to ensure activities are conducted in a manner that promotes fire prevention. It promulgates the plant processes for controlling transient combustibles, ignition sources, fire barrier penetrations and the use of fire suppression water systems for non-emergency, non-fire fighting purposes.

4.2.2.6.1 Controlling Transient Combustible

SA-1-100 requires minimizing the amount of transient fire loading introduced inside or adjacent to safety-related areas or systems and in the Turbine Building during maintenance, modifications, or operations activities. It also requires that the amount of transient material introduced into an area will not exceed allowable combustible limits specified in the Combustible Loading Analysis.

The Combustible Loading Analysis includes an allowance equal to the following total quantities of transients to be introduced into plant areas:

- 100 pounds of ordinary combustibles
- 5 gallons of flammable liquids
- 55 gallons of combustible liquids

It also requires that debris, scrap, rags, oil spills or other waste combustibles resulting from the work activity are removed either after completion of the activity or at the end of the shift, whichever occurs first.

4.2.2.6.2 Controlling Ignition Sources

SA-1-100 requires Hot Work Permits for work using flame, heat or spark producing equipment with the exception of permanent welding areas, shop areas and laboratories for work normally performed within those areas. The Hot Work Permit process requires the following precautions:

- The work area will be personally examined ensuring that the floors and surroundings have been swept clean and that combustibles, floors and construction form work (if any) have been wetted down.
- If the work requires cutting, welding or other spark-producing equipment, all combustibles will be located at least 30 feet to 40 feet from the activity or protected with metal guards, flameproof curtains or covers (not ordinary tarpaulins).

  All floor and wall openings within 40 feet of the activity shall be covered tightly or a fire watch will be established for lower elevations. Floor drains shall not be covered.

  A responsible person shall be assigned to watch for sparks in the area, as well as on floors below. The area shall be roped off or posted.
- A fire watch shall remain for at least one-half hour after work has stopped.

Additional restrictions are identified below for the Control Room, Cable Spreading Rooms, Switchgear Rooms and Data Acquisition System (DAS) Computer Rooms:

- No welding or burning shall be performed above the lowest cable tray in the Cable Spreading Rooms unless the work-controlling documents are reviewed by POSRC and authorized by the Superintendent - Nuclear Operations.
- Alternatives to welding and burning should be explored in all other areas of the Cable Spreading Rooms, Switchgear Rooms and Control Room.
- The duration of welding or burning in the Cable Spreading Rooms, Switchgear Rooms and Data Acquisition System (DAS) Computer Rooms should be minimized because the Halon 1301 system will have to be impaired to prevent unwanted discharge. Welding in these areas shall be pre-approved by the Supervisor - Safety and Fire Protection Unit.

SA-1-100 also prohibits the use of open flame or combustion smoke for leak testing of cable penetrations and fire barriers or for the determination of air flow.

4.2.2.6.3 Controlling Fire Barrier Penetrations

SA-1-100 requires a Fire System/Fire Barrier Impairment Permit for the impairment of an Appendix R fire area boundaries. It also requires an evaluation of compensatory actions such as continuous or periodic fire watches, testing of fire detection and suppression systems or the installation of a temporary fire barrier.

4.2.3 Plant Walkdown

In the course of this study, several walkdowns were conducted to gather the following specific information:

- Ignition Source Walkdowns
- Detailed Fire Modeling Walkdowns
- Fire Barrier Walkdowns

4.2.3.1 Ignition Source Walkdowns

These walkdowns in conjunction with plant drawings are used to develop compartment layout drawings which show the locations and type of ignition sources, major equipment and fire protection features within each compartment. Fire barriers and propagation paths associated with each compartment are also noted. The scope of the walkdowns include a total of 222 compartments in the following areas:

- Auxiliary Building
- Diesel Generator Rooms
- Turbine Building
- Intake Structure
- Yard Areas

Note that the containments and compartments with high radiation are evaluated through the use of drawings and were not physically walkdown.

4.2.3.2 Detailed Fire Modeling Walkdowns

A detailed walkdown of each compartment evaluated for detailed fire modeling was performed. The results of these walkdowns are discussed in Section 4.6.2.

4.2.3.3 Fire Barrier Walkdowns

Walkdowns for all fire barriers which establish the compartment groupings associated with each initiating event that are subdivisions of an Appendix R Fire Area were performed. This was done to ensure that there is no concentration of combustibles near the barriers and to determine the quality of the barrier and the number of penetrations (doors, ventilation openings, and piping and electrical penetrations) through the barrier. This is further discussed in Sections 4.3.1 and 4.3.3.

4.3 Fire Growth and Propagation

4.3.1 Identification of Fire Areas to be Analyzed

Fire ignition frequencies are calculated for all fire compartments that contain PRA components and/or cables. Adjacent compartments to compartments containing PRA related components or cables are also evaluated to access the adequacy of the barrier to contain a fire from propagating This approach assures that any reduction in the mitigation capacity of the plant is evaluated. The ignition frequency for a total of 243 compartments determined to contain PRA components or cables and nine adjacent compartments are calculated. Some areas, such as the Turbine Building, Containment, Intake Structure, Warehouse Area and Yard are broken down into separate PRA Compartments so that they could be evaluated in a more appropriate manner.

A compartment is a well-defined enclosed room, not necessarily with Appendix R fire barriers. One or more compartments are contained within an Appendix R Fire Area. A Fire Area is defined as an area with Appendix R barriers. See Table 4.3.1.

A screening and binning process is used to evaluated the fire risk associated with these compartments. Compartments with no PRA components or cables, or low functional impact are screened. However, these compartments are still assessed as ignition sources in the cross-zone propagation described in Section 4.3.3. Compartments with low fire ignition frequency are also screened. These compartments are also assessed in the cross-zone analysis, but as potential targets not sources.

Each compartment which is not screened is assigned to one or more initiating event designators. The initiating events are assigned an appropriate frequency and set of plant impacts. These impacts are used to modify the base CCPRA in order to determine the fire risk impact.

The compartments in Table 4.3.1 below which have brackets around the room designator are screened because they have no PRA Components or Cables, a low functional impact or a low fire hazard.

Table 4.3.1
Appendix R Fire Area to CCFPRA Compartment

Fire Area	General Description	Unit 1 Initiator	CCFPRA Compartments Within the Fire Area
1	21 ECCS Pump Room	AUX10C	A101
2	22 ECCS Pump Room	AUX10C	A102
3	12 ECCS Pump Room	AUX10E	A118
4	11 ECCS Pump Room	AUX10E	A119
5	11 Charging Pump Room	AUX10B	A115A
6	12 Charging Pump Room	AUX10B	A115B
7	13 Charging Pump Room	AUX10B	A115C
8	22 Charging Pump Room	Low Impact	[A105B]
9	23 Charging Pump Room	Low Impact	[A105C]
10	-10'/-15' Hallways & General Areas	AUX10A	A100, A103, A104, [A105A], A106, A108, [A110], [A111], A113, [A116]
11	-15, -10, 5', 27', 45' and 69' General Areas and Miscellaneous Areas	See Table 4.3.1.1	A107, [A109], [A112], A114, [A117], [A120], A122, A200, A202, A203, A206, A207, [A208], [A209], [A210], [A211], A212, A213, [A214], A215, A216, A216A, A217, A218, A219, A220, A221, A222, A223, A224, A227, [A308], A309, A310, [A315], A316, A319, A320, [A321], A322, A323, A324, [A325], A326, [A327], [A328], A408, A410, [A412], A413, [A417], [A418], A419, A424, [A425], A426, A428, A512, A520, A523, A524, A525, A526, A527, [A528], A530, A531, A533, A586, A587, A588, A589, A590, A591, A592, A593, A594, A595, A596, A597, SHFT
12	Unit 2 CCW Pump Room	AUX20B	A201
13	Unit 2 5' Fan Room	AUX20B	A204
14	Unit 1 5' Fan Room	A225F(x)	A225
15	Unit 1 CCW Pump Room	Low Fire Hazard	[A228]
16	Unit 1 Cable Spreading Room & 1C Chase	A306F(x)	A306, CC1C
16A	Unit 1 Battery Rooms	See Table 4.3.1.1	A301, A304
16B	Hallway outside Unit 1 CSR	FCA300	A300
17	Unit 2 Cable Spreading Room & 2C Chase	A302F(x)	A302, CC2C
17A	Unit 2 Battery Rooms	See Table 4.3.1.1	A305, A307
17B	Hallway Outside Unit 2 CSR	FCA300	A303
18	Unit 2 27' Switchgear Room	A311F(x)	A311
18A	Unit 2 Purge Air Room	FIA312	A312
19	Unit 1 27' Switchgear Room	A317F(x)	A317
19A	Unit 1 Purge Air Room	FIA318	A318
20	Cable Chase 1A	Low Fire Hazard	[CC1A]
21	Cable Chase 1B	Low Fire Hazard	[CC1B]
22	Cable Chase 2A	Low Fire Hazard	[CC2A]
23	Cable Chase 2B	Low Fire Hazard	[CC2B]
24	Control Room Complex	FI1Cxx,FICxxx FI2Cxx and A405Fx	[A400], A401, [A402], [A403], A404, A405, A406, A415, A431, A432, [A434], [A436], [A437], [A438], A442, [A443], A444
25	Unit 2 45' Switchgear Room	A407F(x)	A407

Table 4.3.1
(Cont'd)
Appendix R Fire Area to PRA Compartment

Fire Area	General Description	Unit 1 Initiator	CCFPRA Compartments Within the Fire Area
26	Unit 2 East Electrical Penetration Room	Low Impact	[A409]
27	Unit 2 West Electrical Penetration Room	FIA414	A414
28	2B Diesel Generator Room	FIA416	A416
29	Unit 2 RWT Room	Low Impact	[A440]
30	1B Diesel Generator Room & RC Waste Room	See Table 4.3.1.1	A420, A421
31	2A Diesel Generator Room	FIA422	A422
32	Unit 1 West Electrical Penetration Room	FIA423	A423
33	Unit 1 East Electrical Penetration Room	FIA429	A429
34	Unit 1 45' Switchgear Room	A430F(x)	A430
35	Unit 2 Horizontal Chase	Low Fire Hazard	[A517]
36	Unit 1 Horizontal Chase	Low Fire Hazard	[A518]
37	Unit 1 69' Electrical Room	A529F1	A529
38	Unit 2 69' Electrical Room	Low Impact	[A532]
39	Unit 1 Service Water Pump Room	A226F(x)	A226
40	Unit 2 Service Water Pump Room	AUX20B	A205
41	69' Misc. Waste Evap Room	No PRA Comps	[A536], [A537]
42	Unit 1 AFW Pump Room	T603F(x)	T603
43	Unit 2 AFW Pump Room	FIT605	T605
44	Unit 1 RWT Pump Room	FIA439	A439
AB-1	Auxiliary Building Stairtower	Low Fire Hazard	[AB-1]
AB-2	Auxiliary Building Stairtower	FIMAB2	AB-2
AB-3	Auxiliary Building Stairtower	Low Fire Hazard	[AB-3]
AB-4	Auxiliary Building Stairtower	Low Fire Hazard	[AB-4]
AB-5	Auxiliary Building Stairtower	Low Fire Hazard	[AB-5]
AB-E	Auxiliary Building Elevator Shaft	Low Fire Hazard	[AB-E]
IS	Intake Structure	INTKF(x)	IS
CNTMT	Unit 1 and Unit 2 Containment Buildings	See Section 4.3.1.5	C121, C123, C229, C230
Yard	Outside yard area and buildings	See Table 4.3.1.1	Transformers, Fire Pump House, Well Water Pump House, Tanks, Switchyard Area, Voltage Regulators, 1A & 0C Diesel Generator Buildings, 13KV Switchgear Houses
TB	Unit 1 and Unit 2 Turbine Building	TB(x)	Turbine Building, Water Treatment, Warehouse, [A559]

When calculating ignition frequencies for the fire propagation analysis it is appropriate to exclude certain plant rooms on the basis that they would not be the source of a fire with sufficient intensity to result in the damage of other plant areas. The rooms listed in Sections 4.3.1.2, 4.3.1.3 and 4.3.1.4 are screened due to absence of PRA components or cables, low functional impact and insufficient fire intensity, respectively. No additional evaluation is performed for these rooms. The Containment Building is also screened due to its low fire hazard. See Section 4.3.1.5.

4.3.1.1 Subdivided Fire Areas

Some Appendix R Areas have been divided into compartments or compartment groupings and are represented by two or more initiating events compartments. This subdivding enables more realistic fire modeling. Each common barrier between initiating event groupings is analyzed for fire spread from either direction. The description of the barrier effectiveness assessment is in Section 4.3.3. The following Fire Areas have multiple initiating events:

Table 4.3.1.1
Subdivided Fire Areas

Fire Area	PRA Initiator	PRA Compartment
11	AUX10D	A107, A114
	AUX10E	A122 (Also includes A118 (Area 3) and A119 (Area 4)
	AUX20A	A200, A202, A203, A207, A212, A213, A215, A216, A216A, A217, A218, A219, A220, A222, A223, A224, SHFT
	FCA206	A206, A310
	FCA221	A221, A326
	FIA309	A309
	FCA319	A319, A320, A322, A323, A324
	A419Fx	A408, A410, A413, A419, A424, A426, A428
	FCA227 (Note 1)	A227. A316
	A512F(x), A524F(x)	A512, A520, A524, A525
	FCA523	A523, A526, A527, A530, A531, A533, A586, A587, A588, A589, A590, A591, A592, A593, A594, A595, A596, A597
16A	FIA301	A301
	FIA304	A304
17A	FIA305	A305
	FIA307	A307
30	FIA420	A420
	FIA421	A421
Yard	F1AEDG	Emergency Diesel Generator 1A Building
	F0CEDG	Emergency Diesel Generator 0C Building and Aux Building Roof
	FFPPHS	Fire Pump House
	FCYRD1 thru 6	Combinations of Transformers U-25000-11, U-25000-12, U-22000-21, U-22000-22, U-4000-11, U-4000-12, U-4000-13, U-4000-21
	FCYRDA	Unit 1 13KV Transformer P-13000-1, Bus 11/12, and Switchgear; Voltage Regulators 1H1102 and 1H1103; Transformers U-4000-11 and U-4000-12,
	FCYRDB	Unit 2 13KV Transformer P-13000-2, Bus 21/22, and Switchgear; Voltage Regulators 2H2102 and 2H2103; Transformers U-4000-21 and U-4000-22
	FCYRDC	Voltage Regulator 1H1101 and Transformer U4000-13
	FCYRDD	Voltage Regulator 2H2101 and Transformer U4000-23; Well Water Pump House

Note 1: Although designated as a Group Compartment Initiator, this compartment is fire modeled and is included in Attachment C.

4.3.1.2 Screened Rooms - No PRA Components

The compartments listed below are screened due to the absence of PRA components and due to the absence of cables having the potential to cause failure of a PRA function. No additional fire propagation evaluation within the compartment is performed for these compartments. However, these compartments are still assessed as ignition sources in the cross-zone propagation described in Section 4.3.3.

Table 4.3.1.2
Screened Rooms - No PRA Components or Cables

Room	Fire Area	Room Description
A112	11	12 Reactor Coolant Waste Receiver Tank Room
A116	10	North Hallway by Personnel Elevator
A208	11	Waste Gas Surge Tank Room
A209	11	Decon Room
A210	11	RCP Seal Decon Room
A325	11	Personnel Elevator Area
A327	11	Spent Fuel Pool Cooling Demineralizer Room
A328	11	Spent Fuel Pool Cooling Filter Room
A400	24	Control Room Vestibule
A402	24	Toilet - Control Room
A403	24	Janitor Storage Adj To Control Room Toilet
A412	11	Cask Washdown Pit
A417	11	Cask Loading Area
A425	11	Personnel Elevator Area - 45'
A434	24	Turbine Building Passage To AB-1 & TSC
A436	24	Technical Support Center
A437	24	Technical Support Annex
A438	24	Shift Supervisor's Office
A443	24	Reserve Battery Room Passage Way
A528	11	Personnel Elevator Area
A536	41	Miscellaneous Waste Evaporator Control Panel Room
A537	41	Miscellaneous Waste Evaporator Room

4.3.1.3 Screened Rooms - Low Functional Impact

The compartments identified in Table 4.3.1.3 are screened due to low functional impact. The functions impacted by these compartments are individually examined and qualitatively screened based on the knowledge of risk resulting from similar impacts previously evaluated. Note that the equipment and cables included in the base CCPRA are identified for these compartments and the loss of this equipment and cables are evaluated to make this determination. It is estimated that the risk resulting from fire in these compartments is well below 1E-7.

Examples of the type of functions impacted by the screened rooms are listed below. The analyses of these screened compartments did not show any indication that a reactor or turbine trip would result. However, all cables which could cause a plant trip through various secondary and supporting systems have not been identified. As a result, fire in compartments containing PRA identified equipment are assumed to result in a trip.

- Cable faults whose sole impact is to challenge the associated load breaker or breakers of a single bus. If the associated load breaker opens on demand, then the supplying bus is protected. If the load breaker fails to open then the bus is lost. Since an independent breaker failure is required, the failure probability of the associated bus is small. Therefore, the resulting plant risk is small.
- Failures of standby equipment; such as, the loss of an EDG and/or Unit 2 safety-related air compressors. In this case both normal 4KV power and normal instrument air are not impacted.

Table 4.3.1.3
Screened - Low Functional Impact

Room	Fire Area	Description
A105A	10	21 Charging Pump Room
A105B	8	22 Charging Pump Room
A105C	9	23 Charging Pump Room
A109	11	12 Reactor Coolant Waste Monitor Tank Room
A110	10	RCW Pump Area North/South Hall
A111	10	Cryogenics - Waste Processing Control Room
A117	11	Personnel Elevator Equipment Room
A120	11	Unit 2 Recirculation Pipe Tunnel
A211	11	Unit 2 West Piping Penetration Room
A214	11	Unit 2 Volume Control Tank Room
A321	11	Unit 2 West Piping Penetration Room
A409	26	Unit 2 East Electrical Penetration Room
A418	11	Solid Waste Processing Area
A440	29	Unit 2 RWT Room
A532	38	Unit 2 69' Electrical Room
A559	ACA*	Plant Computer Room

* The Access Control Area (ACA) is part of the Turbine Building Fire Area.

4.3.1.4 Screened Rooms - Low Fire Ignition Frequency

The following compartments are screened due to low fire ignition frequency. The detailed results of the fire hazard evaluation for each compartment below is addressed in Section 4.6.2. The evaluation of each compartment is included as an attachment to Section 4. Note that these compartments are assessed in the cross-zone analysis as potential targets but not as sources. See Section 4.3.3.

Table 4.3.1.4
Screened Rooms - Low Fire Ignition Frequency

Room	Fire Area	Description
A228	15	Unit 1 CCW Pump Room
A308	11	North/South Passage
A315	11	Unit 1 Main Steam Piping Penetration Room
CC1C	20	Cable Chase 1C
CC1C	21	Cable Chase 1C
AB-1	AB-1	Auxiliary Building Stairtower
AB-3	AB-3	Auxiliary Building Stairtower
AB-4	AB-4	Auxiliary Building Stairtower
AB-5	AB-5	Auxiliary Building Stairtower
AB-E	AB-E	Auxiliary Building Elevator Shaft

4.3.1.5 Screened Rooms - Containment

The FIVE methodology does not include ignition source information for the containment location because of the small number of fire events and the conclusion, by previous fire PRAs, that such fires were not risk significant.

FIVE indicates that risk significant fires are unlikely because:

1. A hot gas layer is unlikely to form in most areas of containment which can damage cables.

2. A large percentage of past fires were reactor coolant pump (RCP) fires which are unlikely to occur in the future due to oil collection system design improvements.

However, FIVE cautions that plant-unique features may not provide the equivalent protection against fire found in the examined plants. FIVE recommends a qualitative assessment to determine if a further, more detailed, analysis of the containment is needed. FIVE cites two issues that should prompt a detailed analysis:

1. Plant experience indicating that containment fires have occurred on a recurring basis.

2. The potential for redundant trains of critical equipment within containment to be exposed to the same fire plume or be in a confined space subject to damage by a hot gas layer.

There have been no recorded fires in the Unit 1 Containment during plant operation.

Electrical penetrations enter into the Calvert Cliffs containment on opposite (east and west) sides. Inside the containment, the cable routings, and their associated fire load concentrations, remain segregated in an east-west division as the cables travel to their end location.

Although the containment is a single Appendix R fire area, the Interactive Cable Analysis (ICA, see Section 4.2.2.2) subdivides the containment into three zones. East, west, and a twenty foot wide buffer area centered on the containment north-south center-line. The buffer zone has little fire loading. (Four cable east-west cable trays traverse the buffer zone. These trays have covers to prevent flame spread across the east west zones.)

The principle fire loads in each zone, east and west, are cables and reactor coolant pump (RCP) lube oil. The cables themselves are not ignition sources and do not propagate fire. The RCP lube oil system is encapsulated and uses an oil collection system to divert and control any leakage. In addition, the chosen oil has characteristics that minimize fire hazard. So, the concentration of loading is such that any credible fire and resulting plume is limited to the east or west zone.

The ICA analyzed all containment components necessary to place the reactor in cold shutdown based on a safety function (for example, reactivity control). The analysis shows that cold shutdown is possible using equipment in either zone, east or west, even with the added loss of the north-south buffer. A separate analysis, ES199602166, reviewed instrument tubing interactions.

In summary:

1. There is no plant specific history of, or known susceptibility to fires inside containment.

2. The separation of redundant critical equipment and cable trains is such that no single fire (plume or hot gas layer) will damage both trains. When necessary, shielding or other measures prevent flame spread across the redundant trains.

Figure 4.3 - 1
Auxiliary Building -10 & -15' Level

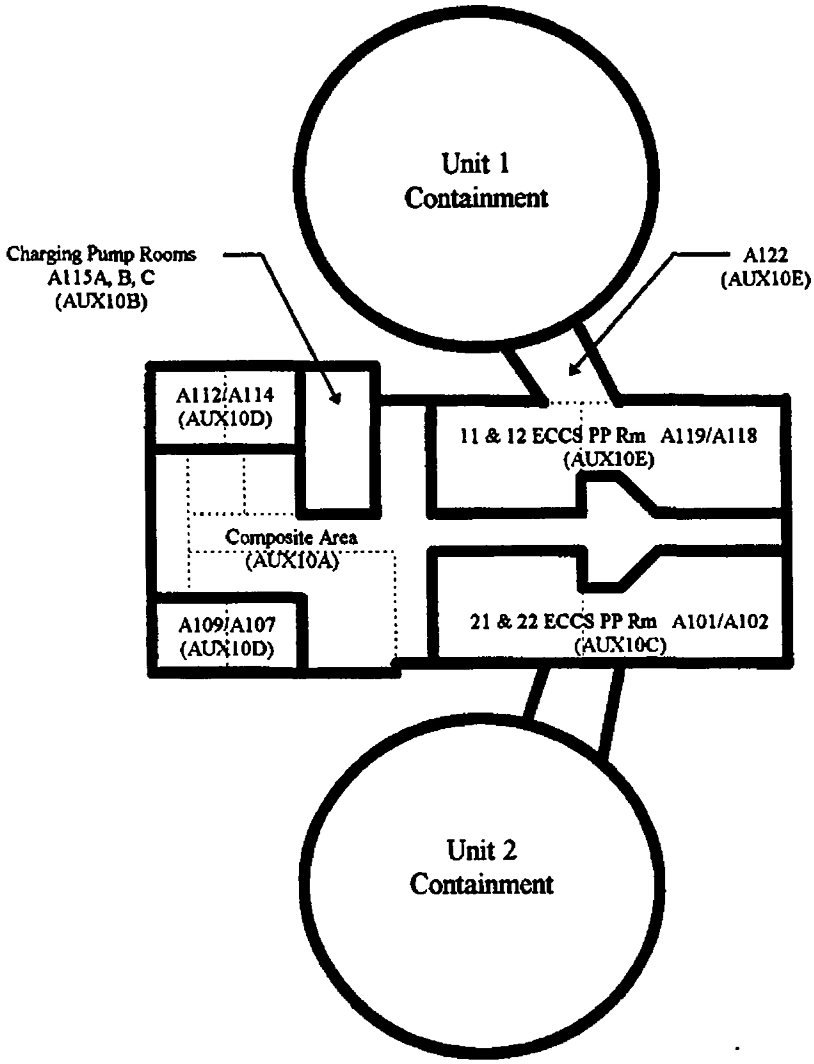


Figure 4.3 - 2
Auxiliary Building 5' Level

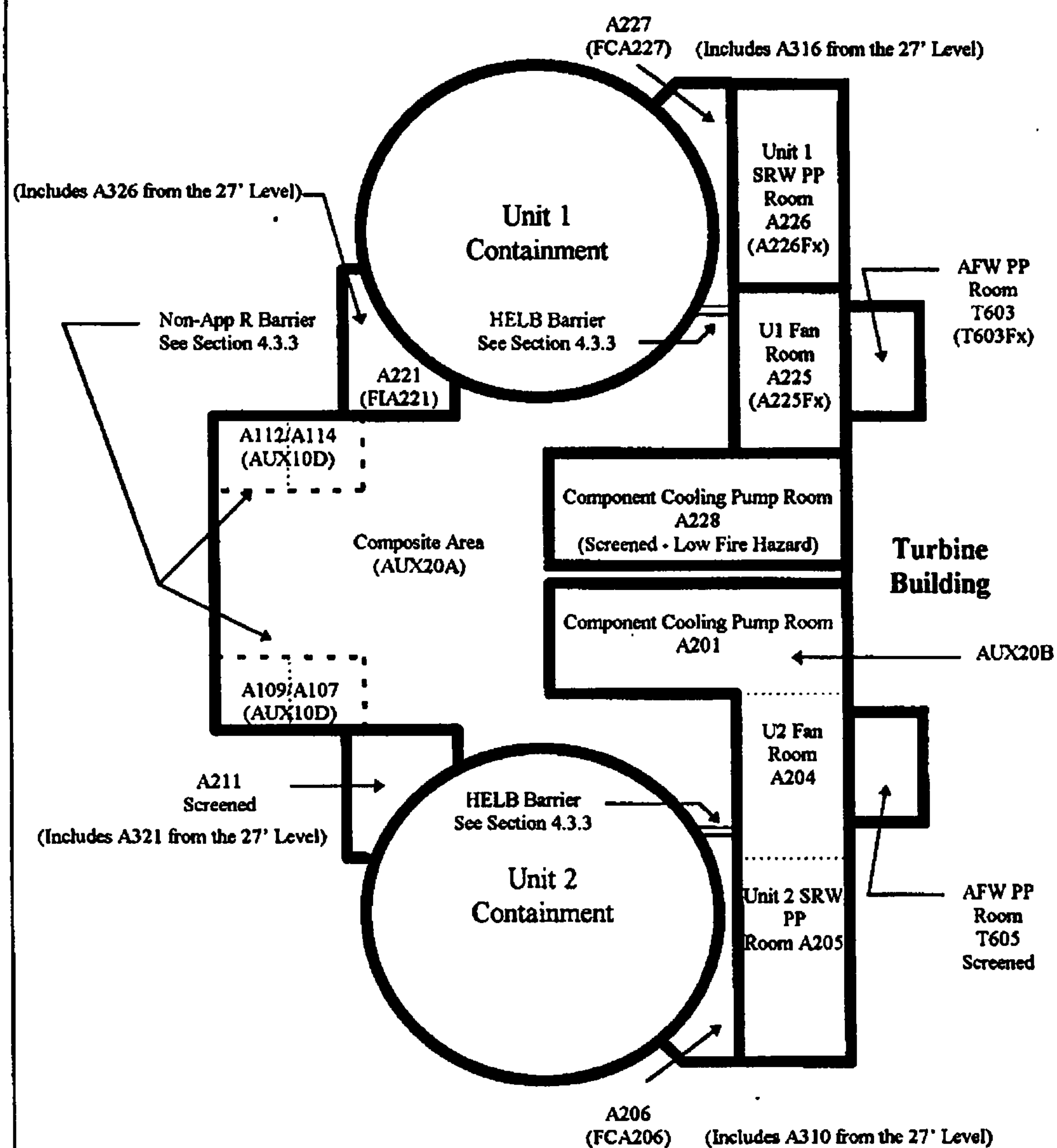


Figure 4.3 - 3
Auxiliary Building 27' Level

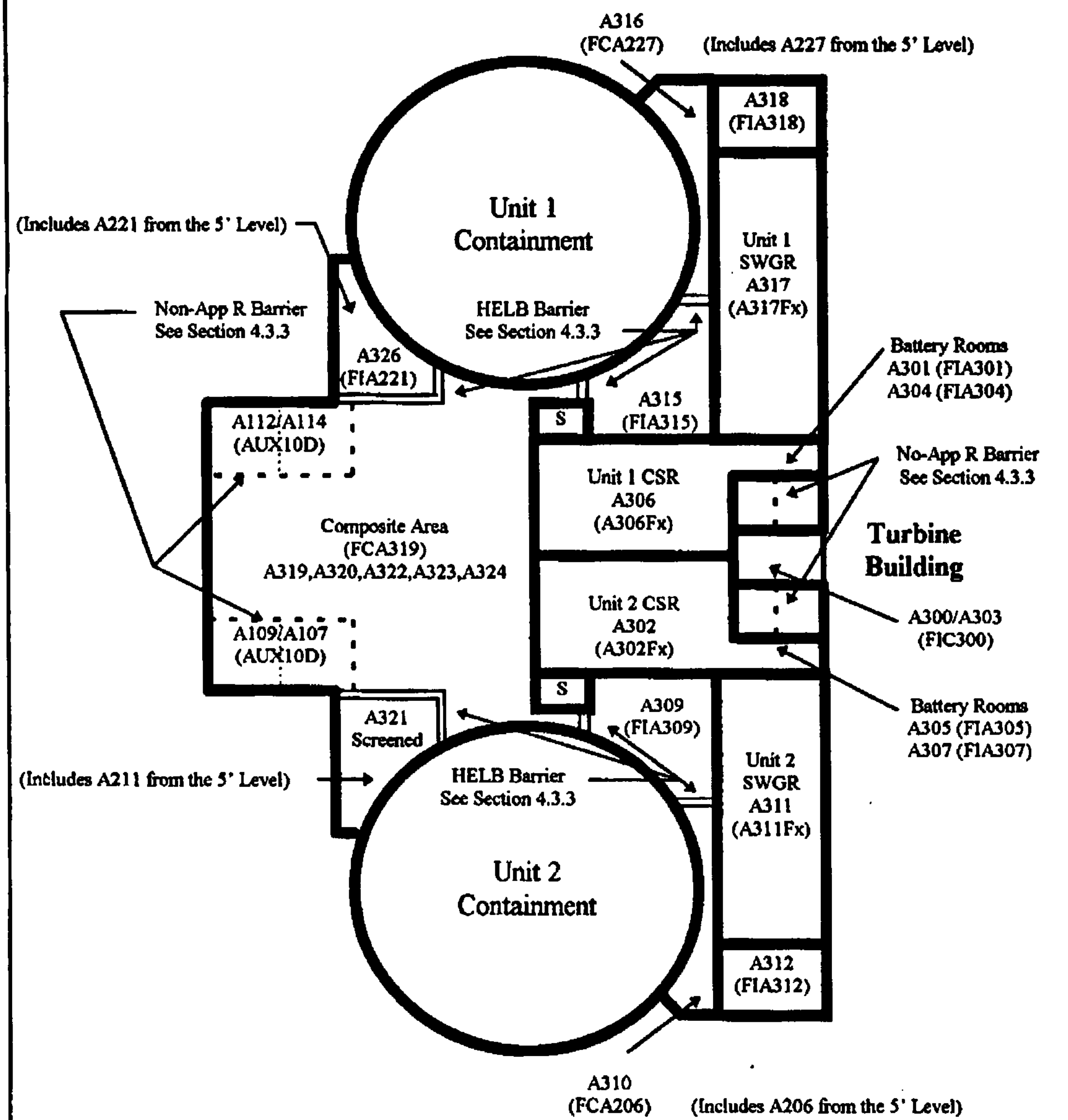


Figure 4.3 - 4
Auxiliary Building 45' Level

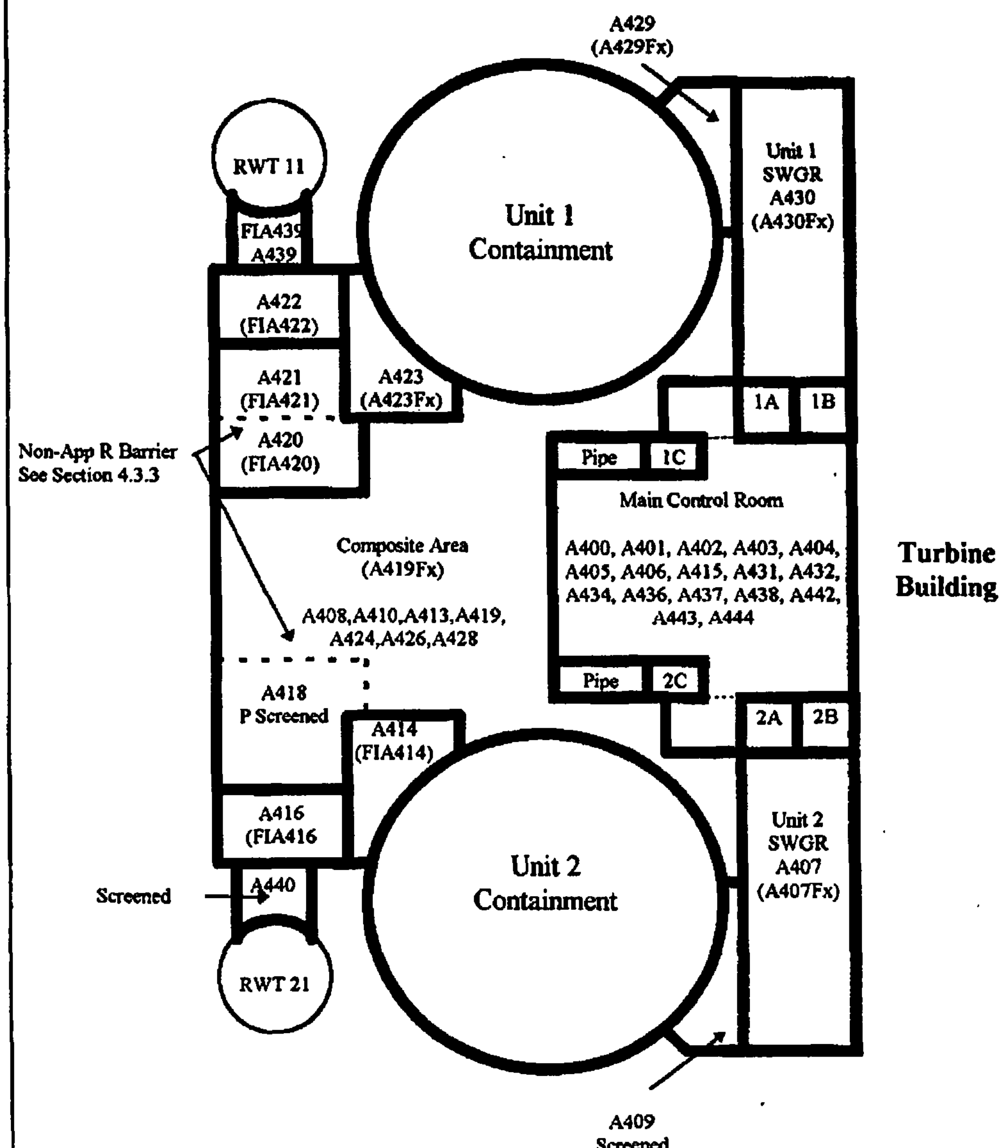
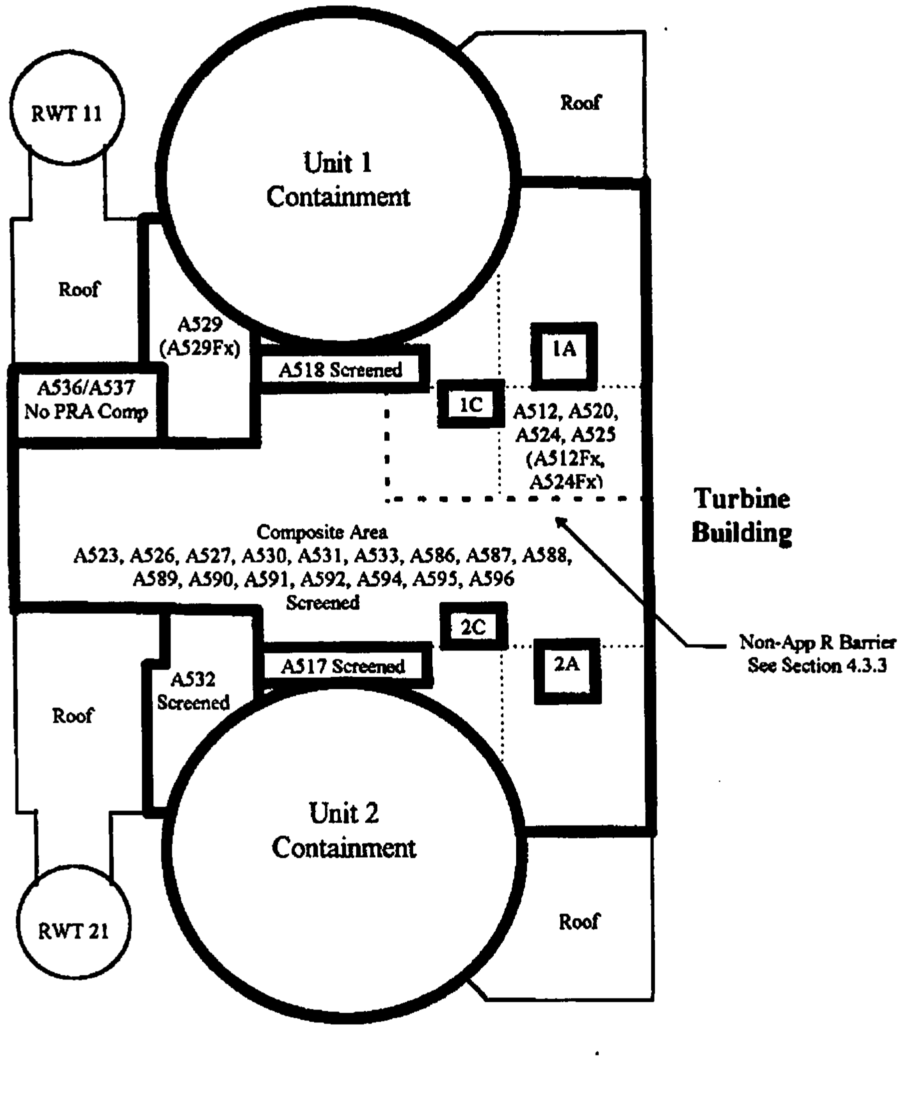


Figure 4.3 - 5
Auxiliary Building 69' Level



4.3.2 Quantification of Fire Ignition Frequencies

The process of determining fire frequencies for each of the fire compartments involved the following steps:

1. Identify all potential fire sources/components in a selected location, i.e., the total number of each components type in the location.
2. Select generic fire frequencies from the EPRI Fire Events Database applicable to the fire sources/components in the selected location. See Table 4.3.2
3. Determine the location weighting factor for the Area/Compartment. This weighting factor is used to translate generic fire frequencies for a location to specific, single-unit fire frequencies. The weighting factors are designed to account for the relative amount of ignition sources in CCNPP compared to the "average' plant. The following location weighting factors are used:

Table 4.3.2a
Fire Ignition Room Weighting Factors

Plant Location	Weighting Factors	CCNPP
Auxiliary Building	number of units / number of auxiliary buildings	2/1 = 2
Diesel Generator Room	number of diesels / number of diesel rooms per site	5/7 = 0.714
Switchgear Room	number of units / number of switchgear rooms	2/13 = 0.154
Battery Room	number of units / number of battery rooms	2/9 = 0.22
Control Room	number of units / number of control rooms	2/1 = 2
Cable Spreading Room	number of units / number of cable spreading rooms	2/2 = 1
Intake Structure	number of units / number of intake structures	2/1 = 2
Turbine Building	number of units / number of turbine buildings	2/1 = 2
Radwaste Area	number of units / number of radwaste areas	2/1 = 2
Transformer Yard	number of units / number of switchyards	2/1 = 2
Plant-Wide Components	number of units	2

4. Determine source weighing factors, i.e., the number of components in the selected location versus the total number of components in similar types of locations. The following codes contained in Table 4.3.2 are used to establish the weighting factors:

   A. No ignition source weighting factor is necessary.

   B. Obtain the ignition source weighting factor by dividing the number of ignition sources in the fire compartment by the number in the selected location.

   C. Obtain the ignition source weighting factor by calculating the inverse of the number of compartments in the locations. Exclude any areas contained in locations other than in this table.

D. Obtain the ignition source weighting factor by summing the factors for ignition sources which are allowed in the zone and divided by the number of zones in the locations in this table. For example, if cigarette smoking is prohibited do not include cigarette smoking factor in the calculation. The factors are:

Cigarette Smoking	2	(Not applicable at CCNPP)
Extension Cord	4	
Heater	3	
Candle	1	(Not applicable at CCNPP)
Overheating	2	
Hot pipe	1	

Note that candles are not allowed in accordance with SA-1-100 (See Section 4.2.2.6) and cigarette smoking is prohibited throughout the analyzed areas.

E. Obtain the ignition source weighting factor by dividing the weight (or BTUs) of cable insulation in area by the total weight (or BTUs) of cable insulation in Appendix R fire areas, not including fire areas in either the Radwaste area or the containment. Cable insulation weights (or BTUs) are provided in Appendix R combustible loading.

F. Obtain the ignition source weighting factor by dividing the number of ignition sources in the fire area by the total number in all the locations in this table.

G. Obtain the ignition source weighting factor by dividing the number of ignition sources in the fire area by the total number in all plant locations, including locations that were not specified in this table.

5. Calculate fire ignition frequency for each source and location.

6. The fire compartment frequency for each ignition source is calculated by multiplying:

- The generic frequency for an ignition source present in the compartment
- The weighting factor for the location determined in Step 3
- The weighting factor for that ignition source as calculated in Step 4

The calculation is repeated for each ignition source and the results are summed to obtained the total fire frequency for that fire compartment.

Table 4.3.2b
Fire Ignition Sources

Plant Location	Fire Ignition Source	Ignition Source Weighting Factor Method	Number of Fires	Generic Fire Frequency
Auxiliary Building	Electrical cabinets	B	15	1.9E-2
	Pumps	B	15	1.9E-2
Diesel Generator Room	Diesel generators	A	65	2.6E-2
	Electrical cabinets	A	6	2.4E-3
Switchgear Room	Electrical cabinets	A	19	1.5E-2
Battery Room	Batteries	A	4	3.2E-3
Control Room	Electrical cabinets	A	12	9.5E-3
Cable Spreading Room	Electrical cabinets	A	4	3.2E-3
Intake Structure	Electrical cabinets	A	3	2.4E-3
	Fire Pumps	A	5	4.0E-3
	Other Pumps	A	4	3.2E-3
Turbine Building	T/G Exciter	B	5	4.0E-3
	T/G Oil	B	17	1.3E-2
	T/G Hydrogen	B	7	5.5E-3
	Electrical cabinets	B	16	1.3E-2
	Other pumps	B	8	6.3E-3
	Main feedwater pumps	A	10	4.0E-3
	Boiler	B	2	1.6E-3
Radwaste Area	Miscellaneous components	A	11	8.7E-3
Transformer Yard	Yard transformers (propagating to Turbine Building)	A	5	4.0E-3
	Yard transformers (LOOP)	A	2	1.6E-3
	Yard transformers (Others)	F	19	1.5E-2
Plant-Wide Components	Fire protection panels	F	3	2.4E-3
	CEA-MG sets	F	7	5.5E-3
	Non-qualified cable run	E	8	6.3E-3
	Junction box/splice in non-qualified cable	E	2	1.6E-3
	Junction box in qualified cable	E	2	1.6E-3
	Transformers (Dry)	F	10	7.9E-3
	Transformers (Wet) (1)	F	n/a	1.9E-4
	Battery chargers	F	5	4.0E-3
	H2 Recombiner	G	41	8.6E-2
	Hydrogen Tanks	G	4	3.2E-3
	Misc. Hydrogen Fires	C	4	3.2E-3
	Air compressors	F	6	4.7E-3
	Ventilation Subsystems	F	12	9.5E-3
	Elevator motors	F	8	6.3E-3
	Dryers	F	11	8.7E-3
	Transients	D	13	1.3E-3
	Cable fire caused by welding	C	4	5.1E-3
	Transient fires caused by welding and cutting	C	24	3.1E-2

Note 1: Based on an EPRI study on the economic risk for electrical equipment containing PCBs.

4.3.3 Treatment of Cross-Zone Fire Spread

This section addresses the evaluation of a fire starting in one compartment and spreading to other compartments. This evaluation includes an assessment of both fire suppression and barrier effectiveness.

The frequency of cross-zone propagation is determined by using the following equation:

$$\textbf{Cross-Zone Propagation Frequency} = I * N * B$$

where I = Barrier Challenge Frequency
N = Suppression Failure Rate
B = Barrier Failure Probability

The zones in this cross-zone evaluation are represented by initiating events. Each PRA fire initiating event represents a fire ignition frequency for a compartment, a set of compartments or a fire scenario within a compartment. For the cross-zone evaluation, the barriers between the initiating event compartment or compartment grouping is assessed. Barriers within an initiating event group are assumed to be ineffective. For initiating events which represent a fire scenario, the set of fire scenarios for a compartment is used to represent that compartment. For example, Initiating Events A419F1, A419F2 ... A419F7 are represented by the designator A419Fx.

Only severe fires are considered for the determination of the barrier challenge frequency. If the compartment or compartment group represented by the initiating event is fired modeled, then only the fire scenarios which could result in the loss of the entire compartment are used as the basis for the barrier challenge frequency. If no compartment lost fire scenarios are identified, then the barrier challenge frequency is considered to be zero. For non-fire modeled compartments, the compartment ignition frequency is multiplied by a severity factor. See Section 4.3.3.2. For non-fire modeled compartment groups, the individual compartment fire frequencies are added and then multiplied by a severity factor.

The suppression failure rate is determined for each compartment of origin. The suppression failure rate of the compartment which the fire is spreading to is assumed to be unavailable. Although conservative, this assumption avoids the evaluation of common failure modes between the compartments' fire suppression systems.

All barriers, both Appendix R and non-Appendix R, between the zones represented by the initiating events were determined and assessed. The failure assessment considers the number of doors, dampers and penetration seals. All credited non-Appendix R barriers are being added to a plant inspection and/or control program.

Compartments are screened from further evaluation if the propagation likelihood is <1.0E-07.

The following sections provide additional detail as to the approach used for the cross-zone analysis. The quantification of fire-induced core damage frequency is obtained by propagating fire-induced failures through a modified version of the CCNPP PRA as described in Section 4.6. The results of the Barrier Analysis are addressed in Section 4.6.6.

4.3.3.1 Barrier Inspection Evaluation

This section addresses the inspection and configuration control of the compartment boundaries. Barriers without an adequate inspection and/or configuration control program are assumed to be ineffective. Barriers are considered to have an adequate program if they meet one of the following criteria.

- Barriers inspected to meet Appendix R requirements.
  STP-F-591-1, Inspection of Fire Doors, Watertight Doors, and Dampers in Fire Rated Barriers and STP-F-592-1, Penetration Fire Barrier Inspection, are currently performed on a periodicity of 18 months for all Appendix R barriers.

- Barriers walked down and determined to be HELB barriers.
  HELB and other adequate barriers are being added to an inspection program. Table 4.3.3a provides a list of the non-Appendix R barriers credited. These barriers are analyzed as having the same failure probabilities as those stated for Appendix R barriers. See Section 7, Improvement 5.

- Barriers with no penetrations
  Barriers walked down and determined to be constructed of a fire resistant material, such as concrete, with no penetrations, doors or ventilation openings. These barriers are being added to a configuration control program. Table 4.3.3b provides a list of these barriers. Also see Section 7, Improvement 5.

Table 4.3.3a
Barriers in an Inspection Program

Fire Area	PRA Initiator to PRA Initiator	Barrier ID	From Room Description	To Room Description
11	AUX20A/FIA211	<<2BARR-203/211>>	U2 5' PIPING AREA	U2 WEST PIPING PEN RM
	AUX20A/FIA206	<<2BARR-203/206>>	U2 5' PIPING AREA	U2 EAST PIPING PEN RM
	AUX20A/FIA221	<<1BARR-221/224>>	U1 WEST PIPING PEN RM	U1 5' PIPING AREA
	FCA319/FIA309	<<0BARR-308/309>>	CORRIDOR	U2 MSIV RM
	FCA319/FIA315	<<0BARR-308/315>>	CORRIDOR	U1 MSIV RM
	FIA221/A419F(x)	<<1BARR-324/419>>	U1 LTDWN HT XCHGR RM	CASK LDG & TRUCK BAY
	FCA319F(x)/A419F(x)	<<2BARR-322/419>>	U2 LTDWN HT XCHGR RM	CASK LDG & TRUCK BAY
	FIA221/A419F(x)	<<1BARR-326/419>>	U1 WEST PIPING PEN RM	CASK LDG & TRUCK BAY
	FCA512/A419F(x)	<<1BARR-428/524>>	U1 45' PIPING AREA	U1 MN PLANT EXH EQ RM
	FCA512/A419F(x)	<<1BARR-428/525>>	U1 45' PIPING AREA	U1 CNTMT ACCESS AREA
TB	TBMFW1/TBMFW2	<<0BARR-601/606>>	U1 12' Turbine Building	U2 12' Turbine Building
	TBMFW1/TBMFW2	<<0BARR-607/608>>	U1 12'TB Heater Bay	U2 12' TB Heater Bay
	TBMFW1/TBMFW2	<<1BARR-L27A/L27B>>	U1 27' Turbine Building	U2 27' Turbine Building
YARD	F1AEDG/F0CEDG	<<0BARR-DG004/SB004>>	1A DG TRENCH	SBO CABLE TRAY AREA
	F1AEDG/F0CEDG	<<0BARR-DG203/SB204>>	1A DG FAN ROOM	SBO HVAC EQUIP AREA
	F1AEDG/F0CEDG	<<0BARR-DG204/SB204>>	1A DG MAINT SHOP	SBO HVAC EQUIP AREA

Table 4.3.3b
Barriers in a Control Program

Fire Area	PRA Initiator to PRA Initiator	Barrier ID	From Room Description	To Room Description
11	AUX10D/AUX20A	<<1BARR-112/216>>	12 RC Waste Rec Tank RM	U1 RC M/U Pump RM
		<<1BARR-114/221>>	11 RC Waste Rec Tank RM	U1 West Piping PEN RM
11	AUX20A/FCA319	<<0BARR-200/222>>	East/West Hallway	Hot EQ Shop
		<<2BARR-216A/323>>	U2 RC M/U Pump RM	Valley Alley
11	AUX20A/SHFT	<<0BARR-212/SHFT>>	North/South Passageway by VCT	Equipment Shaft
		<<2BARR-203/SHFT>>	U2 Piping Area	Equipment Shaft
11	AUX10D/FCA319F	<<1BARR-114/323>>	11 RC Waste Rec Tank RM	Valve Alley
		<<2BARR-107/323>>	11 RC Waste Mon Tank RM	Valve Alley
		<<2BARR-109/323>>	12 RC Waste Mon Tank RM	Valve Alley
11	AUX20A/A308	<<0BARR-200/308>>	East/West Hallway	Corridor
		<<1BARR-202/308>>	North/South Passageway by MCC	Corridor
		<<2BARR-224/308>>	U1 Piping Area	Corridor
		<<2BARR-203/308>>	U2 Piping Area	Corridor
11	AUX20A/FCA319F	<<0BARR-200/319>>	East/West Hallway	Corridor
		<<0BARR-200/320>>	East/West Hallway	SFP Cooling Pump RM
		<<0BARR-200/323>>	East/West Hallway	Valve Alley
		<<0BARR-200/327>>	East/West Hallway	SFP Cooling DEMIN
		<<0BARR-200/328>>	East/West Hallway	SFP Cooling FILTER
		<<0BARR-212/319>>	North/South Passageway by VCT	Corridor
		<<2BARR-214/322>>	U2 VCT RM	U2 Letdown Heat EX
		<<1BARR-217/323>>	U1 BAST RM	Valve Alley
		<<1BARR-218/324>>	U1 VCT RM	U1 Letdown Heat EX
		<<1BARR-218/323>>	U1 VCT RM	Valve Alley
		<<1BARR-219/325>>	Vestibule 5' Elevation	Vestibule 27' Elevation
		<<1BARR-220/319>>	U1 Degasifier Pump RM	Corridor
		<<1BARR-220/324>>	U1 Degasifier Pump RM	U1 Letdown Heat EX
		<<2BARR-209/320>>	DECON RM	SFP Cooling Pump RM
		<<2BARR-210/320>>	Hot Machine Shop	SFP Cooling Pump RM
		<<2BARR-213/319>>	U2 Degasifier Pump RM	Corridor
		<<2BARR-214/323>>	U2 VCT RM	Valve Alley
		<<2BARR-215/323>>	U2 BAST RM	Valve Alley
11	FCA319F/A321	<<2BARR-321/322>>	U2 W Piping PEN RM	U2 Letdown Heat Exchanger RM
11	AUX10D/FCA319	<<1BARR-112/323>>	12 RC Waste Rec Tank RM	Valve Alley
16A	FIA301/FIA304	<<1BARR-301/304>>	U1 Battery RM 11	U1 Battery RM 12
17A	FIA305/FIA307	<<1BARR-305/307>>	U2 Battery RM 21	U2 Battery RM 22
11	FCA319/FCA419F(x)	<<0BARR-319/413>>	Corridor	U2 NSSS Sample RM
		<<0BARR-319/419>>	Corridor	CASK Loading & Truck Bay
		<<0BARR-319/424>>	Corridor	U1 NSSS Sample RM
11	AUX10D/A418	<<2BARR-107/418>>	11 RC Waste Mon Tank RM	Solid Waste Proc
		<<2BARR-109/418>>	12 RC Waste Mon Tank RM	Solid Waste Proc
11	FCA319/A412	<<0BARR-320/412>>	SFP Cooling Pump RM	Cask Washdown Pit
11	FCA319/FCA419F(x)	<<0BARR-320/426>>	SFP Cooling Pump RM	Passageway
		<<1BARR-324/424>>	U1 Letdown Heat Exch RM	U1 NSSS Sample RM
11	FIA221/FCA419F(x)	<<1BARR-326/424>>	U1 West Piping PEN RM	U1 NSSS Sample RM
11	FCA319/FCA419F(x)	<<1BARR-327/426>>	SFP Cooling DEMIN	Passageway
		<<1BARR-328/426>>	SFP Cooling FILTER	Passageway
11	FIA420/FIA421	<<1BARR-420/421>>	RC Coolant Waste Evap Tank	1B EDG RM
1A/0C	F1AEDG/F0CEDG	<<0BARR-DG102/SB103>>	1A EDG RM	SBO Switchgear RM
		<<0BARR-DG102/SB104>>	1A EDG RM	SBO Control RM
11	FCA419F(x)/FCA512F(x)	<<0BARR-410/525>>	Passageway	U1 CNMT ACCESS AREA
11	FCA419F(x)/FCA523	<<0BARR-410/523>>	Passageway	North/South Passage
		<<0BARR-410/527>>	Passageway	U2 CNMT Access Area
		<<0BARR-410/530>>	Passageway	CASK Handling Area
		<<0BARR-426/530>>	Passageway	CASK Handling Area
		<<0BARR-419/531>>	Cask Loading & Truck Bay	Waste Proc Area FAN
		<<0BARR-419/533>>	Cask Loading & Truck Bay	New Fuel Storage Area
		<<1BARR-520/523>>	SFP Area HVAC Equip RM	North/South Passage
		<<1BARR-520/591>>	SFP Area HVAC Equip RM	Anti-C Dress Out

4.3.3.2 Barrier Challenge Frequency Determination (I)

For non-fire modeled compartments, the compartment ignition frequency is multiplied by a severity factor which is based on the EPRI Fire PRA Implementation Guide. The Implementation Guide uses a severity factor to estimate the likelihood that a fire damages other components than that of the ignition source component. This same factor is used in this analysis to estimate the likelihood that a fire will be severe enough to challenge a barrier. With the exception of diesel fuel oil components, a Fire Severity Factor of 0.20 is used to bound the severity impact of all plant compartments. For compartments containing diesel fuel, such as the Emergency Diesel Generator Rooms, a Severity Factor of 0.40 is used in the calculation of the barrier challenge frequency.

For those compartments that have been modeled, the analyzed ignition frequency of fire scenarios which result in the lost of the compartment is used as the barrier challenge frequency. In those instances where no lost of compartment fire scenarios are identified, a frequency of zero is used.

Some compartments are fire modeled for the sole purpose of identifying whether a severe fire which would challenge a barrier is possible. In addition to the fire modeling techniques described in Section 4.3.4, this fire modeling considers:

1. Ignition sources to determine if the compartment has ignition sources which could challenge a barrier.
2. Combustion concentration to determine if the compartment configuration is such that a fire could occur in a location that would challenge the barrier.
3. Fire Brigade response to determine if timely response to manually suppress the fire prior to barrier challenge is likely, given failure of automatic suppression.

The following compartments listed in Table 4.3.3c are modeled for barrier assessment and are determined to have a low fire hazard likelihood, such that a fire is unlikely to propagate into adjacent areas. These non-Appendix R barriers are assumed to not be challenged by a fire in the compartment. Regardless of a barrier failure probability for these compartment, any fire would not be sufficiently severe so as to propagate to an adjacent compartment. For these compartments the barrier challenge frequency is zero.

Table 4.3.3c
Rooms Fire Modeled for Barrier Effectiveness

Fire Area	PRA Initiator	Compartments
11	AUX10D(1)	A107, A109
11	AUX10D(2)	A112, A114
11	AUX20A	A200, A202, A203, A207, A212, A213, A215, A216, A216A, A217, A219, A220, A222, A223, A224, SHFT
11	FIA206	A206, A310
11	FIA221	A221, A326
11	FIA227	A227, A316
11	FIA309	A309
11	FCA319	A319, A320, A322, A323, A324, A325, A327, A328
11	FIA418	A418
11	FCA523	A523, A526, A527, A530, A531, A533, A586, A587, A588, A589, A590, A591, A594, A595, A596
16A	FIA301	A301
16A	FIA304	A304
17A	FIA305	A305
17A	FIA307	A307
30	FIA420	A420
41	FIA536	A536, A537
YARD	F1AEDG	DG1A
	F0CEDG	DG0C, ARF
Turbine	TBALLB	Turbine

4.3.3.3 Compartment Barrier Failure Probability Determination

Failure rates are calculated for barriers containing penetrations by using the probabilities contained in the EPRI PRA Implementation Guide for fire dampers, fire rated hatches and electrical and mechanical penetrations, including fire rated cable tray and conduit seals, and piping and sleeve seals. Only barriers that are in an inspection and/or control program are evaluated. All other barriers are assumed to be ineffective.

Barrier Attribute	Failure Probability
Doors	7.4E-3
Dampers	2.7E-3
Walls	1.2E-3
Penetration Seals	1.2E-3

The equation below is utilized to probabilistically calculate the contribution of penetrations to the barrier failure rate.

$$B_{\text{Barrier Failure}} = 1 - [(1 - B_{\text{Wall}})(1 - B_{\text{Door}})^{\#\text{ Doors}}(1 - B_{\text{Damper}})^{\#\text{ Dampers}}(1 - B_{\text{Penetrations}})^{\#\text{ Penetrations}}]$$

4.3.3.4 Automatic Suppression Systems Failure Probability Determination (N).

The actuation of a properly designed and installed automatic fire suppression system is assumed to effectively mitigate a potential challenge to a fire barrier. FIVE provides nominal unavailabilities for various automatic fire suppression system types that are used in CCFPRA. See Section 4.5 for these values.

The suppression failure rate is determined for each compartment of origin. The suppression failure rate of the compartment which the fire is spreading to is assumed to be unavailable. Although conservative, this assumption avoids the evaluation of common failure modes between the compartments' fire suppression systems.

Halon suppression requires effective isolation, which would also need to be evaluated. However Halon was not credited in the propagation analysis. All the plant compartments that have a halon suppression systems are fire modeled and none were found to have scenarios which result in the loss of the compartment. Therefore the barrier challenge frequency for these compartments is set to zero.

4.3.4 Detailed Fire Modeling

Compartments which exhibited unacceptable conditional CDFs when fully burned or those which have a large system functional loss potential are evaluated to determine targets and fire sources. This evaluation results in the development of credible fire scenarios.

The FIVE methodology was used to estimate the environmental conditions that could develop at a target as a result of a specified exposure fire scenario. Temperature and heat flux are used. If the estimated maximum environmental condition does not exceed the damage threshold of a target, then the target is screened. If the estimated maximum environmental condition is exceeded, then the appropriate functional impact is determined and the impacts of each scenario are coded into the CCFPRA.

This approach, consistent with the guidance in the EPRI Fire PRA Implementation Guide (Ref. 4-3), involves the following steps:

Step 1: Ignition Source Evaluation
Step 2: Detailed Fire Modeling of Fixed Ignition Sources
Step 3: Detailed Fire Modeling of Transient Ignition Sources
Step 4: Assessment of Fire Suppression
Step 5: Assessment of Compartment Fire-Induced Core Damage Frequency
Step 6: Refinement of Fire Modeling (as necessary)

Each of these steps are discussed in the subsequent sections.

Compartment walkdowns are not addressed as a separate step as they are an iterative process, apply to more than one step, and depend upon the configuration and contents of the room. Walkdowns are generally performed for evaluation of ignition sources (i.e., separate fire scenarios); evaluation of spacial considerations for fire damage modeling; and verification of final fire damage modeling assumptions and inputs. Walkdowns are also performed to calculate the compartment fire ignition frequency total and to identify compartment boundaries (e.g., fire doors, fire dampers); however, these issues are used as inputs to the fire damage modeling process. See Section 4.2.3.

4.3.4.1 Step 1: Ignition Source Evaluation

The first step in the fire damage modeling process is to review (typically via a walkdown), the ignition sources in the compartment to dismiss ignition sources that do not impact the plant and do not propagate. This step also includes identifying ignition source characteristics, such as oil-filled pumps, open top electrical cabinets, and spacial distances to critical targets, for further analysis in the fire damage modeling. Ignition sources that cannot damage other equipment and do not impact equipment modeled in the PRA can be dismissed. Such ignition sources are typically fixed. In some cases, transient sources can be dismissed because they are determined not to be realistically representative of the compartment during power operation. Examples of excluded ignition sources are battery-operated emergency lights, fluorescent light fixtures, and fire protection panels.

Given an electrical cabinet fire, the functions of the cabinet are assumed disabled. However, sealed electrical cabinets and panels are also possible to dismiss from further analysis if they are separated from other cabinets by an air gap and failure of the equipment inside the cabinet does not impact equipment modeled in the PRA. Electrical cabinets which will not impact surrounding equipment given a postulated cabinet fire, but which contain PRA functions are retained as fire scenarios and are functionally evaluated.

Based on guidance in the EPRI Fire PRA Implementation Guide, which references cabinet fire experiments conducted by Sandia National Laboratories, sealed electrical cabinets that are separated from other electrical cabinets by an air gap can be assumed to result in damage only to the cabinet in question and it is assumed that no damage occurs to other cabinets. The size of an "air gap" is not defined in the Fire Implementation Guide. The definition used here is that no metal-to-metal contact exists that would support sufficient heat conduction between cabinets, and is typically considered to be eight millimeters of air. An exception to this empirically-based rule is that the immediately adjacent cabinets do not contain sensitive equipment. "Sensitive equipment" is defined as solid state electronic equipment.

The primary targets are typically horizontal cable trays overhead of ignition sources. Targets may also be vertical cable trays or other equipment located radially from an ignition source. During an initial walkdown of a compartment, it is resource efficient to identify the nominal distances from the fire source to the nearest target in either direction, such as two feet to the nearest horizontal cable tray, or three feet radially to the nearest vertical cable tray. If subsequent fire damage modeling, as discussed in Steps 2 and 3, show that the configuration for a particular postulated fire scenario does not result in target damage, distances to additional targets are inconsequential. However, if fire damage modeling results in damage to the nearest targets, drawings (i.e., equipment location drawings, cable tray drawings) and additional walkdowns are used to find distances to additional targets, as necessary.

4.3.4.1.1 Ignition Source Frequencies

Through a review of the compartment ignition frequency evaluation (see Section 4.3.2), it is typically straightforward to breakdown the total compartment ignition frequency by individual sources. For example, if the compartment ignition frequency tabulation shows a contribution of 8E-03/yr for electrical cabinets and the room contains 20 electrical cabinets, the ignition frequency is apportioned to the individual cabinet as 8.00E-03/20 = 4.00E-04/yr. Resolution in the industry data analysis for ignition frequencies does not afford a reasonable basis to assign different ignition frequencies to like components in a compartment based on distinguishing characteristics between one like compartment and another.

Simply stated, all electrical cabinets in the compartment will result in equivalent ignition frequencies, all pumps in the compartment will result in equivalent frequencies, and so on. This equal weighting approach is deviated from in the Control Room evaluation. Since the Control Room panels has significant size variations, a size weighting factor is used. See Attachment 4-I.

Although transient ignition frequencies are also identified in the compartment frequency analysis, the development of specific transient fire scenarios frequencies is described in Step 3, Section 4.3.4.3.

4.3.4.2 Step 2: Detailed Fire Modeling of Fixed Ignition Sources

The detailed fire modeling for fixed ignition sources uses inputs from the previous step to analyze unique fire scenarios, including identification of specific fixed ignition sources, associated characteristics, and distances to targets.

4.3.4.2.1 Fire Damage Worksheets

The fire damage modeling uses worksheets as provided in the FIVE methodology. The results derived from these worksheets is the critical distance from the ignition source for which damage to targets will occur. Targets outside this distance are not damaged by the postulated fire.

- Target In-Plume: This worksheet is used to assess damage for targets located overhead of the fire source in the fire plume. The geometry of the fire plume is essentially a cone with a radius as a function of height that can be estimated by $r = 0.2z$, where z is the height of the target above the fire source. These cases result in the most severe damage conditions.
- Target Outside-Plume: This worksheet is used to assess target damage for targets located overhead of the fire source but outside the cone of the fire plume. This worksheet addresses target damage due to potential ceiling jet and hot gas layer (HGL). Such cases are less severe than the in-plume case. If the in-plume case does not result in damage to a particular target, then similar targets at the same elevation or higher and located outside the plume will also not be damaged.
- Radiant Exposure: This worksheet is used to assess target damage for targets located radially from the fire source. These cases typically result in the least severe conditions.

The worksheets require the input of the following information:

- geometry of the room (length, width, height)
- distance from source to target
- general location of the fire source (in corner, against wall, in center of room)
- damage threshold of target
- heat release rate of fire source
- radiant/convective heat release fraction
- heat loss fraction

The first three items are obtained from drawings and walkdowns. The remaining inputs are determined as follows.

4.3.4.2.2 Damage Thresholds

Damage thresholds are based on empirical evidence and are summarized in the FIVE methodology and the EPRI Implementation Guide for various targets. With respect to IEEE 383 cables, the recommended damage thresholds for in-plume and outside-plume targets are 700 degrees Fahrenheit for cable failure and 932 degrees Fahrenheit for cable ignition. Cable ignition creates a situation where a cable tray fire now exists and fire propagation to additional trays overhead must be treated. With respect to the radiant heat calculations, the damage threshold is expressed in units of Btu/sec/ft^2. The FIVE methodology and the EPRI Implementation Guide both suggest that a value of 1 Btu/sec/ft^2 is a representative value for IEEE 383 cables.

4.3.4.2.3 Heat Release Rate

The heat release rate of the fire source is a widely varying parameter. Industry tests, summarized in the FIVE methodology and the EPRI Implementation Guide, have been performed to aid in the selection of suitable heat release rate for various fire sources. Given the variability of heat release rate as a function of fuel package geometry, fuel type, test conditions and fire duration. Judgment must be used to allow consistent selection of heat release rates for similar sources and configurations. A heat release rate of 65 Btu/sec is used for electrical cabinets containing IEEE 383 cable or equivalent. For cabinets containing non rated cable, 8.5E-4 times the Btu loading in the cabinet is used. The EPRI Implementation Guide provides the following recommendations for a number of fire source configurations, with the following being the most common configurations:

- Vented electrical cabinet (65 Btu/sec)
- Open top electrical cabinet (8.50E-04 x Btu loading in the cabinet)
- Motor windings (< 65 Btu/sec)
- Motor oil spill (135 Btu/sec/ft^2); an equation for estimating the area of the spill area is also provided in the FIVE methodology

Some situations may arise in which judgment is necessary to select an appropriate heat release rate. Such judgment are made using experience, common sense and consideration of insights in the EPRI Implementation Guide. Compartment specific application of heat release rates is addressed in the attachments to Section 4.

4.3.4.2.4 Heat Release Fraction

The heat release fraction is a measure of the fraction of heat generated by the postulated fire that is released via convection and that released via radiation. The FIVE methodology and the EPRI Implementation Guide suggests a conservative approach in which the convective fraction is 80 percent and the radioactive fraction is 40 percent. The 0.80 convective heat release fraction is employed in the In-Plume and Outside-Plume worksheets. The 0.40 radioactive heat release fraction is employed in the Radiant Exposure worksheets.

4.3.4.2.5 Heat Loss Fraction

The heat loss fraction is a measure of the heat lost to the compartment boundaries and other heat sinks in the room. The FIVE methodology suggests that a heat loss fraction of 0.7 is a conservative value. The EPRI Implementation Guide recognizes that 0.7 is a conservative value and that 0.85 may be more realistic for many configurations. The value of 0.85 is used in CCFPRA (given that most compartments have concrete or block walls which act as better heat sinks).

4.3.4.2.6 Pre-Calculated Critical Distances

Worksheets are completed for general cases to pre-calculate critical distances. As an example, a compartment may contain numerous electrical cabinets with louvers rising up the back to a height of seven feet. This case would be analyzed using the In-Plume and the Radiant Exposure worksheets to determine the critical distances around such a fixed ignition source, such as damage to cables located three feet overhead, or damage to equipment located one foot radially. This information is then used to identify the configurations in the compartment that fall within these distances; fixed fire ignition and target scenarios that fall outside these pre-calculated distances are categorized as fire scenarios involving damage to just the fixed component itself. Fixed ignition and target scenarios that fall inside these pre-calculated distances are then inspected to identify the need to perform Outside-Plume cases.

If fire damage modeling results in damage to these nearest targets, equipment location, cable trays and conduit drawings, are used to find distances to additional targets, as necessary. Using the $r = 0.2z$ equation for the cone of a fire plume, a few concentric circles are drawn over the key fixed ignition sources to show the zone of damage.

4.3.4.2.7 Results

The result of this step is a list of the fixed ignition sources and their associated fire-induced damage states. This information is presented in a table that contains a comprehensive list of the fire ignition sources in a compartment. The table summarizes ignition sources, ignition source frequencies, adjacent cabinet and cable tray considerations, damage states and associated comments necessary to describe modeling details or assumptions. These fire scenarios for each fire modeled compartment is included in the attachments to Section 4.

4.3.4.3 Step 3: Detailed Fire Modeling of Transient Ignition Sources

Operating experience, as indicated in the EPRI FEDB, reveals that transient fires typically involve small amounts of combustibles. In two industry events, the only items involved in the fire incident were an extension cord. In another event, a small muslin cloth used to collect radiation samples was accidentally ignited and burned. With respect to hot work induced fires, the majority of the events involved ignition of transient combustibles, only a few incidents involved sparks impacting cable insulation.

Given the variability of transient related fires, the FIVE methodology categorizes transients into the following categories:

- Hot work induced cable fires
- Hot work induced transient fires
- Other transient induced fires

Hot work induced cable fires should be dismissed as a fire damage scenario for the following two reasons: 1) The number of hot work induced cable fires is very small and the incidents are minor in nature, 2) The plant is equipped with IEEE 383 cables which does not support fire propagation. This approach is consistent with NSAC-181, Fire PRA Requantification Studies. Review of the EPRI FEDB indicates that transient combustible fires are almost always ignited by transient ignition sources; fixed ignition sources do not play a significant role in the creation of transient combustible fires.

4.3.4.3.1 Characterization of Transient Fire

The appropriate approach to modeling transient fires is to perform the fire damage modeling assuming the fire is a transient fuel package that may be located anywhere in the compartment, consistent with the EPRI Implementation Guide. The EPRI Implementation Guide provides test data on the heat release rate for a variety of fuel packages, including human occupancy refuse, maintenance refuse, and radiation protection clothing. Selection of the fuel package that represents a generic transient fire in a given compartment requires judgment by the analyst. Based on a review of the transient fuel test summarized in the EPRI Implementation Guide, a representative fire in most areas of the plant can be best characterized as a maintenance refuse package of 100 Btu/sec with a fire duration of not greater than 15 minutes. Review of housekeeping procedures, in conjunction with walkdowns and interviews of fire protection personnel are used to reasonably characterize unique fuel packages to individual fire compartments.

Probabilistically, the fire may be located in any open area on the compartment floor. Consistent with the EPRI Implementation Guide, the conservative approach is to construct various transient fire scenarios impacting each PRA modeled component.

Once the locations are determined, the In-Plume, Outside-Plume, and Radiant Exposure worksheets are used to determine the critical damage distances. The height of the fire above the floor is a parameter required for completion of the In-Plume and Outside-Plume worksheets. Typically, a representative fire is determined to be a trash can fire, three feet from the floor. Such a fire is postulated to be more conservative than the example fire in the FIVE methodology, which places the fire on the floor, due to the closer proximity to targets, such as cable. Distance and location of transient fires, however, is largely determined on a compartment by compartment basis.

4.3.4.3.2 Frequency of a Transient Fire

Once the critical distances are calculated and the associated worst-case equipment damage state is determined, the next step is to determine the frequency of the transient fire scenario. The previously calculated transient ignition frequency is based on the FEDB and merely indicates that a transient ignition of some type may be located in the compartment. The frequency does not indicate that a fuel package is involved in a fire or that such a postulated fire is even near critical equipment. The FIVE methodology recommends a general formula for estimating the fire induced core damage frequency due to transient combustible fires. A modified version of this formula (with the Conditional Core Damage Probability term removed) is used to calculate the ignition frequency of a transient fire.

$$F_t = F_{it} * u * P_{fs}$$

where: F_t = transient fire scenario frequency (Note that FIVE uses this term to represent the induced core damage frequency for the compartment. This is calculated separately in the Fire CCPRA)

F_{it} = transient ignition frequency for the compartment

u = probability that transient combustible located in the range of target components

P_{fs} = probability of fire suppression failure given that the system would actuate prior to target damage

4.3.4.3.3 Transient Combustible in Range of Targets (*u*)

This parameter assesses the probability that given the modeled transient combustible fuel package exists in the compartment, the transient combustible fire will occur near targets in that compartment. Per the FIVE methodology, this parameter is estimated as a ratio of exposure area to total floor area, as follows:

$$u = (A_s + A_{sr})/A_F$$

where: A_s = Exposed surface area of cable trays overhead

A_{sr} = Floor area around targets within the critical radial separation distance determined from Radiant Exposure worksheet.

A_F = Floor Area where combustibles could be placed.

As with many of these parameters in the fire damage modeling analysis, estimation of the *u* parameter requires some judgment. For example, if the fire damage worksheets indicate that no overhead cable tray is close enough to the floor to be impacted by the modeled transient fire, then the contribution from A_s in the calculation of *u* is set to 0.0. In the determination of the A_{sr} contribution, the question arises whether to calculate the area with respect to the particular location in the compartment designated as the worst-case or to calculate the area with respect to all the critical targets in the compartment. If there is only a single worst-case spot in the compartment, then the appropriate approach may be to determine A_{sr} with respect to this single location. If there are a number of potential worst-case locations in the compartment and it requires a judgment call or assignment of a representative damage state to characterize the worst-case damage, then it would be appropriate to calculate A_{sr} with respect to more than one (or potentially all) target in the compartment.

The floor area in the denominator should be that floor area where it is possible for a transient combustible fuel package to be located (i.e., the area of floor mounted equipment should be subtracted from the entire floor area - this may be a best guess percentage estimate).

4.3.4.4 Step 4: Assessment of Fire Suppression

The FIVE methodology and the EPRI Implementation Guide both provide guidance on the treatment of fire suppression. These issues may be addressed in initial fire damage modeling analyses or may be deferred until after the core damage frequency quantification runs to determine if consideration of fire suppression is even warranted (see Step 6).

Fire suppression is divided into the following general categories:

1. Installed automatic suppression systems
2. Installed manually actuated suppression
3. Fire brigade response (including Control Room operators and Hot Work Firewatches)

4.3.4.4.1 Automatic Suppression Systems

Before automatic suppression is credited, a determination of whether the suppression system will actuate prior to target damage is made. This is performed by using the Thermally Thick and Thin Target worksheet supplied in the FIVE methodology. These worksheets assess whether target damage will occur prior to the detectors (e.g., fusible links in sprinkler heads) actuating. The completion of these worksheets require identification of the spacial geometry of the detector with respect to the fire source and target, and the actuation temperature of the detector. If the worksheets show that the detectors will actuate prior to target damage, then automatic suppression can be credited. Another consideration regarding actuation delay applies to a Halon system. Halon systems are designed with a built-in actuation delay following detection. Note that localized suppression zones in a compartment indicate that automatic fire suppression can only be credited for the individual fire scenarios postulated in that localized suppression zone.

If detection and actuation is determined to occur prior to target damage, the next consideration is the failure probability of the suppression system. The FIVE methodology provides generic failure rates for suppression system types. If multiple systems exist then the failure probabilities may be multiplied together. These is no case where such a multiplication is done in CCFPRA.

The final issue regarding automatic suppression is fire suppression induced equipment failure. This issue is discussed separately at the end of this step.

4.3.4.4.2 Manually Actuated Suppression Systems

The discussion above for automatic suppression systems applies to manually actuated suppression systems with one major exception - manual response time versus time available prior to target damage must be considered. The manual response time is the time between fire detection to target damage. For example, if an installed detector will indicate a fire in 20 seconds following fire initiation and the time to target damage is 60 seconds following fire initiation, the time available for manual response is 40 seconds.

Typically, manually actuated systems are located in very few areas and do not enter into fire damage modeling analyses. In addition, the available time for manual response for most postulated fires is on the order of seconds to minutes, such that the human error probability for failure to actuate the system before target damage approaches 1.0. No credit for manual suppression systems is taken in CCFPRA.

4.3.4.4.3 Fire Brigade Response

Issues regarding fire brigade responses are similar to that of manually actuated suppression systems. The allowable response times before target damage occurs are comparatively small. However, the EPRI Implementation Guide does provide guidance on estimate damage times in certain cases such as:

- damage resulting from fire propagation from one cable tray to the next
- damage resulting from a fire in one cabinet impacting sensitive equipment in an adjacent cabinet

As in the manually actuated suppression system case, the prudent approach is to not credit fire brigade response in the initial fire damage modeling, but to defer such consideration until after core damage frequency quantification runs to determine if consideration of fire brigade response is warranted.

The fire brigade response to a Control Room fire is included in the CCPRA. See Section 4.6.1 Top Event VL. In addition, qualitative assessment of the fire brigade is included as part of the cross-zone analysis. See Section 4.3.3.2.

4.3.4.4.4 Suppression Induced Equipment Damage

The FIVE methodology and the EPRI Implementation Guide do not treat the issue of suppression induced equipment damage probabilistically. Response to the IPEEE Program regarding this issue is addressed deterministically under the Fire Risk Scoping Study issues of the internal fires facet of the IPEEE. The issue is treated by walkdowns and plant design reviews to determine if configurations exist anywhere in the plant where actuation of a suppression system could simultaneously disable redundant safe shutdown trains. See Section 4.9.2, Generic Issue-57, "Effects of Fire Protection System Actuation on Safety-Related Equipment."

4.3.4.5 Step 5: Fire-Induced Core Damage Frequency

The previous four steps represent the fire damage modeling portion of the assessment of fire induced plant risk. The results of the first four steps are a list of individual fire scenarios. These fire scenarios are characterized by the following:

- scenario frequency (e.g., frequency of Pump X fire, frequency of generic transient fire)
- associated fire-induced damage (e.g., Pump X and cable tray ABC123 disabled)

This information is used in the accident sequence analysis (i.e., core damage event tree modeling and quantification) to determine the associated core damage accident sequence frequency given the postulated fire-induced damage. See Section 4.6.

The core damage sequence frequencies of all the postulated fire scenarios of a given compartment are summed to determine the compartment fire-induced core damage frequency.

4.3.4.6 Step 6: Refinement of Fire Modeling

The approach to fire damage modeling is generally performed with suitably (not excessively) conservative assumptions and parameters. As such, initial fire damage modeling will result in certain compartments that require additional analysis following accident sequence quantification.

Options to consider in the refinement of fire modeling include the following:

- Determine more realistic Heat Release Rates
- Credit fire suppression and/or fire brigades
- Explicitly credit time delays in fire propagation among cable trays
- Determine more realistic fire-induced equipment damage states
- Relax conservatisms in transient combustible fire scenario frequency.

4.4 Evaluation of Component Fragilities and Failure Modes

For the majority of the compartments, a fire which consumes the entire compartment is assumed. Under these conditions, the equipment, cables and human actions within the compartment are evaluated for functional impact.

For compartments where specific fire scenarios are developed, the specific ignition sources and key targets are evaluated. Again, the equipment, cables and human actions within the bounds of the scenario are evaluated for functional impact

FIVE provides a means to make conservative estimates about conditions that could develop to a target as a result of a specified fire. These conditions are then compared with the target damage threshold criteria and if the criteria are not exceeded, the specified target is assumed to have no direct fire impact.

4.4.1 Cable Analysis

The fire related vulnerabilities for the functions credited in the CCFPRA are obtained by examining the electrical diagrams associated with the base CCPRA model's basic events. The base CCPRA model is very comprehensive and includes the explicit treatment of key electric relays and logic devices. The fire related impacts are obtained by evaluating the associated cables with each of the basic events. This required the integration of the existing Appendix R cabling data and supplemental cable routing research for "non-Appendix R" components. The plant circuit and raceway database (CRS) is used to obtain raceway routing information for the cables identified through this process. Each raceway containing at least one cable associated with a PRA component failure (basic event) is identified and spatially located in the plant using the fire compartment designators.

It should be noted that CRS only supplies the trays and conduits associated with each cable. A query was developed to spatially locate the cable by locating the trays and conduits. This process is conservative in nature because a cable may be in a tray for a short distance before it jumps into another tray. If a tray actually went through more compartments, then these fire compartments are identified as having a cable of concern even though the cable is not in them. This conservative cable to compartment identification

process likely results in more top events impacted in many postulated compartment fires. Although this is conservative, the resources and time necessary to identify each cable route was not available.
The cable failure modes which are considered in this evaluation are open circuits, short circuits and hot shorts. An open circuit condition could result in component failure due to loss of motive power or critical signal failure. A short circuit failure mode could also result in component failure due to loss of motive power or critical signal failure. A hot short failure mode involves the cross energizing of conductors within a cable (conductor to conductor). Cable to cable failures are not considered in this analysis. A conductor to conductor hot short has the potential to cause undesired spurious actuation. The open, short and hot short failure modes are conservatively treated with a conditional failure probability of 1.0. If multiple failure modes of cables due to a single fire can occur, the most severe functional impact is assessed.

The resulting data provides a link between cables associated with PRA equipment and the base CCPRA model basic event. An existing relationship provides the association between basic event identifiers and model top events. The functional impacts which result from a postulated fire in a compartment can then be determined.

In most cases the impacted top events are further evaluated to determine whether the top event failed or is degraded as a result of the cable impact. This evaluation is used as input to modify the base CCPRA to reflect the various fire scenarios.

4.5 Fire Detection and Suppression

The consequences of a postulated fire event may be minimized if fire suppression activities are available and effective. The evaluation of fire detection and suppression effectiveness was performed on a case by case bases. See Section 4.3, Fire Growth and Propagation.

The reliability values for automatic suppression systems provided in the EPRI FIVE methodology are used in CCFPRA.

System Type	Unavailability of System
Wet Pipe Sprinkler	2.0E-2
Preaction Sprinkler	5.0E-2
Deluge Sprinkler	5.0E-2
CO_2	4.0E-2
Halon	5.0E-2

The EPRI FIVE methodology also provides for the crediting of manually actuated suppression systems and fire brigade response. The crediting of these systems requires that a timing analysis be performed to verify that fire suppression would occur prior to cable or equipment damage. The CCFPRA only credited manual fire suppression actions in support of the Control Room analysis. This is discussed in Section 4.6.2.

4.6 Analysis of Plant Systems, Sequences and Plant Response

The quantification of fire-induced core damage frequency (CDF) is obtained by propagating fire-induced failures through a modified version of CCNPP's PRA (CCPRA). This modified version is constructed from the General Transient module of CCNPP's updated internal events Level 1 PRA. The resulting model is referred to as the "CCFPRA." The term plant model is used below and refers to a set of logic rules developed using PLG's RISKMAN Workstation Software (Ref 4-19) which represent CCNPP Unit 1.

Each fire scenario (or initiating event) has a specific impact on the functions (or top events) contained in the plant model. Some of these impacts are complicated enough to warrant unique functions not contained in the CCPRA. Additional functions unique to the Fire CCPRA are described in Section 4.6.1.

The Fire CCPRA unlike the Seismic CCPRA has only a few new functions added. Although few new functions are added, many new boundary conditions for those functions (or split fractions) are required. These additional split fractions are described in Section 4.6.2.

One of the largest contributors of risk due to a fire is the effect of the fire on human action success likelihoods. A human action's likelihood of success varies depending on the location of the fire and the path the operator must travel to complete the action (See Section 4.6.3). For each of these areas, new human action failure probabilities are developed:

- Auxiliary Building
- Turbine Building
- Intake Structure
- Outside & 0C EDG Building (due to similar impacts these are combined)
- Control Room

Based on the human actions value changes, new split fraction values are developed for each of the fire areas listed above. Depending on the point of fire initiation, the appropriate split fraction is questioned.

Each fire scenario is evaluated to determine the extent of the fire impact on the plant model. The scenario is then assigned an initiating event designator and a frequency. This is described in Section 4.6.4.

The incorporation of all of this information into the CCFPRA generates a model considerably larger than the base CCPRA model. The base CCPRA model has five linked events trees consisting of 1377 assignment statements (rules). The CCFPRA model has eight linked events trees consisting of 1642 assignment statements for the Control Room related scenarios, and eight linked event trees consisting of 1588 assignment statements for the non-Control Room related scenarios.

The larger size of the CCFPRA also forces a different approach to the handling of sequence truncation limits. The CCFPRA results, as with the IPE results, are post-sequence processed to reduce truncation errors. All success split fractions not directly related to the likelihood of core damage or the determination of a Plant Damage State (PDS) are eliminated. This makes the sequences similar to cutsets. Given this approach, one of the key factors in determining a truncation limit is the amount of sequences associated with an initiating event. Most fire initiating events have at least 100 sequences.

Those that have less than 100 sequences have an unaccounted frequency less than 1E-7. All initiating events produce at least one sequence regardless of the unaccounted frequency limit. It should be noted that the CCFPRA has well over 180,000 sequences for the 177 initiating events contained within the plant model. See Table 4.6a for a listing of the unaccounted frequencies and sequence count for each initiating event.

4.6.1 Fire Top Events

The new top events developed for the CCFPRA all relate to the presents of smoke in the control room. The amount of smoke in the control room can have these effects:

- None - The level of smoke is barely perceptible and only mildly degrades operator performance.
- Noticeable - The level of smoke will degrade operator performance, but the control room is habitable.
- Severe - The level of smoke is excessive and the operators are forced to abandon the control room and man the auxiliary (aux) shutdown panel.

These categories in additional to establishing functional impacts due to the evacuation of the Control Room also determine which the Control Room related Human Action Model to use.

Top Event CL - Cable Spreading Room Dampers Close of a CSR Fire.

The cable spreading rooms (CSRs) and Control Room (CR) are connected through a common ventilation system. If the CSR dampers close following a CSR fire, then the level of smoke in the Control Room is considered none. If the dampers fail, the level of smoke is considered noticeable. For most CSR fires, the failure likelihood is driven by hardware failures, but some fires will either disable required cabling or fail a necessary support system thereby guarantying the failure of this function.

Top Event CR - Operator Abandons the Control Room

This top event considers these abandonment scenarios:

- Control Room Panel Fire - The panel fire is not suppressed in a timely fashion. This causes excessive smoke generation and forces the abandonment of the CR.
- A Large Turbine Building Fire - The operators fail to or cannot isolate the CR outside air intakes causing thick smoke to enter the CR forcing abandonment.
- A Outside Transformer Fire - The operators fail to or cannot isolate the CR outside air intakes causing thick smoke to enter the CR forcing abandonment. This scenario differs from the Large Turbine Building Fire scenario in that it is less likely that the smoke from a transformer fire will reach the intake.
-

Control Room Abandonment has both a human action impact and an equipment impact.

Human Action

All human actions which do not contain an AOP-9 compliment are considered failed. The essential impact of this is that the only actions left are those associated with AFW Pumps 11, 12, 13 and AFW flow control.

Equipment Impact

Given the Control Room is abandoned, the Unit 1 operators through implementing AOP-9A are considered to secure this key Unit 1 equipment:

- Steam Generator Feed Pumps 11 & 12
- High Pressure Safety Injection Pumps 11, 12, & 13
- Turbine Building SRW Headers 11 & 12
- Containment Spray Pumps 11 & 12
- Containment Air Coolers 13 & 14
- Battery Charger Feeds 12, 13, 22, and 23
- Off-site power feeds to the 4 KV Buses

The equipment is secured either during abandonment or after manning the auxiliary shutdown panel. The equipment/functions listed above are not recovered in AOP-9A. In addition, to this equipment/function loss, the operators shed almost all electrical loads, and re-power the critical loads. The re-powering of critical loads is addressed in Top Event VB.

Top Event VL - Fire Brigade Suppresses a Control Room Panel Fire following Control Room Evacuation prior to further Control Room Panel Loss

This function models the likelihood that the fire brigade prevents fire spread beyond those panels initially impacted by the Control Room panel fire. Failure of this is assumed to not only result in the failure of Top Event VB, re-loading key equipment, due to spurious equipment operation, but the failure is assume to result in the failure of all containment isolation equipment. The method of the containment isolation equipment failure is determined by examining all other plant model fire scenarios and using the worst case failure modes. For example, both Hydrogen Purge motor operated valves are considered to spuriously open. Although there are AOP-9A actions available to mitigate these failures, quantification of the likelihood of successful implementation is not included in the CCFPRA.

Top Event VB - Operator re-loads Key Equipment Shed as part of AOP-9 following Control Room Abandonment

This function models the likelihood that the operator will re-power all equipment shed in AOP-9 and slated for re-powerment in AOP-9. Failure of this function is assumed to result in the failure of all electrical power. Without electrical power, the batteries will deplete. Failure is likely to result in core damage.

Top Event FW - Operator mans the Auxiliary Shutdown Panel following Control Room Abandonment

Once the Control Room is abandoned, the operators are directed to man the auxiliary shutdown panel. The auxiliary shutdown panel is consider successfully manned when the operators have control of the turbine driven AFW Pumps. Failure to man the auxiliary shutdown panel causes the failure of the AFW Pumps which results in core damage.

4.6.2 Fire Split Fractions

Although most fire scenarios have an equipment impact similar to a support system failure, some fire scenarios cause unique impacts which require new failure likelihoods while other can be successfully modeled using existing split fractions.

For example, there is a function in the plant model which requires the Control Room HVAC to provide adequate ventilation to a common header (Top Event HH). This top event models the functionality of both CR HVAC Unit 11 and CR HVAC Unit 12. If a fire fails Unit 11 it is equivalent to a loss of power to a single Unit. In this case a new split fraction is not required as the power loss split fraction which already exist is selected. Some impacts are not as straightforward, those impacts are listed below:

Fire Faults Electrical Equipment

A fire induced equipment failure is assumed to cause any electrical equipment to fault. If the associated equipment's breaker fails to open, then the bus which feeds that equipment is assumed to fault as well. Although these types of failures are considered in the internal flooding analysis, these failures are not in the base general transient model upon which the fire plant model is built. This effect is consider for all 13 KV, 4 KV, and 480 VAC equipment. Additionally, the 13 KV and 4 KV breakers are powered from 125 VDC. If this DC power source is not available, then the entire electrical facility (either A or B, but not both) is considered failed.

EDG Mission Time

The EDG mission times vary based on the perceived difficult of recovery of NSR power to the 4 KV buses. The three EDG mission times are:

- 24 hour Mission: This mission time is used when either a Switchyard Fire occurs, or the operators are forced to evacuate the Control Room. When the Control Room is evacuated the operators purposely remove off-site power from the 4 KV Buses.
- 4 hour Mission: This mission time is used when one or more of the U-4000 transformers is unavailable as a result of the fire.
- 2 hour Mission: This mission time is used when all of the U-4000 transformers are available.

This contrasts with the internal events analysis which has EDG mission times of 1, 2, 4, 11, and 24 hour based on the duration of the Loss-of-Off-site Power Event.

Switchgear (SWGR) Room Ventilation Damper Recovery

The SWGR room dampers must remain open to ensure adequate ventilation to the SWGR rooms (A311, A317, A407, and A430). In the case of a fire in the SWGR rooms, the dampers by design will close. Most fires in the SWGR rooms are considered recoverable. The exception is those fires involving oil cooled transformers. Due to the PCBs within the oil, the operators will not re-establish ventilation following these fires.

Another challenge to the SWGR room dampers, occurs as a result of outside fires. A large turbine building fire or a large outside transformer fire could result in smoke migration through the ventilation intakes of the SWGR rooms. If operators action is not take to recover or prevent this, then SWGR room ventilation will be lost.

Control Room Damper Isolation Prior to Smoke Entering the Control Room

In the event smoke reaches the CR. Inlet Dampers, the operators must isolation the dampers prior to smoke entering the Control Room in such quantities as to reduce the operators likelihood of successfully performing human actions. This action is questioned following a large turbine building fire, and a large outside transformer fire. Failure to perform this action is assumed to result in CR evacuation.

Cable Spreading Room Damper Isolation following CSR Fire

In the event of a CSR fire, the CSR Inlet and Outlet Dampers are designed to isolate. If the dampers do not isolate, then smoke will reach the Control Room. Smoke in the Control Room increases the failure rate of most human actions significantly.

Operator Isolate PORV using the PORV Feeder Breakers (per AOP-9)

A fire on 1C06 or Control Room evacuation prevents the use of the normal PORV isolation techniques. These events prevent the operator's use of the PORV override or the PORV block MOVs. When this occurs the operator is forced to isolate the PORVs using the PORV Feeder Breakers. As the PORVs fail closed on loss of power, this is an alternative method which is credited in AOP-9. Even with the success of this human action, there is still a noticeable likelihood that the PORVs will fail to isolate following relief.

Operator Locally Trips RCPs (per AOP-9)

Some Control Room fires prevent the operators from remotely securing the RCPs. When this occurs the operators must locally secure the RCPs per AOP-9. Even following the success of this action, there is still some likelihood that the RCP seal will fail.

Auxiliary Shutdown Panel Human Actions

If the operator evacuates the Control Room or if key Control Room functions are lost, then the operator must implement certain actions from the auxiliary shutdown panel including:

- Manually aligning AFW Pump 13
- Manually aligning the Turbine Driven AFW Pump Room Emergency Ventilation
- Manually controlling the AFW flow rate
- Manually starting the Turbine Driven AFW Pumps
- Manually opening the Turbine Driven AFW Pump Steam Admission Valves
- Manually aligning a long term AFW Water Supply

Although these actions already exist in the general transient plant model, when the operator mans the auxiliary shutdown panel the likelihood of human action failure increases significantly.

HPSI Loop Header MOVs

The four HPSI LOOP motor operated valves (MOVs) associated with the main header are all powered from MCC 104R. The four HPSI LOOP MOVs associated with the auxiliary header are all powered from MCC 114R. In the general transient model, split fractions already exist to account for the failure of four MOV associated with either the main or auxiliary headers. But in certain fire scenarios only two-of-four MOVs associated with a header may fail. To support this condition, new split fractions are developed.

Hydrogen Purge MOVs and other Containment Isolation Equipment

The failure of the hydrogen purge MOVs will cause a breach in containment integrity. Since these valves are not normally opened, even if power were to be lost in a sequence which results in core damage, there is only a small chance that containment integrity would be lost. But some fire scenarios can cause the MOVs to spuriously open. As a result, new split fraction were required.

4.6.3 Evaluation of Human Recovery Actions

The CCFPRA, uses a hybrid human action reliability analysis method which takes into account the best features of the SLIM-MAUD and HCR (Human Cognitive Reliability) models.

The base CCPRA Human Action analysis uses the SLIM-MAUD methodology's Success Likelihood Index (SLI) and Performance Shaping Factors (PSFs) and uses the HCR model to determine the conversion constants which convert the SLI to a failure probability. The EPRI Operator Reliability Experiments (ORE) were then used to validate the failure probabilities generated by CCPRA's human action model. In addition, the specific operator action in question is divided into three phases: identification, diagnosis and performance. This allows the potential operator errors to be characterized in terms of various influencing factors. Each of the aforementioned phases is also characterized by its possible types of cognitive behavior, namely, knowledge-based, rule-based or skill-based. The combination of the above characteristics and careful use of expert judgment and calibration methods are believed to result in more realistic human action failure probabilities as compared with conventional methodologies.

The base CCPRA human actions are modified as described in Section 4.6.1.1 to reflect the impact of fire and smoke on the performance of actions already included in the PRA. If additional recovery actions are required, these actions are interviewed using the above methodology and added to the CCFPRA. See Section 4.6.1.2 for these actions.

To address the direct fire impact on the operator actions, the plant location of where each action is performed is determined (called the destination room(s)). Also determined is the most likely path an operator will travel in order to reach the destination room(s) (given that the action required a plant operator to perform a function outside of the Control Room). If a fire occurs in the destination room or along the travel route, the human action in question is analyzed in detail as to whether the human action should be set to guaranteed failure or degraded. This determination is done by analyzing such things as the location and severity of ignition sources and combustible loading relative to key targets in a specific area and the timing associated with the human action.

For fires occurring in the Control Room, a more detailed analysis is given a loss of one or several control panels. Furthermore, credit is taken for the operators utilizing certain functions on the auxiliary shutdown panel if a fire destroyed the operator's ability to complete the action from the Control Room. The functional impacts associated with operator actions in implementing the abnormal operating procedure for the evacuation of the Control Room (AOP-9a) is also considered.

Since the ventilation system of the Cable Spreading Room (CSR) is common to that of the Control Room, any fires that originated in the CSR could result in smoke in the Control Room. The CSR ventilation system is demanded to isolate on sensing smoke and Halon is demanded to discharge.

On a failure of the CSR ventilation system to isolate, the human action failure probabilities for smoke in the Control Room are used. In the case where a fire originated in the CSR and ventilation isolates the smoke from the Control Room, the human action failure probabilities generated for a turbine building fire, a clear Control Room condition (Human Action Models 1 through 6 in Section 4.6.1.1 depending on the specific human action), are conservatively used. This selection of the turbine building binning is made since access to the CSR is either through the turbine building or via stairtower from the CR. Top Event CL, CSR dampers close on a fire demand, has been added to the CCFPRA to allow the selection of the appropriate human actions.

For certain actions that are not required until late (about 6 hours) in the transient; such as, supplying an alternate water source for AFW, the baseline human action failure probabilities are utilized. For these actions, it is assumed that the specific fire will be suppressed before the operators will even be required to identify, diagnose, or perform the specific human action in question. The only exception to this occurs during a large Turbine Building fire. Given that the operators may have some difficulty getting to particular external locations due to severe damage done to the Turbine Building, the degraded human action failure probabilities generated for the turbine building bin will be used for such long term actions.

4.6.3.1 Approach to Modifying Base CCPRA Human Actions

This section provides additional detail on how the selected human action performance shaping factors are modified to account for the fire.

The human action methodology used in the CCFPRA modifies critical performance shaping factors (PSFs) depending on the location of the fire and its associated smoke, the timing requirements of the action, and the availability of indication. The number and degree of the impact associated with the PSFs varies not only with the physical characteristics stated above, but also with the phases of the action (identification, diagnosis and performance) and the cognitive behavior used in each phase (knowledge-based, rule-based or skill-based). The performance shaping factors which could be impacted include:

Performance Shaping Factors Impacted by Fire

PSF	Description
VT1	Time Rush index
VE1	Training and experience in identifying the need for the required action
VE2	Training and experience in diagnosing what needs to be done
VE3	Training and experience in carrying out the required action
VA1	Adequacy of manning in the Control Room
VA2	Distraction due to unnecessary personnel in the Control Room
VA3	Adequacy of manning outside the Control Room
VA4	Barriers to communication and coordination between the Control Room and others
VA5	Barriers to communication and coordination within the Control Room
VA6	Barriers to communication and coordination outside the Control Room
VI1	Adequacy of initial indication available
VI2	Adequacy of back-up indication available
VD3	Number of proceeding and concurrent unrelated actions
VL1	Difficulties of access, quality and location of equipment in the Control Room
VL2	Difficulties of access to equipment outside the Control Room

The result of addressing the various physical parameters is 11 different human action models. The parameters for these models are shown in Table 4.6.3.1a. Each model also addresses the variations in the characteristics of the human action (3 cognitive process types and 3 phases). Since the performance phase is always assumed to be skill-based, this results in 7 sub-models for each base model. An example showing which of the PSFs are impacted when considering a human action performed when the Control Room is clear of smoke and has multiple indication available, and with smoke at the action performance location (Human Action Fire Model 5) is shown in Table 4.6.3.1b.

Table 4.6.3.1a
Human Action Fire Models

Model (Least to Most Severe)	Control Room Status	Location	Location Status	Indication
1	Clear	Control Room	Clear	Multiple
2	Clear	Control Room	Clear	Single
3	Clear	Other	Clear	Multiple
4	Clear	Other	Clear	Single
5	Clear	Other	Smoke	Multiple
6	Clear	Other	Smoke	Single
7	Smoke: CR Evacuated	Auxiliary Shutdown Panel	Clear	n/a
8	Smoke: action timeframe > 15 min	Control Room	Smoke	n/a
9	Smoke: action timeframe-> 15 min	Other	Clear	n/a
10	Smoke: action timeframe-< 15 min	Control Room	Smoke	n/a
11	Smoke: action timeframe < 15 min	Other	Clear	n/a

Table 4.6.3.1b
Human Action Fire Model 5
Performance Shaping Factors Impacted

	Skill-Based			Rule-Based		Knowledge-Based	
PSF	**Identification**	**Diagnosis**	**Performance**	**Identification**	**Diagnosis**	**Identification**	**Diagnosis**
VT1	X	X	X	X	X	X	X
VE1	X					X	
VE2		X					X
VE3			X				
VA2	X	X	X	X	X	X	X
VA3			X				
VA4	X	X	X	X	X	X	X
VA6			X				
VI1	X	X		X	X	X	X
VI2		X		X	X	X	X
VD3	X	X	X	X	X	X	X
VL2			X				

A base value for the percentage of smoke degradation for each PSF was established as a benchmark to evaluate each Human Action Model. See Section 4.6.3.2 for the development of the base value. In the human action fire models above, the appropriate PSFs for those actions whose conditions are ranked as being most severe are degraded more that the base value. Whereas, the PSFs for those actions whose conditions are ranked as being less severe are degraded less than the base value.

It should be noted that the weighting factor associated with a specific PSF is assumed to remain the same for a fire scenario. The weighting factor accounts for how important a PSF is in a transient. The PSF is assumed to have the same importance in a fire situation although the ability associated with a specific PSF may be degraded. The way to account for this degradation is by adjusting the survey response for the specific PSF rather than adjusting the weighting factor linked to the PSF.

The conversion constant ('a' constant) that is used to convert the Success Likelihood Index (SLI) to a failure probability is also assumed to remain constant in a fire scenario. The human action methodology for CCFPRA uses seven 'a' constants to calculate failure probabilities for different cognitive binning of the specific action in question. These same 'a' constants are utilized to calculate the fire human action failure probabilities since the 'a' constant is just a conversion constant. The specific impact of the fire will be seen in the degradation of the specific PSFs as stated earlier.

This is best illustrated through as example.

Example of a Human Action modified for the Fire CCPRA:

Data Designator	Description	Baseline CCPRA Failure Probability
BHEF11	Operator returns AFW Turbine-Driven Pump to pre-test configuration w/in 10 minutes of loss of MFW	5.21E-03

The first step in analyzing a particular human action is to determine where the action needs to be completed. This particular action requires the auxiliary operator in the AFW Turbine Driven Pump Room (Room T603) to place the AFW Turbine Driven Pump to its pre-test configuration within 10 minutes. This 10 minute timeframe includes the time for the operators in the Control Room to identify that MFW has been lost, direct the auxiliary operator to complete the desired task, and for the auxiliary operator to physically complete the task.

If a fire occurs in Room T603 this action is set to guaranteed failure.

This particular destination room also includes a common access area just outside the AFW Turbine Driven Pump Room near the main feedwater pumps. If a fire occurs in this area has the potential to fail this particular action since the operators will not be able to access Room T603. However, for this particular human action, the operators will already be stationed in Room T603 during the testing of the pump so there will be no need for the operators to traverse through main feedwater pump area. The human action will conservatively uses the failure probability associated with a turbine building fire (Human Action Model 5, smoke at the action location) for any turbine building fire though the affect of the fire on the operators ability to return the pumps to pre-test configuration will be minimal.

This action is determined to have a rule-based identification phase, a knowledge-based diagnosis phase, and a skill-based performance phase. The following table represents the options available to adjust the Performance Shaping Factors (PSFs) for the Rule-based identification phase only. The relevant PSFs that are assumed to be degraded due to a fire are VT1 (time rush), VA2 (distraction of number of people who are in the control room), VA4 (communication between the control room and the auxiliary operator), VI1 (initial indications), VI2 (secondary indications) and VD3 (number of concurrent actions in progress and overall distractions).

Rule-Based Identification

Binning Status			Percentage of Degradation Used					
CR Status	Location Status	Indication Status	PSF VT1	PSF VA2	PSF VA4	PSF VI1	PSF VI2	PSF VD3
C	C	Multiple	10%	10%	N/A	5%	5%	10%
C	C	Single	10%	10%	N/A	20%	20%	10%
C	S	Multiple	20%	10%	20%	5%	5%	20%
C	S	Single	20%	10%	20%	20%	20%	20%

C - Clear S - Smoke

The highlighted row above shows the appropriate values selected. The particular action in question is assumed to have a clear control room with smoke in the area that the auxiliary operator will be performing the desired task. The Control Room has multiple indications that MFW has been lost and that AFW needs to be established. The average percentage of degradation derived for a rule-based identification action is determined to be 20%. Since this particular action falls about midrange in the hierarchical listing of human action models described earlier, the 20% factor is directly applied to the critical PSFs such as time rush, communication, and concurrent actions in progress. The other percentages of degradation applied to the remaining PSFs are generic factors applied to all cognitive phases. It is important to note that this example only shows the percentages of degradation applied to one of the phases of the human action in question. A similar process is repeated for the knowledge-based diagnosis and skill-based performance phases with differing percentages of degradation. Once all three phases have been evaluated, a failure probability can be generated. For this particular example, a failure probability of 1.54E-02 is generated and is utilized in the fire model for this particular human action for all turbine building fires.

The plant model also uses this action if there is a fire in the auxiliary building, intake structure, SBO Diesel building, outside areas, or Control Room. For a fire occurring in the auxiliary building, intake structure, SBO Diesel building, or any outside areas, the auxiliary operator in the AFW Turbine Driven Pump Room will not be affected by smoke. Therefore, the first row on Table 4.6.3.1c for the rule-based identification phase for a clear control room, clear destination room and multiple indications available (Human Action Model 1). Note that a similar process is again followed from the diagnosis and performance phases. This particular binning for a clear Control Room, clear destination room, and multiple indication available falls on the lower range (less severe) of the hierarchical list of models discussed earlier. Since this action under these conditions now bins to a less severe model, a PSF percentage of degradation of 10% is used for critical PSFs such as time rush, communication, and concurrent actions in progress. A human action failure probability of 8.74E-03 is generated for the remaining areas except for the Control Room fire.

Since Control Room fires can have a severe impact on human actions, one would expect that the number generated for a Control Room fire (as pertaining to this particular human action) to use a PSF percentage of degradation greater than 20%. In fact, for a Control Room fire, a PSF percentage of degradation of 40% is used for short term actions for the critical PSFs. This yields a human action failure probability of 5.80E-02 for this particular action.

4.6.3.2 Basis for the Degree of Human Action Degradation

This section provides the basis for the benchmark values used for the percentage of performance shaping factor degradation in the Human Action Fire Models.

The impact of fire is evaluated by first calculating the probabilities of failure assuming a SLI of 10 (perfect values for all PSFs) for each phase and cognitive behavior. The results yield the most favorable failure probability that could be obtained if all PSFs are set to their most favorable value. The failure probabilities are then increased by a factor of 10 and fed back into the SLIM-MAUD equation solving for the SLI required to obtain the increased failure probability. This factor of 10 is based on expert opinion and is assumed to be the average increase, for each phase, to the failure probability due to a fire situation. The resulting SLI that is obtained for the increased failure probability is then converted to a percentage of degradation for use as an average guideline or benchmark to adjust various PSFs that will be affected in a fire scenario. By treating the different human action base fire models in a hierarchical manner, the rank ordering of the physical conditions ensure the appropriate percentage of degradation is assigned. Degradation severities were assigned using expert opinion.

4.6.3.3 Fire Recovery Human Actions

The consequences of fire-induced damage range from total equipment failure to spurious actuation. In many instances, it is reasonable to credit post-fire recovery actions. The base CCPRA human actions are modified as described in Section 4.6.3.1 to reflect the impact of fire and smoke on the performance of actions already included in the PRA. If additional recovery actions are required, these actions are interviewed using the above methodology and added to the CCFPRA. Listed below are those additional actions added to the base CCPRA to recover from the impacts of various fire scenarios.

Data Designator	Description	Failure Probability
BHEF1H	Operator transfers control of AFW Turbine Driven Pumps to the Auxiliary Shutdown Panel within 45 minutes of a Control Room fire at 1C04, 1C05 and 1C06	1.30E-02
BHEPVA	Operator isolates PORVs by manually opening the PORV feeder breakers within 1 hour following a Control Room fire at 1C04, 1C05 and 1C06	1.86E-02
BHESLY	Operator trips all Unit 1 RCPs from Metal Clad within 45 minutes of fire in Switchgear Room	4.06E-03
BHEHF4	Operator restores ventilation to the 45 ft. Switchgear Room within 45 minutes following halon actuation due to a Switchgear Room fire	2.63E-02
BHEHF5	Operator restores ventilation to the 27 ft. Switchgear Room within 45 minutes following halon actuation due to a Switchgear Room fire	2.69E-02
BHEVB1	Operator restores specific electrical loads after successfully stripping the loads when evacuation of the Control Room was initiated (See Note 1)	1.00E-01
BHEVL1	Fire Brigade suppresses a fire in a Control Room Panel within 25 minutes of arriving (prior to more panels being damaged)	2.49E-02
BHERH3	Operator starts Containment Air Coolers and Containment Spray Pumps within 5 hours of a Control Room fire on 1C05 and 1C06	4.19E-04
BHECR2	Operator places Control Room HVAC in recirc mode within 5 minutes of an outside fire (See Note 2)	5.00E-03

Note 1- This particular operator action failure probability is an estimate provided by the human action analyst. The operators strip various electrical loads when they evacuate the Control Room. These electrical loads are manually reestablished once the operators have successfully shifted control of the plant to the auxiliary shutdown panel. Since this action is being utilized for a variety of electrical loads with a multitude of varying parameters, operator interviews would not be efficient or practical. Therefore, a conservative failure probability of 1.00E-01 is assigned. This failure probability is deemed conservative due to the fact that the operators have explicit guidance for each of the loads that needs to be reestablished.

Note 2- This particular operator action failure probability is an estimate. The estimate is based on certain commitments that were made by Operations to change procedures and implement the appropriate training of the operators. See Section 7.2.

4.6.4 Initiating Events

Several types of initiating events are used in the Fire CCPRA. These include:

Single Compartment Initiator

This initiator type is used for single compartments which are assumed to completely burn due to a fire from that compartment. These compartments which have acceptable functional impact when lost.

An example of this initiator type is FIA414, Unit 2 West Electrical Penetration Room, where the last four digits typically represent the compartment.

Group Compartment Initiator

This initiator type is used for compartments which have minimal impact and where conservatively grouping compartments with similar functional impact has a negligible impact on CDF. Note that these groups are modeled as if all functions associated with all rooms within the group are lost.

An example of this initiator type is AUX10C, Unit 2 ECCS Pump Rooms. Also included in this category are initiators coded as FCAxxx.

Fire Scenario Initiator

This initiator type is used for compartments which if failed by a large fire may have an unacceptable impact on CDF. Fire scenarios are developed using the detailed fire modeling process described in Section 4.3.4.

An example of this initiator type is A311F1. These initiators are typically referred to as A311Fx. Each compartment analyzed using this technique is described fully in Section 4.6.2.

Cross-Zone Initiator

This initiator type is used for compartments which are grouped to address cross-zone fire spread. See Section 4.3.3 for the analysis approach and Section 4.6.6 for the results of this analysis.

For each initiator, the functional impact is modeled. This impact considers:

- the equipment and cable lost due to the fire damage
- equipment impact due to fire suppression and ventilation isolation
- human action impact due to fire, smoke and confusion

Equipment, cable and human action impacts are mapped to their associated functions. The impacted functions are modeled by either failing the associated top events or by modifying the top events failure probability (selecting the appropriate split fraction(s)). For each fire initiating event the top events which are impacted are identified by a two letter code, e.g. "AA" is the top event for 4KV Bus 11 remains energized. Those top events which are impacted but not failed are annotated with an asterisk (*).

Table 4.6b provides a complete list of top events used by the CCFPRA.

4.6.5 Compartment Analysis

This section addresses the analysis and results of all the Compartments. Compartments which are conservatively modeled by burning the entire compartment are listed in Table 4.6.2a. Compartment groups are listed in Table 4.6.2b. Compartments which are fire modeled are listed in Table 4.6.2c and are described in detail in attachments to Section 4.

Table 4.6.5a
Single Compartment Fire Initiators

Initiator	Frequency	Description	Functional Impacts	CDF
FIA301	8.48E-04	Battery Room 11 (Note 1)	DA, XA, TA, TB, F7*, TH,	2.98E-06
FIA304	7.92E-04	Battery Room 12 (Note 1)	DB, XB	5.78E-08
FIA305	8.84E-04	Battery Room 21 (Note 1)	DC, XC, NR, GW, GZ, F9	3.52E-06
FIA307	8.14E-04	Battery Room 22 (Note 1)	DD, XD	5.18E-08
FIA309	4.84E-04	Unit 2 Main Steam Piping Area	QF, GH, DM*, PG, GZ	3.65E-09
FIA312	4.51E-04	Unit 2 Purge Air Room	QD, QF, QE, Y3, Y4, DM*, F9	1.12E-07
FIA414	3.28E-03	Unit 2 West Electric Penetration Room	GF, GH, M3, HL, Y4, H9, NR*, FO*, SH*	3.43E-08
FIA416	2.23E-02	2B Diesel Generator Room	GF, Y4	1.02E-07
FIA420	1.58E-03	Reactor Coolant Waste Evaporator Room	GG, AB*, Y4, FC*, CV, DL*,SG*	1.56E-08
FIA421	2.23E-02	1B Diesel Generator Room	GG, Y3, Y4,	1.39E-07
FIA422	2.23E-02	2A Diesel Generator Room	GH, Y3, RA*, RB*	4.00E-07
FIA439	1.24E-03	Unit 1 RWT Pump Room	RE, RW, RA*, RB*	4.39E-08
FIMAB2	3.90E-04	Auxiliary Building Stairtower AB-2	CV	2.29E-09
FIT605	6.97E-04	Unit 2 AFW Pump Room	Y3, Y4, GW, GZ, F9	8.88E-09
CDF Subtotal		**Single Compartment Initiators**		**7.47E-06**

Note 1: Battery fluid, although corrosive, is not flammable or combustible. The battery casings are in contact with the fluid and are unlikely to be an ignition source. Therefore, the only plausible ignition source considered for these compartments is maintenance refuse.

Table 4.6.5b
Group Compartment Fire Initiators

Initiator	Frequency	Description	Functional Impacts	CDF
AUX10A	4.04E-03	-10'/-15' Hallways	Y4, FC, CV, HW, WJ	8.58E-08
AUX10B	3.19E-03	Unit 1 Charging Pump Rooms	FC, CV	3.48E-08
AUX10C	5.88E-03	Unit 2 ECCS Pump Rooms	Y3, Y4, GW	4.22E-08
AUX10D	7.85E-04	Reactor Coolant Waste Tank Rooms	Y4, FC*, CV, HW*, DL*, SG*, HA	1.05E-08
AUX10E	6.23E-03	Unit 1 ECCS Pump Rooms and Recirc Tunnel	Y3, Y4, CV, VM, V1, V2, MV, HA, HB, HW, CS, CT, SR*, TE, TW	4.49E-07
AUX20A	2.21E-02	5' Multi-compartment Area (Fire Area 11)	Y3, Y4, GW, FF, FC*, F1*, CV, V1, HB, V5, HW, DL*, CT, SH*, GG	2.83E-06
AUX20B	7.84E-03	Unit 2 Service Water, Component Cooling and Radiation Exhaust Rooms	Y3, Y4, GW, GZ, FO, F9, NR*, NS*	1.13E-07
FCA206	9.29E-04	Unit 2 East Electrical Penetration Room	DM*, F9*, SR*	3.27E-09
FCA221	3.46E-04	Unit 1 West Piping Pentration Rooms	DL*	2.62E-09
FCA300	6.20E-04	Hallways Outside the Control Room	HR	1.87E-08
FCA319	4.19E-03	27' Multi-compartment Area (Fire Area 11)	NR*, CV, DL*, SR*	4.62E-08
FCA523	1.23E-02	69' Multi-compartment Area (Fire Area 11)	GG, GH, H9, H6, XW, HH*	1.22E-07
CDF Subtotal		**Group Compartment Initiators**		**3.76E-06**

Table 4.6.5c
Fire Modeled Compartments

Attachment	Room	Fire Area	Description	CDF
A	A225	14	Unit 1 Radiation Exhaust Equipment Room	3.99E-09
B	A226	39	Unit 1 Service Water Pump Room	3.47E-08
C	A227	11	Unit 1 East Piping Penetration Rooms	6.82E-07
D	A228	15	Unit 1 Component Cooling Pump Room	0.00E-00
E (CSR)	A302	17	Unit 2 Cable Spreading Room	6.81E-07
	A306	16	Unit 1 Cable Spreading Room	6.72E-06
E1	A308	11	North/South Passage Way	0.00E-00
F (SWGR)	A311	18	Unit 2 27' Switchgear Room	2.22E-07
	A317	19	Unit 1 27' Switchgear Room	4.28E-06
	A407	25	Unit 2 27' Switchgear Room	1.19E-07
	A430	34	Unit 1 45' Switchgear Room	1.54E-06
G	A315	11	Unit 1 Main Steam Isolation Valve Room	0.00E-00
H	A318	19A	Unit 1 Purge Air Room	1.15E-09
I (MCR)	A405	24	Main Control Room	2.53E-05

Table 4.6.5c
Fire Modeled Compartments (Continued)

J	A419	11	Cask and Equipment Loading Area-Truck Bay	1.86E-07
K	A423	32	Unit 1 West Electric Penetration Room	6.78E-08
L	A429	33	Unit 1 East Electric Penetration Room	7.10E-08
M	A512 A520 A524 A525	11 11 11 11	Control Room HVAC Room Spent Fuel Vent Room Unit 1 Main Vent Fan Room Unit 1 Containment Access Are	1.91E-09 0.00E-00 2.37E-08 0.00E-00
N	A529	37	Unit 1 69' Electrical Room	9.18E-08
O	AB	AB-1,3,4,5,E	Stairtowers	0.00E-00
P	CC-A&B	20,21,22,23	Cable Chases 1A,1B,2A,2B	9.67E-07
Q	CC-C	16,17	Cable Chases 1C,2C	0.00E-00
R	INTAKE	IS	Intake Structure	1.22E-08
S	T603	42	Unit 1 AFW Pump Room	4.76E-07
T	TB	TB	Turbine Building	1.66E-05
U	YARD	YARD	Transformers, Tanks and Independent Structures	3.53E-06
CDF Subtotal			**Fire Modeled Compartments Total**	**6.16E-05**

4.6.6 Cross-Zone Analysis

This section provides the results of the cross-zone analysis described in Section 4.3 for propagation frequencies > 1.0E-07/yr. The quantification of fire-induced core damage frequency (CDF) is obtained by propagating fire-induced failures through a modified version of the CCNPP PRA as described in Section 4.6.

For many of the compartment combinations, the functional impact of the loss of one of the compartments is a subset of the other compartment. For these compartment combinations, no new initiating event is developed. The risk impact is determined by simply multiplying the cross-zone propagation frequency by the conditional core damage frequency of the bounding compartment. These compartment combinations are shown in Table 4.6.6a as having an "unassigned" initiating event.

In six cases, the impact of fire damage is not a subset by an existing initiator and a new Cross-Zone Fire Initiator is used.

Table 4.6.6a
Cross-Zone Fire Inititators Analysis

PRA Initiator	Room Pairs	Propagation Frequency	Functional Impacts	CDF
FCCMB1	FIA421/FIA422 FIA421/FIA439	1.06E-05 1.56E-05	GG, GH, Y3, RE, RW	9.40E-10
FCCMB2	FIA416/FIA418	8.53E-07	GF, Y4, SH*	1.50E-12
FCCMB3	AUX10A/AUX10C	1.56E-06	Y3, Y4, GW, FC, CV, HW, WJ	7.64E-11
FCCMB4	AUX10A/AUX10E	1.61E-06	Y3, Y4, FC, CV, MV, HW, SR*, TE, TW	1.42E-10

Table 4.6.6a
Cross-Zone Fire Inititators Analysis (Continued)

PRA Initiator	Room Pairs	Propagation Frequency	Functional Impacts	CDF
FCCMB5	AUX10A/AUX10D	5.75E-05	Y4, FC, CV, HA, HW, SG*, WJ	1.05E-09
FCCMB6	A423F(x)/FIA421 A423F(x)/FIA422	1.91E-06 9.77E-06	E1, E2, E3, E4 (Back up Bus fails), GG, M1, HR, HL, Y3, XW, QZ*, NR*, NS*, H6, RS*, RR, PS, PV*, FT, CV, RE, RW, WY*, SG, SH, GH	1.19E-07
Unassigned	A105B/AUX10A	2.14E-06	Represented by AUX10A	4.54E-11
Unassigned	A105C/AUX10A	1.15E-06	Represented by AUX10A	2.44E-11
Unassigned	A117/AUX10A	1.92E-06	Represented by AUX20A	4.08E-11
Unassigned	A117/AUX10E	1.78E-07	Represented by AUX10E	1.28E-11
Unassigned	A117/AUX20A	1.78E-07	Represented by AUX20A	2.28E-11
Unassigned	A120/AUX20A	2.10E-06	Represented by AUX20A	2.69E-10
Unassigned	A440/FIA416	1.31E-05	Represented by FIA416	5.97E-11
Unassigned	A536/FIA420	4.65E-06	Represented by FIA420	4.59E-11
Unassigned	AB-1/AUX10A	7.00E-07	Represented by AUX10A	1.49E-11
Unassigned	AB-E/AUX10A	2.07E-05	Represented by AUX10A	4.39E-10
Unassigned	AUX10A/AUX10B	7.28E-05	Represented by AUX10A	7.94E-10
Unassigned	AUX10A/AUX20A	2.07E-05	Represented by AUX20A	2.65E-09
Unassigned	AUX10A/FIMAB2	2.89E-06	Represented by AUX10A	1.69E-11
Unassigned	AUX10B/AUX20A	1.22E-07	Represented by AUX20A	1.56E-11
Unassigned	FIA414/FIA416	9.81E-06	Represented by FIA414	4.47E-11
Unassigned	FIA420/FIA421	2.14E-07	Represented by FIA420	1.33E-12
CDF Subtotal			**Cross-zone Analysis Total**	**1.26E-07**

4.6.7 Unit 1 Fire Risk Assessment Results

The results of the CCFPRA are summarized below in Table 4.6.7a by analysis type. The majority of the contribution is as expected from the compartments which are fire modeled. However, it should be noted that the compartments which are not fire modeled, the single and group compartment initiators, would likely be substantially reduced with additional effort. The cross-zone fire analysis results in an insignificant contribution to risk.

The top 100 sequences quantified are shown in Table 4-6c. The dominant issues apparent from these sequences are identified in the following sections. Table 4-6d provides the descriptions and values of the split fractions used in the top 100 sequences.

Table 4.6.7a
CCFPRA Results Summary

Analysis Event Type	CDF Contribution	Percentage
Single Compartment (Initiators)	7.47E-6	10.2%
Group Compartment (Initiators)	3.76E-6	5.2%
Fire Modeled Compartments	6.16E-5	84.4%
Cross-zone Analysis	1.26E-7	0.2%
Total CDF	**7.30E-5**	**100%**

4.6.7.1 Large Turbine Building Fire (Initiating Event TBALLB)

A large turbine building (TB) fire through equipment failure causes the loss of off-site power. The large fire in the TB also prevents several key human actions:

- Human Action to align the 0C EDG: This is considered failed as access to the SWGR rooms is more difficult. Even if the 0C EDG is aligned it will probably be aligned to Unit 2 as Unit 1 has the 1A EDG.

- Human Actions to align make-up to the SRW head tanks: This is considered failed as access to fire protection and condensate feeds to the SRW head tanks is by procedure through the turbine building. Even if the operators could use fire protection to directly feed the head tanks located in the auxiliary building, it is unlikely that the fire protection system would have adequate pressure due to the turbine building fire fighting activities. The final avenue of make-up is using the SW system to directly feed the SRW system. The normal route of access is through the turbine building, but access through the auxiliary building is possible. No credit is taken for this alternate make-up route.

- Human Action to Open the AFW Pump Room Doors: This action is considered failed as the AFW Pump room doors open directly into the turbine building (the fire location).

- Human Action to Align AFW Pump 23 to Unit 1: This action is considered failed as Unit 2 is considered to be in the same dire straights as Unit 1.

- Human Action to align Portable Fans to the SWGR Rooms is lost: This action is considered failed as the portable fans are located in the TB.

These impacts alone are not sufficient to cause core damage. But any of these independent failures combination with the impacts listed above will cause core damage:

- 1A EDG Fails (Top Event GE) with high SRW pre-existing Leakage (Top Event RL): When the SRW System has high pre-existing leakage, the head tanks can not maintain sufficient inventory to prevent SRW pump failure over a full 24 hr mission. The failure of SRW causes the EDGs cooled by SRW to be lost. This station blackout condition causes the failure of all of the Unit 1 AFW Pumps. The motor driven pumps fail due to loss of power while the turbine driven AFW Pumps fail due to loss of room cooling. As mentioned above, the human action to open the turbine driven AFW pump room doors (for cooling) is assumed failed.

- Operator Fails to Open SWGR Dampers (Top Event HF): The large turbine building fire is considered to cause both SWGR room dampers to close. The mechanism of closure is either due to smoke entering the outside air intake or through the failure of the Halon control panel. If the operators fail to open these dampers, then a Unit 1 blackout is assumed.

- Operator fails to provide long term water for AFW following CST 12 Depletion (Top Event F3): Following the depletion of CST 12, operator action is required to align another water source for AFW. Although the tank farm is removed from a turbine building fire, it does have an impact on the success of this action by restricting operator access to the AFW Turbine Driven Pump Room. Normally when the turbine driven pumps are operating, an operator is stationed in the pump room. Even if the control room fails to align an alternate water source prior to the AFW Pumps losing Net Positive Suction Head (NPSH), the operator stationed in the pump room would notice pump cavitation. This provides an addition check in preventing the failure of the operating AFW pumps due to loss of NPSH. The presents of a turbine building fire eliminates this redundant check there by increasing the likelihood of failure.

- Operator fails to Isolation Control Room Inlet Dampers (Top Event CR): Failure to isolate this inlet dampers is assumed to result in control room evacuation. Given that there is a large fire in the turbine building limiting local access to the turbine driven AFW pumps, no credit is taken for the operators manning the aux shutdown panel. As a result, core damage is assumed.

- 1B EDG Fails (Top Event GG) with a Unit 1 SWGR HVAC Failure (Top Event HS): This case is similar to the first case, but Unit 1 power is lost due to a loss of SWGR ventilation. The Unit 1 SWGR Room must be cooled to prevent the SR electrical busses from being lost. This cooling can be supplied be either the portable fans or the SWGR HVAC system. As the portable fans are lost in this scenario, only the HVAC system is available. The loss of the 1B EDG cause the failure of one of the SWGR HVAC headers, while the remaining header independently fails (mostly likely as a result of header unavailability), Without SWGR HVAC, a Unit 1 blackout is assumed.

4.6.7.2 Fire on Auxiliary Feedwater (1C04) and Reactivity Control (1C05)

A fire on 1C04 disables MFW, OTCC, and AFW. If the operator does not man the auxiliary shutdown panel (1C43) and establish AFW flow in a timely fashion core damage occurs. In the short term, the most likely failure scenarios are the aux shutdown control is not established (Top Event FW), or that both of the turbine driven pumps fail (Top Events TF and TG). In the long term, the most likely failure scenario is the failure to establish a long term water source of AFW (Top Event F3).

4.6.7.3 Battery Room Fire (Initiating Events FIA305 & FIA301)

The loss of a battery is a significant event in the general transient evaluation, and even more so when a fire is the cause of the loss. Recovery from a battery loss requires several local human actions:

- Local AFW Flow Control (Top Events HX and UQ): The loss of DC power causes MFW to be lost and the AFW flow control valves to fail open. Without operator action, the S/G will overfill which would cause the turbine driven AFW pumps to fail. Additionally in attempting to rectify the overfill situation the operators may underfeed the S/Gs.

- Local Trip the RCP (Top Event SL): The loss of DC power isolates cooling to the RCPs. If the operators do not trip the RCPs locally, then the RCP Seals may fail prior to the pump stopping. Subsequent safety injection failures (Top Event HA) could lead to core damage.

Although the likelihood of a battery loss is much lower due to a fire then through other mechanisms, the likelihood of human action failure increases dramatically during a fire. For this reason, the battery room fires are significant.

4.6.7.4 Control Room Panel Fire which Leads to Evacuation

Every Control Panel Fire can lead to Control Room evacuation. If the fire is not suppressed (Top Event CR), then the operators are forced to evacuate. Once the Control Room is evacuated, the operators are required to load shed most of the electrical loads, and manually re-start these loads (Top Events VB). If the re-start is not done, then site is in a self-induced station blackout condition. This condition will eventually result in the loss of the 125 VDC batteries. This causes a loss of all indication. This is assumed to result in core damage. Even if the operators, successfully re-load the buses, a failure of either EDG 1A or EDG 2B will leave the site with only one-of-two facilities (A or B). This will eventually cause 2-of-4 batteries to be lost which will cause an SSSA condition (See 3.1.5.6 in the Seismic Sequence Analysis for more information on this condition). An SSSA condition with the auxiliary shutdown panel as the operators only source of indication is assumed to result in core damage. The key to mitigating all Control Room evacuation scenarios is that the operators man the auxiliary shutdown panel and establish AFW flow prior to core damage from lack of decay heat removal.

4.6.8 Unit 2 Fire Assessment

Since a Unit 2 specific PRA model has not been developed, an assessment of the differences between units with respect to fire is performed. The following sections describe the methodology and results.

4.6.8.1 Methodology

The main difference between Unit 1 and Unit 2 is a SRW dependency on the EDGs. Both Unit 1 EDGs are SRW-cooled, while only one of two Unit 2 EDGs are SRW-cooled. Because of this difference, even if the fire impacted Unit 1 and Unit 2 equipment in the exact same fashion, Unit 2 would be affected differently than Unit 1. As a result, the first step is to determine the change in risk for Unit 2 assuming the fire impacts on Unit 2 are the same as those impacts on Unit 1 (See 4.6.8.1.1).

The second step is to determine which Unit 2 fire scenarios do not have a Unit 1 equivalent fire scenario (See 4.6.8.1.2). The change in risk is estimated for each Unit 1 fire scenario which does not have a functionally equivalent Unit 2 impact.

4.6.8.1.1 Emergency Diesel Generator Difference

The most significant non-cable routing difference between Unit 1 and Unit 2 is the lack of a SRW dependency on Emergency Diesel Generator 1A. EDG 1A is a newly-installed safety-related self-cooled diesel generator and backs up Unit 1's safety-related 4KV Bus 11. The Unit 2 equivalent bus is 4KV Bus 24. It is backed up by SRW-dependent EDG 2B.

The EDGs tie into the 4KV buses which in turn power the 480V, 125VDC, and vital inverters which power the 120VAC vital buses. At the 125VDC and 120VAC levels, multiple trains and cross ties between the units electrical buses eliminates the impact of the SRW dependency. That is, at the lower voltage levels, Unit 2 has half of its trains ultimately powered from the non-SRW-dependent EDG 1A, just as Unit 1 has. However, at levels above these (4KV and 480V), there are no Unit 2 buses backed by EDG 1A and the SRW dependency exists.

The most significant impact of the increased SRW dependency of Unit 2 is on the motor driven AFW pump. Unit 2's motor driven AFW pump (AFW Pump 23) is aligned to 4KV Bus 24, which is backed up by SRW dependent EDG 2B. Unit 1's motor driven pump is aligned to EDG 1A-backed 4KV Bus 11, and does not have the SRW dependency. However, this does not mean that Unit 2 AFW is entirely SRW dependent. Unit 1 AFW may be cross-connected to Unit 2, but this requires a human action. Alignment of Unit 1 AFW PP 13 to Unit 2 is only possible if Unit 1 already has adequate decay heat removal through use of MFW or the Unit 1 turbine driven pumps. If Unit 1 does not have MFW or at least one Unit 1 turbine driven AFW, then Unit 1 AFW Pump 13 is not available to Unit 2.

A Unit 2 Fire CDF is evaluated by quantifying a modified version of the Unit 1 CCFPRA. For the Unit 2 model, the diesel generators are 'rewired' to the 4 KV Buses.

4.6.8.1.2 Cable Routing Differences

Three types of cable routing comparisons are performed: fire modeled to fire modeled compartments, non-fire modeled to fire modeled compartments and non-fire modeled to non-fire modeled compartments. These comparisons are limited to top event level and are described below. That is, if like top events are impacted in scenarios or compartments which are being compared, then no additional review is performed. If differences are noted where there is a loss of a Unit 2 function which is not included in the like Unit 1 assessment, then additional review is performed to determine if the top event which is not bounded is only degraded. Non-bounded differences which can not be resolved are quantified.

Fire Modeled to Fire Modeled Compartment Comparisons

If the Unit 2 compartment and its equivalent Unit 1 compartment are both fire modeled, then the Unit 2 impacts on top events are compared to Unit 1 impacts for the equivalent fire scenarios. Fire impacts for the following compartments are evaluated in this way.

1. A311 and its equivalent compartment A317 (27' switchgear rooms)
2. A407 and its equivalent compartment A430 (45' switchgear rooms)
3. A306 and its equivalent compartment A302 (cable spreading rooms)

Note that many Unit 2 compartments have an impact on the Unit 1 fire analysis. This is true for the three pairs of compartment listed above. For example, both cable spreading rooms are evaluated in the Unit 1 CCFPRA. Therefore two comparisons must be made when comparing Unit 1 with Unit 2.

Cable Spreading Room Comparisons

Compartment	Evaluation Perspective	Comparison Compartment
A302 Unit 2 Cable Spreading Room	Unit 2	A306 as modeled in Unit 1's CCFPRA
A306 Unit 1 Cable Spreading Room	Unit 2	A302 as modeled in Unit 1's CCFPRA

Non-Fire Modeled to Fire Modeled Compartment Comparisons

Unit 2 impact compartments A414, A409 and A532 are not fire modeled for the Unit 1 analysis. The equivalent Unit 1 compartments A423, A429 and A529 are fire modeled. Using knowledge from the equivalent Unit 1 compartments and additional information from walkdowns, cable raceways, in A414, A409 and A532 that would be impacted by fire are identified. The impact of burning the identified raceways in the Unit 2 compartments is compared to the impact of similar fire scenarios developed for the Unit 1 compartments.

Non-Fire Modeled to Non-Fire Modeled Comparisons

For the compartments that are not fire modeled in the Unit 1 analysis, a list of Unit 2 impacts is generated for each compartment. The Unit Fire Initiating Event that includes the equivalent Unit 1 Compartment is selected and a list of Unit 1 impacts caused by the fire initiating event is generated. The list of Unit 2 top impacts are then compared to the Unit 1 list.

4.6.8.3 Unit 2 Fire Assessment Results

The estimated Unit 2 fire CDF is 1.1E-05 per reactor year. The emergency diesel generators configuration differences account for 97% of the difference is risk between the units. This contribution was estimated using the CCFPRA with only the diesel generators 'rewired' so that the SRW-dependent diesels supply the 4KV buses on the Unit 1 side. Following this, the risk due to the rewired diesels and other cable routing differences was estimated. This combination increased the estimated risk by an additional 3% to the calculated risk yielding the total of 1.1E-04.

The large turbine building fire contributes about a quarter to the risk of Unit 2 as opposed to 17% for Unit 1 and is responsible for about a half of the overall increase in risk. This contribution results from the degradation of SRW makeup, both hardware and human actions. Both Unit 1 and 2 CCFPRAs consider the probability that the SRW system leakage will be high enough to result in the failure SRW within 24 hours. Such a failure results in the loss of Emergency Diesel Generators 2A and 2B (and 1B for Unit 1).

4.7 Analysis of Containment Performance

This section describes the results of the Level II containment performance analysis. CCFPRA explicitly evaluates the fire impact on the containment mitigation and isolation functions. Therefore, the methodology used to determine the containment failure modes is essentially the same as that used in the Calvert Cliffs IPE Summary Report (Ref. 4-6). The plant damage states (PDSs) are determined for each initiator. Like the IPE, these PDSs are related to key PDSs. The key PDSs are those for which the accident progression was determined. The contribution of these key PDSs to containment failure is shown in Table 4-7. Note that this approach assumes that the accident progression for a given CCFPRA key PDS is not significantly difference than that of the IPE.

The two dominate key PDSs are HRIF and HGIP. These key PDSs represent a high pressure core melt with the containment isolated and either no containment cooling (F) or containment sprays available (P). Both of these PDSs were also identified as high contributors in the IPE submittal. A new key PDS, HRWF, is also identified. This PDS represents a high core melt with large early containment failure and no containment cooling. Fire sequences which could result in spurious actuation of containment isolation valves are conservatively mapped to this PDS. Many of the spurious actuation sequences result from fires where the Control Room is evacuated. It also should be noted IPE submittal has a larger early small containment failure contribution. The IPE's contribution is primarily due to a steam generator tube rupture which is not a concern in the CCFPRA.

The impact of fire on interfacing LOCAs is also evaluated. All cables which could affect the probability of having an interfacing LOCA (V-sequence) are routed and assessed. It is determined that even though some cable failures could cause an increase in the likelihood of a V-sequence, none of these cables are impacted by a plausible fire scenario. Therefore, with the exception of the Control Room noted in the paragraph above, a V-sequence is not initiated by a fire initiating event.

4.8 Treatment of Fire Risk Scoping Study Issues

This section provides the results of the evaluation of the Fire Risk Scoping Study issues for CCNPP. The discussion of these issues for CCNPP is presented here consistent with the NRC recommendation that resolution of the FRSS issues be coordinated with the Fire IPEEE. These issues are identified in NUREG-1407 as well as in NRC Generic Letter 88-20 Supplement 4.

Sandia National Laboratories, as part of the Fire Protection Research Project, undertook two tasks in what is now referred to as the Fire Risk Scoping Study (FRSS). The tasks were to review and update the perspective of fire risk in light of the information developed through the Fire Protection Research Project, and to identify and perform initial investigations of any potential unaddressed issues of fire risk. As a result of that study, the NRC identified six issues to be addressed in any future fire evaluation methodology. Subsequent to NUREG-1407, Generic Letter 88-20, Supplement 4, was also issued which addressed the six issues contained within the FRSS. These issues are:

- Seismic/Fire Interactions
- Fire Barrier Qualifications
- Manual Fire Fighting Effectiveness
- Total Environment Equipment Survival
- Control Systems Interactions

- Improved Analytical Codes

The approach taken to address these issues was similar to that described in FIVE regarding each of the above issues. A summary of the results of the investigation into these issues is provided in the following sections. The last issue, namely "Improved Analytical Codes", involves some questions regarding the available fire models for use in the Fire IPEEE. These questions are considered resolved with the development of the FIVE methodology and no additional evaluation of this last issue is addressed.

4.8.1 Seismic/Fire Interactions

The Seismic/Fire Interactions involves three specific concerns as follows:

- Seismically-Induced Fires
- Seismic Actuation of Fire Suppression Systems
- Seismic Degradation of Fire Suppression Systems

4.8.1.1 Seismically-Induced Fires

Description of the Issue

A verification that hydrogen or other flammable gas or liquid storage vessels in areas with seismic safe shutdown or safety-related equipment are not subject to leakage under seismic conditions.

Resolution of Issue

This issue is considered resolved. At CCNPP an evaluation, was performed which determined there are no seismic induced flammable gas or liquid leakage potentials that could impact safe shutdown equipment. This issue is further addressed in Section 3.1.3.3 of this summary report.

4.8.1.2 Seismic Degradation of Fire Suppression

Description of the Issue

A verification that the design of water suppression systems considers the effects of inadvertent suppressions system actuation and discharge on that equipment credited as part of the seismic safe shutdown path in a margins assessment that was not previously reviewed relative to the internal flooding analysis or concerns as those discussed in IE Information Notice 83-41.

Resolution of Issue

This issue is considered resolved. At CCNPP an evaluation was performed on the design of both water suppression and gas fire suppression systems to consider the effects of inadvertent actuation and discharge on safe shutdown equipment. As a result of this evaluation, it is determined that there are no impacts to safe shutdown due to inadvertent actuation of water suppression systems. However, should the switchgear room Halon systems spuriously activate, they could isolate the ventilation to both the 45' and 27' Switchgear rooms. This would require timely operator action to restore ventilation and this has been factored into the CCFPRA. See Section 4.8.4.1. Additionally, the Cable Spreading Room Halon system could be activated by fire in the Control Room resulting in the isolation of the ventilation system to both cable spreading rooms.

4.8.1.3 Seismic Actuation of Fire Suppression Systems

Description of Issue

A verification that fire suppression systems have been structurally installed in accordance with good industrial practices and reviewed for seismic considerations such that suppression system piping and components will not fall and damage safe shutdown path components nor is it likely that leaking or cascading of the suppressant will result.

Resolution of Issue

This issue is considered resolved. Walkdowns were performed that verified fire suppression systems and components are structurally installed with due consideration for seismic events.

4.8.2 Fire Barrier Qualification

Description of the Issue

Fire barriers and components such as fire dampers, fire penetration seals and fire doors for fire barriers considered in the Fire PRA are included in a plant surveillance and maintenance program. Additionally fire penetration seals have been installed and maintained to address concerns in IE Notice 88-04; and fire dampers have been inspected, installed and maintained to address concerns in IE Notices 83-69 and 89-52.

Resolution of Issue

This issue is addressed and considered resolved. At CCNPP the barriers required for Appendix R separation are included in a surveillance and maintenance program. The fire penetration seals associated with these barriers have been inspected and evaluated to satisfy concerns raised in IE Notice 88-04. Fire dampers have been inspected and evaluated to satisfy concerns raised in IE Notice's 83-69 and 89-52. Additionally, non Appendix R Fire Barriers which are credited in the CCFPRA are being added to an inspection or control program to ensure the barrier integrity is maintained. See Section 4.3.3.1 and Section 7.

4.8.3 Manual Firefighting Effectiveness

Description of the Issue

The Sandia Fire Risk Scoping Evaluation addresses a number of attributes of an acceptable fire brigade training and preparedness program. An evaluation of these attributes such as reporting fires, brigade personal and training, practice and drills and records should be performed.

Resolution of Issue

This issue has been evaluated and is considered resolved. The CCFPRA takes moderate credit for manual firefighting. Such action is credited for Control Room fire scenarios where operators and/or the fire brigade is successful in extinguishing the fire. Credit for manual firefighting was also taken into account in the analysis for determining fire barrier failure probabilities. See Section 4.3.3.2.

Nevertheless, the effectiveness of manual firefighting activities at CCNPP was reviewed using Table 4-3 of the EPRI FPRA Guide. Based on a review of the established program for Fire Prevention, Fire Fighting and Strategies, periodic assessment of the program, and the performance during drills and actual fire events; the plant fire brigade and manual firefighting capability is considered to be effective.

4.8.4 Total Environment Equipment Survival

Description of the Issue

The Total Environment Equipment Survival issue involves three specific concerns as follows:

- Potential Adverse Effects of Plant Equipment by Combustion Products
- Spurious or Inadvertent Fire Suppression Activation
- Operator Action Effectiveness

4.8.4.1 Potential Adverse Effects of Plant Equipment by Combustion Products

Description of the Issue

The immediate effects of smoke and non-thermal fire effects on plant equipment are addressed indirectly by conservatively assuming all equipment within the fire area is disabled by the postulated fire. In cases where a fire scenario does not screen out (i.e. fire modeling is required), explicit treatment of smoke and non-thermal fire effects is not possible due to lack of industry tests and studies. However, it is assumed that the conservative treatment of fire ignition frequencies, propagation factors and target damage adequately compensates for any uncertainty.

Resolution of Issue

This issue is considered to be adequately addressed in CCFPRA. The CCFPRA addresses the impact smoke has on all human actions, on inadvertent Halon actuation and on Control Room evacuation. The impact of smoke on human actions is extensively discussed in Section 4.6.3. The degradation of recovery actions is a significant risk contribution to CCPRA. For the issue of inadvertent Halon actuation, smoke from external sources outside the protected compartment could result in the inadvertent actuation of Halon and the coincident isolation of ventilation. The compartments found to be of concern are the switchgear rooms and cable spreading rooms. Human actions to recover ventilation to these rooms are explicitly modeled. The most interesting aspect of this issue is the potential for the loss of ventilation to critical areas as a result of a fire in a non-critical area such as the turbine building or an outside transformer fire. The evacuation of the Control Room due to smoke is also explicitly modeled. See Section 4.6. By addressing these key issues above, it is believed that the most significant aspects of smoke are capture in CCFPRA.

4.8.4.2 Spurious or Inadvertent Fire Suppression Activation

Description of the Issue

Verify that the design of fire suppression systems considers the effects, if appropriate, of inadvertent suppression system actuation and discharge on equipment credited for safe shutdown, for concerns such as those discussed in NRC I & E Information Notice 83-41.

Resolution of Issue

This issue is considered resolved. See Section 4.8.1.2. As a result of this evaluation, it is determined that there are no impacts to safe shutdown due to inadvertent actuation of water suppression systems. However, should the Switchgear Room Halon system spuriously activate, it could isolate the ventilation to both the 45' and 27' Switchgear Rooms or the cable spreading rooms. This would require timely operator action to restore ventilation and this has been factored into the CCFPRA. See Section 4.8.4.1.

4.8.4.3 Operator Action Effectiveness

Description of the Issue

There are safe shutdown procedures identifying the steps for planned shutdown when necessary in the event of a fire, and operators receive training on these procedures. If in the performance of these procedures operators are expected to pass through or perform manual actions in areas that contain fire or smoke, suitable SCBA equipment and other protective equipment are available for operators to perform their function.

Resolution of Issue

This issue has been evaluated and is considered resolved. Operator actions required at CCNPP in the event of a fire are delineated in AOP-09, Alternate Safe Shutdown/Control Room Evacuation procedures. See Section 4.2.2.4. All licensed and non-licensed operators are trained together at least once each training cycle on these procedures. Additionally, all operator actions are evaluated for possible degradation due to fire impacts on the operators ability to perform their function(s) as described in Section 4.6.1.

4.8.5 Control Systems Interactions

Description of the Issue

Safe shutdown circuits are physically independent of, or can be isolated from, the control room for a fire in the control room fire area.

Resolution of Issue

This issue has been evaluated and is considered resolved. The CCNPP Appendix R analysis included a specific assessment of the consequences of a bounding worst case Control Room fire. Such a fire is conservatively assumed to disable all Control Room controls and functions and require operator evacuation of the Control Room area. The evaluation of the scenario, as described in the Appendix R Interactive Cable Analysis (References 4-8 and 4-9), confirmed that safe plant shutdown can be achieved by using available controls and indications located on the Auxiliary Shutdown Panels located in the 45' Switchgear room for each unit. The assessment also confirmed that the circuits associated with these features are located remote to the Control Room or can otherwise be isolated from the Control Room. Additionally, certain safe shutdown components are provided with local/remote switching capabilities to provide local control under certain fire scenarios.

4.8.6 Improved Analytical Codes

The fire modeling methodology for CCNPP uses FIVE worksheets that are based on correlation's used as the foundation for COMPBRN IIIe. The worksheets allow for even more conservative assumptions than COMPBRN IIIe to reduce the complexity and number of variables required for calculation. This utilizes look-up tables for quantifying the potential fire exposure to targets.

Though some limited uses of the look-up tables have been shown to be less conservative than experimental data, the difference is small and other conservatism's are considered to more than compensate for these differences.

4.9 USI A-45 and other Safety Issues

The purpose of this report section is to describe how the CCNPP IPEEE evaluates the Decay Heat Removal Safety Function based on the results from the CCNPP Fire PRA. The evaluation was performed to identify potential decay heat removal vulnerabilities for fire induced initiating events, and to determine if the risk associated with the loss of decay heat removal can be reduced in a cost-effective manner.

The results of this evaluation support closure of Unresolved Safety Issue (USI) A-45.

4.9.1 USI A-45 Background

The primary objectives of USI A-45 are to evaluate the adequacy of the decay heat removal systems, determine the benefit of providing an alternate means of decay heat removal, and assess the benefit and cost of alternative measures (See Section 3.2.1)

4.9.1.1 Evaluation

The CCNPP IPE addressed the issue of decay heat removal (DHR). With the exception of external events, the CCNPP IPE concluded that USI A-45 is resolved for Calvert Cliffs. This section provides an evaluation of the CCNPP DHR functions with respect to the CCFPRA.

During the development of the IPEEE, several fire scenarios revealed potential vulnerabilities. These potential vulnerabilities are addressed by introducing new fire recovery human actions and developing procedure changes to address the control of transient ignition sources in the cable chases and to control and in some cases inspect critical fire barriers. These improvements are identified in Section 7. As a result of these improvements, no other weaknesses are identified.

A further review of fire initiating event contributions in the CCFPRA reveals that there are no additional unique decay heat removal vulnerabilities for events initiated at power for CCNPP that were not already identified in the CCNPP IPE. Based on these results, the CCFPRA has effectively evaluated the DHR function at CCNPP and USI A-45 is resolved.

4.9.2 Generic Issue 57, "Effects of Fire Protection System Actuation"

4.9.2.1 Generic Issue 57 Background

NUREG-1407 requires that the IPEEE address Generic Issue-57, "Effects of Fire Protection System Actuation on Safety-Related Equipment". The NRC staff expects that if a vulnerability in this area is identified through the IPEEE, that the issue will be addressed rather than waiting for the GI-57 resolution. It is expected that during walkdowns information will be collected on whether actuated fire protection systems would spray safety-related equipment and institute protective measures to prevent damage.

4.9.2.1 Evaluation

BGE has reviewed GI-57 and related Information Notices for comparisons with site configurations and to identify problem areas. It was determined that single failure of a fire protection system would lead to fire suppressant release in only two cases: Halon systems (smoke detector actuated), and deluge systems (manually-actuated or heat detector actuated). Halon release will not cause equipment damage, although the ventilation dampers to that room will have to be re-opened. Deluge system actuation will release water onto the protected equipment. Deluge systems are installed in the large outdoor transformers. These deluge systems have, in the past, inadvertently actuated with no affect to operability of equipment. The impact of the loss of ventilation on the cable spreading rooms and switchgear rooms is explicitly modeled in the CCFPRA. As a result of BGE's review of GI-57, several changes were implemented.

1) E-406, Electrical Design Standard (Ref 4-20), was changed to require conduit seals. Holes are also required to be drilled in the bottom of junction boxes in the event of water entry.

2) All compartments protected by automatic sprinklers with open pathways were previously walked down in November 1996 to verify and update of results conducted on 1984 in response to Information Notice 83-41 (Ref 4-21). Electrical equipment in those compartments that could be damaged by water intrusion were identified. As a result of the 1984 walkdowns, modifications (FCR 84-1014, Ref. 4-23) were performed to protect equipment from minor water leakage between fire areas, through sealing conduits and cabinets, and equipping isolation switches with water shields.

The issue and associated information notices are concerned with a non-fire scenario that actuates suppression and causes equipment damage. Most plant systems are wet pipe systems. If a wet pipe system inadvertently actuated (i.e., due to high temperatures or equipment failure) spray down is not considered to be a problem due to electrical installation configuration.

Mechanical equipment (such as pumps, valves, and piping) are generally close cased so that water intrusion is not possible. Insulation is provided for hot metal components so that cold sprinkler water will not result in thermal shock. Likewise, electrical equipment is protected by casing or water shields. Cable and conduit is protected by insulation, covered trays, and seals.

It is concluded that no additional actions are required to address the issue of spurious actuation of fire suppression systems. These findings were verified during the compartment walkdowns performed in conjunction with CCFPRA detailed fire modeling efforts and are documented in the Section 4 compartment evaluation attachments.

4.10 References

4-1 USNRC, Generic Letter No. 88-20, Supplement 4, "Individual Plant Examination of External Events (IPEEE) for Severe Accident Vulnerabilities", June 28, 1991

4-2 USNRC, NUREG-1407, "Procedural and Submittal Guidance for the Individual Plant Examination of External Events for Severe Accident Vulnerabilities", Final Report, June 1991

4-3 EPRI "Fire PRA Implementation Guide," EPRI TR-105928, Final Report, December 1995

4-4 EPRI "Fire Events Data Base for U.S. Nuclear Power Plants, NSAC-178L, December 1991

4-5 EPRI "Fire-Induced Vulnerability Evaluation (FIVE), EPRI TR-10037, Final Report, April 1992

4-6 BGE, "Calvert Cliffs Individual Plant Examination Summary Report," December 1993

4-7 BGE, Calvert Cliffs Nuclear Power Plant Fire Hazards Analysis Summary Document, Revision 0, June 4, 1997

4-8 BGE, Interactive Cable Analysis for Calvert Cliffs Nuclear Power Plant Unit 1, Revision 6, October 18, 1996

4-9 BGE, Interactive Cable Analysis for Calvert Cliffs Nuclear Power Plant Unit 2, Revision 5 October 18, 1996

4-10 CCNPP Fire Protection Evaluation Program

4-11 CCNPP, Nuclear Program Directive, "Fire Protection Program," SA-1, December 18, 1996

4-12 USNRC NUREG-1335, "Individual Plant Examination: Submittal Guidance," August 1989

4-13 USNRC NUREG/CR-4840, "Procedures for the External Event Core Damage Frequency Analysis for NUREG-1150, November 1990

4-14 BGE, Calvert Cliffs Nuclear Power Plant Units 1 and 2 Combustible Loading Re-Analysis Calculation, July 1993

4-15 EPRI Economic Risk Management Models for Electrical Equipment Containing PCBs

4-16 "The SFPE Handbook of Fire Protection Engineering", 2nd Edition, Society of Fire Protection Engineering

4-17 BGE, Independent Spent Fuel Storage Installation Updated Environmental Report, Revision 1

4-18 BGE, Independent Spent Fuel Storage Installation Updated Evaluation Report, Revision 4

4-19 PLG RISKMAN PRA Workstation Software, Revision 8.0

4-20 BGE, Calvert Cliffs Nuclear Power Plant, E-406, 61-406-A SEC.A05.4 Sheet 1, Revision 0, March 25, 1992

4-21 USNRC Information Notice 83-41, "Actuation of Fire Suppression System Causing Inoperability of Safety-Related Equipment," June 22 1983

4-23 BGE, Calvert Cliffs Nuclear Power Plant, Facility Change Request 84-1014, modifications as a result of USNRC IE Notice 83-41 walkdowns

TABLE 4.6a

Unaccounted Frequencies & Sequence Count

MFF	IE	Unaccounted	Frq	Truncation Limit	Sequence Count	Unaccounted Greater Than 1E-7	Sequence Count Less Than 100
AUX	A225F1	6.77E-08	1.19E-04	1.00E-13	458		
AUX	A225F2	6.76E-08	1.19E-04	1.00E-13	466		
AUX	A225F3	4.09E-07	2.85E-04	1.00E-12	283	Yes	
AUX	A225F4	1.24E-08	2.52E-06	1.00E-13	136		
AUX	A318F8	9.94E-08	6.28E-06	1.00E-12	113		
AUX	A419F1	3.92E-08	3.92E-05	1.00E-13	103		
AUX	A419F2	5.55E-08	3.92E-05	1.00E-13	798		
AUX	A419F3	5.77E-09	1.92E-07	1.00E-13	1516		
AUX	A419F5	8.01E-10	1.92E-07	1.00E-14	4		Yes
AUX	A423F1	2.93E-06	2.11E-03	1.00E-11	638	Yes	
AUX	A423F2	5.60E-08	7.12E-05	1.00E-13	237		
AUX	A423F3	5.68E-08	7.12E-05	1.00E-13	238		
AUX	A423F4	3.76E-08	5.92E-06	1.00E-12	20		Yes
AUX	A429F1	1.32E-06	2.84E-04	1.00E-11	551	Yes	
AUX	A512F1	1.18E-07	1.93E-04	1.00E-13	955	Yes	
AUX	A512F2	1.11E-07	1.72E-04	1.00E-13	829	Yes	
AUX	A524F1	1.56E-07	1.10E-04	1.00E-12	795	Yes	
AUX	A524F2	4.41E-08	1.03E-05	1.00E-12	469		
AUX	A524F3	1.82E-07	1.19E-04	1.00E-12	1005	Yes	
AUX	A529F1	2.01E-06	9.91E-04	1.00E-11	942	Yes	
AUX	AUX10A	6.36E-06	4.04E-03	1.00E-11	894	Yes	
AUX	AUX10B	4.92E-06	3.19E-03	1.00E-11	371	Yes	
AUX	AUX10C	5.95E-06	5.88E-03	1.00E-11	408	Yes	
AUX	AUX10D	7.56E-07	7.85E-04	1.00E-12	1127	Yes	
AUX	AUX10E	2.15E-05	6.23E-03	1.00E-10	500	Yes	
AUX	AUX20A	3.68E-05	2.21E-02	1.00E-10	2338	Yes	
AUX	AUX20B	5.99E-06	7.84E-03	1.00E-11	1110	Yes	
AUX	FCA206	6.86E-07	9.29E-04	1.00E-12	266	Yes	
AUX	FCA221	4.23E-07	3.46E-04	1.00E-12	275	Yes	
AUX	FCA227	3.26E-06	4.26E-04	1.00E-10	747	Yes	
AUX	FCA319	4.96E-06	4.19E-03	1.00E-11	441	Yes	
AUX	FCA523	2.13E-05	1.23E-02	1.00E-10	128	Yes	
AUX	FCCMB1	7.65E-08	2.62E-05	1.00E-12	110		
AUX	FCCMB2	1.68E-09	8.53E-07	1.00E-14	14		Yes
AUX	FCCMB3	9.58E-09	1.56E-06	1.00E-13	71		Yes
AUX	FCCMB4	9.83E-09	1.61E-06	1.00E-13	143		
AUX	FCCMB5	2.00E-07	5.75E-05	1.00E-12	110	Yes	
AUX	FCCMB6	6.02E-07	1.17E-05	1.00E-10	115	Yes	
AUX	FIA309	5.82E-07	4.84E-04	1.00E-12	373	Yes	
AUX	FIA312	2.94E-06	4.51E-04	1.00E-11	1215	Yes	
AUX	FIA414	2.86E-05	3.28E-03	1.00E-10	281	Yes	

TABLE 4.6a

Unaccounted Frequencies & Sequence Count (Continued)

MFF	IE	Unaccounted	Frq	Truncation Limit	Sequence Count	Unaccounted Greater Than 1E-7	Sequence Count Less Than 100
AUX	FIA416	9.13E-06	2.23E-02	1.00E-11	1041	Yes	
AUX	FIA420	3.11E-06	1.58E-03	1.00E-11	227	Yes	
AUX	FIA421	2.74E-05	2.23E-02	1.00E-10	144	Yes	
AUX	FIA422	2.70E-05	2.23E-02	1.00E-10	451	Yes	
AUX	FIA439	2.22E-06	1.24E-03	1.00E-11	459	Yes	
AUX	FIMAB2	4.01E-07	3.90E-04	1.00E-12	225	Yes	
CR	A302F1	6.60E-07	9.04E-06	1.00E-11	2938	Yes	
CR	A302F2	4.01E-07	8.28E-05	1.00E-12	454	Yes	
CR	A302F3	2.01E-08	2.34E-06	1.00E-13	326		
CR	A302F4	9.89E-08	3.12E-06	1.00E-12	1690		
CR	A302F5	1.05E-06	3.98E-05	1.00E-11	863	Yes	
CR	A302F6	7.84E-08	2.34E-06	1.00E-12	1914		
CR	A302F7	2.56E-07	4.32E-05	1.00E-12	624	Yes	
CR	A302F8	1.80E-06	1.58E-04	1.00E-11	289	Yes	
CR	A302F9	1.54E-08	2.01E-06	1.00E-13	113		
CR	A302FA	4.75E-07	9.31E-06	1.00E-11	1113	Yes	
CR	A302FB	2.82E-08	2.45E-06	1.00E-13	181		
CR	A302FC	2.62E-07	3.48E-06	1.00E-12	4867	Yes	
CR	A302FM	2.31E-08	3.71E-06	1.00E-13	38		Yes
CR	A302FN	1.24E-06	6.29E-04	1.00E-12	1548	Yes	
CR	A306F1	3.18E-07	1.58E-05	1.00E-12	200	Yes	
CR	A306F2	1.88E-05	1.16E-04	1.00E-10	1993	Yes	
CR	A306F3	2.25E-07	2.41E-05	3.00E-13	1901	Yes	
CR	A306F4	5.83E-07	2.01E-04	1.00E-12	2340	Yes	
CR	A306F5	6.69E-07	2.01E-04	1.00E-12	373	Yes	
CR	A306F6	1.66E-09	1.74E-07	1.00E-14	1447		
CR	A306F7	1.33E-07	4.37E-06	1.00E-12	5851	Yes	
CR	A306F8	7.61E-07	7.59E-06	1.00E-11	2360	Yes	
CR	A306F9	5.36E-08	5.58E-06	1.00E-13	417		
CR	A306FA	2.78E-06	4.07E-05	1.00E-11	1269	Yes	
CR	A306FB	2.49E-08	7.36E-07	1.00E-13	49		Yes
CR	A306FC	1.14E-07	8.52E-07	1.00E-12	6214	Yes	
CR	A306FD	3.01E-06	3.76E-05	1.00E-11	5213	Yes	
CR	A306FE	9.76E-08	1.47E-06	1.00E-12	146		
CR	A306FF	4.19E-06	3.76E-05	1.00E-10	373	Yes	
CR	A306FG	6.75E-09	7.36E-07	1.00E-14	584		
CR	A306FH	2.24E-06	3.76E-05	1.00E-11	693	Yes	
CR	A306FI	6.91E-09	7.36E-07	1.00E-14	931		
CR	A306FJ	6.16E-08	3.22E-06	1.00E-12	263		
CR	A306FK	5.24E-08	3.68E-06	1.00E-13	1060		
CR	A306FL	2.47E-07	1.43E-06	1.00E-12	1328	Yes	

TABLE 4.6a

Unaccounted Frequencies & Sequence Count (Continued)

MFF	IE	Unaccounted	Frq	Truncation Limit	Sequence Count	Unaccounted Greater Than 1E-7	Sequence Count Less Than 100
CR	A306FM	6.53E-07	7.11E-06	1.00E-11	4833	Yes	
CR	A306FN	1.76E-06	6.262E-04	1.00E-12	1326	Yes	
CR	A306FO	9.47E-08	2.18E-05	1.00E-13	258		
CR	A405F1	6.83E-06	1.97E-04	1.00E-10	1383	Yes	
CR	A405F2	3.33E-06	1.58E-04	1.00E-10	714	Yes	
CR	A405F3	2.06E-06	3.94E-05	1.00E-10	238	Yes	
CR	A405F4	4.17E-06	1.18E-04	1.00E-10	551	Yes	
CR	A405F5	1.76E-06	5.91E-05	1.00E-10	339	Yes	
CR	A405FM	5.40E-06	3.55E-04	1.00E-10	1242	Yes	
CR	A405FN	1.21E-05	1.34E-03	1.00E-10	3074	Yes	
CR	FCA300	4.09E-06	6.20E-04	3.00E-12	656	Yes	
CR	FI1C03	3.82E-06	7.88E-05	1.00E-10	1227	Yes	
CR	FI1C04	7.98E-06	1.18E-04	1.00E-10	3168	Yes	
CR	FI1C06	3.20E-06	1.18E-04	1.00E-11	1352	Yes	
CR	FI1C07	4.82E-06	1.58E-04	1.00E-10	905	Yes	
CR	FI1C09	2.14E-06	7.88E-05	1.00E-10	407	Yes	
CR	FI1C10	1.95E-06	7.88E-05	1.00E-10	401	Yes	
CR	FI1C13	8.69E-07	7.88E-05	1.00E-11	4690	Yes	
CR	FI1C17	3.92E-06	7.88E-05	1.00E-10	403	Yes	
CR	FI1C18	1.32E-05	7.88E-05	1.00E-10	1538	Yes	
CR	FI1C19	1.83E-05	7.88E-05	1.00E-09	363	Yes	
CR	FI1C20	4.09E-06	7.88E-05	1.00E-10	654	Yes	
CR	FI1C34	6.85E-07	3.94E-05	1.00E-11	1293	Yes	
CR	FI2C05	4.18E-06	2.36E-04	1.00E-10	963	Yes	
CR	FI2C09	2.18E-06	7.88E-05	1.00E-10	403	Yes	
CR	FI2C13	1.90E-06	7.88E-05	1.00E-10	391	Yes	
CR	FI2C17	2.34E-06	7.88E-05	1.00E-10	403	Yes	
CR	FIA301	5.98E-05	8.48E-04	5.00E-10	307	Yes	
CR	FIA304	4.69E-06	7.92E-04	1.00E-11	469	Yes	
CR	FIA305	3.44E-05	8.84E-04	1.00E-09	319	Yes	
CR	FIA307	5.14E-06	8.14E-04	1.00E-11	378	Yes	
CR	FIC18A	1.24E-06	3.94E-05	1.00E-10	467	Yes	
CR	FIC18B	1.26E-06	3.94E-05	1.00E-10	324	Yes	
CR	FIC19C	5.77E-07	3.94E-05	1.00E-11	1337	Yes	
CR	FIC20A	1.24E-06	3.94E-05	1.00E-10	377	Yes	
CR	FIC20B	1.26E-06	3.94E-05	1.00E-10	324	Yes	
CR	FIC24A	3.00E-06	7.88E-05	1.00E-10	587	Yes	
INTK	INTKF1	6.81E-08	8.10E-05	1.00E-13	2089		
INTK	INTKF2	7.12E-08	8.10E-05	1.00E-13	4782		
INTK	INTKF3	1.04E-07	1.62E-04	1.00E-13	2366	Yes	
INTK	INTKF4	5.00E-08	8.10E-05	1.00E-13	405		

TABLE 4.6a

Unaccounted Frequencies & Sequence Count (Continued)

MFF	IE	Unaccounted	Frq	Truncation Limit	Sequence Count	Unaccounted Greater Than 1E-7	Sequence Count Less Than 100
INTK	INTKF5	5.41E-08	8.10E-05	1.00E-13	251		
TB	A226F1	3.04E-07	1.34E-04	1.00E-12	344	Yes	
TB	A226F2	2.77E-07	1.34E-04	1.00E-12	298	Yes	
TB	A226F3	1.97E-07	6.67E-05	1.00E-12	1572	Yes	
TB	A311F1	6.22E-07	4.62E-05	1.00E-11	522	Yes	
TB	A311F2	3.22E-06	6.97E-04	1.00E-11	1213	Yes	
TB	A311F3	7.48E-07	9.18E-04	1.00E-12	1236	Yes	
TB	A317F1	2.96E-06	2.03E-04	1.00E-11	5450	Yes	
TB	A317F2	9.47E-06	3.46E-04	1.00E-10	896	Yes	
TB	A317F3	1.22E-06	4.50E-05	1.00E-11	1265	Yes	
TB	A317F4	1.36E-05	2.84E-03	1.00E-10	237	Yes	
TB	A317F5	1.17E-06	1.23E-04	1.00E-11	479	Yes	
TB	A317F6	1.17E-06	1.23E-04	1.00E-11	479	Yes	
TB	A317F7	1.38E-05	7.44E-04	1.00E-10	549	Yes	
TB	A317F8	8.98E-07	3.19E-05	1.00E-11	960	Yes	
TB	A317F9	3.06E-06	3.19E-05	1.00E-10	421	Yes	
TB	A317FA	3.06E-06	5.63E-04	1.00E-11	1223	Yes	
TB	A317FB	1.04E-05	2.54E-04	1.00E-10	685	Yes	
TB	A317FC	9.91E-07	1.60E-04	1.00E-11	343	Yes	
TB	A407F1	6.83E-07	4.62E-05	1.00E-11	424	Yes	
TB	A407F2	1.53E-06	2.32E-04	1.00E-11	399	Yes	
TB	A407F3	1.99E-06	4.65E-04	1.00E-11	357	Yes	
TB	A407F4	5.22E-07	5.93E-04	1.00E-12	746	Yes	
TB	A430F1	8.05E-07	5.80E-05	1.00E-11	917	Yes	
TB	A430F2	6.86E-06	4.14E-04	1.00E-10	274	Yes	
TB	A430F3	7.26E-08	9.24E-05	1.00E-13	926		
TB	A430F4	6.26E-06	3.03E-04	1.00E-10	504	Yes	
TB	A430F5	5.60E-07	5.77E-04	1.00E-12	360	Yes	
TB	A430F6	2.63E-06	4.55E-04	1.00E-11	2448	Yes	
TB	A430F7	2.80E-06	4.96E-04	1.00E-11	1808	Yes	
TB	A430F8	3.91E-06	2.89E-03	1.00E-11	1452	Yes	
TB	FIT605	5.81E-07	6.97E-04	1.00E-12	868	Yes	
TB	T603F1	1.63E-07	8.83E-05	1.00E-12	652	Yes	
TB	T603F2	7.93E-08	1.94E-05	1.00E-12	1447		
TB	T603F3	5.78E-07	8.83E-05	1.00E-11	1398	Yes	
TB	T603F4	3.03E-07	1.94E-05	1.00E-11	1130	Yes	
TB	T603F5	2.59E-07	2.17E-05	1.00E-11	412	Yes	
TB	T603F6	4.44E-08	5.74E-05	1.00E-13	282		
TB	TBALLB	3.15E-06	2.91E-04	1.00E-10	5596	Yes	
TB	TBMFW1	3.94E-05	3.92E-03	1.00E-10	1407	Yes	
TB	TBMFW2	2.28E-05	3.92E-03	1.00E-10	2393	Yes	

TABLE 4.6a

Unaccounted Frequencies & Sequence Count (Continued)

MFF	IE	Unaccounted	Frq	Truncation Limit	Sequence Count	Unaccounted Greater Than 1E-7	Sequence Count Less Than 100
YRD	F0CEDG	1.03E-05	2.50E-02	1.00E-11	1546	Yes	
YRD	F1AEDG	9.11E-06	2.63E-02	1.00E-11	2852	Yes	
YRD	FCYRD1	1.45E-07	6.52E-06	1.00E-11	600	Yes	
YRD	FCYRD2	5.77E-07	5.86E-05	1.00E-11	1503	Yes	
YRD	FCYRD3	2.55E-08	1.75E-06	1.00E-12	584		
YRD	FCYRD4	3.02E-07	4.78E-05	1.00E-12	933	Yes	
YRD	FCYRD5	2.31E-08	1.65E-06	1.00E-12	1382		
YRD	FCYRD6	1.23E-07	3.86E-05	1.00E-12	136	Yes	
YRD	FCYRDA	4.85E-05	1.12E-02	1.00E-10	1405	Yes	
YRD	FCYRDB	3.86E-05	1.14E-02	1.00E-10	1110	Yes	
YRD	FCYRDC	5.99E-06	3.27E-03	1.00E-11	1003	Yes	
YRD	FCYRDD	1.29E-06	4.24E-03	1.00E-12	1869	Yes	
YRD	FFPPHS	8.74E-06	1.81E-02	1.00E-11	720	Yes	
				TOTAL:	184075		

TABLE 4.6b
Top Events

Top Event	Description
AA	4KV Bus 11 energized
AB	4KV Bus 14 energized
AC	4KV Bus 21 energized
AD	4KV Bus 24 energized
AE	4KV Bus 12 energized
AF	4KV Bus 13 energized
AL	Auxiliary Feedwater maintains system integrity from the 1st hour to the 24th hour
AQ	Op starts boration given control rods stuck
AU	Fraction of time a non-PORV challenging MTC exists over core life
AV	Large Negative Moderator Temperature Coefficient (MTC) Allows ATWS Mitigation
BS	Turbine Bypass Valves modulate
BV	Turbine Bypass Valves quick open (4-of-4)
CA	OP aligns condensate or Fire Protection to either Service Water or Component Cooling Water given more than enough time
CB	OP aligns condensate or Fire Protection to either Service Water or Component Cooling Water within 95-119 minutes of plant trip
CD	OP aligns condensate or Fire Protection to Component Cooling Water within 24 hours, given CKV-105 fails
CI	OP starts stand-by NSR Chilled Water Pump
CL	Cable Spreading Room Dampers close on a CSR fire
CR	Operator abandons the Control Room due to fire or smoke
CS	CNTMT Spray Header 11 operates as required
CT	CNTMT Spray Header 12 operates as required
CV	2of3 Charging Pumps operate to mitigate an ATWS or to support OTCC
CX	ESFAS Logic Cabinet A cooling operates
CY	ESFAS Logic Cabinet B cooling operates
DA	125VDC Bus 11 energized for 4 hrs
DB	125VDC Bus 12 energized for 4 hrs
DC	125VDC Bus 21 energized for 4 hrs
DD	125VDC Bus 22 energized for 4 hrs
DL	Safety Injection provides sufficient flow
DM	Demineralized water supplies water to Service Water and Component Cooling Water
DV	ADVs modulate, given the ADVs quick open
DW	ADVs quick open
E1	120VAC Vital Panel 11 energized for 4 hrs
E2	120VAC Vital Panel 12 energized for 4 hrs
E3	120VAC Vital Panel 13 energized for 4 hrs
E4	120VAC Vital Panel 14 energized for 4 hrs
E5	208/120VAC Instrument Bus 11 energized
E6	208/120VAC Instrument Bus 12 energized
EA	CSAS Channel A actuates
EB	CSAS Channel B actuates
ES	48VDC Power supply 1R01A energized

TABLE 4.6b
Top Events (Continued)

Top Event	Description
EW	120VAC Vital Panel 11 energized long term
EX	120VAC Vital Panel 12 energized long term
EY	120VAC Vital Panel 13 energized long term
EZ	120VAC Vital Panel 14 energized long term
F1	Auxiliary Feedwater delivers adequate flow
F3	Auxiliary Feedwater has adequate inventory
F7	Auxiliary Feedwater Pump 13 provides adequate flow
F9	Auxiliary Feedwater Pump 23 supplies Unit 1 with Auxiliary Feedwater
FC	Auxiliary Feedwater Turbine Pump Room cooling operates
FF	Operator aligns Auxiliary Feedwater Turbine Pump Room Emergency Cooling Fans
FG	Operator recovers all Auxiliary Feedwater Pumps from testing
FH	Operator starts Auxiliary Feedwater Pump 13 from the Control Room
FJ	Operator recovers one Auxiliary Feedwater Turbine Pump from testing
FN	Operator aligns N2 or starts Salt Water Air Compressors to Auxiliary Feedwater CVs
FO	Operator starts U-2 Salt Water Air Compressors and aligns air to U-2 Auxiliary Feedwater CVs
FP	Fuel Oil Storage Tank 21 provides sufficient fuel oil to EDG 2A & 2B
FQ	Fuel Oil Storage Tank 11 provides sufficient fuel oil to EDG 1B & 0C
FT	Feedwater trips (SGIS)
FW	Operator mans the Auxiliary Shutdown Panel following Control Room abandonment
GE	EDG 1A starts & provides power to 4KV Bus 11
GF	EDG 2B starts & provides power to 4KV Bus 24
GG	EDG 1B starts & provides power to 4KV Bus 14
GH	EDG 2A starts & provides power to 4KV Bus 21
GJ	EDG 0C starts & provides power to a 4KV Bus
GW	Salt Water & Service Water HDR 22 operate
GZ	Salt Water HDR 21 & Service Water HDR 21 operate
H3	OP starts standby Switchgear HVAC train within 45 min.
H4	OP starts standby Control Room HVAC
H5	OP alternates supply to 4KV busses with 13KV on loss of normal 13KV Bus feed to the 4KV bus
H6	OP cross-connects MCC 104 & 114
H9	OP x-connects U2 MCCs and aligns one U2 vital 120VAC Bus to the back-up instrument bus
HA	HP Safety Injection aux header operates
HB	HP Safety Injection main header operates
HC	Control Room HVAC dampers remain open
HF	27' Switchgear Room HVAC dampers remain open
HG	45' Switchgear Room HVAC dampers remain open
HH	CR/CSR HVAC header operates
HL	Unit 2 Cable Spreading Room HVAC dampers remain open
HP	Operator recovers CR/CSR dampers within 3 hrs of detection
HR	Unit 1 Cable Spreading Room HVAC dampers remain open
HS	SWGR HVAC header operates
HU	OP re-throttles Auxiliary Feedwater flow given no flow exists in 3 out of 4 Auxiliary Feedwater flow paths

TABLE 4-6b
Top Events (Continued)

Top Event	Description
HV	OP locally opens ECCS Cooler Salt Water CVs
HW	HP Safety Injection 12 provides adequate flow to Aux Hdr
HX	Operator controls Auxiliary Feedwater flow from Control Room
HZ	OP locally ventilates both switchgear rooms with temporary fans
I1	Salt Water AC Header 11 supplies adequate air
I2	Salt Water AC Header 12 supplies adequate air
IA	Steam Generator Isolation Signal Ch A actuates
IB	Steam Generator Isolation Signal Ch B actuates
IC	Operator places Component Cooling Water and Service Water Throttle valves in the Open position within 10 minutes of loss of 1Y01 or 1Y02 (Pre-Trip)
ID	120VAC Vital Panel fails (Common Cause Inverter failure) after occurrence of a 120VAC Inverter Initiating Event
IG	Operator successfully controls Pressurizer Level on loss of 1Y01 or 1Y02 (due to L12V1 or 2 initiating event)
IH	OP starts both Salt Water Air Compressors during a (LOOP or Non-LOOP)
IL	Salt Water Air Compressor 11 Header maintains integrity
IN	Salt Water Air Compressor 12 Header maintains integrity
IP	SR & NSR CA systems do not become contaminated and common Salt Water Air Compressors piping maintains integrity
IZ	Salt Water Air Compressor Headers 11 and 12 supply adequate air
JA	ESFAS Sequencer 11 actuates, no LOCA, all support available
JB	ESFAS Sequencer 14 actuates
K3	Component Cooling Water HX 11 cools Component Cooling Water
K4	Component Cooling Water HX 12 cools Component Cooling Water
K5	Component Cooling Water maintains inventory
KE	Component Cooling Water Pump 11 or 13 operates
KH	OP starts Component Cooling Pump 13, given OP opens Component Cooling HX 12
KI	OP aligns stand-by Component Cooling Water HX within 2 hours of a RAS
KJ	OP opens Component Cooling HX 12 and starts Component Cooling Pump 13 given HX 11 flow path fails
KL	Component Cooling Water HX flow paths remain open
KM	Component Cooling Water HX 11 flow path remains open
KN	Component Cooling Water HX 12 flow path remains open
KS	Component Cooling Water flow path to the RCP Seals exists
KX	Component Cooling Water Pump 11 operates
KY	Component Cooling Water Pump 12 operates
KZ	Component Cooling Water Pump 13 operates
LF	Op initiates low pressure feed using Condensate
M1	480VAC MCC 104R energized
M2	480VAC MCC 114R energized
M3	480VAC MCC 204R energized
M7	480VAC MCC 106T energized
M8	480VAC MCC 116T energized
MC	Main Condensate provides adequate flow
MF	Operator aligns Salt Water to Service Water within 150 minutes of plant trip
MG	Operator aligns Salt Water to Service Water within 114-124 minutes of plant trip

TABLE 4-6b

Top Events (Continued)

Top Event	Description
MH	Operator recovers failed steam admission line to Auxiliary Feedwater turbine driven pumps
MN	MFW is adequate after reactor trip
MP	MFW ramps back after reactor trip
MS	Main Steam Isolation Valves shut on SGIS
MT	Op trips Steam Generator Feed Pumps or throttles flow after S/G high level trip w/in 30 minutes
MV	MOV 659 and 660 remain open
MX	Operator cross-connects MCC 104 & 114
N1	480VAC Bus 11A energized
N2	480VAC Bus 11B energized
N3	480VAC Bus 14A energized
N4	480VAC Bus 14B energized
N5	480VAC Bus 21A energized
N6	480VAC Bus 21B energized
N7	480VAC Bus 24A energized
N8	480VAC Bus 24B energized
NR	Non-safety related IA/PA supplies Unit 1 air loads
NS	Non-safety related IA/PA supplies Unit 2 air loads
OP	Grid fails to remain energized following a plant trip and over the next 24 hours
OT	Operator aligns for OTCC and opens PORVs
PG	Fire Protection provides adequate make-up to Service Water/Component Cooling Water
PH	Operator isolates PORVs on lowering RCS pressure
PN	At least one PORV re-closes
PS	Both PORVs open as required
PT	Reactor Vessel Failure from PTS
PV	Both PORVs re-close on low RCS pressure
Q1	OP recovers Auxiliary Feedwater at onset (w/in 10 minutes) of a spurious AFAS block
Q5	Operators fail Auxiliary Feedwater Pump Alignment actions (global failure)
Q6	Operators fail Auxiliary Feedwater flow control actions (global failure)
QC	U-4000-11 Service Transformer Provides Adequate Current
QD	U-4000-21 Service Transformer Provides Adequate Current
QE	U-4000-12 Service Transformer Provides Adequate Current
QF	U-4000-22 Service Transformer Provides Adequate Current
QQ	OP Recovers spurious ESFAS
QZ	OP recovers Auxiliary Feedwater following a spurious AFAS block
R3	Salt Water header 11 is unavailable
R4	Salt Water header 12 is unavailable
RA	RAS Channel A actuates
RB	RAS Channel B actuates
RE	East RWT Header operates
RH	Operator manually starts Safety Injection equipment following SIAS failure
RI	Operator recovers ADV control after long term failure of Reactor Regulating System

TABLE 4-6b

Top Events (Continued)

Top Event	Description
RL	Service Water maintains adequate inventory
RQ	Late Manual Reactor Trip successful.
RR	Reactor Reg. System Channel X controls the Atmospheric Dump Valves (ADVs) and Turbine Bypass Valves (TBVs)
RS	Reactor Trip actuates
RU	Pressurizer SRVs open as required on high RCS pressure
RV	Safety Relief Valves (SRVs) re-close on low RCS pressure
S1	Salt Water train 11 supplies flow
S2	Salt Water train 12 supplies flow
S3	Service Water train 11 supplies flow
S4	Service Water train 12 supplies flow
SA	SIAS Channel A actuates
SB	SIAS Channel B actuates
SC	Common Salt Water discharge header operates as required
SG	Hydrogen purge line operates
SH	Penetrations less than or equal to 4" function
SI	Penetrations greater than 4" function
SL	RCP seals remain intact
SP	OP starts a stand-by Component Cooling pump within 30 minutes to prevent RCP seal challenge
SR	Cntmt Normal Sump Drain Line isolates on a LOCA
ST	OP starts 13 Salt Water Pump within 30 minutes
SV	Main Steam Safety Relief Valves open on demand (8of16)
SW	Main Steam Relief Valves close
T1	Condensate Storage Tanks 11 and 21 do not spuriously make up to main condensate
TA	Service Water Turbine Building header 11 provides cooling
TB	Service Water Turbine Building header 12 provides cooling
TE	East Containment Sump Supply Header operates
TF	Auxiliary Feedwater TURB PP 11 provides adequate flow
TG	Auxiliary Feedwater Pump 12 provides adequate flow
TH	Unavailable Salt Water HDR is recovered within 4 hrs
TT	Turbine Stop and Control Valves shut
TW	West Contmt Sump Supply Header operates
TX	Turbine Trip Bus energized
UA	UV Channel A actuates
UB	UV Channel B actuates
UQ	OPs do not underfill S/Gs when Auxiliary Feedwater flow control is lost
V1	ECCS Pump Room Air Cooler 11 operates
V2	ECCS Pump Room Air Cooler 12 operates
V5	ECCS Pump Room Exhaust Fans Operate as Required
VB	Operator re-loads key equipment shed as part of AOP-9 following Control Room abandonment
VC	Condenser Vacuum maintained
VH	Service Water Header 11 is unavailable

TABLE 4-6b

Top Events (Continued)

Top Event	Description
VI	Service Water Header 12 is unavailable
VL	Fire Brigade suppresses a Control Room panel fire following Control Room evacuation prior to further Control Room panel loss
VM	OPS starts ECCS Coolers and aligns CVs open from the Control Room
W3	Salt Water header 21 is unavailable
W4	Salt Water header 22 is unavailable
WJ	OPS secures sampling line-up
WY	Containment Air Coolers (CACs) operate as required (2 of 4)
XA	125VDC Bus 11 energized long term
XB	125VDC Bus 12 energized long term
XC	125VDC Bus 21 energized long term
XD	125VDC Bus 22 energized long term
XW	OP supplies a 120VAC Vital Panel from 208/120 VAC Instrument Bus
Y1	13KV Bus 11 energized
Y2	13KV Bus 21 energized
Y3	13KV/4KV Facility A related components do not fault
Y4	13KV/4KV Facility B related components do not fault
Z1	Demineralized Water Storage Tank (DWST) fails due to tornado missile
Z2	Condensate Storage Tank (CST) #21 fails due to tornado missile
Z3	Condensate Storage Tank (CST) #11 fails due to tornado missile
Z4	#11 Pretreated Water Storage Tank (PTWST) fails due to a tornado missile
Z5	#12 Pretreated Water Storage Tank (PTWST) fails due to a tornado missile
Z6	EDG 0C fails due to tornado missile
Z7	EDG 2A fails due to a tornado missile
Z8	EDG 1B fails due to a tornado missile
Z9	#11 Fuel Oil Storage Tank (FOST) fails due to a tornado missile
ZA	#11 Refueling Water Storage Tank (RWT) fails due to a tornado missile
ZB	Control Room HVAC and U1 Switchgear Room HVAC fail due to tornado
ZC	Unit 1 Service Water Head Tanks fail due to a tornado missile
ZD	#21 Refueling Water Storage Tank (RWT) fails due to a tornado missile
ZE	EDG 2B fails due to a tornado missile
ZF	Unit 2 Switchgear HVAC fails due to a tornado
ZG	Unit 2 Service Water Head Tanks fail due to tornado missile
ZH	Tank Farm receives point strike and the Fire Protection Building fails

TABLE 4.6c

Top 100 Sequences

IE	MFF	Initiator	Core Damage Frequency	PDS	Sequence	Sequence Importance	Cumulative Importance
1	Turbine	TBALLB	2.61E-06	HRIF	GE3*RL9	3.58%	3.58%
2	Control Room	FI1C04	1.49E-06	HHIP	FW1*(1-(HA4*HBD*HWW))	2.04%	5.62%
3	Turbine	TBALLB	1.46E-06	HRIO	CR2	1.99%	7.61%
4	Turbine	TBALLB	1.46E-06	HRIF	HF3	1.99%	9.60%
5	Control Room	FIA305	1.29E-06	HHIP	HXA*UQ5	1.77%	11.37%
6	Control Room	FIA305	1.09E-06	MBIP	SLR*HA4	1.49%	12.86%
7	Control Room	A405FN	1.02E-06	HRIF	CR1*VB1*(1-(RL9+R31+WY4+HX7*SRA+SW3*PT6*HX7))	1.39%	14.26%
8	Turbine	TBALLB	1.00E-06	HHIP	F33*(1-(RL9*VM2*V11+R31*RL9+RL9*V14+R31*GGD))	1.37%	15.63%
9	Control Room	FIA301	9.92E-07	HHIP	HXA*UQ5*(1-(MG2*HB4*HWR+IH3*HB4*HWR))	1.36%	16.99%
10	Control Room	FI1C19	7.84E-07	HRIF	CR1*(1-(VL1))	1.07%	18.06%
11	Control Room	FI1C18	7.82E-07	HRIF	CR1*(1-(VL1+HX7*SRA+HSV*HX7*SG4))	1.07%	19.13%
12	Control Room	A306F2	6.32E-07	HHIP	Q51*(1-(HSN*HZ1+HB4*HWR))	0.87%	20.00%
13	Control Room	A306F2	6.23E-07	HHIP	FH2*MH2*(1-(MG2*HB4*HWR+IH3*HB4*HWR+HHH*HB4*HWR+Q11*HB4*HWR))	0.85%	20.85%
14	Turbine	TBALLB	5.92E-07	HRIF	GGD*HSJ	0.81%	21.66%
15	Control Room	A306F2	5.75E-07	HHIP	HX5*UQ5*(1-(IH3*MG2*HB4*HWR+HHH*MG2*HB4*HWR+HHH*Q11*IH3*HB4*HWR+HHH*IH3*CB2*HB4*HWR))	0.79%	22.45%
16	Turbine	TBALLB	5.52E-07	HRIF	GE3*HSO	0.76%	23.21%
17	Control Room	FI1C04	5.34E-07	HHIP	TF1*TGE*(1-(H41*HA4*HBD*HWW))	0.73%	23.94%
18	Turbine	TBALLB	5.24E-07	HRIF	GE3*GGD	0.72%	24.66%
19	Control Room	FIA301	5.00E-07	HHIP	FH5*TF1*TGC	0.69%	25.34%
20	Control Room	FI1C19	4.75E-07	HRIF	XW2*GZI*FCH	0.65%	25.99%
21	Control Room	A405FN	4.50E-07	HRIF	CR1*GF3*(1-(RL9+R31+WY4+HX7*SRA+SW3*PT6*HX7))	0.62%	26.61%
22	Control Room	FI1C03	4.30E-07	HHIP	TF1*TG5	0.59%	27.20%
23	Control Room	A306F2	4.28E-07	HHIP	FH2*HU2*(1-(HHH*Q11*IH3*MG2*HB4*HWR))	0.59%	27.78%
24	Turbine	TBALLB	4.28E-07	HHIP	GE3*TF2	0.59%	28.37%
25	Control Room	A306F2	4.21E-07	HHIP	FH2*HX5*(1-(MG2*HB4*HWR+IH3*HB4*HWR+HHH*HB4*HWR+Q11*HB4*HWR+CB2*HB4*HWR))	0.58%	28.95%
26	Control Room	A405FN	4.17E-07	HRIF	CR1*GE3*(1-(RL9+R31+WY4+HX7*SRA+SW3*PT6*HX7))	0.57%	29.52%
27	Control Room	A405FN	4.07E-07	HRIF	CR1*W41*(1-(RL9+R31+WY4+HX7*SRA+SW3*PT6*HX7))	0.56%	30.08%

TABLE 4.6c

Top 100 Sequences (Continued)

IE	MFF	Initiator	Core Damage Frequency	PDS	Sequence	Sequence Importance	Cumulative Importance
28	Control Room	FI1C13	4.04E-07	HRIF	CR1*HSU*(1-(HX7*SRA))	0.55%	30.63%
29	Control Room	FI1C19	3.94E-07	HRIF	HSI*H91*GZI*HX3*F1R	0.54%	31.17%
30	Control Room	FIC18A	3.92E-07	HRIF	CR1*(1-(VL1+HSV*HH8*HX7*SRA))	0.54%	31.71%
31	Control Room	FIC20A	3.91E-07	HRIF	CR1*(1-(VL1+SRA))	0.54%	32.24%
32	Control Room	FI1C04	3.76E-07	HHIP	FH7*HX6*F1R	0.52%	32.76%
33	Turbine	TBALLB	3.64E-07	HRIF	HSB	0.50%	33.26%
34	Aux. Bldg.	AUX20A	3.63E-07	MRIO	NR1*SL2	0.50%	33.75%
35	Control Room	A405FN	3.40E-07	HRWF	CR1*VL1*(1-(HX7*SRA+SW3*PT6*HX7))	0.47%	34.22%
36	Control Room	A405FN	3.37E-07	HRIO	CR1*FFA*(1-(HX7*SRA+SW3*PT6*HX7))	0.46%	34.68%
37	Control Room	FI1C19	3.27E-07	HRIF	XW2*GZI*HX5*F1R*V5E	0.45%	35.13%
38	Turbine	TBALLB	3.19E-07	HRIF	CR2*RL9	0.44%	35.56%
39	Control Room	FI1C18	3.11E-07	HHIP	IH1*HX2*F1R	0.43%	35.99%
40	Control Room	A405FN	3.02E-07	HRIF	CR1*GGD*HSU	0.41%	36.40%
41	Control Room	FI1C04	2.96E-07	HHIP	F3E	0.41%	36.81%
42	Control Room	FI1C19	2.94E-07	HRIF	XW2*Q11*GZI*HX5*F1R	0.40%	37.21%
43	Control Room	FIA301	2.88E-07	MBIP	SLQ*HB4*HWR	0.39%	37.61%
44	Control Room	A405FM	2.83E-07	HRIF	CR1*VB1*(1-(RL9))	0.39%	37.99%
45	Turbine	TBALLB	2.79E-07	HRIF	R41*HSJ	0.38%	38.38%
46	Control Room	FIA305	2.61E-07	HBIP	HXA*UQ5*HA4	0.36%	38.73%
47	Control Room	A306FF	2.49E-07	MRIO	PVZ*PHZ	0.34%	39.08%
48	Turbine	TBALLB	2.47E-07	HRIF	R41*GE3	0.34%	39.41%
49	Control Room	FI1C19	2.44E-07	HHIP	Q11*HX5*UQ5	0.33%	39.75%
50	Yard	FCYRDA	2.38E-07	HHIP	GE3*TF1*TGA*F9E	0.33%	40.07%
51	Aux. Bldg.	AUX20A	2.34E-07	HRIF	N1Z*N3Z*N4Z*N5Z*F1R	0.32%	40.39%
52	Aux. Bldg.	AUX20A	2.34E-07	HRIF	N1Z*N3Z*N5Z*N64*F1R	0.32%	40.71%
53	Control Room	FIA301	2.30E-07	HHIP	FH5*FF2*FCH	0.31%	41.03%
54	Control Room	A306F2	2.29E-07	HHIP	FH2*TF1*TGC	0.31%	41.34%
55	Turbine	A317F2	2.23E-07	HHIP	TF1*TGA*F9E	0.31%	41.65%
56	Turbine	A317F2	2.13E-07	HHIP	TF1*TGA*FO2	0.29%	41.94%
57	Control Room	FI1C19	2.10E-07	HRIF	XW2*Q11*QZ4*GZI*V5E	0.29%	42.23%
58	Control Room	FIC18B	2.03E-07	HRIF	CR1*HSU	0.28%	42.51%

TABLE 4.6c

Top 100 Sequences (Continued)

IE	MFF	Initiator	Core Damage Frequency	PDS	Sequence	Sequence Importance	Cumulative Importance
59	Control Room	FIC20B	2.03E-07	HRIF	CR1*HSU	0.28%	42.78%
60	Control Room	FI1C04	1.94E-07	HHIP	FH7*TF1*TGH	0.27%	43.05%
61	Control Room	FI1C03	1.94E-07	HHIP	F3E*(1-(CR1+H41*MG2*HA4*HBD*HWW))	0.27%	43.32%
62	Control Room	FI2C05	1.88E-07	HRIF	CR1*VB1*(1-(RL9))	0.26%	43.57%
63	Yard	FCYRDA	1.83E-07	HHIP	GE3*TF1*TGA*FO2	0.25%	43.82%
64	Control Room	FI1C18	1.80E-07	HRIF	GH3*GJG*XW2*HX3*F1R	0.25%	44.07%
65	Control Room	FI1C19	1.79E-07	HRIF	H31*HSR*H91*GZI*HX3*F1R	0.25%	44.31%
66	Turbine	A317F2	1.78E-07	HHIP	IH1*TF1*TGA	0.24%	44.56%
67	Control Room	A405FN	1.77E-07	HRIO	CR1*FW1*(1-(HX7*SRA+SW3*PT6*HX7))	0.24%	44.80%
68	Control Room	A405F1	1.73E-07	HRIF	CR1*GF5*(1-(RL9))	0.24%	45.04%
69	Control Room	FIA301	1.71E-07	HHIP	MG2*FH5*HXA*F1R	0.23%	45.27%
70	Control Room	FI1C07	1.68E-07	MBIP	NR1*SLZ	0.23%	45.50%
71	Control Room	FIA305	1.66E-07	HHIP	F73*TF9*TGC	0.23%	45.73%
72	Control Room	FIA305	1.64E-07	ATWS	RS4	0.23%	45.95%
73	Turbine	A317FB	1.64E-07	HHIP	TF1*TGA*F9E	0.22%	46.18%
74	Control Room	FIA301	1.62E-07	HHIP	IH3*FH5*HXA*F1R	0.22%	46.40%
75	Turbine	TBALLB	1.60E-07	HRIF	CR2*GF5	0.22%	46.62%
76	Control Room	FI1C18	1.59E-07	HRIF	W31*GJG*XW2*HX3*F1R	0.22%	46.84%
77	Turbine	A317F9	1.58E-07	MRIO	SLY*(1-(JB3+GGD+R41))	0.22%	47.05%
78	Control Room	A306FD	1.58E-07	MRIO	PVZ*PHZ*(1-(WY3+HW7))	0.22%	47.27%
79	Control Room	A405F1	1.57E-07	HRIF	CR1*VB1*(1-(RL9))	0.21%	47.49%
80	Turbine	A317FB	1.57E-07	HHIP	TF1*TGA*FO2	0.21%	47.70%
81	Aux. Bldg.	FCA227	1.51E-07	HHIP	NR1*IH1	0.21%	47.91%
82	Turbine	TBALLB	1.49E-07	HHIP	GE3*HX2	0.20%	48.11%
83	Turbine	TBALLB	1.46E-07	HRIF	CR2*VB1	0.20%	48.31%
84	Control Room	A405FN	1.43E-07	HRIF	CR1*R41*HSU	0.20%	48.51%
85	Yard	FCYRDA	1.42E-07	HHIP	GE3*IH1*TF1*TGA	0.19%	48.70%
86	Turbine	A317F9	1.42E-07	HRIO	QZ2*(1-(JB3+GGD+R41))	0.19%	48.89%
87	Turbine	T603F3	1.42E-07	MGIP	F73*F9X	0.19%	49.09%
88	Turbine	A317F7	1.38E-07	ATWS	RS4	0.19%	49.28%
89	Control Room	A405FN	1.36E-07	HRIO	CR1*TF2*TGG	0.19%	49.46%

TABLE 4.6c

Top 100 Sequences (Continued)

IE	MFF	Initiator	Core Damage Frequency	PDS	Sequence	Sequence Importance	Cumulative Importance
90	Control Room	FI1C18	1.35E-07	HHIP	TF1*TGA*FO2	0.18%	49.65%
91	Control Room	A306F2	1.32E-07	HBIP	Q51*HB4*HWR	0.18%	49.83%
92	Turbine	A317F1	1.31E-07	HHIP	TF1*TGA*F9E	0.18%	50.01%
93	Turbine	A317FB	1.31E-07	HHIP	IH1*TF1*TGA	0.18%	50.19%
94	Turbine	TBALLB	1.28E-07	HHIP	F72*TFA	0.18%	50.36%
95	Control Room	FIA301	1.28E-07	HBIP	MG2*HXA*UQ5*HB4*HWR	0.17%	50.54%
96	Yard	FCYRDA	1.27E-07	HRIF	DA1*XW2*HX3*F1R	0.17%	50.71%
97	Turbine	TBALLB	1.27E-07	HHIP	GE3*FF1	0.17%	50.89%
98	Control Room	A405F2	1.26E-07	HRIF	CR1*VB1*(1-(RL9))	0.17%	51.06%
99	Control Room	FI1C07	1.26E-07	HRIF	CR1*VB1*(1-(RL9))	0.17%	51.23%
100	Turbine	A317F1	1.25E-07	HHIP	TF1*TGA*FO2	0.17%	51.40%

TABLE 4.6d

Split Fraction Descriptions

PLANT MODEL SF	SPLIT FRACTION DESCRIPTION	AUX BLDG FIRE MFF	TB FIRE MFF	INTAKE FIRE MFF	OUTSIDE MFF FIRE	CONT RM FIRE MFF
CB2	OP aligns condensate or FP to either SRW or CCW within 95-119 minutes of plant trip, EOP-8 or LOCA	1.07E-01	1.07E-01	5.96E-02	5.96E-02	3.07E-01
CR1	Operations abandons Control Room, given Control Room Panel Fire	1.02E-02	1.02E-02	1.02E-02	1.02E-02	1.02E-02
CR2	Control Room HVAC goes into recirculation, given fire which enters HVAC	5.00E-03	5.00E-03	5.00E-03	5.00E-03	5.00E-03
DA1	125VDC Bus 11 energized for 4 hrs, LOOP and all supports available	5.75E-04	5.75E-04	5.75E-04	5.75E-04	5.75E-04
F1R	AFW delivers adequate flow, given SBO with S/G overfill	5.74E-01	5.74E-01	4.79E-01	4.79E-01	7.06E-01
F33	AFW has adequate inventory (includes op action for tank switchover) given no indication	2.11E-03	3.47E-03	2.11E-03	2.11E-03	2.11E-03
F3E	AFW has adequate inventory (includes op action for tank switchover), shutdown from the Auxiliary Shutdown Panel - otherwise boundary conditions per F31, F32 or F3D	2.51E-03	2.51E-03	2.51E-03	2.51E-03	2.51E-03
F72	AFW Pump 13 provides adequate flow, given Operators recover from all testing (FG=S), 4KV Bus 11 losses offsite power from U-4000-11, the Shutdown Sequencer Ch. A (JA) is successful, and the operators fail to manually start the AFW Pump locally or remotely	9.23E-03	9.23E-03	9.23E-03	9.23E-03	9.23E-03
F73	AFW Pump 13 provides adequate flow, given Operators recover from all testing (FG=S), 4KV Bus 11 provided offsite power from U-4000-11 (QC=S), and the operators fail to manually start the AFW Pump locally or remotely (FH=F).	8.95E-03	8.95E-03	8.95E-03	8.95E-03	8.95E-03
F9E	AFW Pump 23 supplies Unit 1 with AFW, given either (1)13 AFW PP not? & at least one of 11 & 12 AFW PPs fails or (2) 12 & 13 AFW PPs succ and 11 fails; Also DC Bus 11 avail, U2 MFW not avail, Op action to recover testing succ	5.72E-02	6.30E-02	5.72E-02	5.72E-02	7.61E-02
F9X	AFW Pump 23 supplies Unit 1 with AFW, given AFW PPs 11, 12, & 13 fail, DC Bus 11 avail, Op action to recover from testing sucessful, U2 MFW succ/fail	1.76E-01	1.79E-01	1.76E-01	1.76E-01	1.87E-01
FCH	AFW PP RM cooling operates, given a SBO condition,EOP8	3.66E-03	9.60E-03	3.66E-03	3.66E-03	3.02E-02
FF1	Operator aligns AFW Turbine Pump Room Emergency Cooling Fans, ASA	1.86E-02	1.06E-02	1.06E-02	1.06E-02	4.69E-02
FF2	Operator aligns AFW Turbine Pump Room Emergency Cooling Fans, EOP-8	9.32E-02	5.32E-02	5.32E-02	5.32E-02	2.35E-01
FFA	Operator aligns AFW Turbine Pump Room Emergency Cooling Fans, given CR evacuation and S/D from Aux S/D Panel	2.47E-02	2.47E-02	2.47E-02	2.47E-02	2.47E-02
FH2	Operator starts AFW Pump 13 locally, ASA	3.82E-02	5.41E-02	3.82E-02	3.82E-02	1.28E-01
FH5	Operator starts AFW Pump 13, given fire in the CSR and dampers isolate fire	3.82E-02	3.82E-02	3.82E-02	3.82E-02	3.82E-02

TABLE 4.6d

Split Fraction Descriptions (Continued)

PLANT MODEL SF	SPLIT FRACTION DESCRIPTION	AUX BLDG FIRE MFF	TB FIRE MFF	INTAKE FIRE MFF	OUTSIDE MFF FIRE	CONT RM FIRE MFF
FH7	Operator starts AFW Pump 13 locally, CR evacuated to Aux Shutdown Panel	1.28E-01	1.28E-01	1.28E-01	1.28E-01	1.28E-01
FO2	Operator starts U-2 SWACs and aligns air to U-2 AFW CVs, given: support available to at least one Unit 2 SWAC, a U1 SWAC is succ and other U1 SWAC succ or not?, FN (Op aligns N2 or starts SWACs) & IH (Ops starts both SWACs) are not questioned or succ	4.39E-02	6.02E-02	4.39E-02	4.39E-02	1.23E-01
FW1	Auxiliary Feedwater control transferred to the Auxiliary Shutdown Panel (Control Room Evac)	1.30E-02	1.30E-02	1.30E-02	1.30E-02	1.30E-02
GE1	EDG 1A starts & provides power to 4KV Bus 11, LOOP < 1 hour (INIT=LOOP1), ASA	2.81E-02	2.81E-02	2.81E-02	2.81E-02	2.81E-02
GE3	EDG 1A starts & provides power to 4KV Bus 11, LOOP initiating events between 2 and 4 hours, and 4KV electrical power losses which occur post trip, (ASA)	4.10E-02	4.10E-02	4.10E-02	4.10E-02	4.10E-02
GF3	EDG 2B starts & provides power to 4KV Bus 24, LOOP initiating events between 2 and 4 hours, and 4KV electrical power losses which occur post trip, (ASA)	4.42E-02	4.42E-02	4.42E-02	4.42E-02	4.42E-02
GF5	EDG 2B starts & provides power to 4KV Bus 24, LOOP > 11 hours, (ASA)	1.10E-01	1.10E-01	1.10E-01	1.10E-01	1.10E-01
GGD	EDG 1B starts & provides power to 4KV Bus 14, LOOP initiating events between 2 and 4 hours, and 4KV electrical power losses which occur post trip, EDG 2B succeeds or not ?d, (ASA)	4.39E-02	4.39E-02	4.39E-02	4.39E-02	4.39E-02
GH3	EDG 2A starts & provides power to 4KV Bus 21, LOOP initiating events between 2 and 4 hours, and 4KV electrical power losses which occur post trip, EDG 1B & 2B not ?d or succeed, ASA	4.51E-02	4.51E-02	4.51E-02	4.51E-02	4.51E-02
GJG	EDG 0C starts & provides power to a 4KV Bus, LOOP > 11 hours, EDG 1A not ?d, ASA	2.31E-01	2.36E-01	2.31E-01	2.92E-01	4.21E-01
GZI	SW HDR 21 & SRW HDR 21 operate, given one or more of the preceding SW & SRW HDRs fail (including SW/SRW 22 HDR - GW), Unit 1 Inventory (RL) not questioned, Demin fails, Unit 2 SW provides makeup to SRW	5.49E-01	6.49E-01	5.49E-01	5.49E-01	9.64E-01
H31	OP starts stand-by Switchgear HVAC train within 45 min., no LOOP	1.59E-02	8.57E-03	8.57E-03	8.57E-03	3.92E-02
H41	OP starts stand by Control Room HVAC, no LOOP	2.29E-01	2.29E-01	2.29E-01	2.29E-01	4.91E-01
H91	OP x-connects U2 MCCs and aligns one U2 vital 120VAC Bus to the back-up instrument bus, LOOP	1.01E-01	7.37E-02	7.37E-02	7.37E-02	2.07E-01
HA4	HPSI aux header operates, low sump temperature, no LLOCA, EOP8	4.82E-02	4.82E-02	4.82E-02	4.82E-02	2.02E-01
HB4	HPSI main header operates, low sump temperature, no LLOCA, aux header not questioned, EOP8	4.79E-02	4.79E-02	4.79E-02	4.79E-02	2.01E-01

TABLE 4.6d

Split Fraction Descriptions (Continued)

PLANT MODEL SF	SPLIT FRACTION DESCRIPTION	AUX BLDG FIRE MFF	TB FIRE MFF	INTAKE FIRE MFF	OUTSIDE MFF FIRE	CONT RM FIRE MFF
HBD	HPSI main header operates, low sump temperature, no LLOCA, aux header fails, (ASA)	1.12E-01	1.12E-01	1.12E-01	1.12E-01	1.73E-01
HF3	27' Switchgear Room HVAC dampers remain open, given an outside/TB fire produces smoke which challanges the outside air intake	5.00E-03	5.00E-03	5.00E-03	5.00E-03	5.00E-03
HH8	CR/CSR HVAC header operates, LOOP, OP fails to start stand-by A/C, one A/C header provided power with required support DA(DC)=S and JA(JD)=S for that header	5.04E-01	5.04E-01	5.04E-01	5.04E-01	5.04E-01
HHH	CR/CSR HVAC header operates, no LOOP, OP fails to start stand-by A/C and support to NSR chiller & one A/C header lost	5.01E-01	5.01E-01	5.01E-01	5.01E-01	5.01E-01
HSB	SWGR HVAC HDR OPERATES given both HVAC trains have support; electrical power is momentarily lost to 4KV Bus 11 and/or 14 due to LOOP or XFMER failure; Operator can start standby train; Instrument Air available or not.	1.25E-03	1.25E-03	1.25E-03	1.25E-03	1.25E-03
HSI	SWGR HVAC HDR OPERATES given HVAC train 12 is failed due to loss of support; HVAC train 11 has support; 4KV XFMER 11 and Bus 11 remain energized; Operator can start standby train; Instrument Air available or not questioned.	4.32E-02	4.32E-02	4.32E-02	4.32E-02	4.32E-02
HSJ	SWGR HVAC HDR OPERATES given HVAC train 12 is failed due to loss of support; HVAC train 11 has support; 4KV XFMER 11 failure or LOOP causes momentary loss of 4KV Bus 11; Operator can start standby train; Instrument Air available or not questioned.	4.63E-02	4.63E-02	4.63E-02	4.63E-02	4.63E-02
HSN	SWGR HVAC HDR OPERATES given HVAC train 11 is failed due to loss of support; HVAC train 12 has support; 4KV XFMER 21 and Bus 12 remain energized; Operator can start standby train; Instrument Air available or not questioned.	4.32E-02	4.32E-02	4.32E-02	4.32E-02	4.32E-02
HSO	SWGR HVAC HDR OPERATES given HVAC train 11 is failed due to loss of support; HVAC train 12 has support; 4KV XFMER 21 failure or LOOP causes momentary loss of 4KV Bus 12; Operator can start standby train; Instrument Air available or not questioned.	4.63E-02	4.63E-02	4.63E-02	4.63E-02	4.63E-02
HSR	SWGR HVAC HDR OPERATES given HVAC train 12 is failed due to loss of support; HVAC train 11 has support; 4KV XFMER 11 and 4KV Bus 11 remain energized; Operator fails to start standby train; Instrument Air available or not question.	5.01E-01	5.01E-01	5.01E-01	5.01E-01	5.01E-01

TABLE 4.6d

Split Fraction Descriptions (Continued)

PLANT MODEL SF	SPLIT FRACTION DESCRIPTION	AUX BLDG FIRE MFF	TB FIRE MFF	INTAKE FIRE MFF	OUTSIDE MFF FIRE	CONT RM FIRE MFF
HSU	SWGR HVAC HDR OPERATES given HVAC train 12 fails due to loss of support; HVAC train 11 has support; 4KV XFMER 11 failure or LOOP causes momentary loss of 4KV Bus 11; Operator fails to start standby train; Instrument Air available or not questioned.	5.04E-01	5.04E-01	5.04E-01	5.04E-01	5.04E-01
HSV	SWGR HVAC HDR OPERATES given HVAC train 11 is failed due to loss of support; HVAC train 12 has support; 4KV XFMER 21 failure or LOOP causes momentary loss of 4KV Bus 12; Operator fails to start standby train; Instrument Air available or not questioned.	5.04E-01	5.04E-01	5.04E-01	5.04E-01	5.04E-01
HU2	OP re-throttles AFW flow given no flow exists in 3 out of 4 AFW flow paths, EOP-8	3.56E-03	3.56E-03	3.56E-03	3.56E-03	2.93E-02
HW7	HPSI 12 provides adequate flow to Aux Hdr, given 11 HPSI not ?d, 13 not ?d or succeeds, low sump temp, EOP8	1.55E-01	1.41E-01	1.41E-01	1.41E-01	3.53E-01
HWR	HPSI 12 provides adequate flow to Aux Hdr, given 11 HPSI fails, 13 not ?d or succeeds, high sump temp, EOP8	4.66E-01	3.39E-01	3.39E-01	3.39E-01	8.58E-01
HWW	HPSI 12 provides adequate flow to Aux Hdr, given 11 & 13 HPSI fail, high sump temp, EOP8	5.15E-01	3.77E-01	3.77E-01	3.77E-01	8.77E-01
HX2	Operator controls AFW flow locally due to CR AFW Flow control support unavailable for either flowpath where flow exists	2.10E-02	1.25E-02	1.25E-02	1.25E-02	4.94E-02
HX3	Operator controls AFW flow locally due to CR AFW Flow control support unavailable for either flowpath where flow exists, and no S/G level ind avail, EOP-08	6.59E-01	5.60E-01	5.60E-01	5.60E-01	8.22E-01
HX5	Operator controls AFW flow locally due to CR AFW Flow control support unavailable for either flowpath where flow exists, S/G Level Ind avail, EOP-08	2.10E-02	1.25E-02	1.25E-02	1.25E-02	4.99E-02
HX6	Operator controls AFW flow locally due to CR AFW Flow control support unavailable for either flowpath where flow exists, shutdown from Auxiliary Shutdown Panel	3.53E-02	3.53E-02	3.53E-02	3.53E-02	3.53E-02
HX7	Operator controls AFW flow locally due to CR AFW Flow control support unavailable for either flowpath where flow exists, and no S/G level ind avail, EOP-08, given shutdown from the Auxilairy Shutdown Panel	8.08E-01	8.08E-01	8.08E-01	8.08E-01	8.08E-01
HXA	Operator controls AFW flow locally due to CR AFW Flow control support unavailable for either flowpath where flow exists, S/G Level Ind avail, EOP-08 and CSR fire that is isolated	1.25E-02	1.25E-02	1.25E-02	1.25E-02	1.25E-02
HZ1	OP locally ventilates both Swgr Rms using temporary fans, no LOOP	3.73E-02	5.84E-02	3.73E-02	3.73E-02	1.04E-01
IH1	OP starts both SWACs during a (LOOP or Non-LOOP), non-EOP-08	3.41E-02	5.03E-02	3.41E-02	3.41E-02	1.13E-01

TABLE 4.6d

Split Fraction Descriptions (Continued)

PLANT MODEL SF	SPLIT FRACTION DESCRIPTION	AUX BLDG FIRE MFF	TB FIRE MFF	INTAKE FIRE MFF	OUTSIDE MFF FIRE	CONT RM FIRE MFF
IH3	OP starts both SWACs during a LOOP or NON-LOOP, EOP8	1.70E-01	2.52E-01	1.70E-01	1.70E-01	5.66E-01
JB3	ESFAS Sequencer 14 actuates, Sequencer 11 failed, no LOCA, (ASA)	4.40E-02	4.40E-02	4.40E-02	4.40E-02	4.40E-02
MG2	Operator aligns SW to SRW within 114-124 minutes of plant trip, EOP-8	1.82E-01	2.82E-01	1.82E-01	1.82E-01	5.97E-01
MH2	Operator recovers failed steam admission line to AFW turbine driven pumps, EOP-8	9.52E-03	9.52E-03	9.52E-03	9.52E-03	6.75E-02
N1Z	480VAC Bus 11A energized, during a fire or flood (7 bkrs challenged)	8.64E-04	8.64E-04	8.64E-04	8.64E-04	8.64E-04
N3Z	480VAC Bus 14A energized, Buses 11A & 11B fail, all support available, fire or flood (6 bkrs challenged)	2.50E-01	2.50E-01	2.50E-01	2.50E-01	2.50E-01
N4Z	480VAC Bus 14B energized, Buses 11A, 11B & 14A fail, all support available , fire or flood (8 bkrs challenged)	2.85E-01	2.85E-01	2.85E-01	2.85E-01	2.85E-01
N5Z	480VAC Bus 21A energized, given all four Unit 1 480V Buses (N1-N4) fail, ASA, fire or flood (3 bkrs challenged)	3.00E-01	3.00E-01	3.00E-01	3.00E-01	3.00E-01
N64	480VAC Bus 21B energized, given three of previous questioned buses (Unit 1 480V Buses N1-N4 and 21A - N5) and others succeed or not questioned, ASA	2.84E-01	2.84E-01	2.84E-01	2.84E-01	2.84E-01
NR1	Non-safety related IA/PA supplies Unit 1 air loads, no LOOP, ASA including PA Compressor 21	1.04E-02	1.04E-02	1.04E-02	1.04E-02	1.04E-02
PHZ	Operator isolates PORVs on lowering RCS pressure, both PORVs spuriously open, both MCCs available, given CSR fire where dampers isolate CSR	3.72E-02	3.72E-02	3.72E-02	3.72E-02	3.72E-02
PT6	Reactor Vessel Failure from PTS, Large SLB	4.00E-03	4.00E-03	4.00E-03	4.00E-03	4.00E-03
PVZ	Both PORVs re-close on low RCS pressure, both PORVs spuriously open, both MCCs available, CSR fire where dampers isolate CSR	1.78E-01	1.78E-01	1.78E-01	1.78E-01	1.78E-01
Q11	OP recovers AFW at onset (w/in 10 minutes) of a spurious AFAS block (Cowboy)	2.37E-01	2.37E-01	2.37E-01	2.37E-01	5.30E-01
Q51	Operators fail AFW Pump Alignment actions (global failure)	8.88E-04	8.88E-04	8.88E-04	8.88E-04	6.62E-03
QZ4	OP recovers AFW following a spurious AFAS block, given PORVs open on SSSA	7.47E-03	7.47E-03	7.47E-03	7.47E-03	4.26E-02
R31	SW header 11 is unavailable	2.07E-02	2.07E-02	2.07E-02	2.07E-02	2.07E-02
R41	SW header 12 is unavailable	2.07E-02	2.07E-02	2.07E-02	2.07E-02	2.07E-02
RL9	SRW maintains adequate inventory, no support available	2.19E-01	2.19E-01	2.19E-01	2.19E-01	2.19E-01
RS4	Reactor Trip actuates, one or more of 120VAC Vital Panel 11&12 fails and one of the 125VDC Buses fails	1.86E-04	1.86E-04	1.86E-04	1.86E-04	1.86E-04
SG4	Hydrogen purge line operates, no support available	1.15E-03	1.15E-03	1.15E-03	1.15E-03	1.15E-03

TABLE 4.6d

Split Fraction Descriptions (Continued)

PLANT MODEL SF	SPLIT FRACTION DESCRIPTION	AUX BLDG FIRE MFF	TB FIRE MFF	INTAKE FIRE MFF	OUTSIDE MFF FIRE	CONT RM FIRE MFF
SL2	RCP seals remain intact given RCPs are secured by OP when CCW fails, post trip, 125VDC Bus 11 & 21 (DA & DC) and 120VAC Buses 11 and 12 (E1 & E2) available, not EOP-08	1.58E-03	1.58E-03	1.58E-03	1.58E-03	2.21E-03
SLQ	RCP seals remain intact given RCPs are locally secured by the OPs, 125VDC Bus 21 and 120VAC Buses 11 & 12 available, CSR fire which is isolated	1.97E-03	1.97E-03	1.97E-03	1.97E-03	1.97E-03
SLR	RCP seals remain intact given RCPs are secured by a loss of power (including LOOP), given CSR fire which is isolated (SL9 with CL=S)	6.09E-03	6.09E-03	6.09E-03	6.09E-03	6.09E-03
SLY	RCP seals remain intact, given Operators secure RCPs from the Metal Clad	5.56E-03	5.56E-03	5.56E-03	5.56E-03	5.56E-03
SLZ	RCP seals remain intact, given RCPs are never secured	1.02E-01	1.02E-01	1.02E-01	1.02E-01	1.02E-01
SRA	Cntmt Normal Sump Drain Line isolates on a LOCA, no support available	2.01E-03	2.01E-03	2.01E-03	2.01E-03	2.01E-03
SW3	Main Steam Relief Valves close, all banks re-close & modulate when TBVs (or TBVs & ADVs) fail to quick open, ADVs and TBVs fail to modulate long term	1.46E-01	1.46E-01	1.46E-01	1.46E-01	1.46E-01
TF1	AFW TURB PP 11 provides adequate flow given PP 13 successful or not?ed, OP actions successful	3.17E-02	3.17E-02	3.17E-02	3.17E-02	3.17E-02
TF2	AFW TURB PP 11 provides adequate flow, given PP 13 successful or not?ed, TURB Rec fails	3.59E-02	3.59E-02	3.59E-02	3.59E-02	3.59E-02
TF9	AFW TURB PP 11 provides adequate flow given PP 13 fails, OP actions successful	4.30E-02	4.30E-02	4.30E-02	4.30E-02	4.30E-02
TFA	AFW TURB PP 11 provides adequate flow given PP 13 fails, TURB Rec fails	4.76E-02	4.76E-02	4.76E-02	4.76E-02	4.76E-02
TG5	AFW Pump 12 works, AFW PP 11 fails and 13 suc, Turb pump maint recovery succeeds	1.28E-01	1.39E-01	1.28E-01	1.28E-01	1.72E-01
TGA	AFW Pump 12 works, AFW PP 11 fails and 13 fails, Op actions succeed	2.86E-01	3.23E-01	2.86E-01	2.86E-01	4.39E-01
TGC	AFW Pump 12 works, AFW PP 11 fails and 13 fails, Turb Rec from maint success, EOP8	3.55E-01	3.87E-01	3.55E-01	3.55E-01	4.87E-01
TGE	AFW Pump 12 works, shutdown from Aux Shut Panel given either 1. AFW PP 11 successful & 13 fails, EOP8 - OR - 2. AFW PP 11 fails and 13 succ, Turb pump maint recovery succeeds (comb of TG4 & TG5 for remote s/d)	1.45E-01	1.45E-01	1.45E-01	1.45E-01	1.45E-01
TGG	AFW Pump 12 works, shutdown from Aux Shut Panel - otherwise given TG3, 6, 7, or 8 boundary conditions	2.78E-01	2.78E-01	2.78E-01	2.78E-01	2.78E-01
TGH	AFW Pump 12 works, shutdown from Aux Shut Panel - otherwise given TG9, A, B, or C boundary conditions	4.06E-01	4.06E-01	4.06E-01	4.06E-01	4.06E-01
UQ5	OPs do not underfill S/Gs when AFW flow control is lost, given: S/G Level Ind avail, remote flow control failed, EOP-8	5.31E-02	4.04E-02	4.04E-02	4.04E-02	1.17E-01

TABLE 4.6d

Split Fraction Descriptions (Continued)

PLANT MODEL SF	SPLIT FRACTION DESCRIPTION	AUX BLDG FIRE MFF	TB FIRE MFF	INTAKE FIRE MFF	OUTSIDE MFF FIRE	CONT RM FIRE MFF
V11	ECCS Pump Room Air Cooler 11 operates, LOCA <0.02 sq. ft, operator recovery fails, (ASA)	1.10E-01	1.10E-01	1.10E-01	1.10E-01	1.10E-01
V14	ECCS Pump Room Air Cooler 11 operates, all operator recoveries succeed, (ASA)	1.17E-02	1.17E-02	1.17E-02	1.17E-02	1.17E-02
V5E	ECCS Pump Room Exhaust Fans Operate as Required, MCC 104R or 114R avail, temp loss of power (QC &/or QD=S) to available MCC, Op actions to align ECCS coolers fail or V1 & V2 fail or comb, EOP-08	1.81E-01	1.81E-01	1.81E-01	1.81E-01	5.90E-01
VB1	Electrical Realignment on AOP-9 performed succesfully on Control Room Evacuation	1.00E-01	1.00E-01	1.00E-01	1.00E-01	1.00E-01
VL1	Fire Brigade supresses Control Room Panel fire within 30 minutes (prior to fire involving further panels)	1.00E-02	1.00E-02	1.00E-02	1.00E-02	1.00E-02
VM2	OPS starts ECCS Coolers and aligns CVs open from the Control Room, EOP-8	3.20E-02	3.20E-02	3.20E-02	3.20E-02	1.61E-01
W31	SW header 21 is unavailable	4.00E-02	4.00E-02	4.00E-02	4.00E-02	4.00E-02
W41	SW header 22 is unavailable	4.00E-02	4.00E-02	4.00E-02	4.00E-02	4.00E-02
WY3	CACs operate as required (2of4), no LOOP with support for 2 of 4 CACs failed	1.28E-02	1.28E-02	1.28E-02	1.28E-02	1.28E-02
WY4	CACs operate as required (2of4), LOOP with support for 2 of 4 CACs failed	1.33E-02	1.33E-02	1.33E-02	1.33E-02	1.33E-02
XW2	OP supplies a 120VAC Vital Panel from 208/120 VAC Instrument Bus, given MCC 104R is available	1.01E-01	7.37E-02	7.37E-02	7.37E-02	2.07E-01

Table 4.7
Frequencies of Major Containment Failure Categories

			HRIF	HGIP	MBIO	MRIO	MRIF	HRWF	ATWS	MCIF	LBIO	HRSF
Containment Failure Category	**Percentage**	**Totals**	3.67E-05	2.42E-05	3.00E-06	2.52E-06	2.22E-06	2.34E-06	1.16E-06	5.20E-07	4.25E-07	1.24E-07
Intact Containment	37.1%	2.67E-05		2.18E-05	2.32E-06	1.12E-06			1.06E-06		3.51E-07	
Late Containment Failure	56.5%	4.07E-05	3.49E-05	9.66E-07	6.27E-07	1.38E-06	2.21E-06		4.77E-08	4.51E-07	7.09E-08	
Early Small Containment Failure	1.7%	1.21E-06	6.61E-07	3.14E-07	3.60E-08	7.55E-09	6.65E-09		8.15E-09	5.35E-08	1.27E-09	1.20E-07
Early Large Containment Failure	6.5%	4.67E-06	1.18E-06	1.06E-06	1.80E-08	5.04E-09	4.44E-09	2.34E-06	4.42E-08	1.51E-08	1.27E-09	4.23E-09
Small Containment Bypass	0.0%	0.00E+00										
Large Containment Bypass	0.0%	0.00E+00										

A225	**Unit 1 Radiation Exhaust Equipment Room (Fan Room)**	**Location:**	5' Auxiliary Building
		Fire Area:	14
		CDF:	3.99E-9

This compartment has exhaust ventilation equipment including two charcoal filters. It also contains piping associated with the Main Steam System, and the main and Auxiliary Feedwater Systems.

The compartment is approximately fifty-eight feet long and thirty feet wide with 1,725 square feet of area. Floor mounted equipment and interferences occupy approximately 925 square feet, leaving 800 square feet of open floor. The ceiling height is approximately twenty feet for a room volume of 34,500 cubic feet. The compartment has a concrete floor, concrete walls and a concrete ceiling.

Fire Analysis Results

Thirteen fire scenarios are identified for A225. Ten are the result of fixed ignition sources and three are due to transient ignition sources. Five scenarios are screened due to low functional impact. The screening is based on the same criteria described in Section 4.3.1.3. The remaining eight scenarios identified in Table 4-A-1 are represented by four fire initiating events identified in Table 4-A-2. The consolidation of fire scenarios is based on an assessment of the functional impact and ignition frequency of each scenario. The frequency of each initiator is the sum of the frequencies of all the fire scenarios it represents.

Table 4-A-1
A225 Fire Scenarios Summary

Scenario	Fire Scenario Description	Equipment Damaged
1	11 AFW Pump Room Exhaust Subsystem Fire + Transient Fire	No other effects
2	12 AFW Pump Room Exhaust Subsystem Fire + Transient Fire	No other effects
3	11 ECCS Pump Room Exhaust Subsystem Fire + Transient Fire	No other effects
4	12 ECCS Pump Room Exhaust Subsystem Fire + Transient Fire	No other effects
10	Control Panel 1C107 Fire	No other effects
11	Transient Fire Location 1	Conduits: 1A1576, 1A0366, 1A0278, 1A0367, 1A2901, 1A2794
12	Transient Fire Location 2	Cable Trays: 1AC09, 1AC19
13	Transient Fire Location 3	Cable Trays: 1AA61

Table 4-A-2
A225 Fire Analysis Results

Initiating Event	Fire Scenario	Frequency	Ignition Source	Functional Impact	CDF
A225F1	1,2	1.19E-4	11 or 12 AFW Rm Exhaust Fans	FF, FC	4.58E-10
A225F2	3,4	1.19E-4	11 or 12 ECCS Pump Rm Exhaust Fans	FF, V5	4.63E-10
A225F3	10	2.85E-4	Control Panel 1C107	FF, CV	2.95E-09
A225F4	11,12,13	2.52E-6	Maintenance Refuse	Y4, I1, I2, S3, S4, TA, TB, KY, FF, CV, HW, CT, SR*, TH	1.23E-10

A225 Fire Ignition Frequency

Both fixed and transient ignition frequencies are determined for the 5' Fan Room.

Fixed Ignition Frequency

The fixed ignition frequency is determined by starting with the compartment fixed ignition frequency results of Section 4.3.2 and then developing a scenario specific ignition frequency as described in Section 4.3.4.3. The sources and parameters used for A225 are:

Source	Generic Fire Frequency	Location Weighting	Source Weighting	Total Sources in Aux Bldg	Room Specific Frequency
Ventilation Subsystem	9.5E-3	2	9	331	5.17E-4
Electrical Cabinet	1.9E-2	2	1	135	2.81E-4

The nine ventilation subsystems are considered to be motors for fire modeling. Non-motor driven components that could contribute to the ignition (shaft bearings and belts) are considered less damaging than the motor. Although the belts are a plausible ignition source, they do not represent significant loading or heat release rate.

Some units house internal charcoal elements: These elements are considered benign with regard to room fire modeling because they are contained within the plenum housing.

All the motors have a 65 Btu heat release rate and are treated as radiant sources. A radiant exposure worksheet calculates the associated critical damage distance. In all cases, there is no cabling or equipment damage associated with a motor failure.

Since there are nine ventilation subsystems, each subsystem is assigned one ninth of the compartment specific frequency of 5.74E-5. Initiating Events A225F1 and A225F2 contain two ventilation units each. The fixed frequency contribution for these initiating events is therefore:

$$A225F1 \text{ or } A225F2_{\text{Fix Ignition}} = 5.74E\text{-}5 * 2 = 1.15E\text{-}4/yr$$

Panel 1C107 is the sole electrical cabinet in the fan compartment and is a totally enclosed cabinet with no ventilation openings, grills or louvers. For this cabinet, a fire will not propagate beyond the confines of the cabinet itself because combustion products suppress fire growth. As an ignition source, sealed enclosures have a 65 Btu heat release rate with a damage range analyzed using a radiant exposure worksheet. No other components or cables are within the damage range.

Since there is only one electrical cabinet or panel in the compartment, the ignition frequency for 1C107 is simply the fixed ignition frequency value for electrical cabinets in A225.

$$A225F3_{\text{Fix Ignition}} = 2.81E\text{-}4/yr$$

Transient Ignition Frequency

The transient ignition frequency is determined by starting with the compartment transient ignition frequency results of Section 4.3.2 and then developing a scenario specific ignition frequency as described in Section 4.3.4.3. The sources and parameters used for A225 are:

Source	Generic Fire Frequency	Location Weighting	Source Weighting	Total Sources in Aux Bldg	Room Specific Frequency
Transient fire - other	1.3E-3	2	10	232	1.12E-4

The likelihood a hot work induced transient fire is very small since a Hot Work Permit is required for this compartment. This means that there is continual fire watch during any hot work activities and for a period of at least thirty minutes after work is complete. See Section 4.2.2.6.2 for additional information on controlling ignition sources.

The formula for calculating the damage frequency due to transient combustible fires is contained in Section 4.3.4.3.2. The floor area of the 5' Fan Room is approximately 1,725 square feet. Floor mounted equipment and interferences occupy approximately 925 square feet of floor area, leaving 800 square feet of open floor (=A_F).

Each ventilation unit is a transient ignition source target. Assuming a radiant damage distance of approximately three feet extended as a semi-circle around each unit motor, the floor area where a transient could cause damage equals about fourteen square feet. Therefore:

$$u = (0\ \text{ft}^2 + 14\ \text{ft}^2) / 800\ \text{ft}^2 = 0.0175$$

and

$$F_t = 1.12\text{E-}04 * 0.0175 * 1 = 1.96\text{E-}06$$

As above, A225F1 and A225F2 contain two ventilation units each. Therefore:

A225F1 or A225F2$_{transient}$ = 1.96E-06 * 2 = 3.92E-6/yr

Similarly, panel 1C107 is a transient target. Using a 1.8 foot damage distance, the area around the panel perimeter is estimated at twenty-five square feet. Therefore:

$$u = (0\ \text{ft}^2 + 25\ \text{ft}^2) / 800\ \text{ft}^2 = 0.03125$$

and

$$F_t = 1.12\text{E-}04 * 0.03125 * 1 = 3.5\text{E-}06$$

A225F3$_{transient}$ = 3.50E-6/yr

Based on the transient fire worksheets for this compartment, the Critical Damage Distance to the lowest overhead target is approximately eleven feet from the floor when the fire is in a corner location. Three corners have cable trays or conduits that are potential targets. Each location presents a six square foot overhead target. Therefore:

$$u = (6\ \text{ft}^2 + 0\ \text{ft}^2) / 800\ \text{ft}^2 = 0.0075$$

and

$$F_t = 1.12\text{E-}04 * 0.0075 * 1 = 8.4\text{E-}07$$

However, the three cable target transient fire scenarios are consolidated into one initiating event.

A225F4$_{Transient}$ = 8.41E-7 * 3 = 2.52E-6/yr

Total Ignition Frequency

The total ignition frequency is the sum of fixed and transient frequencies (as applicable).

$$A225F1 \text{ or } A225F2 = 1.15E\text{-}4 + 3.92E\text{-}6 = 1.19E\text{-}4/yr$$

$$A225F3 = 2.81E\text{-}4 + 3.5E\text{-}6 = 2.85E\text{-}4/yr$$

$$A225F4 = 0 + 2.52E\text{-}6 = 2.52E\text{-}6/yr$$

Fire Suppression

Although smoke detection and a wet pipe sprinkler system provide fire suppression mitigation for this area, these are not credited in the fire analysis.

Fire Suppression Induced Equipment Failures

Based on the approach described in Section 4.3.4.4.4, equipment failure due to the inadvertent actuation of the automatic fire suppression system is assumed not to occur. Cable and conduit, pumps and other PRA equipment in the room are not considered to be susceptible to water damage.

A226	Unit 1 Service Water Pump Room	Location:	5' Auxiliary Building
		Fire Area:	39
		CDF:	3.47E-08

This compartment contains the three Service Water (SRW) Pumps, two SRW heat exchangers and their associated control valves for Unit 1. It also the contains the Unit 1 Auxiliary Feedwater (AFW) motor driven pump and associated flow control valves. Both safety-related Saltwater System Air Compressors (SWAC) are also located in this compartment.

This room is located on the 12' elevation at the east end of the Auxiliary Building, but access is from the Turbine Building. The room entry is through a water tight door on the east side of the room. This compartment is a relatively large open room eighty-two feet long by thirty feet wide for a total room area of approximately 2,460 square feet. The height of the room is twenty-two feet for a total room volume of 54,120 cubic feet. The room has a sealed (painted) floor which provides a smooth slick surface. The room is divided approximately half way up by open grating which separates the SRW pumps below from the SWACs above.

This area has products of combustion and flame detection and a wet pipe sprinkler system.

Fire Analysis Results

Thirteen fire scenarios were identified for A226. Nine are the result of fixed ignition sources and four are due to combined transient and fixed ignition sources. These scenarios are represented by three fire initiating events. The consolidation of fire scenarios is based on an assessment of the functional impact and ignition frequency of each scenario. The frequency of each initiator is the sum of the frequencies of all the fire scenarios it represents.

Table 4-B-1
A226 Fire Scenario Summary

Scenario	Initiating Equipment	Scenario Description	Size	Suppression	Effect
1	11 SWAC	Oil Spill	Small	N/A	SW Air Compressor 11, SRW Pump 12
2	12 SWAC	Oil Spill	Small	N/A	SW Air Compressor 12, SRW Pump 13
3	11 SWAC	Oil Spill	Large	Yes	SW Air Compressor 11, SRW Pump 12
4	12 SWAC	Oil Spill	Large	Yes	SW Air Compressor 12, SRW Pump 13
5	11 SWAC	Oil Spill	Large	No	SW Air Compressors 11 and 12; ZB1AC21; SRW Pumps 11, 12 and 13; AFW Pump 13

Table 4-B-1
A226 Fire Scenario Summary (Continued)

Scenario	Initiating Equipment	Scenario Description	Size	Suppression	Effect
6	12 SWAC	Oil Spill	Large	No	SW Air Compressors 11 and 12; ZB1AC21; SRW Pumps 11, 12 and 13; AFW Pump 13
7	12 SRW	Oil Spill	Large	No	SW Air Compressors 11 and 12; ZB1AC21; SRW Pumps 11, 12 and 13; AFW Pump 13
8	13 SRW	Oil Spill	Large	No	SW Air Compressors 11 and 12; ZB1AC21; SRW Pumps 11, 12 and 13; AFW Pump 13
9	13 AFW	Oil Spill Oil Spill	Large Large	Yes No	SW Air Compressors 11 and 12; ZB1AC21; SRW Pumps 11, 12 and 13; AFW Pump 13
10	13 AFW	Motor Failure, Transient, Oil Spill	N/A N/A Small	N/A N/A N/A	13 AFW
11	11 SRW	Motor Failure, Transient, Oil Spill Oil Spill Oil Spill	N/A N/A Small Large Large	N/A N/A N/A Yes No	11 SRW
12	12 SRW	Motor Failure, Transient, Oil Spill Oil Spill	N/A N/A Small Large	N/A N/A N/A Yes	12 SRW
13	13 SRW	Motor Failure, Transient, Oil Spill Oil Spill	N/A N/A Small Large	N/A N/A N/A Yes	13 SRW

Table 4-B-2
A226 Fire Analysis Results

Initiating Event	Fire Scenario	Frequency	Ignition Source	Functional Impact	CDF
A226F1	1, 3	1.34E-4	11SWAC small spill; 11SWAC large spill with suppression	Y3, Y4, I1, S4	3.40E-9
A226F2	2, 4	1.34E-4	12SWAC small spill; 12SWAC large spill with suppression	Y3, Y4, I2, (S3, S4: SRW PP 13 only)	3.12E-9

Table 4-B-2
A226 Fire Analysis Results (Continued)

Initiating Event	Fire Scenario	Frequency	Ignition Source	Functional Impact	CDF
A226F3	5, 6, 7, 8, 9	6.67E-5	11SWAC large spill no suppression; 12SWAC large spill no suppression; 12SRW large spill no suppression; 13SRW large spill no suppression; 13AFW large spill	Y3, Y4, I1, I2, S3, S4, F7	2.82E-8
Screened	10, 11, 12, 13	N/A	13AFW motor, transient, small spill; 11SRW motor, transient, small spill, large spill; 12SRW motor, transient, small spill, large spill with suppression; 13SRW motor, transient, small spill, large spill with suppression	None	

Fire Ignition Frequency

Both fixed and transient ignition frequencies were determined for the Service Water Pump Room.

Fixed Ignition Frequency

The fixed ignition frequency is determined by starting with the compartment fixed ignition frequency results of Section 4.3.2 and then developing a scenario specific ignition frequency as described in Section 4.3.4.3. The sources and parameters used for A226 are:

Source	Generic Fire Frequency	Location Weighting	Source Weighting	Total Sources in Aux Bldg	Room Specific Frequency
Air Compressor	4.7E-3	2	2	35	5.37E-4
Pump	1.9E-2	2	4	54	2.81E-3
Ventilation Subsystems	9.5E-3	2	2	331	1.15E-4

Note: Failure of the ventilation systems in the SRW Pump Room has been screened for plant impact. Fire modeling showed that upon ignition, neither unit had any impact other than damage to the units themselves. Therefore, the ventilation systems are not discussed here.

The fire ignition frequency for devices with oil (compressors and pumps) does not indicate that each ignition results in a large fully developed fire if not suppressed. Review of the EPRI Fire Events Database shows that approximately fifty percent of pump fire events involve lubricating oil fires and fifty percent represent motor winding fires or other miscellaneous ignitions. The Fire PRA Implementation Guide indicates that approximately eighty-eight percent of oil spill fires are "small" spills and twelve percent are "large" spills. This information is used to calculate scenario frequencies.

An additional factor used in the calculations is the fire suppression failure rate. Per the FIVE methodology, the wet pipe sprinkler system failure rate is 0.02.

SALTWATER AIR COMPRESSORS (SWAC)

This area contains two air compressors located on the upper level grated floor, 11 SWAC and 12 SWAC. Two SRW pumps are located essentially directly below (13 SRW Pump below 12 SWAC and 12 SRW pump below 11 SWAC). An electrical motor fire or a lubricating oil spill fire can be postulated for these compressors. As there is no equipment closer than a couple feet to each compressor other than its own electrical feed, a motor fire would not damage any other equipment other than disable the compressor (an electric motor is modeled as a 65 Btu/s radiant heat case, yielding a critical damage distance of 1.4 ft). The most conservative fire is that associated with an oil spill. As such, 11 and 12 SWAC fires are modeled as postulated oil spill and ignition fires. However, this compartment has an area-wide automatic sprinkler system that, if successfully actuated, will minimize fire induced damage to other equipment, as discussed below.

The air compressors contain approximately 0.8 gallons of oil. The large oil spill fire for each compressor is postulated to be a spill of the entire 0.8 gallon oil inventory. Given the small size of these compressors and the relatively small inventory of oil in each one, the effects of the small oil spill fire are assumed to be confined to the compressor itself.

11 SWAC, Small Oil Spill: Fire induced damage limited to unavailability of the compressor itself and 12 SRW pump directly below.

Scenario 1 = 5.37E-4 ÷ 2 * 50% * 82% = 1.10E-4

12 SWAC, Small Oil Spill: Fire induced damage limited to unavailability of the compressor itself and 13 SRW pump directly below.

Scenario 2 = 5.37E-4 ÷ 2 * 50% * 82% = 1.10E-4

The oil is modeled as having the combustion characteristics of transformer oil and the spill characteristics of DTE 797 lubricating oil (these correspond to representative parameters provided in the FIVE methodology).

The one complicating configuration issue is that the air compressors are located on the upper level on the grate floor. Any oil spill could be postulated to be dispersed as it falls to and through the grate floor and hits the concrete floor below. The fire models available in pertinent literature do not provide guidance as to the treatment of such a configuration. Common sense indicates that modeling the oil as dispersed into small pools and widely spread droplets would yield less conservative results than modeling an oil pool fire. As such, the air compressor oil spill fire is modeled as 1) an oil pool fire, that 2) occurs on the concrete floor below, and 3) disables the SRW pump below. This assumption addresses the configurational issue of each compressor being located above a SRW pump.

The postulated spill is treated as a confined spill. The idealized fire modeled in FIVE calculates a fire duration of just eight seconds if an unconfined oil spill is postulated. Information provided in the Fire PRA Implementation Guide indicates that qualified cables would need to be exposed to hot gases for a "significant time" to result in damage. In the case of ignition, NUREG/CR-4527 states that tests show that direct flame impingement for ten minutes is necessary to ignite qualified cables. Therefore, an idealized eight second fire is non-conservative in that damage to able targets would not be indicated.

The duration of an oil spill fire is linearly related to the thickness of the oil pool and, thus, the spill area. Therefore, to lengthen the fire duration, it is necessary to lessen the spill area by postulating a confined spill. This is consistent with the Fire PRA Implementation Guide which states that it is unrealistic to assume that the only factor affecting spill size is the viscosity of the oil. Depressions in the floor, floor drains, obstructions in the floor (e.g., berms, columns, equipment pedestals) and housekeeping practices that direct cleaning up of spills, all support the conclusion that an idealized unconfined spill is an unrealistic assumption for a representative fire. As such, the spill size was adjusted to obtain a fire duration of approximately one minute. This results in a spill surface area of 12.5 square feet and a corresponding peak fire intensity of 1,434 Btu/sec. Postulation of longer durations is not considered justified due to the very small spill area that would be required to be postulated; such a spill would not be consistent with the location of the compressor (i.e., on an elevated grated floor).

A walkdown of the compartment and a review of arrangement and cable tray drawings determined that the nearest, and only, cable tray ZB1AC21 is under the grated floor and toward the south-west corner. However, the majority of a large oil spill from either of the overhead compressors would most likely be confined between the pedestals of 12 and 13 SRW pumps, placing this tray at least eight feet laterally from the edge of the oil spill on the floor. However, directly overhead of the postulated oil spill are located a number of electrical conduits running generally north and south approximately ten feet above the floor. Two of these conduits contain the electrical feeds for 12 and 13 SRW Pumps. In addition, a run of various electrical conduits, also running north and south and approximately ten feet above the floor, are located a couple feet laterally from the edge of the postulated oil spill. This run of conduits contains the electrical feed for 11 SRW Pump and cabling to 13 AFW Pump. All the previously mentioned targets are inside the oil spill fire plume (based on plume cone radius of $r = 0.2z$). Therefore, a postulated large oil spill fire from either of the above air compressors has the potential to simultaneously disable all three SRW pumps and 13 AFW Pump.

The in-plume fire damage worksheet for this configuration shows that all the previously mentioned targets are within the critical damage distance (i.e., damage to the cabling is indicated). The radiant exposure worksheet for this configuration shows that the critical damage distance from the edge of the oil spill (assuming a damage criteria of 3.75 Btu/sec/ft^2 for motors, per the Fire PRA Implementation Guide) is about three and a half feet. Given the distances between the SRW pumps and the AFW pump, it is judged that radiant induced damage to the other pumps (i.e., other than the pump directly below the SW air compressor) may occur. An oil spill from 12 SWAC may place 12 SRW within the critical damage distance (approximately three feet); the motor of the AFW pump is outside the critical damage distance.

11 SWAC, Large Oil Spill Without Suppression: Fire induced damage includes damage to both SWACs, 11, 12, and 13 SRW Pumps, and 13 AFW Pump.

Scenario 5 = 5.37E-4 ÷ 2 * 50% * 18% * 0.02 = 4.83E-7

12 SWAC, Large Oil Spill Without Suppression: Fire induced damage includes damage to both SWACs, 11, 12, and 13 SRW Pumps, and 13 AFW Pump.

Scenario 6 = 5.37E-4 ÷ 2 * 50% * 18% * 0.02 = 4.83E-7

In the above analysis, the transient thermal response of the targets is not explicitly considered and is assumed to be negligible; the targets are assumed to instantly reach the critical temperature. Considering the existence of fire suppression in the room, it is appropriate to explicitly model the transient thermal response of the targets. If the actuation time of the wet pipe sprinklers (based on transient thermal response) is less than the time to target damage, then the damage to the targets can be prevented (given successful actuation of the sprinklers).

The automatic sprinkler heads in the area are equipped with link type elements and have a temperature rating of 212°F. No specific information has been identified regarding the time constants of the heads. The FIVE methodology recommends time constants in the range of 60-120 seconds for fusible link type heads; a value of one hundred seconds is assumed here. The sprinkler heads for the lower level are approximately twelve feet above the floor and are equidistant throughout the room on approximate ten foot centers. Based on a plume cone radius of $r = 0.2z$, at least one sprinkler head, and most likely more than one, will be in the plume of the postulated oil spill fire. Using the FIVE fire modeling worksheets for transient thermal response, it was determined that suppression system actuation would occur in approximately twenty seconds, whereas damage to nearby pumps would occur in thirty-six seconds and to the cabling overhead in forty seconds (this estimate is conservative as it does not account for the shielding provided by the conduit).

11 SWAC, Large Oil Spill With Suppression: Fire induced damage includes damage to the compressor itself and 12 SRW pump directly below.

Scenario 3 = 5.37E-4 ÷ 2 * 50% * 18% * (1 – 0.02) = 2.37E-5

12 SWAC, Large Oil Spill With Suppression: Fire induced damage includes damage to the compressor itself and 13 SRW pump directly below.

Scenario 4 = 5.37E-4 ÷ 2 * 50% * 18% * (1 – 0.02) = 2.37E-5

SERVICE WATER PUMPS (SRW):

A pump fire may include an electrical motor fire or a lubricating oil spill fire. As there is no equipment closer than six feet to each pump other than its own electrical feed, a motor fire would not damage any other equipment other than disable the pump (an electric motor is modeled as a 65 Btu/s radiant heat case, yielding a critical damage distance of 1.4 ft). The most conservative fire is that associated with an oil spill. As such, the SRW Pumps are modeled as postulated oil spill and ignition fires. This compartment has an area-wide automatic sprinkler system that, if successfully actuated, will minimize fire induced damage to other equipment, as discussed below.

The SRW pumps water pumps each contain approximately one gallon of oil. The large oil spill fire for each SRW pump is postulated to be a spill that includes the entire one gallon oil inventory. As no guidance is available in industry literature regarding the size of a "small" oil spill, judgment is used here to select a spill volume of one quart to represent a small oil spill fire for these pumps.

Like the compressor oil fires, the oil is modeled as having the combustion characteristics of transformer oil and the spill characteristics of DTE 797 lubricating oil (these correspond to representative parameters provided in the FIVE methodology).

Like the compressor oil spill fires, the postulated SRW pump oil spills are treated as confined. As such, the spill size was adjusted to obtain a fire duration of approximately one minute. This results in a spill surface area of approximately four square feet and a corresponding peak fire intensity of 448 Btu/sec for the small oil spill. The large oil spill is approximately 15.5 square feet and 1,779 Btu/sec.

11 SRW Pump is located at the north end of the compartment and far removed from the other pumps. The nearest pump is 15-20 feet to the south with no intervening combustibles or overhead cable trays. As such, any oil spill fire associated with 11 SRW Pump would only damage the pump itself. Oil spill fire effects for the other pumps are discussed below.

11 SRW Pump, Small or Large Oil Spill, With or Without Suppression; Motor: Fire induced damage is limited to unavailability of the SRW pump itself.

$$\text{Scenario } 11_{fix} = [2.81\text{E-}3 \div 4 * 50\%] + [2.81\text{E-}3 \div 4 * 50\%] = 7.03\text{E-}4$$

A walkdown of the compartment and a review of arrangement and cable tray drawings determined that the nearest, and only, cable tray (ZB1AC21) is under the grate floor and toward the south-west corner. This target is outside the damage influence of 12 SRW; although, it may be within the damage influence of an oil spill fire of 13 SRW. Whether cable tray ZB1AC21 is damaged by an oil spill fire of 13 SRW Pump depends on the configuration of the oil spill (e.g., which side of the pump pedestal the oil spill propagates). This analysis assumes that cable tray ZB1AC21 is damaged following an oil spill fire of 13 SRW Pump if damage to other overhead targets is indicated.

As discussed earlier, overhead of the postulated oil spill are located a number of electrical conduits running generally north and south approximately ten feet above the floor. Conduits containing the electrical feeds for the SRW pumps and the AFW pump are located in these conduit runs. Depending on the configuration of the oil spill, some or all of these conduits may be in the fire plume. This analysis assumes that for an oil spill fire of 12 or 13 SRW Pump the overhead electrical feeds for all pumps in the room are within the fire plume. Therefore, a postulated large oil spill fire from 12 or 13 SRW Pump has the potential to simultaneously disable all three SRW pumps and 13 AFW Pump.

12 SRW Pump, Large Oil Spill Without Suppression: Fire induced damage includes damage to 11, 12, and 13 SRW Pumps, 13 AFW Pump, and both SWACs.

Scenario 7 = 2.81E-3 ÷ 4 * 50% * 18% * 0.02 = 1.26E-6

13 SRW Pump, Large Oil Spill Without Suppression: Fire induced damage includes damage to 11, 12, and 13 SRW Pumps, 13 AFW Pump, and both SWACs.

Scenario 8 = 2.81E-3 ÷ 4 * 50% * 18% * 0.02 = 1.26E-6

In the case of the small oil spill fire, the in-plume fire damage worksheet and the radiant exposure worksheet show that damage to adjacent equipment or overhead cabling is not indicated. As such, in the case of small oil spill fires of the pumps, the only impact is damage to the pump itself.

For the large oil spill, the in-plume fire damage worksheet for this configuration shows that all the previously mentioned targets are within the critical damage distance (i.e., damage to the cabling is indicated). The radiant exposure worksheet for this configuration shows that the critical damage distance from the edge of the oil spill (assuming a damage criteria of 3.75 Btu/sec/ft^2 for motors, per the Fire PRA Implementation Guide) is about four feet. Given the distances between the SRW pumps and the AFW pump, it is judged that radiant induced damage to the other pumps may occur.

Like the compressor oil spill fire, the transient thermal response of the targets in the above analysis is not explicitly considered and is assumed to be negligible; the targets are assumed to instantly reach the critical temperature. This is conservative. Considering the existence of fire suppression in the room, it is appropriate to explicitly model the transient thermal response of the targets. If the actuation time of the wet pipe sprinklers (based on transient thermal response) is less than the time to target damage, then the damage to the targets can be prevented (given successful actuation of the sprinklers).

The automatic sprinkler heads in the area are equipped with link type elements and have a temperature rating of 212°F. No specific information has been identified regarding the time constants of the heads. The FIVE methodology recommends time constants in the range of 60-120 seconds for fusible link type heads; a value of one hundred seconds is assumed here. The sprinkler heads for the lower level are approximately twelve feet above the floor and are generally equidistant throughout the room on approximate ten foot centers. Based on a plume cone radius of $r = 0.2z$, at least one sprinkler head will be in the plume of the postulated oil spill fire. Using the FIVE fire modeling worksheets for transient thermal response, it was determined that suppression system actuation would occur in approximately eighteen seconds, whereas damage to nearby pumps would occur in forty-four seconds and damage to the cabling overhead would occur in thirty seconds. These results indicate that the suppression system will actuate prior to, and prevent, damage to other equipment.

12 SRW Pump, Small Oil Spill With or Without Suppression; Large Oil Spill With Suppression; Motor

$$\text{Scenario } 12_{fix} = 2.81\text{E-}3 \div 4 * 50\% * \{82\% + [18\% * (1 - 0.02)]\} + [2.81\text{E-}3 \div 4 * 50\%] = 7.01\text{E-}4$$

13 SRW Pump, Small Oil Spill With or Without Suppression; Large Oil Spill With Suppression; Motor

$$\text{Scenario } 13_{fix} = 2.81\text{E-}3 \div 4 * 50\% * \{82\% + [18\% * (1 - 0.02)]\} + [2.81\text{E-}3 \div 4 * 50\%] = 7.01\text{E-}4$$

AUXILIARY FEEDWATER PUMP (AFW):

The AFW pump fire is modeled in the same manner as the SRW pumps. The AFW pump contains approximately 1.4 gallons of oil. The large oil spill fire for the AFW pump is postulated to be a spill that includes the entire oil inventory. As no guidance is available in industry literature regarding the size of a "small" oil spill, judgment is used here to select a spill volume of one quart to represent a small oil spill fire for these pumps.

Like the compressor and SRW pump oil spill fires, the oil is modeled as having the combustion characteristics of transformer oil and the spill characteristics of DTE 797 lubricating oil (these correspond to representative parameters provided in the FIVE methodology).

Like the compressor and SRW pump oil spill fires, the postulated AFW pump oil spills are treated as confined. As such, the spill size was adjusted to obtain a fire duration of approximately one minute. This results in a spill surface area of approximately four square feet and a corresponding peak fire intensity of 448 Btu/sec for the small oil spill. The large oil spill is approximately twenty-two square feet and 2,525 Btu/sec.

Cable tray ZB1AC21 is located approximately twenty feet away and outside the damage influence of the AFW pump oil spill fire. However, as discussed earlier, overhead of the postulated oil spill are located a number of electrical conduits running generally north and south approximately ten feet above the floor. Conduits containing the electrical feeds for the SRW pumps and the AFW pump are located in these conduit runs. Depending on the configuration of the oil spill, some or all of these conduits may be in the fire plume. If the AFW pump oil spill fire occurs against the east wall, near which the AFW pump is located, these conduits would be outside the fire plume and most likely undamaged by the postulated fire. This analysis assumes that the oil spill occurs on the side of the AFW pump toward the center of the room, such that the overhead electrical feeds for all pumps in the room are within the fire plume. Therefore, the postulated large oil spill fire of 13 AFW Pump has the potential to simultaneously disable all three SRW pumps and 13 AFW Pump.

In the case of the small oil spill fire, the in-plume fire damage worksheet and the radiant exposure worksheet show that damage to adjacent equipment or overhead cabling is not indicated. As such, in the case of small oil spill fires of the pumps, the only impact is damage to the pump itself.

13 AFW Pump, Small Oil Spill With or Without Suppression; Motor: Fire induced damage limited to unavailability of the AFW pump itself.

Scenario 10_{fix} = [2.81E-3 ÷ 4 * 50% * 82%] + [2.81E-3 ÷ 4 * 50%] = 6.39E-4

For the large oil spill, the in-plume fire damage worksheet for this configuration shows that all the previously mentioned targets are within the critical damage distance (i.e., damage to the cabling is indicated). The radiant exposure worksheet for this configuration shows that the critical damage distance from the edge of the oil spill (assuming a damage criteria of 3.75 Btu/sec/ft^2 for motors, per the Fire PRA Implementation Guide) is about four and a half feet. Given the distances between the SRW pumps and the AFW pump, it is judged that radiant induced damage to the other pumps may occur.

Like the compressor and SRW pump oil spill fires, the transient thermal response of the targets in the above analysis is not explicitly considered and is assumed to be negligible; the targets are assumed to instantly reach the critical temperature. This is conservative. Considering the existence of fire suppression in the room, it is appropriate to explicitly model the transient thermal response of the targets. If the actuation time of the wet pipe sprinklers (based on transient thermal response) is less than the time to target damage, then the damage to the targets can be prevented (given successful actuation of the sprinklers).

The automatic sprinkler heads in the area are equipped with link type elements and have a temperature rating of 212°F. No specific information has been identified regarding the time constants of the heads. The FIVE methodology recommends time constants in the range of 60-120 seconds for fusible link type heads; a value of one hundred seconds is assumed here. The sprinkler heads for the lower level are approximately twelve feet above the floor and are generally equidistant throughout the room on approximate ten foot centers. Based on a plume cone radius of r = 0.2z, at least one sprinkler head will be in the plume of the postulated oil spill fire. Using the FIVE fire modeling worksheets for transient thermal response, it was determined that suppression system actuation would occur in approximately fourteen seconds, whereas damage to nearby pumps would occur in sixteen seconds and damage to the cabling overhead would occur in thirteen seconds. These results do not provide confidence that the suppression system will prevent damage to the overhead electrical feeds.

13 AFW Pump, large Oil Spill With or Without Suppression: Fire induced damage includes damage to 11, 12, and 13 SRW Pumps, and both SWACs.

Scenario 9 = 2.81E-3 ÷ 4 * 50% * 18% = 6.32E-5

Transient Ignition Frequency

The transient ignition frequency is determined by starting with the compartment transient ignition frequency results of Section 4.3.2 and then developing a scenario specific ignition frequency as described in Section 4.3.4.3. The sources and parameters used for A226 are:

Source	Generic Fire Frequency	Location Weighting	Source Weighting	Total Sources in Aux Bldg	Room Specific Frequency
Cable fire - welding	5.1E-3	2	1	232	4.40E-5
Transient - welding	3.1E-2	2	1	232	2.67E-4
Transient - other	1.3E-3	2	9	232	1.01E-4

Note that welding related transients are not considered for damage.

Approximately twenty percent of the SRW Pump Room floor area is obstructed and would not allow for the placement of combustibles. Therefore, total available floor area:

$$A_F = 2460 \text{ ft}^2 * 80\% = 1968 \text{ ft}^2$$

The only plausible transient targets are the pumps or compressors themselves and these are susceptible to radiant damage at a distance of 1.8 feet. Each pump pedestal is of a similar dimension: The area formed by extending 1.8 feet around each pedestal perimeter is approximately sixty square feet ($=A_{sr}$). Therefore:

$$u = (A_s + A_{sr}) / A_F \quad (0 \text{ ft}^2 + 60 \text{ ft}^2) / 1968 \text{ ft}^2$$

$$= 3.05\text{E-}2$$

$$\text{and} \qquad F_t = 1.01\text{E-}4 * 3.05\text{E-}2 * 1$$

$$\text{Pump}_{trans} = 3.08\text{E-}6$$

The area surrounding each SWAC is estimated to be twenty square feet ($=A_{sr}$). Therefore:

$$u = (0 \text{ ft}^2 + 20 \text{ ft}^2) / 1968 \text{ ft}^2$$

$$= 1.02\text{E-}2$$

$$\text{and} \qquad F_t = 1.01\text{E-}4 * 1.02\text{E-}2 * 1$$

$$\text{Compressor}_{trans} = 1.03\text{E-}6$$

The SWAC transient frequency is not combined with other room fire scenarios. As an individual scenario it is subsumed by the normal equipment failure frequency. Therefore, the SWAC transient frequency is not considered for impact.

Total Ignition Frequency

Fixed scenarios and transient scenarios with like impacts were added together.

$$\text{Scenario 10} = \text{Scenario } 10_{\text{fix}} + \text{Pump}_{\text{trans}} = 6.39\text{E-4} + 3.08\text{E-6} = 6.42\text{E-4}$$
$$\text{Scenario 11} = \text{Scenario } 11_{\text{fix}} + \text{Pump}_{\text{trans}} = 7.03\text{E-4} + 3.08\text{E-6} = 7.06\text{E-4}$$
$$\text{Scenario 12} = \text{Scenario } 12_{\text{fix}} + \text{Pump}_{\text{trans}} = 7.01\text{E-4} + 3.08\text{E-6} = 7.04\text{E-4}$$
$$\text{Scenario 13} = \text{Scenario } 13_{\text{fix}} + \text{Pump}_{\text{trans}} = 7.01\text{E-4} + 3.08\text{E-6} = 7.04\text{E-4}$$

Initiating event frequencies are then summed using the individual scenario frequencies.

A226F1 = Scenario 1 + Scenario 3 = 1.10E-4 + 2.37E-5 = 1.34E-4

A226F2 = Scenario 2 + Scenario 4 = 1.10E-4 + 2.37E-5 = 1.34E-4

A226F3 = Scenario 5 + Scenario 6 + Scenario 7 + Scenario 8 + Scenario 9
= 4.83E-7 + 4.83E-7 + 1.26E-6 + 1.26E-6 + 6.32E-5 = 6.67E-5 *

Scenarios 10, 11, 12, and 13 total impact is screened.

Fire Suppression

The compartment contains both flame and smoke detectors for alarm purposes. Five of the six smoke detectors are located in the ceiling and spaced generally equidistant down the center of the room; the sixth is located under the grate floor at the north end. The three flame detectors are located under the grate floor, one above each of the three SRW pumps.

The compartment is equipped with a wet pipe sprinkler system. One subsystem is located in the ceiling of the upper elevation and the other subsystem is located under the grate floor. There are approximately thirty to forty heads generally uniformly spaced on each elevation. The wet pipe sprinkler heads are equipped with thermal fusible links rated at 212° F.

Fire Suppression Induced Equipment Failure

The compartment has several floor drains which would remove accumulated fire system spray. In addition, the pump motors are pedestal mounted and compressor motors are on the upper level grating. All the motors appear to be a drip-proof or closed design with qualified connections. All cabling is assumed to be impervious to water. Therefore, suppression induced damage is judged unlikely in this room.

A227/A316 East Piping Penetration Rooms

Location:	5' Auxiliary Building 27' Auxiliary Building
Fire Area:	11
CDF:	7.34E-7

This compartment is approximately eighty-nine feet in length with a width of about sixteen feet, for a compartment area of 1,424 square feet. The height of this area is thirty-eight feet from the ceiling, giving the compartment a room volume of 54,112 cubic feet. This area contains several very small pumps and motor-operated valves which are insignificant ignition sources. There are no cable trays to provide an ignition target within the critical damage distance of these ignition sources. Conduit is not located near any potential ignition sources.

Grating separates the two rooms. At the 5' level is a three hour fire door. There is also a three hour non-rated emergency escape hatch which provides an adequate fire barrier. At the 27' level is a normally closed non-rated three hour fire door. The hatch at the 27' level is a one and a half hour door, which is considered adequate. Barriers between A316, A317 and A429 contain dampers that are adequate fire barriers as installed for original construction. Propagation beyond this compartment is unlikely due to the configuration of the area and the low combustible loading.

A227 <u>**Fire Analysis Results**</u>

No fire scenarios are identified for this room. Given the configuration of the room, damage due to a pump or motor-operated valve fire is limited to the component itself. Plausible fire scenarios were disregarded due to the low fire frequencies and limited fire damage. The worst case postulated transient fire would cause no damage to conduits. The plant impact due to fire frequency is bounded by the individual component failure rates used in the internal events CCPRA, and as a result, this room has no additional equipment impact due to fire.

However, these rooms are destination locations for several human actions associated with the recovery of AFW flow control functions. Smoke and increased temperatures in these rooms will degrade these actions.

Table 4-C-1
A227/A316 Fire Analysis Results

Initiating Event	Fire Scenario	Frequency	Ignition Source	Major Impact	CDF
FCA227	None	4.62E-04	No equipment damage; however, Human Actions are degraded due to smoke and increased temperature in the room.	HX*, UQ*, FI*	2.53E-05

Suppression Systems

Flame detectors are installed at the 5' level throughout the room, and smoke detectors are installed at the 27' level. The compartment has installed wet pipe fire suppressions at both levels.

Fire Suppression Induced Equipment Failures

Based on the approach described in Section 4.3.4.4.4, equipment failure due to the inadvertent actuation of the automatic fire suppression system is assumed not to occur. Conduit, valves and other PRA equipment in the room are not considered to be susceptible to water damage.

A228	**Unit 1 Component Cooling Water Pump Room**	**Location:**	5' Auxiliary Building
		Fire Area:	15
		CDF:	Screened - Low Fire Ignition Frequency

Three Component Cooling Water Pumps (CCW 11, 12 and 13) and the two CCW Heat Exchangers (11 and 12) are located in Room A228. During normal operation one pump and one heat exchanger provide the necessary cooling. The major cooling loads for this system are the RCPs, the Letdown Heat Exchanger, and the Liquid Waste Evaporators.

The CCW Pump Room is approximately eighty-four feet long by approximately thirty-nine feet wide for an area of 3,265 square feet. The height of the compartment is twenty feet for a approximate room volume of 65,000 cubic feet. The compartment has a concrete floor, walls, and ceiling.

A228 Fire Analysis Results

Three fire scenarios are identified for this room, one for each pump. Given the configuration of the room, damage due to a pump fire is limited to the pump itself. The pump scenarios are screened due to the low fire frequencies and limited fire damage. The worst case postulated transient fire would cause no damage to cable trays or conduits and the only other targets are the CCW pumps. The plant impact due to fire frequency is bounded by the individual component failure rates used in the internal events CCPRA, and as a result, this room has no additional equipment impact due to fire. This room is screened due to low ignition frequency.

Fixed Ignition Sources

In the CCW Pump Room, the battery-operated emergency lighting, general area lighting, and small junction boxes are excluded from the fire modeling process on the basis that they would not be the source of fire intensity sufficient to result in other equipment damage. Several control and relief valves, which have a metal housing, are also excluded. A fire in these items would be small and confined within the boundaries of the item itself. Therefore. the only plausible ignition sources in the CCW Pump Room are the three CCW pumps.

Suppression Systems

The CCW Pump Room is equipped with a wet pipe automatic sprinkler fire suppression and smoke detectors. Compartment fire modeling does not take credit for the suppression.

Fire Suppression Induced Equipment Failures

Based on the approach described in Section 4.3.4.4.4, equipment failure due to the inadvertent actuation of the automatic fire suppression system is assumed not to occur. Cable and conduit, pumps and other PRA equipment in the room are not considered to be susceptible to water damage. In the event of a suppression actuation, the motors are of a design that prevents water spray intrusion and their connections and wiring also resist water intrusion. In addition, cabling in the room is not affected by water spray.

A306	**Unit 1 Cable Spreading Room**	**Location:**	27' Auxiliary Building
		Fire Area:	16
		CDF:	6.72E-6

The Unit 1 Cable Spreading Room (CSR) is the merging point for cables traveling to and from the Control Room. It also contains the following PRA related equipment:

- Unit 1 120VAC Vital AC Distribution Panels (1Y01, 1Y02, 1Y03 and 1Y04) and their associated inverters
- 125VDC Distribution Panels (1D01 and 1D02), associated DC sub-panels, and four associated chargers
- Unit 1 120VAC Instrument Buses (1Y09 and 1Y10)
- Unit 1 Engineering Safety Features Actuation System logic and actuation cabinets
- Unit 1 Auxiliary Feedwater Actuation System logic and actuation cabinets

The cable spreading room is rectangular with approximate dimensions forty-five feet wide, sixty-five feet long, and sixteen feet high. The nominal room area is 2,925 square feet. The room is separated from Unit 2 Cable Spreading Room by a common east-west wall which is a one hour fire rated barrier; a steel door is located in this wall at the west end.

The room contains numerous vertical floor mounted electrical cabinets aligned in rows. The majority of the cabinets are enclosed steel cabinets with sealed conduit entries. Some of the cabinets are designed with grates or vents in the top and/or sides (e.g., inverters). Numerous cable trays traverse horizontally in the ceiling. No motor driven equipment of significance exist in either compartment. Deterministic fire modeling will involve numerous cabinet fire scenarios with potential damage to overhead cable trays; depending on the details of the analysis, the individual cabinet fires may result in localized damage (i.e., damage only to the cabinet itself) or damage to abutted cabinets and/or cable trays due to radiant and conductive heat transfer.

Fire Analysis Results

One hundred and nineteen fire scenarios were identified for Unit 1 Cable Spreading Room. Ninety-four are the result of fixed ignition sources and twenty-five are due to transient ignition sources. These scenarios are represented by twenty-four fire initiating events. The consolidation of fire scenarios is based on an assessment of the functional impact and ignition frequency of each scenario. The frequency of each initiator is the sum of the frequencies of all the fire scenarios it represents.

The tables below list individual fire scenarios. For each scenario the initiating component is damaged.

Table 4-E-1
A306 Fixed Ignition Fire Scenarios Summary

Scenario	Fire Scenario Description	Trays and Panels Damaged by Fire
C1	Reserve Battery Charger	1AK30, 1AH23
C2	23 Battery Charger	No other effects
C3	24 Battery Charger	No other effects
C4	11 Battery Charger	No other effects
C5	12 Battery Charger	No other effects
C6	120VAC Instrument Transformer 11	No other effects
C7	120VAC Instrument Bus 11	No other effects
C8	120VAC Instrument Transformer 12	No other effects
C9	120VAC Instrument Bus 12	No other effects
C10	1B DG Logic Panel	No other effects
C11	2A DG Logic Panel	No other effects
C12	125VDC Bus 11	1PNL1D11, 1PNL1D12, 1PNL1D13
C13	125D Distribution Panel 1D11	1PNL1D01, 1PNL1D12, 1PNL1D13
C14	125D Distribution Panel 1D12	1PNL1D01, 1PNL1D11, 1PNL1D13
C15	125D Distribution Panel 1D13	1PNL1D01, 1PNL1D11, 1PNL1D12
C16	125VDC Bus 12	1PNL1D14
C17	125D Distribution Panel 1D14	1BUS1D02
C18	125D Distribution Panel 1D15	1PNL1D16, 1PNL1D17
C19	125D Distribution Panel 1D16	1PNL1D15, 1PNL1D17
C20	125D Distribution Panel 1D17	1PNL1D15, 1PNL1D16
C21	1 ESFAS ACT RELAY Panel ZA	1PNL1C67L
C22	1 ESFAS ACT Panel ZA	1PNL1C67, 1PNL1C91
C23	1 ESFAS ACT RELAY Panel ZB	1PNL1C68L
C24	1 ESFAS ACT Panel ZB	1PNL1C68, 1PNL1C94
C25	1 ESFAS Sensor Panel ZD	1PNL1C67L, 1PNL1C92
C26	1 ESFAS Sensor Panel ZE	1PNL1C91, 1PNL1C93
C27	1 ESFAS Sensor Panel ZF	1PNL1C92, 1PNL1C94
C28	1 ESFAS Sensor Panel ZG	1PNL1C68L, 1PNL1C93

Table 4-E-1
A306 Fixed Ignition Fire Scenarios Summary (Continued)

Scenario	Fire Scenario Description	Trays and Panels Damaged by Fire
C29	AFAS A ACT Cabinet	1AG51, 1AG49, 1AG33, 1PNL1C100D, 1PNL1C100F
C30	AFAS B ACT Cabinet	1AG51, 1AG49, 1AG33, 1PNL1C100E, 1PNL1C100G
C31	AFAS D Sensor Cabinet	1AG51, 1AG49, 1AG33, 1PNL1C100A, 1PNL1C100E, 1PNL1C100F
C32	AFAS E Sensor Cabinet	1AG51, 1AG49, 1AG33, 1PNL1C100B, 1PNL1C100D, PNL1C100G
C33	AFAS F Sensor Cabinet	1AG51, 1AG49, 1AG33, 1PNL1C100A, 1PNL1C100D
C34	AFAS G Sensor Panel	1AG51, 1AG49, 1AG33, 1PNL1C100B, 1PNL1C100E
C35	120VAC Inverter Backup Bus	No other effects
C36	120VAC Inverter 11	1AL06, 1AJ01, 1AL76, 1AJ07, 1AL93, 1AJ13, 1AL86
C37	120VAC Distribution Panel 11	No other effects
C38	REACTOR CLNT SYS CHANNELS	1AG26, 1AK77
C39	120VAC Inverter 12	1AH91, 1AL40, 1AK24
C40	120VAC Distribution Panel 12	No other effects
C41	120VAC Distribution Panel	1AG23
C42	120VAC Inverter 13	1AF96, 1WW55, 1AL22, 1AG02
C43	120VAC Distribution Panel 13	No other effects
C44	120VAC Regulating X 11	No other effects
C45	120VAC Inverter 14	1AH52, 1AF25
C46	120VAC Distribution Panel 14	No other effects
C47	120VAC Computer Inverter 11	1AJ23, 1AL59, 1AJ30, 1AF98
C48	1R01B Instrument Power Supply	1PNL1R01B
C49	1R01B Instrument Power Supply	1PNL1R01A
C50	Reactor Trip Switchgear Cabinet A	1AH33, 1AH26, 1PNL1Q01B
C51	Reactor Trip Switchgear Cabinet B	1AH33, 1AF11, 1AH26, 1AF04, 1PNL1Q01A, 1PNL1Q01C
C52	Reactor Trip Switchgear Cabinet C	1AH33, 1AF11, 1AH26, 1AF04, 1PNL1Q01B, 1PNL1Q01D
C53	Reactor Trip Switchgear Cabinet D	1AH33, 1AE96, 1AF03, 1AH26, 1AH10, 1PNL1Q01C, 1PNL1Q01E
C54	Reactor Trip Switchgear Cabinet E	1AH26, 1AF03, 1AE96, 1AH33, 1PNL1Q01D
C55	11 MT EHC Panel	1AL90, 1AJ09, 1AL04, 1AJ15
C56	TURB AUX SUPERVISORY INST	1AJ15

Table 4-E-1
A306 Fixed Ignition Fire Scenarios Summary (Continued)

Scenario	Fire Scenario Description	Trays and Panels Damaged by Fire
C57	Shutdown CCP Panel 1Q02B SEC3	1AG64, 1AF87, 1AG69, 1PNL1Q02A/S2, 1PNL1Q02B/S4
C58	Shutdown CCP Panel 1Q02B SEC4	1AF10, 1AG65, 1AG70, 1PNL1Q02B/S3, 1PNL1Q02B/S5
C59	Shutdown CCP Panel 1Q02B SEC5	1AF10, 1AG65, 1AG70, 1PNL1Q02B/S4, 1PNL1Q02B/S6
C60	Shutdown CCP Panel 1Q02B SEC6	1AG65, 1AG70, 1PNL1Q02B/S5, 1PNL1Q02B/S7
C61	Shutdown CCP Panel 1Q02B SEC7	1AG65, 1AG70, 1AF03, 1PNL1Q02B/S6, 1PNL1Q02B/S8
C62	Regulating CCP Panel 1Q02C SEC-8	1AG70, 1AF03, 1AG65, 1PNL1Q02B/S7, 1PNL1Q02B/S9
C63	Regulating CCP Panel 1Q02C SEC 9	1AG70, 1AG65, 1PNL1Q02B/S8, 1PNL1Q02B/S10
C64	Regulating CCP Panel 1Q02C SEC 10	1AE95, 1AG71, 1AG66, 1PNL1Q02B/S9, 1PNL1Q02B/S11
C65	Regulating CCP Panel 1Q02C SEC 11	1AE95, 1AG71, 1AG66, 1PNL1Q02B/S10, 1PNL1Q02B/S12
C66	Regulating CCP Panel 1Q02C SEC 12	1AF77, 1AG71, 1AF73, 1PNL1Q02B/S11, 1PNL1Q02B/S13
C67	Regulating CCP Panel 1Q02C SEC 13	1AF77, 1AG71, 1AF73, 1AG66, 1PNL1Q02B/S12, 1PNL1Q02B/S14
C68	Regulating CCP Panel 1Q02C SEC 14	1AF77, 1AG71, 1AF73, 1PNL1Q02B/S13, 1PNL1Q02B/S15
C69	Regulating CCP Panel 1Q02C SEC 15	1AG72, 1AG67, 1PNL1Q02B/S14, 1PNL1Q02B/S16
C70	Regulating CCP Panel 1Q02C SEC 16	1AG72, 1AG67, 1PNL1Q02B/S15
C71	Transformer/Generator Relay Panel	1AG73, 1AF40, 1AF37, 1PNL1C40B
C72	Transformer/Generator Relay Panel A	1AG73, 1AF40, 1AF37, 1AK97, 1AL95, 1PNL1C40A, 1PNL1C40C
C73	Transformer/Generator Relay Panel B	1AF40, 1PNL1C40B, 1PNL1C40D
C74	Transformer/Generator Relay Panel C	1AF33, 1AL95, 1AK97, 1AF62, 1AG73, 1AF35, 1PNL1C40C, 1PNL1C40E
C75	Transformer/Generator Relay Panel D	1AF35, 1AF62, 1AF32, 1AF31, 1AF30, 1AF33, 1AF29, 1AF28, 1AG73, 1PNL1C40D, 1PNL1C40F
C76	Transformer/Generator Relay Panel E	1AF32, 1AF28, 1AF29, 1AF31, 1AF30, 1PNL1C40E, 1PNL1C40G
C77	Transformer/Generator Relay Panel F	1AF30, 1AF31, 1AF28, 1AF32, 1AF29, 1PNL1C40F
C78	Annunciator Logic Panel 1K03	1AF25, 1AH32
C79	125D Reserve Charger Disconnect 1D50	No other effects
C80	125D Reserve Charger Disconnect 1D55	No other effects
C81	125D Reserve Charger Disconnect 1D56	No other effects
C82	125D Reserve Charger Disconnect 1D58	< No other effects >

Table 4-E-1
A306 Fixed Ignition Fire Scenarios Summary (Continued)

Scenario	Fire Scenario Description	Trays and Panels Damaged by Fire
C83	Battery Transfer Switch	No other effects
C84	Annunciator Logic Cabinet	1PNL1K02, 1J207
C85	Annunciator 12 Logic Panel	1PNL1K01
C86	CEDS CEA Control Panel	1PNL1Q03-2, 1PNL1Q03A
C87	CEDS CEA Control Panel	1PNL1Q03-1, 1PNL1Q03-3
C88	CEDS CEA Control Panel	1PNL1Q03-2, 1PNL1Q03-4
C89	CEDS CEA Control Panel	1PNL1Q03-3, 1PNL1Q03B
C90	CEDS CEA Control Panel	1PNL1Q03-1
C91	CEDS CEA Control Panel	1PNL1Q03-4
C92	Transformer 1X21	No other effects
C93	120I Bus 1Y09/1Y10 TIE BK	No other effects
C94	Unit XFRM Net Power Output Panel	< No other effects >

Table 4-E-2
A306 Transient Fire Scenarios Summary

Scenario	Description	Trays and Panels Damaged by Fire
T1	ESFAS	(Component only)
T2	AFAS	(Component only)
T3	1Q02(s)	(Component only)
T4	Any 1Q03 section	(Component only)
T5	Any 1C40 panel	(Component only)
T6	Any 1Q01 section	(Component only)
T7	1Y09 OR 1Y10	(Component only)
T8	1T11	(Component only)
T9	1T14	(Component only)
T10	1R01A/B	(Component only)
T11	Single battery charger	(Component only)

Table 4-E-2
A306 Transient Fire Scenarios Summary (Continued)

Scenario	Description	Trays and Panels Damaged by Fire
T12	Single inverter	(Component only)
T13	DC21 (1D15, 1D16, or 1D17)	(Component only)
T14	DC12 (1D02 or 1D14)	(Component only)
T15	1K01/1K02	(Component only)
T16	1K03	(Component only)
T17	EDG Logic Panels (1C70, 1C69/2C70)	(Component only)
T18	Single Vital AC panel	(Component only)
T19	Both 1X08 and 1X09	(Component only)
T20	1D11, 1D12, 1D13, OR 1D01	(Component only)
T21	1Y03 and 1Y04A	(Component only)
T22	Reserve charger	(Component only)
T23	Reserve switchgear	(Component only)
T24	Back up Bus	(Component only)
T25	Computer Inverter	(Component only)

Table 4-E-3
Cable Spreading Room Fire Analysis Results

Initiating Event	Fire Scenario	Freq	Ignition Source	Functional Impact	CDF
A306F1	C1, C57-59, C66-68, C78	1.58E-5	Reserve Battery Charger, Shutdown CCP Panel 1Q02B SEC3, Shutdown CCP Panel 1Q02B SEC4, Shutdown CCP Panel 1Q02B SEC5, Regulating CCP Panel 1Q02C SEC 12, Regulating CCP Panel 1Q02C SEC 13, Regulating CCP Panel 1Q02C SEC 14, Annunciator Logic Panel 1K03	HR, XW, QZ*, QQ, RS*, TX*, DW, DV, BS, PV*, MN, IA*, IB*, CV, SA*, SB*, EA*, EB*, RA*, RB*, PA*, PB*	3.02E-9
A306F2	C11-15, C47, T11, T17, T20	1.16E-4	2A DG Logic Panel, 125VDC Bus 11, 125D Distribution Panel 1D11, 25D Distribution Panel 1D12, 125D Distribution Panel 1D13, 120VAC Computer Inverter 11, Single battery charger, Transient damage to EDG Logic Panels (1C70, 1C69/2C70), 1D11, 1D12, 1D13, OR 1D01	DA, GH, HR, XW, QQ, NR, NS, KY, KS, RS*, TX*, T1, VC, RQ, MP*, SL*, MN, FT, F1*, LF, CV	4.23E-6
A306F3	C3, C5, C16, C17, T11, T14	2.41E-5	125VDC Bus 12, 125D Distribution Panel 1D14, 24 Battery Charger, 12 Battery Charger, Transient damage to Single battery charger, DC12 (1D02 or 1D14)	DB, HR, XW, QQ	1.01E-8
A306F4	C6, C7, T7	2.01E-4	120VAC Instrument Transformer 11, 120VAC Instrument Bus 11, Transient damage to 1Y090	HR, XW, QQ, E5	2.56E-8
A306F5	C8, C9, T7	2.01E-4	120VAC Instrument Transformer 12, 120VAC Instrument Bus 12, Transient Damage to 1Y10	HR, XW, QQ, E6	6.53E-9
A306F6	T19	1.74E-7	Transient induced 1X08 and 1X09 fires	HR, XW, QQ, E5, E6	2.30E-10
A306F7	C10, C18-20, T13, T17	4.37E-6	1B DG Logic Panel, 125D Distribution Panel 1D15, 125D Distribution Panel 1D16, 125D Distribution Panel 1D17, Transient damage to DC21 (1D15, 1D16, or 1D17), EDG Logic Panels (1C70, 1C69/2C70)	DC, GG, HR, XW,QQ	5.91E-8
A306F8	C21-27, T1	7.59E-6	ESFAS Actuation Relay Panel ZA, ESFAS Actuation Panel ZA, ESFAS Actuation Relay Panel ZB, ESFAS Actuation Panel ZB, ESFAS Sensor Panel ZD, ESFAS Sensor Panel ZE, ESFAS Sensor Panel ZF, Transient damage to ESFAS	ESFAS Spurious Channel A or B actuation, HR, XW, QQ	2.21E-7

Table 4-E-3
Cable Spreading Room Fire Analysis Results (Continued)

Initiating Event	Fire Scenario	Freq	Ignition Source	Functional Impact	CDF
A306F9	C29-34, T2	5.58E-6	AFAS A Actuation Cabinet, AFAS B Actuation Cabinet, AFAS D Sensor Cabinet, AFAS E Sensor Cabinet, AFAS F Sensor Cabinet, AFAS G Sensor Panel, Transient damage to AFAS	HR, XW, QZ*, QQ, S4, F1*, K4, PA*, PB*	6.77E-10
A306FA	C36, C51, C52, T6, T12, T18	4.07E-5	120VAC Inverter 11, Reactor Trip Switchgear Cabinet B, Reactor Trip Switchgear Cabinet C, Transient damage to Single inverter, Single Vital AC panel, Any 1Q01 section	E1, EA*, EB*, GE, GG*, GH, SA*, SB*, H5, HS*, AC*, HH*, HR, XW, QQ, QZ*, NR, NS*, S3, KX, KZ, KS, RS*, TX*, MC, VC, RR, BV, DW, DV, BS, RQ, MP, MN, IA, IB*, FT*, LF, CV, HA, RA, WY*, CS, K3, PA, PV*, PB*, RB*	2.33E-7
A306FB	C37	7.36E-7	120VAC Panel 1Y01	E1, HR, XW, QQ	1.10E-10
A306FC	C38, T21	8.52E-7	120VAC Panel 1Y01-1, and Transient induced 1Y03 and 1Y04A fires	E3, E4, HR, XW, QQ, RS*,TX*, T1	6.17E-8
A306FD	C39, T12, T18	3.76E-5	120VAC Inverter 12, and Transient induced inverter and vital AC panel fires	E2, GG, HS*, HH*, HR, XW, QZ*, QQ, NR*, S2, S4, TA, TB, KY, RS*, TX*, TT*, T1, PS*, PV*, PN, MP, SL*, MN, IA*, IB, FT, LF, CV, SA*, SB*, MV, HB, DL*, EA*, EB*, RA*, RB, CT, SR*, K4, TW, PA*, PB*	5.19E-7
A306FE	C40, C41	1.47E-6	120VAC Panel 12 (1Y02)	E2, HR, XW, QQ,	2.05E-9
A306FF	C42, T12, T18	3.76E-5	120VAC Inverter 13, and Transient induced inverter and vital AC panel fires	E3, GE, GH, HH*, HR, XW, QZ*, QQ, KS, PV*, MN*, IA*, IB*, FT*, CV, SA*, SB*, V1, MV, V5, DL*, EA, WY*, SH*, SR*, TE	4.36E-7
A306FG	C43	7.36E-7	120VAC Panel 13 (1Y02)	E3, HR, XW, QQ	1.39E-10
A306FH	C45, T12, T18	3.76E-5	120VAC Inverter 14, and Transient induced inverter and vital AC panel fires	E4, HR, XW, QZ*, QQ, RS*, TX*, DW, DV, BS, PV*, MN, IB*, CV, SA*, SB*, EA*, EB*, PA*, PB*	8.61E-8

Table 4-E-3
Cable Spreading Room Fire Analysis Results (Continued)

Initiating Event	Fire Scenario	Freq	Ignition Source	Functional Impact	CDF
A306FI	C46	7.36E-7	120VAC Panel 14 (1Y04)	E4, HR, XW, QQ	1.87E-10
A306FJ	C48, C49, T10	3.22E-6	NSR DC Instrument PWR Supply (1R01A), and Transient induced 1R01A/B fires	HR, XW, QQ, ES, BS	3.45E-9
A306FK	C50, C53, C54, C64, C65	3.68E-6	Reactor Trip SWGR A, C or D	HH*, HR, XW, QZ*, QQ, S3, TA, TB, RS*, TX, RR, PV*, SA*, SB*, EA*, EB*, CS, SG*, K3, PA*, PB*	1.69E-9
A306FL	C55, T8	1.43E-6	EHC Panel 11, and Transient induced 1T11 fires	Y1, AA, HR, XW, QQ, NR*, NS*, TA, TB, KS, TX, TT*, BV, DW, BS, RQ, SL*, MN, FT, LF, CV, WY*, CS, K3	3.09E-8
A306FM	C71-77, T5	7.11E-6	Transformer/Generator Relay panels: 1C40A,B,C,D,E,F or G, and Transient induced 1C40 panel fires	OP, GJ, AA, HR, XW, QQ, KZ, RS*, RR, DW, PV*, RQ, SL*,CV, SA*, SB*, HB, EA*, EB*, PA*, PB*	7.67E-7
A306FN	C35, C44, C56, C60-63, C69, C70, C79-94, T3, T4, T9, T15, T16, T22-25	6.26E-4	120VAC Inverter Backup Bus (1Y11), 120VAC Regulator X Transformer 11, Turb Aux Supervisory Inst Panel (1T14), Shutdown Rod Panels 5 or 6, Regulator Rod Panels 8 9,15 or 16, 125VDC Reserve Battery breakers and disconnect switches, Annunciator Panels 1K01 or 1K02, CEA Control Power cabinets, and Transient induced 1Q02, 1Q03, 1T14, 1K01/1K02, 1K03, reserve charger, reserve SWGR, BUB, and computer inverter induced fires	HR, XW, QQ	2.32E-8
A306FO	C2, C4, T11	2.18E-5	Battery Charger 11 or 23, and Transient induced battery charger fires	HR, XA*, XW, QQ	5.69E-10

*Note: The asterisk (*) indicates those top events which are impacted but not failed.

A306 __Analysis__

The Cable Spreading Room damage is evaluated using screening distances. The screening included a radiant, and three (center, along wall, in corner) in-plume cases for both damage and ignition. Targets within the screening range are considered damaged or ignited. The screening calculation uses standard FIVE heat release rates except for open equipment, where the heat release rate is calculated.

The screening distances, ignition source equipment heights, and cable tray heights are then analyzed to determine which trays are damaged or ignited. The results are then checked in the field for accuracy.

Conservative judgment is applied when open cable trays are in stacks where lower trays ignited. In general, the fire propagation continued up the stack, or to a covered tray. No targets are analyzed as outside-plume cases. Targets that are potential outside-plume cases are treated as inside-plume plume targets and analyzed accordingly.

When the ignition source abutted other panels, those panels were also considered damaged.

The table below summarizes the various CSR equipment and configuration cases.

Pre-Calculated Critical Damage Distances

FIRE SOURCE			HEAT RELEASE RATE	Conv HRR Frac	Rad HRR Frac	CRITICAL DAMAGE DISTANCE FROM SOURCE IN FEET						
						Damage				Ignition		
						In-Plume			RADIANT	In-Plume		
TYPE	ITEM	SPECIFIC				LF1	LF2	LF4		LF1	LF2	LF4
Fixed	Electrical Cabinet	Closed	65 BTU/sec	0.8	0.4	N/A	N/A	N/A	1.4	N/A	N/A	N/A
		Vented	65 BTU/sec	0.8	0.4	3.4	4.5	6.0	1.4	2.8	3.7	4.9
	Transformer	Open	65 BTU/sec	0.8	0.4	3.4	4.5	6.0	1.4	2.8	3.7	4.9
	Inverter	Open	170 BTU/sec	0.8	0.4	5.2	N/A	N/A	2.1	4.3	N/A	N/A
	01 Battery Charger	Open	170 BTU/sec	0.8	0.4	N/A	7.0	N/A	2.1	N/A	5.5	N/A
	1C40B and E	Open	270 BTU/sec	0.8	N/A	6.4	N/A	N/A	N/A	5.2	N/A	N/A
	1C40A, C, D, F	Open	190 BTU/sec	0.8	N/A	5.4	N/A	N/A	N/A	4.5	N/A	N/A
	1C40G	Open	130 BTU/sec	0.8	N/A	N/A	6.1	N/A	N/A	N/A	5.0	N/A
Transient	Trash	Refuse	100 BTU/sec	0.8	0.4	4.5	5.9	7.8	1.8	3.7	4.9	6.4

A306 Fire Ignition Frequency

Both fixed and transient ignition frequencies were determined for the Unit 1 Cable Spreading Room.

Fixed Ignition Frequency

The fixed ignition frequency is determined by starting with the compartment fixed ignition frequency results of Section 4.3.2 and then developing a scenario specific ignition frequency as described in Section 4.3.4.3. The sources and parameters used for A306 are:

Source	Generic Fire Frequency	Location Weighting	Source Weighting	Total Sources in Aux Bldg	Room Specific Frequency
Battery Chargers	4.0E-3	2	5	15	2.67E-3
Electrical Cabinets	3.2E-3	1	-	-	3.20E-3
Transformers (Dry)	7.9E-3	2	5	80	9.88E-4
Fire Protection Panels	2.4E-3	2	1	35	1.37E-4

BATTERY CHARGERS

The station battery chargers are analyzed as sealed electrical cabinets since their cooling fans have louvered shutters.

The Reserve Battery Charger, and open top cabinet, is evaluated using a calculated heat release rate based on an open door cabinet with unqualified cable, using the formula:

$$HRR = 8.5E\text{-}04 * \text{fuel loading}$$

The estimated HRR is 170 btu/sec.

ELECTRICAL CABINETS

In accordance with FIVE, cabinets, control panels, and inverters fall into this category. Electrical cabinets represent potential fire sources that are difficult to characterize given the range of potential configurations. To facilitate the analysis, electrical cabinets are categorized consistent with the EPRI Fire PRA Implementation Guide as sealed, vented, and open. The fire modeling for each of these cabinet categories is performed as follows.

- Sealed Cabinets

  Cabinets that are enclosed on all four vertical sides, as well as top and bottom, with no ventilation openings, grills or louvers are classified as sealed. For this type of cabinet it is assumed that the fire would not propagate beyond the contents of the cabinet itself. Damage to adjacent cabinets and nearby metal encased cable trays is analyzed using the air gap approach Combustion products will suppress fire growth; as such, a significant fire will not result to warrant more than the Radiant Heat Worksheet calculation.

Fires in enclosed vertical electrical cabinets are assumed to fail only components within that particular cabinet. It is assumed that no damage occurs to adjacent cabinets, provided that cabinets are separated by a double wall with an air gap, and there is no sensitive equipment contained within the adjacent cabinet. Enclosed cabinets in the cable spreading room are assumed not to propagate a fire beyond the boundaries of the cabinet itself.

- Vented Cabinets

Cabinets with no top ventilation, but having grills or louvers on at least one vertical side, are classified as vented. The analyzed height of fire is assumed to be the height of the top louver. Damage to adjacent cabinets and nearby metal encased cable trays is analyzed using the air gap approach.

Fire damage analysis for targets above these cabinets used the In-plume Worksheets, using a convective Heat Release Rate (HRR) of 80% (subtracting 20% of the HRR associated with radiant heat), per the EPRI Fire PRA Implementation. These cabinets are also analyzed for potential fire damage using the Radiant Heat worksheet for nearby exposed vertical and horizontal (overhead) cable trays; conservatively, 40% of the total HRR is assumed to be radiant.

- Open Cabinets (Inverters)

Cabinets with ventilation on the top of the cabinet (e.g., screened tops, ventilation fans) are categorized as open. The analyzed height of fire is assumed to be the height of the top of the cabinet. Damage to adjacent cabinets and nearby metal encased cable trays is analyzed using the air gap approach.

In the cable spreading room, the only open top cabinets are the inverters. Fire damage analysis for targets above such cabinets is used In-plume Worksheets. Using the HRR for an open door cabinet with non-qualified cable, the calculated HRR for inverters is 170 btu/sec. These cabinets are also analyzed for potential fire damage using the Radiant Heat worksheet for nearby exposed vertical and horizontal (overhead) cable trays.

- 1C40 Panels

The unit protection panels (A through F) are located behind a partial height gypsum wall and locked door. They are full height vertical switchboards typical of station protection equipment including mounted drawout protective relays, lockout devices, indicators, and test facilities. These components represent a small fuel loading (the largest equipment is protective relays which are inside a sealed metal enclosure). There are no barriers or metal extensions between panel sections.

Due to their vertical construction, it is unlikely that a fire in these panels would spread or propagate in a horizontal plane. Conservatively, it is assumed that the fire would be one panel wide (about three feet).

Due to the open configuration, panel heat release rates (HRR) are calculated based on an open door cabinet with unqualified cable, using the formula:

$$HRR = 8.5E\text{-}04 * \text{fuel loading}$$

The HRRs are then binned into one of three ratings, 130, 190, or 270 btu/sec.

Cabinet and control panel (but not inverter) ignition frequencies are multiplied by a 0.2 severity factor: This value bounds various severity factors (except diesel generators) and is, therefore, conservative. It is assumed that only severe fires generate enough smoke to challenge room habitability, but, any fire is sufficient to actuate the room fire suppression.

TRANSFORMERS

Transformers in the cable spreading room have a single slit vent at the top edge and are analyzed consistent with the approach used for vented electrical cabinets. The analyzed height of fire is assumed to be the height of the top louver. Damage to adjacent cabinets and nearby metal encased cable trays is analyzed using the air gap approach.

The damage distance to targets above the transformers is calculated using In-plume Worksheets and a convective Heat Release Rate (HRR). They are also analyzed for potential fire damage using the Radiant Heat worksheet for nearby exposed vertical and horizontal (overhead) cable trays.

Because of dense wiring mass, the transformer severity factor is assumed to be 1.0.

FIRE PROTECTION PANELS

The sole fire protection panel is a small sealed cabinet with sealed conduit entries. It has been excluded from further analysis since any ignition would be small and confined within the cabinet itself.

CALCULATION OF COMPONENT FIXED IGNITION FREQUENCY

Each individual component is assigned an ignition frequency apportioned by dividing the compartment specific frequency for the component type by the total number of each ignition source of that type. The chart below summarizes the calculation and incorporates the applicable severity factor.

A306 Equipment Count and Individual Frequency Summary

Equipment Type	Panel Count	Room Specific Frequency	Severity Factor	Individual Frequency
Battery Chargers	5	2.67E-3	0.2	1.07E-5
Electrical Cabinets or Panels	87	3.20E-3		
(Cabinet or Panel)	(80)		0.2	7.36E-7
(Inverter)	(7)		1.0	3.68E-5
Transformers	5	9.88E-4	1.0	1.98E-4
Fire Protection Panels	1	1.37E-4	0.2	2.74E-6

Transient Ignition Frequency

The transient ignition frequency is determined by starting with the compartment transient ignition frequency results of Section 4.3.2 and then developing a scenario specific ignition frequency as described in Section 4.3.4.3. The sources and parameters used for A306 are:

Source	Generic Fire Frequency	Location Weighting	Source Weighting	Total Sources in Aux Bldg	Room Specific Frequency
Cable fires - welding	5.1E-3	2	1	232	4.40E-5
Transient fires - welding	3.1E-2	2	1	232	2.67E-4
Transient - other	1.3E-3	2	7	232	7.84E-5

The transient fire is modeled as a maintenance refuse fire and, therefore, uses the "Transient - other" as the ignition frequency, F_{it} = 7.84E-05. Fire suppression is not credited in this analysis. Therefore, conservatively, $P_{fs} = 1.0$.

TRANSIENT COMBUSTIBLE IN RANGE OF TARGETS (U) AND RESULTING FREQUENCY

The formula for the Transient Combustible in Range of Targets (u) is:

$$u = (A_s + A_{sr}) / A_F$$

The refuse fire In-Plume worksheets show that the worst case plume (a corner configuration) is eight feet above the source, a three feet tall fire. The lowest overhead CSR cable trays are about ten feet up, but the corner location trays are higher. Therefore, it is judged unlikely that overhead cable damage will result from a postulated transient fire and A_s = zero.

The remaining damage assessment centered on radiant damage to floor mounted equipment. The radiant damage distance for the 100 btu/sec trash can fire is 1.8 feet (using a critical flux value of 1.0 btu/sec/ft^2). The A_{sr} for each component or equipment grouping (depending on the physical arrangement) is estimated by multiplying a 1.8 foot margin times the available floor area around the target component.

The cable spreading room floor area is approximately 2,925 square feet. The floor area displaced by cabinets is calculated by summing the individual cabinet areas using the A_{sr} perimeter dimensions as length and width. The total cabinet area is 496 square feet, so:

$$A_F = 2925\text{ft}^2 - 496\text{ft}^2 = 2{,}429\text{ft}^2$$

and for the CSR, $u = A_{sr} / A_F$

and: $P_{fs} = 1$

$$\text{Since: } F_t = F_{it} * u * P_{fs} = F_{it} * [A_{sr} / A_F] * 1$$

$$= 7.84\text{E-}4 * [A_{sr} / 2429\text{ft}^2]$$

The grouping of similar components resulted twenty-five scenarios, tabularized below.

Table 4-E-4
A306 Transient Fire Scenarios and Frequency Summary

Scenario	Description	A_{sr}	F_{it}	Number of cabinets	Floor Area
T1	ESFAS	76	2.44E-06	1	38
T2	AFAS	36	1.16E-06	1	16
T3	1Q02(s)	99	3.20E-06	1	63
T4	Any 1Q03 section	52	1.69E-06	1	42
T5	Any 1C40 panel	61	1.96E-06	1	8
T6	Any 1Q01 section	52	1.69E-06	1	48
T7	1Y09 OR 1Y10	23	7.56E-07	2	24
T8	1T11	22	6.98E-07	1	10
T9	1T14	10	3.34E-07	1	2
T10	1R01A/B	54	1.74E-06	1	36
T11	Single battery charger	16	5.14E-07	4	32
T12	Single inverter	11	3.68E-07	4	32
T13	DC21 (1D15, 1D16, or 1D17)	28	9.01E-07	1	0
T14	DC12 (1D02 or 1D14)	25	8.14E-07	1	17
T15	1K01/1K02	28	8.91E-07	1	7
T16	1K03	10	3.20E-07	1	7
T17	EDG Logic Panels (1C70, 1C69/2C70)	16	5.23E-07	2	10
T18	Single Vital AC panel	13	4.07E-07	4	10
T19	Both 1X08 and 1X09	5	1.74E-07	1	7
T20	1D11, 1D12, 1D13, OR 1D01	37	1.20E-06	1	31
T21	1Y03 and 1Y04A	4	1.16E-07	1	0
T22	Reserve charger	31	1.01E-06	1	17
T23	Reserve switchgear	13	4.07E-07	6	36
T24	Back up Bus	8	2.62E-07	1	5
T25	Computer Inverter	27	8.72E-07	1	15

Total Ignition Frequency

Equipment and trays damaged in each scenario (transient and fixed) were mapped to corresponding plant model top events. The top events are then binned for impact. The binnings are then used to group the fire scenarios into plant model scenarios. Each plant model scenario is the sum of individual fire scenarios listed below.

Plant model scenario A306F1:

C1	Reserve Battery Charger	1.07E-05
C57	Shutdown CCP Panel 1Q02B SEC3	7.36E-07
C58	Shutdown CCP Panel 1Q02B SEC4	7.36E-07
C59	Shutdown CCP Panel 1Q02B SEC5	7.36E-07
C66	Regulating CCP Panel 1Q02C SEC 12	7.36E-07
C67	Regulating CCP Panel 1Q02C SEC 13	7.36E-07
C68	Regulating CCP Panel 1Q02C SEC 14	7.36E-07
C78	Annunciator Logic Panel 1K03	7.36E-07
	Total frequency for plant model A306F1 is:	1.58E-05

Plant model scenario A306F2:

C11	2A DG Logic Panel	7.36E-07
C12	125VDC Bus 11	7.36E-07
C13	125D Distribution Panel 1D11	7.36E-07
C14	125D Distribution Panel 1D12	7.36E-07
C15	125D Distribution Panel 1D13	7.36E-07
C47	120VAC Computer Inverter 11	1.10E-04
T11	Single battery charger	5.14E-07
T17	EDG Logic Panels (1C70, 1C69/2C70)	5.23E-07
T20	1D11, 1D12, 1D13, OR 1D01	1.20E-06
	Total frequency for plant model A306F2 is:	1.16E-04

Plant model scenario A306F3:

C16	125VDC Bus 12	7.36E-07
C17	125D Distribution Panel 1D14	7.36E-07
C3	24 Battery Charger	1.07E-05
C5	12 Battery Charger	1.07E-05
T11	Single battery charger	5.14E-07
T14	DC12 (1D02 or 1D14)	8.14E-07
	Total frequency for plant model A306F3 is:	2.41E-05

Plant model scenario A306F4:

C6	120VAC Instrument Transformer 11	2.00E-04
C7	120VAC Instrument Bus 11	7.36E-07
T7	1Y09 OR 1Y10 (1Y09)	7.56E-07
	Total frequency for plant model A306F4 is:	2.01E-04

Plant model scenario A306F5:

C8	120VAC Instrument Transformer 12	2.00E-04
C9	120VAC Instrument Bus 12	7.36E-07
T7	1Y09 OR 1Y10 (1Y10)	7.56E-07
	Total frequency for plant model A306F5 is:	2.01E-04

Plant model scenario A306F6:

T19	Both 1X08 and 1X09	1.74E-07
	Total frequency for plant model A306F6 is:	1.74E-07

Plant model scenario A306F7:

C10	1B DG Logic Panel	7.36E-07
C18	125D Distribution Panel 1D15	7.36E-07
C19	125D Distribution Panel 1D16	7.36E-07
C20	125D Distribution Panel 1D17	7.36E-07
T13	DC21 (1D15, 1D16, or 1D17)	9.01E-07
T17	EDG Logic Panels (1C70, 1C69/2C70)	5.23E-07
	Total frequency for plant model A306F7 is:	4.37E-06

Plant model scenario A306F8:

C21	1 ESFAS ACT RELAY Panel ZA	7.36E-07
C22	1 ESFAS ACT Panel ZA	7.36E-07
C23	1 ESFAS ACT RELAY Panel ZB	7.36E-07
C24	1 ESFAS ACT Panel ZB	7.36E-07
C25	1 ESFAS Sensor Panel ZD	7.36E-07
C26	1 ESFAS Sensor Panel ZE	7.36E-07
C27	1 ESFAS Sensor Panel ZF	7.36E-07
T1	ESFAS	2.44E-06
	Total frequency for plant model A306F8 is:	7.59E-06

Plant model scenario A306F9:

C29	AFAS A ACT Cabinet	7.36E-07
C30	AFAS B ACT Cabinet	7.36E-07
C31	AFAS D Sensor Cabinet	7.36E-07
C32	AFAS E Sensor Cabinet	7.36E-07
C33	AFAS F Sensor Cabinet	7.36E-07
C34	AFAS G Sensor Panel	7.36E-07
T2	AFAS	1.16E-06
	Total frequency for plant model A306F9 is:	5.58E-06

Plant model scenario A306FA:

C36	120VAC Inverter 11	3.68E-05
C51	Reactor Trip Switchgear Cabinet B	7.36E-07
C52	Reactor Trip Switchgear Cabinet C	7.36E-07
T12	Single inverter	3.68E-07
T18	Single Vital AC panel	4.07E-07
T6	Any 1Q01 section	1.69E-06
	Total frequency for plant model A306FA is:	4.07E-05

Plant model scenario A306FB:

C37	120VAC Distribution Panel 11	7.36E-07
	Total frequency for plant model A306FB is:	7.36E-07

Plant model scenario A306FC:

C38	REACTOR CLNT SYS CHANNELS	7.36E-07
T21	1Y03 and 1Y04A	1.16E-07
	Total frequency for plant model A306FC is:	8.52E-07

Plant model scenario A306FD:

C39	120VAC Inverter 12	3.68E-05
T12	Single inverter	3.68E-07
T18	Single Vital AC panel (1Y02)	4.07E-07
	Total frequency for plant model A306FD is:	3.76E-05

Plant model scenario A306FE:

C40	120VAC Distribution Panel 12	7.36E-07
C41	120VAC Distribution Panel	7.36E-07
	Total frequency for plant model A306FE is:	1.47E-06

Plant model scenario A306FF:

C42	120VAC Inverter 13	3.68E-05
T12	Single inverter	3.68E-07
T18	Single Vital AC panel 1Y03	4.07E-07
	Total frequency for plant model A306FF is:	3.76E-05

Plant model scenario A306FG:

C43	120VAC Distribution Panel 13	7.36E-07
	Total frequency for plant model A306FG is:	7.36E-07

Plant model scenario A306FH:

C45	120VAC Inverter 14	3.68E-05
T12	Single inverter	3.68E-07
T18	Single Vital AC panel (1Y04)	4.07E-07
	Total frequency for plant model A306FH is:	3.76E-05

Plant model scenario A306FI:

C46	120VAC Distribution Panel 14	7.36E-07
	Total frequency for plant model A306FI is:	7.36E-07

Plant model scenario A306FJ:

C48	1R01B Instrument Power Supply	7.36E-07
C49	1R01B Instrument Power Supply	7.36E-07
T10	1R01A/B	1.74E-06
	Total frequency for plant model A306FJ is:	3.22E-06

Plant model scenario A306FK:

C50	Reactor Trip Switchgear Cabinet A	7.36E-07
C53	Reactor Trip Switchgear Cabinet D	7.36E-07
C54	Reactor Trip Switchgear Cabinet E	7.36E-07
C64	Regulating CCP Panel 1Q02C SEC 10	7.36E-07
C65	Regulating CCP Panel 1Q02C SEC 11	7.36E-07
	Total frequency for plant model A306FK is:	3.68E-06

Plant model scenario A306FL:

C55	11 MT EHC Panel	7.36E-07
T8	1T11	6.98E-07
	Total frequency for plant model A306FL is:	1.43E-06

Plant model scenario A306FM:

C71	Transformer/Generator Relay Panel	7.36E-07
C72	Transformer/Generator Relay Panel A	7.36E-07
C73	Transformer/Generator Relay Panel B	7.36E-07
C74	Transformer/Generator Relay Panel C	7.36E-07
C75	Transformer/Generator Relay Panel D	7.36E-07
C76	Transformer/Generator Relay Panel E	7.36E-07
C77	Transformer/Generator Relay Panel F	7.36E-07
T5	Any 1C40 panel	1.96E-06
	Total frequency for plant model A306FM is:	7.11E-06

Plant model scenario A306FO:

C2	23 Battery Charger	1.07E-05
C4	11 Battery Charger	1.07E-05
T11	Single battery charger	5.14E-07
	Total frequency for plant model A306FO is:	2.18E-05

Plant model scenario A306FN:

C35	120VAC Inverter Backup Bus	7.36E-07
C44	120VAC Regulating X 11	4.00E-04
C56	TURB AUX SUPERVISORY INST	7.36E-07
C60	Shutdown CCP Panel 1Q02B SEC6	7.36E-07
C61	Shutdown CCP Panel 1Q02B SEC7	7.36E-07
C62	Regulating CCP Panel 1Q02C SEC-8	7.36E-07
C63	Regulating CCP Panel 1Q02C SEC 9	7.36E-07
C69	Regulating CCP Panel 1Q02C SEC 15	7.36E-07
C70	Regulating CCP Panel 1Q02C SEC 16	7.36E-07
C79	125D Reserve Charger Disconnect 1D50	7.36E-07
C80	125D Reserve Charger Disconnect 1D55	7.36E-07
C81	125D Reserve Charger Disconnect 1D56	7.36E-07
C82	125D Reserve Charger Disconnect 1D58	7.36E-07
C83	Battery Transfer Switch	7.36E-07
C84	Annunciator Logic Cabinet	7.36E-07
C85	Annunciator 12 Logic Panel	7.36E-07
C86	CEDS CEA Control Panel	7.36E-07
C87	CEDS CEA Control Panel	7.36E-07
C88	CEDS CEA Control Panel	7.36E-07
C89	CEDS CEA Control Panel	7.36E-07
C90	CEDS CEA Control Panel	7.36E-07
C91	CEDS CEA Control Panel	7.36E-07
C92	Transformer 1X21	2.00E-04
C93	120I Bus 1Y09/1Y10 TIE BK	7.36E-07
C94	Unit XFRM Net Power Output Panel	7.36E-07
T3	1Q02(s)	3.20E-06
T4	Any 1Q03 section	1.69E-06
T9	1T14	3.34E-07
T15	1K01/1K02	8.91E-07
T16	1K03	3.20E-07
T22	Reserve charger	1.01E-06
T23	Reserve switchgear	4.07E-07
T24	Back up Bus	2.62E-07
T25	Computer Inverter	8.72E-07

Total frequency for plant model A306FN is: 6.26E-04

<u>Fire Suppression</u>

Although an automatic total flooding halon suppression system is installed in the Cable Spreading Room, the probability of actuation prior to target damage is not evaluated as is the likelihood of fire brigade response to manually suppress the fire prior to target damage.

<u>Fire Suppression Induced Equipment Failure</u>

Further, when suppression actuates, the suppression agent (halon) causes no equipment damage.

A302	**Unit 2 Cable Spreading Room**	**Location:**	27' Auxiliary Building
		Fire Area:	17
		CDF:	6.81E-7

The Unit 2 Cable Spreading Room (CSR) is the merging point for cables traveling to and from the Control Room. It also contains the following PRA related equipment:

- Unit 2 120VAC Vital AC Distribution Panels (2Y01, 2Y02, 2Y03 and 2Y04) and their associated inverters for 2Y03 and 2Y04
- 125VDC Distribution Panels (2D01 and 2D02), associated DC sub-panels, and four associated chargers
- Unit 2 120VAC Instrument Buses (2Y09 and 2Y10)
- Unit 2 Engineering Safety Features Actuation System logic and actuation cabinets
- Unit 2 Auxiliary Feedwater Actuation System logic and actuation cabinets

The cable spreading room is rectangular with approximate dimensions forty-five feet wide, sixty-five feet long, and sixteen feet high. The nominal room area is 2,695 square feet. The room is separated from Unit 1 Cable Spreading Room by a common east-west wall which is a one-hour fire rated barrier and a steel door is located in this wall at the west end.

The room contains numerous vertical floor-mounted electrical cabinets aligned in rows. The majority of the cabinets are enclosed steel cabinets with sealed conduit entries. Some of the cabinets are designed with grates or vents in the top and/or sides (e.g., inverters). Numerous cable trays traverse horizontally in the ceiling. No motor-driven equipment of significance exist in either compartment. Deterministic fire modeling will involve numerous cabinet fire scenarios with potential damage to overhead cable trays; depending on the details of the analysis, the individual cabinet fires may result in localized damage (i.e., damage only to the cabinet itself) or damage to abutted cabinets and/or cable trays due to radiant and conductive heat transfer.

Fire Analysis Results

One hundred and eleven fire scenarios are identified for Unit 2 Cable Spreading Room. Eighty-eight are the result of fixed ignition sources and twenty-three are due to transient ignition sources. These scenarios are represented by twenty-four fire initiating events. The consolidation of fire scenarios is based on an assessment of the functional impact and ignition frequency of each scenario. The frequency of each initiator is the sum of the frequencies of all the fire scenarios it represents.

The tables below list individual fire scenarios. For each scenario the initiating component is damaged.

Table 4-E-5
A302 Fixed Ignition Fire Scenarios Summary

Scenario	Fire Scenario Description	Trays and Panels Damaged by Fire
C1	21 Battery Charger	< No other effects >
C2	22 Battery Charger	< No other effects >
C3	13 Battery Charger	< No other effects >
C4	14 Battery Charger	< No other effects >
C5	120VAC Instrument Transformer 21	< No other effects >
C6	120VAC Instrument Bus 21	< No other effects >
C7	120VAC Instrument Transformer 22	< No other effects >
C8	120VAC Instrument Bus 22	< No other effects >
C9	120VAC Bus 2Y09/2Y10 Tie Breaker	< No other effects >
C10	2B DG Logic Panel	< No other effects >
C11	125VDC Bus 21	2PNL2D15, 2PNL2D16, 2PNL2D17
C12	125VDC Distribution Panel 2D15	2PNL2D01, 2PNL2D17, 2PNL2D16
C13	125VDC Distribution Panel 2D16	2PNL2D01, 2PNL2D15, 2PNL2D17
C14	125VDC Distribution Panel 2D17	2PNL2D01, 2PNL2D15, 2PNL2D16
C15	125VDC Bus 22	2PNL2D14
C16	125VDC Distribution Panel 2D14	2BUS2D02
C17	125VDC Distribution Panel 2D11	2PNL2D12, 2PNL2D13
C18	125VDC Distribution Panel 2D12	2PNL2D11, 2PNL2D13
C19	125VDC Distribution Panel 2D13	2PNL2D11, 2PNL2D12
C20	ESFAS Actuation Relay Panel ZA	2PNL2C67L
C21	ESFAS A Actuation Cabinet	2PNL2C67, 2PNL2C91
C22	ESFAS Actuation Relay Panel ZB	2PNL2C68L
C23	ESFAS B Actuation Cabinet	2PNL2C68, 2PNL2C94
C24	ESFAS D Sensor Cabinet	2PNL2C67L, 2PNL2C92
C25	ESFAS E Sensor Panel	2PNL2C91, 2PNL2C93
C26	ESFAS F Sensor Cabinet	2PNL2C92, 2PNL2C94
C27	ESFAS G Sensor Cabinet	2PNL2C68L, 2PNL2C93
C28	AFAS A Actuation Cabinet	2AH53, 2AH51, 2PNL2C100D, 2PNL2C100F

Table 4-E-5
A302 Fixed Ignition Fire Scenarios Summary (Continued)

Scenario	Fire Scenario Description	Trays and Panels Damaged by Fire
C29	AFAS B Actuation Cabinet	2AH53, 2AH51, 2PNL2C100E, 2PNL2C100G
C30	AFAS D Sensor Cabinet	2AH53, 2AH51, 2PNL2C100A, 2PNL2C100E, 2PNL2C100F
C31	AFAS E Sensor Cabinet	2AH53, 2AH51, 2PNL2C100B, 2PNL2C100D, 2PNL2C100G
C32	AFAS F Sensor Cabinet	2AH53, 2AH51, 2PNL2C100A, 2PNL2C100D
C33	AFAS G Sensor Cabinet	2AH53, 2AH51, 2PNL2C100B, 2PNL2C100E
C34	120VAC Inverter Back up Bus	< No other effects >
C35	120VAC Inverter 21	2AK11, 2AJ46, 2AJ38, 2AJ50,
C36	120VAC Distribution Panel 21	< No other effects >
C37	REACTOR COOLANT SYS CHANN	2AK12, 2AG43, 2AG37
C38	120VAC Inverter 22	2AF62, 2AG59
C39	120VAC Distribution Panel 22	< No other effects >
C40	REACTOR CLNT SYS CH TR-12	2AG87, 2AH37, 2AG79
C41	120VAC Inverter 23	2AG53, 2AH99, 2AG60
C42	120VAC Distribution Panel 23	< No other effects >
C43	120VAC Inverter 24	2AG26, 2AG44, 2AG38
C44	120VAC Distribution Panel 24	< No other effects >
C45	120VAC Computer Inverter	2AG36, 2AH19, 2AG24
C46	Regulating Transformer for Back up Bus	2AH32
C47	2R01A Instrument Power Supply	2PNL2R01B
C48	2R01B Instrument Power Supply	2PNL2R01A
C49	2 RPS ReactorTrip Switchgear Cabinet A	2AH90, 2AF78, 2AH79, 2PNL2Q01B
C50	2 RPS ReactorTrip Switchgear Cabinet B	2AH89, 2AH78, 2PNL2Q01A, 2PNL2Q01C
C51	RPS Unit 2 ReactorTrip Switchgear	2AF72, 2AH89, 2AH71, 2AH78, 2PNL2Q01B, 2PNL2Q01D
C52	2 RPS ReactorTrip Switchgear Cabinet C	2AF72, 2AH89, 2AH71, 2AH78, 2PNL2Q01C, 2PNL2Q01E
C53	2 RPS ReactorTrip Switchgear Cabinet D	2AF68, 2AH89, 2AH71, 2AH78, 2AH61, 2PNL2Q01D
C54	Electro-Hydraulic Control	2AK71, 2AJ48, 2AL05, 2AF80, 2AK09, 2AJ44, 2AJ36
C55	Shutdown CCP Panel 2Q02B SEC3	2AF84, 2AG93, 2AF82, 2AG85, 2PNL2Q02B/S04
C56	Shutdown CCP Panel 2Q02B SEC4	2AF84, 2AG93, 2AF82, 2PNL2Q02B/S03, 2PNL2Q02B/S05

Table 4-E-5
A302 Fixed Ignition Fire Scenarios Summary (Continued)

Scenario	Fire Scenario Description	Trays and Panels Damaged by Fire
C57	Shutdown CCP Panel 2Q02B SEC5	2AF84, 2AG93, 2AF82, 2PNL2Q02B/S04, 2PNL2Q02B/S06
C58	Shutdown CCP Panel 2Q02B SEC6	2AF77, 2AG93, 2AG85, 2PNL2Q02B/S05, 2PNL2Q02B/S07
C59	Shutdown CCP Panel 2Q02B SEC7	2AF77, 2AG92, 2AG84, 2PNL2Q02B/S06, 2PNL2Q02B/S08
C60	Regulating CCP Panel 2Q02C SEC 8	2AG92, 2AG84, 2PNL2Q02B/S07, 2PNL2Q02B/S09
C61	Regulating CCP Panel 2Q02C SEC 9	2AF72, 2AG92, 2AG84, 2PNL2Q02B/S08, 2PNL2Q02B/S10
C62	Regulating CCP Panel 2Q02C SEC 10	2AF72, 2AG92, 2AG84, 2PNL2Q02B/S09, 2PNL2Q02B/S11
C63	Regulating CCP Panel 2Q02C SEC 11	2AG92, 2AG84, 2PNL2Q02B/S10, 2PNL2Q02B/S12
C64	Regulating CCP Panel 2Q02C SEC 12	2AG92, 2AF67, 2PNL2Q02C/S11, 2PNL2Q02C/S13
C65	Regulating CCP Panel 2Q02C SEC 13	2AF67, 2AG84, 2AJ55, 2AG92, 2PNL2Q02C/S12, 2PNL2Q02C/S14
C66	Regulating CCP Panel 2Q02C SEC 14	2AJ55, 2AG83, 2AG91, 2PNL2Q02C/S13, 2PNL2Q02C/S15
C67	Regulating CCP Panel 2Q02C SEC 15	2AG83, 2AG91, 2PNL2Q02C/S14, 2PNL2Q02C/S16
C68	Regulating CCP Panel 2Q02C SEC 16	2AG83, 2AG91, 2PNL2Q02C/S15, 2PNL2Q02C/S17
C69	Regulating CCP Panel 2Q02C SEC 17	2AF61, 2AG91,2PNL2Q02C/S16
C70	Transformer/Generator Relay Panel A	2AJ85, 2AJ31, 2AJ39, 2AG95, 2AG78, 2PNL2C40B
C71	Transformer/Generator Relay Panel B	2AJ31, 2AG95, 2AJ85, 2AJ39, 2PNL2C40A, 2PNL2C40C
C72	Transformer/Generator Relay Panel C	2AG95, 2AK04, 2PNL2C40B, 2PNL2C40D
C73	Transformer/Generator Relay Panel D	2AG95, 2AJ21, 2AJ29, 2AJ27, 2PNL2C40C, 2PNL2C40E
C74	Transformer/Generator Relay Panel E	2AG95, 2AJ21, 2AJ29, 2AJ27, 2PNL2C40D, 2PNL2C40F
C75	Transformer/Generator Relay Panel F	2AH94, 2AJ07, 2AJ14, 2AH82, 2AG95, 2AH93, 2PNL2C40E, 2PNL2C40G
C76	Transformer/Generator Relay Panel G	2AH93, 2AH94, 2AJ07, 2AH82, 2AJ14, 2PNL2C40F
C77	Annunciator Logic Panel	2AH88, 2AF62, 2AH77, 2AH70, 2AJ71
C78	Annunciator 21 Logic Control Panel	2PNL2K02
C79	Annunciator 22 Logic Control Panel	2PNL2K01
C80	CEDS UNIT 2 CEA Control Panel	2PNL2Q03-2, 2PNL2Q03A
C81	CEDS UNIT 2 CEA Control Panel	2PNL2Q03-1, 2PNL2Q03-3
C82	CEDS UNIT 2 CEA Control Panel	2PNL2Q03-2, 2PNL2Q03-4
C83	CEDS UNIT 2 CEA Control Panel	2PNL2Q03-3, 2PNL2Q03B

Table 4-E-5
A302 Fixed Ignition Fire Scenarios Summary (Continued)

Scenario	Fire Scenario Description	Trays and Panels Damaged by Fire
C84	CEDS UNIT 2 CEA Control Panel	2PNL2Q03-1
C85	CEDS UNIT 2 CEA Control Panel	2PNL2Q03-4
C86	Reserve Battery Disconnect Switch 1D57	No other effects
C85	CEDS UNIT 2 CEA Control Panel	2PNL2Q03-4
C87	125V Battery 01 DISC SW #2	No other effects
C88	Transformer 2X21	No other effects

Table 4-E-6
A302 Transient Fire Scenarios Summary

Scenario	Description	Trays and Panels Damaged by Fire
T1	ESFAS	(Component only)
T2	AFAS	(Component only)
T3	2Q02(S)	(Component only)
T4	Any 2Q03 section	(Component only)
T5	Any 2C40 panel	(Component only)
T6	Any 2Q01 section	(Component only)
T7	2Y09 OR 2Y10	(Component only)
T8	2T11	(Component only)
T9	2R01A/B	(Component only)
T10	Single battery charger	(Component only)
T11	Single inverter	(Component only)
T12	DC11 (2D11, 2D12, or 2D13)	(Component only)
T13	DC22 (2D02 or 2D14)	(Component only)
T14	2K01/2K02	(Component only)
T15	2K03	(Component only)
T16	DIESEL LOGIC (2C69)	(Component only)
T17	Single Vital AC panel	(Component only)
T18	2X08 and 2X09	(Component only)
T19	2D15, 2D16, 2D17, OR 2D01	(Component only)
T20	2Y03 and 2Y02A	(Component only)
T21	Reserve Switchgear	(Component only)
T22	Back up Bus	(Component only)
T23	Computer Inverter	(Component only)

Table 4-E-7
Cable Spreading Room Fire Analysis Results

Initiating Event	Fire Scenario	Freq	Ignition Source	Functional Impact	CDF
A302F1	C20-27, T1, T20	9.04E-6	ESFAS Cabinets: 2C67, 2C67L, 2C68, 2C68L, 2C91, 2C92, 2C93, 2C94, Transient induced ESFAS fires	HL, H9, QQ	2.73E-7
A302F2	C35-36, C38-40, C42, C44, T11, T17	8.28E-5	120VAC AC AC Inventor 21, 120VAC AC Distribution Panel 21, 120VAC AC AC Inventor 22, 120VAC AC Distribution Panel 22A, Distribution Panel 23, Distribution Panel 24, Transient induced inverter and/or Vital AC panel fires	HL, H9, QQ	5.49E-9
A302F3	C51-53	2.34E-6	Reactor Trip switchgear C, Reactor Trip switchboard D, Reactor Trip Switchgear E	AC, HL, H9, QQ	3.83E-10
A302F4	C72-74, C77	3.12E-6	Transformer/Generator Relay Panel C, Transformer/Generator Relay Panel D, Transformer/Generator Relay Panel E, Annunciator Logic Panel	OP, AC, HL, H9, QQ	2.42E-8
A302F5	C43, C75	3.98E-5	120VAC Inverter 24, Transformer/Generator Relay Panel F	OP, HL, H9, QQ, NS	9.73E-8
A302F6	C54, C70-71	2.34E-6	Electro-Hydraulic Control, Transformer/Generator Relay Panel A, Transformer/Generator Relay Panel B	OP, AC, HL, H9, QQ, F9	3.53E-8
A302F7	C1-4, T10	4.32E-5	21 Battery Charger, 22 Battery Charger, 13 Battery Charger, 14 Battery Charger, Transient damage to a single battery charger	HL, H9, QQ, XC*, XD*	1.12E-8
A302F8	C41, C45, C49, C58-59	1.58E-4	120VAC Inverter 23, 120VAC Computer Inverter, RPS Reactor Trip Switchgear Cabinet A, Shutdown CCP Panel 2Q02B SEC6, Shutdown CCP Panel 2Q02B SEC7	GF, HL, H9, QQ, S4, T1, F9	3.01E-8
A302F9	C10, C37, T16	2.01E-6	2B DG Logic Panel, 120VAC Distribution Panel 22A, Diesel Logic Cabinet (2C69)	GH, HL, H9, QQ	9.02E-11

Table 4-E-7
Cable Spreading Room Fire Analysis Results (Continued)

Initiating Event	Fire Scenario	Freq	Ignition Source	Functional Impact	CDF
A302FA	C11-14, C28-33, T19	9.31E-6	125VDC Bus 21, 125VDC Distribution Panel 2D15, 125VDC Distribution Panel 2D16, Distribution Panel 2D17, AFAS A Actuation Cabinet, AFAS B Actuation Cabinet, AFAS D Sensor Cabinet, AFAS E Sensor Cabinet, AFAS F Sensor Cabinet, AFAS G Sensor Cabinet, 2D15, 2D16, 2D17, OR 2D01	DC,HL, H9, QQ	1.19E-7
A302FB	C15-16, T13	2.45E-6	125VDC Bus 22, 125VDC Distribution Panel 2D14, DC22 (2D02 or 2D14)	DD, HL, H9, QQ	2.64E-10
A302FC	C17-19, T12	3.48E-6	125VDC Distribution Panel 2D11, 125VDC Distribution Panel 2D12, 125VDC Distribution Panel 2D13, DC11 (2D11, 2D12, or 2D13)	DA, HL, H9, QQ	6.49E-8
A302FM	C9, T7, T9, T18	3.71E-6	120VAC Bus 2Y09/2Y10 Tie Breaker, Transient damage to 2X08 and 2X09, 2Y09 OR 2Y10, or 2R01A/B	No Unit 2 MFW, HL, H9, QQ	4.84E-11
A302FN	C5-8, C34, C55, C60-69, C76, C78-86, C88, T2-6, T8, T14-15, T21-23	6.29E-4	120VAC Inverter Back up Bus, 120VAC Instrument Transformer 21, Shutdown CCP Panel 2Q02B SEC3, 120VAC Instrument Bus 21, Regulating CCP Panel 2Q02C SEC 8, Regulating CCP Panel 2Q02C SEC 17, 120VAC Instrument Transformer 22, Transformer/Generator Relay Panel G, Annunciator 21 Logic Control Panel, Annunciator 22 Logic Control Panel, 120VAC Instrument Bus 22, CEDS CEA Control Panel 1, CEDS CEA Control Panel 2, CEDS CEA Control Panel 3, CEDS CEA Control Panel 4, CEDS CEA Control Panel A, CEDS CEA Control Panel B, Reserve Battery Disconnect Switch 1D57, Transformer 2X21, Transient damage to: 2K01/2K02, 2K03, AFAS, Reserve Switchgear, Back up Bus, Computer Inverter, 2Q02(S), Any 2Q03 section, Any 2C40 panel, Any 2Q01 section, 2T11	HL, H9, QQ	1.99E-8

A302 Analysis

The A302 analysis is similar to A306. Therefore, the descriptions and tables are omitted except when significant differences exist.

A302 Fire Ignition Frequency

Both fixed and transient ignition frequencies are determined for the cable spreading room.

Fixed Ignition Frequency

The fixed ignition frequency is determined by starting with the compartment fixed ignition frequency results of Section 4.3.2 and then developing a scenario specific ignition frequency as described in Section 4.3.4.3. The sources and parameters used for A302 are:

Source	Generic Fire Frequency	Location Weighting	Source Weighting	Total Sources in Aux Bldg	Room Specific Frequency
Battery Chargers	4.0E-3	2	4	15	2.13E-3
Electrical Cabinets	3.2E-3	1	-	-	3.20E-3
Transformers (Dry)	7.9E-3	2	5	80	9.88E-4
Fire Protection Panels	2.4E-3	2	1	35	1.37E-4

THESE COMPONENT DESCRIPTIONS, HEAT RELEASE RATES, AND ANALYSES ARE SIMILAR TO THOSE IN A306.
BATTERY CHARGERS
ELECTRICAL CABINETS
TRANSFORMERS
FIRE PROTECTION PANELS

CALCULATION OF COMPONENT FIXED IGNITION FREQUENCY

Each individual component is assigned an ignition frequency apportioned by dividing the Room Specific Frequency for the component type by the total number of each ignition source of that type. The chart below summarizes that calculation and incorporates the applicable severity factor.

Equipment Count and Individual Frequency Summary (A302)

Equipment Type	Panel Count	Room Specific Frequency	Severity Factor	Individual Frequency
Battery Chargers	4	2.13E-3	0.2	1.07E-5
Electrical Cabinets or Panels	82	3.20E-3		
(Cabinet or Panel)	(75)		0.2	7.80E-7
(Inverter)	(7)		1.0	3.90E-5
Transformers	5	9.88E-4	1.0	1.98E-4
Fire Protection Panels	1	1.37E-4	0.2	2.74E-6

Transient Ignition Frequency

The transient ignition frequency is determined by starting with the compartment transient ignition frequency results of Section 4.3.2 and then developing a scenario specific ignition frequency as described in Section 4.3.4.3. The sources and parameters used for A302 are:

Source	Generic Fire Frequency	Location Weighting	Source Weighting	Total Sources in Aux Bldg	Room Specific Frequency
Cable fires - welding	5.1E-3	2	1	232	4.40E-5
Transient fires - welding	3.1E-2	2	1	232	2.67E-4
Transient - other	1.3E-3	2	7	232	7.84E-5

The transient fire modeling method parallels the method used to model A306. The calculation is presented below.

TRANSIENT COMBUSTIBLE IN RANGE OF TARGETS (U) AND RESULTING FREQUENCY

The Cable Spreading Room floor area is approximately 2,695 square feet. The floor area displaced by cabinets is calculated by summing the individual cabinet areas using the A_{sr} perimeter dimensions as length and width. The total cabinet area is 477 square feet, so:

$$A_F = 2695ft^2 - 477ft^2 = 2{,}218ft^2$$

$$\text{and for the CSR,} \quad u = A_{sr} / A_F$$

$$\text{and:} \quad P_{fs} = 1$$

$$\text{Since:} \quad F_t = F_{it} * u * P_{fs} = F_{it} * [A_{sr} / A_F] * 1$$

$$= 7.84E\text{-}4 * [A_{sr} / 2{,}218ft^2]$$

The grouping of similar components resulted 25 scenarios, tabularized below.

Table 4-E-8
A302 Transient Fire Scenarios and Frequency Summary

Scenario	Description	A_{sr}	F_{it}	Number of cabinets	Floor Area
T1	ESFAS	76	2.67E-06	1	38
T2	AFAS	36	1.27E-06	1	16
T3	2Q02(S)	99	3.50E-06	1	63
T4	Any 2Q03 section	52	1.85E-06	1	42
T5	Any 2C40 panel	61	2.14E-06	1	8
T6	Any 2Q01 section	52	1.85E-06	1	48
T7	2Y09 OR 2Y10	23	8.27E-07	2	24
T8	2T11	29	1.04E-06	1	14
T9	2R01A/B	54	1.91E-06	1	36
T10	Single battery charger	14	4.99E-07	4	19
T11	Single inverter	11	4.03E-07	4	32
T12	DC11 (2D11, 2D12, or 2D13)	32	1.14E-06	1	18
T13	DC22 (2D02 or 2D14)	25	8.91E-07	1	17
T14	2K01/2K02	31	1.10E-06	1	22
T15	2K03	10	3.50E-07	1	7
T16	DIESEL LOGIC (2C69)	13	4.46E-07	2	10
T17	Single Vital AC panel	13	4.46E-07	4	10
T18	2X08 and 2X09	5	1.91E-07	1	7
T19	2D15, 2D16, 2D17, OR 2D01	43	1.51E-06	1	31
T20	2Y03 and 2Y02A	4	1.27E-07	1	0
T21	Reserve Switchgear	47	1.65E-06	2	12
T22	Back up Bus	8	2.86E-07	1	5
T23	Computer Inverter	27	9.55E-07	1	19

Total Ignition Frequency

Equipment and trays damaged in each scenario (transient and fixed) are mapped to corresponding plant model top events. The top events are then binned for impact. The binnings are then used to group the fire scenarios into plant model scenarios. Each plant model scenario is the sum of individual fire scenarios listed below.

Plant model scenario A302F1:

C20	ESFAS Actuation Relay Panel ZA	7.80E-07
C21	ESFAS A Actuation Cabinet	7.80E-07
C22	ESFAS Actuation Relay Panel ZB	7.80E-07
C23	ESFAS B Actuation Cabinet	7.80E-07
C24	ESFAS D Sensor Cabinet	7.80E-07
C25	ESFAS E Sensor Panel	7.80E-07
C26	ESFAS F Sensor Cabinet	7.80E-07
C27	ESFAS G Sensor Cabinet	7.80E-07
T1	ESFAS	2.67E-06
T20	2Y03 and 2Y02A	1.27E-07
	Total frequency for plant model A302F1 is	9.04E-06

Plant model scenario A302F2:

C35	120V Inverter 21	3.90E-05
C36	120V Distribution Panel 21	7.80E-07
C38	120V Inverter 22	3.90E-05
C39	120V Distribution Panel 22	7.80E-07
C40	120V Distribution Panel 22A	7.80E-07
C42	120V Distribution Panel 23	7.80E-07
C44	120V Distribution Panel 24	7.80E-07
T11	Single inverter	4.03E-07
T17	Single Vital AC panel	4.46E-07
	Total frequency for plant model A302F2 is:	8.28E-05

Plant model scenario A302F3:

C51	RPS Reactor Trip Switchgear	7.80E-07
C52	RPS Reactor Trip Switchgear Cabinet C	7.80E-07
C53	RPS Reactor Trip Switchgear Cabinet D	7.80E-07
	Total frequency for plant model A302F3 is:	2.34E-06

Plant model scenario A302F4:

C72	Transformer/Generator Relay Panel C	7.80E-07
C73	Transformer/Generator Relay Panel D	7.80E-07
C74	Transformer/Generator Relay Panel E	7.80E-07
C77	Annunciator Logic Panel	7.80E-07
	Total frequency for plant model A302F4 is:	3.12E-06

Plant model scenario A302F5:

C43	120V Inverter 24	3.90E-05
C75	Transformer/Generator Relay Panel F	7.80E-07
	Total frequency for plant model A302F5 is:	3.98E-05

Plant model scenario A306F7:

C10	1B DG Logic Panel	7.36E-07
C18	125D Distribution Panel 1D15	7.36E-07
C19	125D Distribution Panel 1D16	7.36E-07
C20	125D Distribution Panel 1D17	7.36E-07
T13	DC21 (1D15, 1D16, or 1D17)	9.01E-07
T17	EDG Logic Panels (1C70, 1C69/2C70)	5.23E-07
	Total frequency for plant model A306F7 is:	4.37E-06

Plant model scenario A306F8:

C21	1 ESFAS ACT RELAY Panel ZA	7.36E-07
C22	1 ESFAS ACT Panel ZA	7.36E-07
C23	1 ESFAS ACT RELAY Panel ZB	7.36E-07
C24	1 ESFAS ACT Panel ZB	7.36E-07
C25	1 ESFAS Sensor Panel ZD	7.36E-07
C26	1 ESFAS Sensor Panel ZE	7.36E-07
C27	1 ESFAS Sensor Panel ZF	7.36E-07
T1	ESFAS	2.44E-06
	Total frequency for plant model A306F8 is:	7.59E-06

Plant model scenario A306F9:

C29	AFAS A ACT Cabinet	7.36E-07
C30	AFAS B ACT Cabinet	7.36E-07
C31	AFAS D Sensor Cabinet	7.36E-07
C32	AFAS E Sensor Cabinet	7.36E-07
C33	AFAS F Sensor Cabinet	7.36E-07
C34	AFAS G Sensor Panel	7.36E-07
T2	AFAS	1.16E-06
	Total frequency for plant model A306F9 is:	5.58E-06

Plant model scenario A306FA:

C36	120VAC Inverter 11	3.68E-05
C51	Reactor Trip Switchgear Cabinet B	7.36E-07
C52	Reactor Trip Switchgear Cabinet C	7.36E-07
T12	Single inverter	3.68E-07
T18	Single Vital AC panel	4.07E-07
T6	Any 1Q01 section	1.69E-06
	Total frequency for plant model A306FA is:	4.07E-05

Plant model scenario A306FB:

C37	120VAC Distribution Panel 11	7.36E-07
	Total frequency for plant model A306FB is:	7.36E-07

Plant model scenario A306FC:

C38	REACTOR CLNT SYS CHANNELS	7.36E-07
T21	1Y03 and 1Y04A	1.16E-07
	Total frequency for plant model A306FC is:	8.52E-07

Plant model scenario A306FD:

C39	120VAC Inverter 12	3.68E-05
T12	Single inverter	3.68E-07
T18	Single Vital AC panel (1Y02)	4.07E-07
	Total frequency for plant model A306FD is:	3.76E-05

Plant model scenario A306FE:

C40	120VAC Distribution Panel 12	7.36E-07
C41	120VAC Distribution Panel	7.36E-07
	Total frequency for plant model A306FE is:	1.47E-06

Plant model scenario A306FF:

C42	120VAC Inverter 13	3.68E-05
T12	Single inverter	3.68E-07
T18	Single Vital AC panel 1Y03	4.07E-07
	Total frequency for plant model A306FF is:	3.76E-05

Plant model scenario A306FG:

C43	120VAC Distribution Panel 13	7.36E-07
	Total frequency for plant model A306FG is:	7.36E-07

Plant model scenario A306FH:

C45	120VAC Inverter 14	3.68E-05
T12	Single inverter	3.68E-07
T18	Single Vital AC panel (1Y04)	4.07E-07
	Total frequency for plant model A306FH is:	3.76E-05

Plant model scenario A306FI:

C46	120VAC Distribution Panel 14	7.36E-07
	Total frequency for plant model A306FI is:	7.36E-07

Plant model scenario A306FJ:

C48	1R01B Instrument Power Supply	7.36E-07
C49	1R01B Instrument Power Supply	7.36E-07
T10	1R01A/B	1.74E-06
	Total frequency for plant model A306FJ is:	3.22E-06

Plant model scenario A306FK:

C50	Reactor Trip Switchgear Cabinet A	7.36E-07
C53	Reactor Trip Switchgear Cabinet D	7.36E-07
C54	Reactor Trip Switchgear Cabinet E	7.36E-07
C64	Regulating CCP Panel 1Q02C SEC 10	7.36E-07
C65	Regulating CCP Panel 1Q02C SEC 11	7.36E-07
	Total frequency for plant model A306FK is:	3.68E-06

Plant model scenario A306FL:

C55	11 MT EHC Panel	7.36E-07
T8	1T11	6.98E-07
	Total frequency for plant model A306FL is:	1.43E-06

Plant model scenario A306FM:

C71	Transformer/Generator Relay Panel	7.36E-07
C72	Transformer/Generator Relay Panel A	7.36E-07
C73	Transformer/Generator Relay Panel B	7.36E-07
C74	Transformer/Generator Relay Panel C	7.36E-07
C75	Transformer/Generator Relay Panel D	7.36E-07
C76	Transformer/Generator Relay Panel E	7.36E-07
C77	Transformer/Generator Relay Panel F	7.36E-07
T5	Any 1C40 panel	1.96E-06
	Total frequency for plant model A306FM is:	7.11E-06

Plant model scenario A306FO:

C2	23 Battery Charger	1.07E-05
C4	11 Battery Charger	1.07E-05
T11	Single battery charger	5.14E-07
	Total frequency for plant model A306FO is:	2.18E-05

Plant model scenario A306FN:

C35	120VAC Inverter Backup Bus	7.36E-07
C44	120VAC Regulating X 11	4.00E-04
C56	TURB AUX SUPERVISORY INST	7.36E-07
C60	Shutdown CCP Panel 1Q02B SEC6	7.36E-07
C61	Shutdown CCP Panel 1Q02B SEC7	7.36E-07
C62	Regulating CCP Panel 1Q02C SEC-8	7.36E-07
C63	Regulating CCP Panel 1Q02C SEC 9	7.36E-07
C69	Regulating CCP Panel 1Q02C SEC 15	7.36E-07
C70	Regulating CCP Panel 1Q02C SEC 16	7.36E-07
C79	125D Reserve Charger Disconnect 1D50	7.36E-07
C80	125D Reserve Charger Disconnect 1D55	7.36E-07
C81	125D Reserve Charger Disconnect 1D56	7.36E-07
C82	125D Reserve Charger Disconnect 1D58	7.36E-07
C83	Battery Transfer Switch	7.36E-07
C84	Annunciator Logic Cabinet	7.36E-07
C85	Annunciator 12 Logic Panel	7.36E-07
C86	CEDS CEA Control Panel	7.36E-07
C87	CEDS CEA Control Panel	7.36E-07
C88	CEDS CEA Control Panel	7.36E-07
C89	CEDS CEA Control Panel	7.36E-07
C90	CEDS CEA Control Panel	7.36E-07
C91	CEDS CEA Control Panel	7.36E-07
C92	Transformer 1X21	2.00E-04
C93	120I Bus 1Y09/1Y10 TIE BK	7.36E-07
C94	Unit XFRM Net Power Output Panel	7.36E-07
T3	1Q02(s)	3.20E-06
T4	Any 1Q03 section	1.69E-06
T9	1T14	3.34E-07
T15	1K01/1K02	8.91E-07
T16	1K03	3.20E-07
T22	Reserve charger	1.01E-06
T23	Reserve switchgear	4.07E-07
T24	Back up Bus	2.62E-07
T25	Computer Inverter	8.72E-07

Total frequency for plant model A306FN is: 6.26E-04

Fire Suppression

Although an automatic total flooding halon suppression system is installed in the Cable Spreading Room, the probability of actuation prior to target damage is not evaluated as is the likelihood of fire brigade response to manually suppress the fire prior to target damage.

Fire Suppression Induced Equipment Failure

Further, when suppression actuates, the suppression agent (halon) causes no equipment damage.

A302	**Unit 2 Cable Spreading Room**	**Location:**	27' Auxiliary Building
		Fire Area:	17
		CDF:	6.81E-7

The Unit 2 Cable Spreading Room (CSR) is the merging point for cables traveling to and from the Control Room. It also contains the following PRA related equipment:

- Unit 2 120VAC Vital AC Distribution Panels (2Y01, 2Y02, 2Y03 and 2Y04) and their associated inverters for 2Y03 and 2Y04
- 125VDC Distribution Panels (2D01 and 2D02), associated DC sub-panels, and four associated chargers
- Unit 2 120VAC Instrument Buses (2Y09 and 2Y10)
- Unit 2 Engineering Safety Features Actuation System logic and actuation cabinets
- Unit 2 Auxiliary Feedwater Actuation System logic and actuation cabinets

The cable spreading room is rectangular with approximate dimensions forty-five feet wide, sixty-five feet long, and sixteen feet high. The nominal room area is 2,695 square feet. The room is separated from Unit 1 Cable Spreading Room by a common east-west wall which is a one-hour fire rated barrier and a steel door is located in this wall at the west end.

The room contains numerous vertical floor-mounted electrical cabinets aligned in rows. The majority of the cabinets are enclosed steel cabinets with sealed conduit entries. Some of the cabinets are designed with grates or vents in the top and/or sides (e.g., inverters). Numerous cable trays traverse horizontally in the ceiling. No motor-driven equipment of significance exist in either compartment. Deterministic fire modeling will involve numerous cabinet fire scenarios with potential damage to overhead cable trays; depending on the details of the analysis, the individual cabinet fires may result in localized damage (i.e., damage only to the cabinet itself) or damage to abutted cabinets and/or cable trays due to radiant and conductive heat transfer.

Fire Analysis Results

One hundred and eleven fire scenarios are identified for Unit 2 Cable Spreading Room. Eighty-eight are the result of fixed ignition sources and twenty-three are due to transient ignition sources. These scenarios are represented by twenty-four fire initiating events. The consolidation of fire scenarios is based on an assessment of the functional impact and ignition frequency of each scenario. The frequency of each initiator is the sum of the frequencies of all the fire scenarios it represents.

The tables below list individual fire scenarios. For each scenario the initiating component is damaged.

Table 4-E-5
A302 Fixed Ignition Fire Scenarios Summary

Scenario	Fire Scenario Description	Trays and Panels Damaged by Fire
C1	21 Battery Charger	< No other effects >
C2	22 Battery Charger	< No other effects >
C3	13 Battery Charger	< No other effects >
C4	14 Battery Charger	< No other effects >
C5	120VAC Instrument Transformer 21	< No other effects >
C6	120VAC Instrument Bus 21	< No other effects >
C7	120VAC Instrument Transformer 22	< No other effects >
C8	120VAC Instrument Bus 22	< No other effects >
C9	120VAC Bus 2Y09/2Y10 Tie Breaker	< No other effects >
C10	2B DG Logic Panel	< No other effects >
C11	125VDC Bus 21	2PNL2D15, 2PNL2D16, 2PNL2D17
C12	125VDC Distribution Panel 2D15	2PNL2D01, 2PNL2D17, 2PNL2D16
C13	125VDC Distribution Panel 2D16	2PNL2D01, 2PNL2D15, 2PNL2D17
C14	125VDC Distribution Panel 2D17	2PNL2D01, 2PNL2D15, 2PNL2D16
C15	125VDC Bus 22	2PNL2D14
C16	125VDC Distribution Panel 2D14	2BUS2D02
C17	125VDC Distribution Panel 2D11	2PNL2D12, 2PNL2D13
C18	125VDC Distribution Panel 2D12	2PNL2D11, 2PNL2D13
C19	125VDC Distribution Panel 2D13	2PNL2D11, 2PNL2D12
C20	ESFAS Actuation Relay Panel ZA	2PNL2C67L
C21	ESFAS A Actuation Cabinet	2PNL2C67, 2PNL2C91
C22	ESFAS Actuation Relay Panel ZB	2PNL2C68L
C23	ESFAS B Actuation Cabinet	2PNL2C68, 2PNL2C94
C24	ESFAS D Sensor Cabinet	2PNL2C67L, 2PNL2C92
C25	ESFAS E Sensor Panel	2PNL2C91, 2PNL2C93
C26	ESFAS F Sensor Cabinet	2PNL2C92, 2PNL2C94
C27	ESFAS G Sensor Cabinet	2PNL2C68L, 2PNL2C93
C28	AFAS A Actuation Cabinet	2AH53, 2AH51, 2PNL2C100D, 2PNL2C100F

Table 4-E-5
A302 Fixed Ignition Fire Scenarios Summary (Continued)

Scenario	Fire Scenario Description	Trays and Panels Damaged by Fire
C29	AFAS B Actuation Cabinet	2AH53, 2AH51, 2PNL2C100E, 2PNL2C100G
C30	AFAS D Sensor Cabinet	2AH53, 2AH51, 2PNL2C100A, 2PNL2C100E, 2PNL2C100F
C31	AFAS E Sensor Cabinet	2AH53, 2AH51, 2PNL2C100B, 2PNL2C100D, 2PNL2C100G
C32	AFAS F Sensor Cabinet	2AH53, 2AH51, 2PNL2C100A, 2PNL2C100D
C33	AFAS G Sensor Cabinet	2AH53, 2AH51, 2PNL2C100B, 2PNL2C100E
C34	120VAC Inverter Back up Bus	< No other effects >
C35	120VAC Inverter 21	2AK11, 2AJ46, 2AJ38, 2AJ50,
C36	120VAC Distribution Panel 21	< No other effects >
C37	REACTOR COOLANT SYS CHANN	2AK12, 2AG43, 2AG37
C38	120VAC Inverter 22	2AF62, 2AG59
C39	120VAC Distribution Panel 22	< No other effects >
C40	REACTOR CLNT SYS CH TR-12	2AG87, 2AH37, 2AG79
C41	120VAC Inverter 23	2AG53, 2AH99, 2AG60
C42	120VAC Distribution Panel 23	< No other effects >
C43	120VAC Inverter 24	2AG26, 2AG44, 2AG38
C44	120VAC Distribution Panel 24	< No other effects >
C45	120VAC Computer Inverter	2AG36, 2AH19, 2AG24
C46	Regulating Transformer for Back up Bus	2AH32
C47	2R01A Instrument Power Supply	2PNL2R01B
C48	2R01B Instrument Power Supply	2PNL2R01A
C49	2 RPS ReactorTrip Switchgear Cabinet A	2AH90, 2AF78, 2AH79, 2PNL2Q01B
C50	2 RPS ReactorTrip Switchgear Cabinet B	2AH89, 2AH78, 2PNL2Q01A, 2PNL2Q01C
C51	RPS Unit 2 ReactorTrip Switchgear	2AF72, 2AH89, 2AH71, 2AH78, 2PNL2Q01B, 2PNL2Q01D
C52	2 RPS ReactorTrip Switchgear Cabinet C	2AF72, 2AH89, 2AH71, 2AH78, 2PNL2Q01C, 2PNL2Q01E
C53	2 RPS ReactorTrip Switchgear Cabinet D	2AF68, 2AH89, 2AH71, 2AH78, 2AH61, 2PNL2Q01D
C54	Electro-Hydraulic Control	2AK71, 2AJ48, 2AL05, 2AF80, 2AK09, 2AJ44, 2AJ36
C55	Shutdown CCP Panel 2Q02B SEC3	2AF84, 2AG93, 2AF82, 2AG85, 2PNL2Q02B/S04
C56	Shutdown CCP Panel 2Q02B SEC4	2AF84, 2AG93, 2AF82, 2PNL2Q02B/S03, 2PNL2Q02B/S05

Table 4-E-5
A302 Fixed Ignition Fire Scenarios Summary (Continued)

Scenario	Fire Scenario Description	Trays and Panels Damaged by Fire
C57	Shutdown CCP Panel 2Q02B SEC5	2AF84, 2AG93, 2AF82, 2PNL2Q02B/S04, 2PNL2Q02B/S06
C58	Shutdown CCP Panel 2Q02B SEC6	2AF77, 2AG93, 2AG85, 2PNL2Q02B/S05, 2PNL2Q02B/S07
C59	Shutdown CCP Panel 2Q02B SEC7	2AF77, 2AG92, 2AG84, 2PNL2Q02B/S06, 2PNL2Q02B/S08
C60	Regulating CCP Panel 2Q02C SEC 8	2AG92, 2AG84, 2PNL2Q02B/S07, 2PNL2Q02B/S09
C61	Regulating CCP Panel 2Q02C SEC 9	2AF72, 2AG92, 2AG84, 2PNL2Q02B/S08, 2PNL2Q02B/S10
C62	Regulating CCP Panel 2Q02C SEC 10	2AF72, 2AG92, 2AG84, 2PNL2Q02B/S09, 2PNL2Q02B/S11
C63	Regulating CCP Panel 2Q02C SEC 11	2AG92, 2AG84, 2PNL2Q02B/S10, 2PNL2Q02B/S12
C64	Regulating CCP Panel 2Q02C SEC 12	2AG92, 2AF67, 2PNL2Q02C/S11, 2PNL2Q02C/S13
C65	Regulating CCP Panel 2Q02C SEC 13	2AF67, 2AG84, 2AJ55, 2AG92, 2PNL2Q02C/S12, 2PNL2Q02C/S14
C66	Regulating CCP Panel 2Q02C SEC 14	2AJ55, 2AG83, 2AG91, 2PNL2Q02C/S13, 2PNL2Q02C/S15
C67	Regulating CCP Panel 2Q02C SEC 15	2AG83, 2AG91, 2PNL2Q02C/S14, 2PNL2Q02C/S16
C68	Regulating CCP Panel 2Q02C SEC 16	2AG83, 2AG91, 2PNL2Q02C/S15, 2PNL2Q02C/S17
C69	Regulating CCP Panel 2Q02C SEC 17	2AF61, 2AG91,2PNL2Q02C/S16
C70	Transformer/Generator Relay Panel A	2AJ85, 2AJ31, 2AJ39, 2AG95, 2AG78, 2PNL2C40B
C71	Transformer/Generator Relay Panel B	2AJ31, 2AG95, 2AJ85, 2AJ39, 2PNL2C40A, 2PNL2C40C
C72	Transformer/Generator Relay Panel C	2AG95, 2AK04, 2PNL2C40B, 2PNL2C40D
C73	Transformer/Generator Relay Panel D	2AG95, 2AJ21, 2AJ29, 2AJ27, 2PNL2C40C, 2PNL2C40E
C74	Transformer/Generator Relay Panel E	2AG95, 2AJ21, 2AJ29, 2AJ27, 2PNL2C40D, 2PNL2C40F
C75	Transformer/Generator Relay Panel F	2AH94, 2AJ07, 2AJ14, 2AH82, 2AG95, 2AH93, 2PNL2C40E, 2PNL2C40G
C76	Transformer/Generator Relay Panel G	2AH93, 2AH94, 2AJ07, 2AH82, 2AJ14, 2PNL2C40F
C77	Annunciator Logic Panel	2AH88, 2AF62, 2AH77, 2AH70, 2AJ71
C78	Annunciator 21 Logic Control Panel	2PNL2K02
C79	Annunciator 22 Logic Control Panel	2PNL2K01
C80	CEDS UNIT 2 CEA Control Panel	2PNL2Q03-2, 2PNL2Q03A
C81	CEDS UNIT 2 CEA Control Panel	2PNL2Q03-1, 2PNL2Q03-3
C82	CEDS UNIT 2 CEA Control Panel	2PNL2Q03-2, 2PNL2Q03-4
C83	CEDS UNIT 2 CEA Control Panel	2PNL2Q03-3, 2PNL2Q03B

Table 4-E-5
A302 Fixed Ignition Fire Scenarios Summary (Continued)

Scenario	Fire Scenario Description	Trays and Panels Damaged by Fire
C84	CEDS UNIT 2 CEA Control Panel	2PNL2Q03-1
C85	CEDS UNIT 2 CEA Control Panel	2PNL2Q03-4
C86	Reserve Battery Disconnect Switch 1D57	No other effects
C85	CEDS UNIT 2 CEA Control Panel	2PNL2Q03-4
C87	125V Battery 01 DISC SW #2	No other effects
C88	Transformer 2X21	No other effects

Table 4-E-6
A302 Transient Fire Scenarios Summary

Scenario	Description	Trays and Panels Damaged by Fire
T1	ESFAS	(Component only)
T2	AFAS	(Component only)
T3	2Q02(S)	(Component only)
T4	Any 2Q03 section	(Component only)
T5	Any 2C40 panel	(Component only)
T6	Any 2Q01 section	(Component only)
T7	2Y09 OR 2Y10	(Component only)
T8	2T11	(Component only)
T9	2R01A/B	(Component only)
T10	Single battery charger	(Component only)
T11	Single inverter	(Component only)
T12	DC11 (2D11, 2D12, or 2D13)	(Component only)
T13	DC22 (2D02 or 2D14)	(Component only)
T14	2K01/2K02	(Component only)
T15	2K03	(Component only)
T16	DIESEL LOGIC (2C69)	(Component only)
T17	Single Vital AC panel	(Component only)
T18	2X08 and 2X09	(Component only)
T19	2D15, 2D16, 2D17, OR 2D01	(Component only)
T20	2Y03 and 2Y02A	(Component only)
T21	Reserve Switchgear	(Component only)
T22	Back up Bus	(Component only)
T23	Computer Inverter	(Component only)

Table 4-E-7
Cable Spreading Room Fire Analysis Results

Initiating Event	Fire Scenario	Freq	Ignition Source	Functional Impact	CDF
A302F1	C20-27, T1, T20	9.04E-6	ESFAS Cabinets: 2C67, 2C67L, 2C68, 2C68L, 2C91, 2C92, 2C93, 2C94, Transient induced ESFAS fires	HL, H9, QQ	2.73E-7
A302F2	C35-36, C38-40, C42, C44, T11, T17	8.28E-5	120VAC AC AC Inventor 21, 120VAC AC Distribution Panel 21, 120VAC AC AC Inventor 22, 120VAC AC Distribution Panel 22A, Distribution Panel 23, Distribution Panel 24, Transient induced inverter and/or Vital AC panel fires	HL, H9, QQ	5.49E-9
A302F3	C51-53	2.34E-6	Reactor Trip switchgear C, Reactor Trip switchboard D, Reactor Trip Switchgear E	AC, HL, H9, QQ	3.83E-10
A302F4	C72-74, C77	3.12E-6	Transformer/Generator Relay Panel C, Transformer/Generator Relay Panel D, Transformer/Generator Relay Panel E, Annunciator Logic Panel	OP, AC, HL, H9, QQ	2.42E-8
A302F5	C43, C75	3.98E-5	120VAC Inverter 24, Transformer/Generator Relay Panel F	OP, HL, H9, QQ, NS	9.73E-8
A302F6	C54, C70-71	2.34E-6	Electro-Hydraulic Control, Transformer/Generator Relay Panel A, Transformer/Generator Relay Panel B	OP, AC, HL, H9, QQ, F9	3.53E-8
A302F7	C1-4, T10	4.32E-5	21 Battery Charger, 22 Battery Charger, 13 Battery Charger, 14 Battery Charger, Transient damage to a single battery charger	HL, H9, QQ, XC*, XD*	1.12E-8
A302F8	C41, C45, C49, C58-59	1.58E-4	120VAC Inverter 23, 120VAC Computer Inverter, RPS Reactor Trip Switchgear Cabinet A, Shutdown CCP Panel 2Q02B SEC6, Shutdown CCP Panel 2Q02B SEC7	GF, HL, H9, QQ, S4, T1, F9	3.01E-8
A302F9	C10, C37, T16	2.01E-6	2B DG Logic Panel, 120VAC Distribution Panel 22A, Diesel Logic Cabinet (2C69)	GH, HL, H9, QQ	9.02E-11

Table 4-E-7
Cable Spreading Room Fire Analysis Results (Continued)

Initiating Event	Fire Scenario	Freq	Ignition Source	Functional Impact	CDF
A302FA	C11-14, C28-33, T19	9.31E-6	125VDC Bus 21, 125VDC Distribution Panel 2D15, 125VDC Distribution Panel 2D16, Distribution Panel 2D17, AFAS A Actuation Cabinet, AFAS B Actuation Cabinet, AFAS D Sensor Cabinet, AFAS E Sensor Cabinet, AFAS F Sensor Cabinet, AFAS G Sensor Cabinet, 2D15, 2D16, 2D17, OR 2D01	DC,HL, H9, QQ	1.19E-7
A302FB	C15-16, T13	2.45E-6	125VDC Bus 22, 125VDC Distribution Panel 2D14, DC22 (2D02 or 2D14)	DD, HL, H9, QQ	2.64E-10
A302FC	C17-19, T12	3.48E-6	125VDC Distribution Panel 2D11, 125VDC Distribution Panel 2D12, 125VDC Distribution Panel 2D13, DC11 (2D11, 2D12, or 2D13)	DA, HL, H9, QQ	6.49E-8
A302FM	C9, T7, T9, T18	3.71E-6	120VAC Bus 2Y09/2Y10 Tie Breaker, Transient damage to 2X08 and 2X09, 2Y09 OR 2Y10, or 2R01A/B	No Unit 2 MFW, HL, H9, QQ	4.84E-11
A302FN	C5-8, C34, C55, C60-69, C76, C78-86, C88, T2-6, T8, T14-15, T21-23	6.29E-4	120VAC Inverter Back up Bus, 120VAC Instrument Transformer 21, Shutdown CCP Panel 2Q02B SEC3, 120VAC Instrument Bus 21, Regulating CCP Panel 2Q02C SEC 8, Regulating CCP Panel 2Q02C SEC 17, 120VAC Instrument Transformer 22, Transformer/Generator Relay Panel G, Annunciator 21 Logic Control Panel, Annunciator 22 Logic Control Panel, 120VAC Instrument Bus 22, CEDS CEA Control Panel 1, CEDS CEA Control Panel 2, CEDS CEA Control Panel 3, CEDS CEA Control Panel 4, CEDS CEA Control Panel A, CEDS CEA Control Panel B, Reserve Battery Disconnect Switch 1D57, Transformer 2X21, Transient damage to: 2K01/2K02, 2K03, AFAS, Reserve Switchgear, Back up Bus, Computer Inverter, 2Q02(S), Any 2Q03 section, Any 2C40 panel, Any 2Q01 section, 2T11	HL, H9, QQ	1.99E-8

A302 Analysis

The A302 analysis is similar to A306. Therefore, the descriptions and tables are omitted except when significant differences exist.

A302 Fire Ignition Frequency

Both fixed and transient ignition frequencies are determined for the cable spreading room.

Fixed Ignition Frequency

The fixed ignition frequency is determined by starting with the compartment fixed ignition frequency results of Section 4.3.2 and then developing a scenario specific ignition frequency as described in Section 4.3.4.3. The sources and parameters used for A302 are:

Source	Generic Fire Frequency	Location Weighting	Source Weighting	Total Sources in Aux Bldg	Room Specific Frequency
Battery Chargers	4.0E-3	2	4	15	2.13E-3
Electrical Cabinets	3.2E-3	1	-	-	3.20E-3
Transformers (Dry)	7.9E-3	2	5	80	9.88E-4
Fire Protection Panels	2.4E-3	2	1	35	1.37E-4

THESE COMPONENT DESCRIPTIONS, HEAT RELEASE RATES, AND ANALYSES ARE SIMILAR TO THOSE IN A306.

BATTERY CHARGERS
ELECTRICAL CABINETS
TRANSFORMERS
FIRE PROTECTION PANELS

CALCULATION OF COMPONENT FIXED IGNITION FREQUENCY

Each individual component is assigned an ignition frequency apportioned by dividing the Room Specific Frequency for the component type by the total number of each ignition source of that type. The chart below summarizes that calculation and incorporates the applicable severity factor.

Equipment Count and Individual Frequency Summary (A302)

Equipment Type	Panel Count	Room Specific Frequency	Severity Factor	Individual Frequency
Battery Chargers	4	2.13E-3	0.2	1.07E-5
Electrical Cabinets or Panels	82	3.20E-3		
(Cabinet or Panel)	(75)		0.2	7.80E-7
(Inverter)	(7)		1.0	3.90E-5
Transformers	5	9.88E-4	1.0	1.98E-4
Fire Protection Panels	1	1.37E-4	0.2	2.74E-6

Transient Ignition Frequency

The transient ignition frequency is determined by starting with the compartment transient ignition frequency results of Section 4.3.2 and then developing a scenario specific ignition frequency as described in Section 4.3.4.3. The sources and parameters used for A302 are:

Source	Generic Fire Frequency	Location Weighting	Source Weighting	Total Sources in Aux Bldg	Room Specific Frequency
Cable fires - welding	5.1E-3	2	1	232	4.40E-5
Transient fires - welding	3.1E-2	2	1	232	2.67E-4
Transient - other	1.3E-3	2	7	232	7.84E-5

The transient fire modeling method parallels the method used to model A306. The calculation is presented below.

TRANSIENT COMBUSTIBLE IN RANGE OF TARGETS (U) AND RESULTING FREQUENCY

The Cable Spreading Room floor area is approximately 2,695 square feet. The floor area displaced by cabinets is calculated by summing the individual cabinet areas using the A_{sr} perimeter dimensions as length and width. The total cabinet area is 477 square feet, so:

$$A_F = 2695ft^2 - 477ft^2 = 2{,}218ft^2$$

and for the CSR, $u = A_{sr} / A_F$

and: $P_{fs} = 1$

$$\text{Since: } F_t = F_{it} * u * P_{fs} = F_{it} * [A_{sr} / A_F] * 1$$
$$= 7.84E\text{-}4 * [A_{sr} / 2{,}218ft^2]$$

The grouping of similar components resulted 25 scenarios, tabularized below.

Table 4-E-8
A302 Transient Fire Scenarios and Frequency Summary

Scenario	Description	A_{sr}	F_{lt}	Number of cabinets	Floor Area
T1	ESFAS	76	2.67E-06	1	38
T2	AFAS	36	1.27E-06	1	16
T3	2Q02(S)	99	3.50E-06	1	63
T4	Any 2Q03 section	52	1.85E-06	1	42
T5	Any 2C40 panel	61	2.14E-06	1	8
T6	Any 2Q01 section	52	1.85E-06	1	48
T7	2Y09 OR 2Y10	23	8.27E-07	2	24
T8	2T11	29	1.04E-06	1	14
T9	2R01A/B	54	1.91E-06	1	36
T10	Single battery charger	14	4.99E-07	4	19
T11	Single inverter	11	4.03E-07	4	32
T12	DC11 (2D11, 2D12, or 2D13)	32	1.14E-06	1	18
T13	DC22 (2D02 or 2D14)	25	8.91E-07	1	17
T14	2K01/2K02	31	1.10E-06	1	22
T15	2K03	10	3.50E-07	1	7
T16	DIESEL LOGIC (2C69)	13	4.46E-07	2	10
T17	Single Vital AC panel	13	4.46E-07	4	10
T18	2X08 and 2X09	5	1.91E-07	1	7
T19	2D15, 2D16, 2D17, OR 2D01	43	1.51E-06	1	31
T20	2Y03 and 2Y02A	4	1.27E-07	1	0
T21	Reserve Switchgear	47	1.65E-06	2	12
T22	Back up Bus	8	2.86E-07	1	5
T23	Computer Inverter	27	9.55E-07	1	19

Total Ignition Frequency

Equipment and trays damaged in each scenario (transient and fixed) are mapped to corresponding plant model top events. The top events are then binned for impact. The binnings are then used to group the fire scenarios into plant model scenarios. Each plant model scenario is the sum of individual fire scenarios listed below.

Plant model scenario A302F1:

C20	ESFAS Actuation Relay Panel ZA	7.80E-07
C21	ESFAS A Actuation Cabinet	7.80E-07
C22	ESFAS Actuation Relay Panel ZB	7.80E-07
C23	ESFAS B Actuation Cabinet	7.80E-07
C24	ESFAS D Sensor Cabinet	7.80E-07
C25	ESFAS E Sensor Panel	7.80E-07
C26	ESFAS F Sensor Cabinet	7.80E-07
C27	ESFAS G Sensor Cabinet	7.80E-07
T1	ESFAS	2.67E-06
T20	2Y03 and 2Y02A	1.27E-07
	Total frequency for plant model A302F1 is	9.04E-06

Plant model scenario A302F2:

C35	120V Inverter 21	3.90E-05
C36	120V Distribution Panel 21	7.80E-07
C38	120V Inverter 22	3.90E-05
C39	120V Distribution Panel 22	7.80E-07
C40	120V Distribution Panel 22A	7.80E-07
C42	120V Distribution Panel 23	7.80E-07
C44	120V Distribution Panel 24	7.80E-07
T11	Single inverter	4.03E-07
T17	Single Vital AC panel	4.46E-07
	Total frequency for plant model A302F2 is:	8.28E-05

Plant model scenario A302F3:

C51	RPS Reactor Trip Switchgear	7.80E-07
C52	RPS Reactor Trip Switchgear Cabinet C	7.80E-07
C53	RPS Reactor Trip Switchgear Cabinet D	7.80E-07
	Total frequency for plant model A302F3 is:	2.34E-06

Plant model scenario A302F4:

C72	Transformer/Generator Relay Panel C	7.80E-07
C73	Transformer/Generator Relay Panel D	7.80E-07
C74	Transformer/Generator Relay Panel E	7.80E-07
C77	Annunciator Logic Panel	7.80E-07
	Total frequency for plant model A302F4 is:	3.12E-06

Plant model scenario A302F5:

C43	120V Inverter 24	3.90E-05
C75	Transformer/Generator Relay Panel F	7.80E-07
	Total frequency for plant model A302F5 is:	3.98E-05

Plant model scenario A302F6:

C54	Electro-Hydraulic Control	7.80E-07
C70	Transformer/Generator Relay Panel A	7.80E-07
C71	Transformer/Generator Relay Panel B	7.80E-07
	Total frequency for plant model A302F6 is:	2.34E-06

Plant model scenario A302F7:

C1	21 Battery Charger	1.07E-05
C2	22 Battery Charger	1.07E-05
C3	13 Battery Charger	1.07E-05
C4	14 Battery Charger	1.07E-05
T10	Single battery charger	4.99E-07
	Total frequency for plant model A302F7 is:	4.32E-05

Plant model scenario A302F8:

C41	120V Inverter 23	3.90E-05
C45	120V Computer Inverter	1.17E-04
C49	RPS Reactor Trip Switchgear Cabinet A	7.80E-07
C58	Shutdown CCP Panel 2Q02B SEC6	7.80E-07
C59	Shutdown CCP Panel 2Q02B SEC7	7.80E-07
	Total frequency for plant model A302F8 is:	1.58E-04

Plant model scenario A302F9:

C10	2B DG Logic Panel	7.80E-07
C37	120V Distribution Panel 21A	7.80E-07
T16	DIESEL LOGIC (2C69)	4.46E-07
	Total frequency for plant model A302F9 is:	2.01E-06

Plant model scenario A302FA:

C11	125VDC Bus 21	7.80E-07
C12	125D Distribution Panel 2D15	7.80E-07
C13	125D Distribution Panel 2D16	7.80E-07
C14	125D Distribution Panel 2D17	7.80E-07
C28	AFAS A Actuation Cabinet	7.80E-07
C29	AFAS B Actuation Cabinet	7.80E-07
C30	AFAS D Sensor Cabinet	7.80E-07
C31	AFAS E Sensor Cabinet	7.80E-07
C32	AFAS F Sensor Cabinet	7.80E-07
C33	AFAS G Sensor Cabinet	7.80E-07
T19	2D15, 2D16, 2D17, OR 2D01	1.51E-06
	Total frequency for plant model A302FA is:	9.31E-06

Plant model scenario A302FB:

C15	125VDC Bus 22	7.80E-07
C16	125D Distribution Panel 2D14	7.80E-07
T13	DC22 (2D02 or 2D14)	8.91E-07
	Total frequency for plant model A302FB is:	2.45E-06

Plant model scenario A302FC:

C17	125D Distribution Panel 2D11	7.80E-07
C18	125D Distribution Panel 2D12	7.80E-07
C19	125D Distribution Panel 2D13	7.80E-07
T12	DC11 (2D11, 2D12, or 2D13)	1.14E-06
	Total frequency for plant model A302FC is:	3.48E-06

Plant model scenario A302FM:

C9	120V Bus 2Y09/2Y10 Tie Breaker	7.80E-07
T18	2X08 and 2X09	1.91E-07
T7	2Y09 OR 2Y10	8.27E-07
T9	2R01A/B	1.91E-06
	Total frequency for plant model A302FM is:	3.71E-06

Plant model scenario A302FN:

C34	120V Inverter Back up Bus	7.80E-07
C5	120V Instrument Transformer 21	1.98E-04
C55	Shutdown CCP Panel 2Q02B SEC3	7.80E-07
C6	120V Instrument Bus 21	7.80E-07
C60	Regulating CCP Panel 2Q02C SEC 8	7.80E-07
C61	Regulating CCP Panel 2Q02C SEC 9	7.80E-07
C62	Regulating CCP Panel 2Q02C SEC 10	7.80E-07
C63	Regulating CCP Panel 2Q02C SEC 11	7.80E-07
C64	Regulating CCP Panel 2Q02C SEC 12	7.80E-07
C65	Regulating CCP Panel 2Q02C SEC 13	7.80E-07
C66	Regulating CCP Panel 2Q02C SEC 14	7.80E-07
C67	Regulating CCP Panel 2Q02C SEC 15	7.80E-07
C68	Regulating CCP Panel 2Q02C SEC 16	7.80E-07
C69	Regulating CCP Panel 2Q02C SEC 17	7.80E-07
C7	120V Instrument Transformer 22	1.98E-04
C76	Transformer/Generator Relay Panel G	7.80E-07
C78	Annunciator 21 Logic Control Panel	7.80E-07
C79	Annunciator 22 Logic Control Panel	7.80E-07
C8	120V Instrument Bus 22	7.80E-07
C80	CEDS CEA Control Panel	7.80E-07
C81	CEDS CEA Control Panel	7.80E-07
C82	CEDS CEA Control Panel	7.80E-07
C83	CEDS CEA Control Panel	7.80E-07
C84	CEDS CEA Control Panel	7.80E-07
C85	CEDS CEA Control Panel	7.80E-07
C86	Reserve Battery Disconnect Switch 1D57	7.80E-07
C88	Transformer 2X21	1.98E-04
T14	2K01/2K02	1.10E-06
T15	2K03	3.50E-07
T2	AFAS	1.27E-06
T21	Reserve Switchgear	1.65E-06
T22	Back up Bus	2.86E-07
T23	Computer Inverter	9.55E-07
T3	2Q02(S)	3.50E-06
T4	Any 2Q03 section	1.85E-06
T5	Any 2C40 panel	2.14E-06
T6	Any 2Q01 section	1.85E-06
T8	2T11	1.04E-06
	Total frequency for plant model A302FN is:	6.29E-04

Fire Suppression

Although an automatic total flooding halon suppression system is installed in the Cable Spreading Room, the probability of actuation prior to target damage is not evaluated as is the likelihood of fire brigade response to manually suppress the fire prior to target damage.

Fire Suppression Induced Equipment Failure

When suppression actuates, the suppression agent (halon) causes no equipment damage.

A311	**Unit 2 27' Switchgear Room**	**Location:** **Fire Area:** **CDF:**	27' Auxiliary Building 18 2.22E-7
A317	**Unit 1 27' Switchgear Room**	**Location:** **Fire Area:** **CDF:**	27' Auxiliary Building 19 4.28E-6
A407	**Unit 2 45' Switchgear Room**	**Location:** **Fire Area:** **CDF:**	45' Auxiliary Building 25 1.19E-7
A430	**Unit 1 45' Switchgear Room**	**Location:** **Fire Area:** **CDF:**	45' Auxiliary Building 34 1.54E-6

Four switchgear rooms were analyzed for Unit 1. Two of the rooms (A317 and A430) primarily support Unit 1 equipment and the other two rooms (A311 and A407) primarily support Unit 2 equipment.

A311 Contains the 4KV Buses (21 and 22), 480 volt load centers and their associated transformers (21A, 21B, 22A and 22B), and cabling feeding the related buses and equipment.

A317 Contains the 4KV Buses (11 and 12), 480 volt load centers and their associated transformers (11A, 11B, 12A and 12B), and cabling feeding the related buses and equipment.

A407 Contains the 4KV Buses (23 and 24), 480 volt load centers and their associated transformers (23A, 23B, 24A and 24B), and cabling feeding the related buses and equipment.

A430 Contains the 4KV Buses (13 and 14), 480 volt load centers and their associated transformers (13A, 13B, 14A and 14B), and cabling feeding the related buses and equipment.

In the event of a fire, the inlet and outlet ventilation isolation dampers for the affected room will close prior to actuation of halon suppression shutting of the HVAC system to the affected SWGR Room. The unaffected SWGR Room will continue to have HVAC cooling.

<u>Fire Analysis Results</u>

The results of each switchgear room are addressed sequentially in the following sections.

Unit 2 27' Switchgear Room (A311)

Twenty-eight fire scenarios were identified for A311. Twenty-three are the result of fixed ignition sources and five are due to transient ignition sources. Five scenarios are screened due to low functional impact. The remaining twenty-five scenarios identified in Table 4-F-1 are represented by three fire initiating events shown in Table 4-F-2. The consolidation of fire scenarios is based on an assessment of the functional impact and ignition frequency of each scenario. The frequency of each initiator is the sum of the frequencies of all the fire scenarios it represents.

Table 4-F-1
Unit 2 27' Switchgear Room (A311)
Fire Scenario Summary

Scenario	Fire Scenario Description	Equipment Damaged
01	4.16kV Switchgear Bus 21 Cubicle Fire impacting bus or load (14 cubicles)	Loss of Bus 21 or Individual Cubicles or Load Center; Loss of Ventilation; Dampers Close (recoverable). Note that this Bus 21 failure does not impact cable trays.
02	4.16kV Switchgear Bus 21 Bus Fire with cable tray impact	Loss of Bus 21; Loss of Ventilation; Dampers Close (recoverable); 2AD74, 2AD73, 2AB13, 2AB14, 2AB15, 2AF21, 2AF22, 2AD93, 2AB02, 2AB03
04	4.16kV Switchgear Bus 22 Bus Fire with cable tray impact	Loss of Bus 22; Loss of Ventilation; Dampers Close (recoverable); 2AM51, 2AD72, 2AD71, 2AF23, 2AF24, 2AB04, 2AB05, 2AB06, 2AB17, 2AB16, 2AB15
07	Minor Transformer 21A Liquid Fire	Loss of Bus 21A (Transformer only); Loss of Ventilation; Dampers Close (not recoverable); 2A0654, 2A0655, 2A0656
08	Severe Transformer 21A Liquid Fire	Loss of Bus 21A (Transformer and Switchgear cabinet); Loss of Ventilation; Dampers Close (not recoverable); 2A0654, 2A0655, 2A0656, 2AF16, 2AF17, 2AB09, 2AB10
09	Minor Transformer 21B Liquid Fire	Loss of Bus 21B (Transformer only); Loss of Ventilation; Dampers Close (not recoverable); 2A0657, 2A0658, 2A0659
10	Severe Transformer 21B Liquid Fire	Loss of Bus 21B (Transformer and Switchgear cabinet); Loss of Ventilation; Dampers Close (not recoverable); 2A0657, 2A0658, 2A0659, 2AF17, 2AB09
11	Minor Transformer 22A Liquid Fire	Loss of Bus 22A (Transformer only); Loss of Ventilation; Dampers Close (not recoverable); 2A0660, 2A0661, 2A0662
12	Severe Transformer 22A Liquid Fire	Loss of Bus 22A (Transformer and Switchgear cabinet); Loss of Ventilation; Dampers Close (not recoverable); 2A0660, 2A0661, 2A0662, 2AF18, 2AF19, 2AB08, 2AB07
13	Minor Transformer 22B Liquid Fire	Loss of Bus 22B (Transformer only); Loss of Ventilation; Dampers Close (not recoverable); 2A0654, 2A0655, 2A0656
14	Severe Transformer 22B Liquid Fire	Loss of Bus 22B (Transformer and Switchgear cabinet); Loss of Ventilation; Dampers Close (not recoverable); 2A0654, 2A0655, 2A0656, 2AF19, 2AB58
15	480V SWGR 21A Bus Fire	Loss of Bus 21A; Loss of Ventilation; Dampers Close (recoverable); 2A0461, 2A2125, 2A0460, 2A0459, 2A0458, 2A2127, 2A0462, 2A1449, 2A0323, 2A1448, 2A0318, 2AF17, 2AF16, 2AF15, 2AB09, 2AB10, 2AB11
16	480V SWGR 21A Cubicle Fire	Loss of Individual Cubicle or Load Center; Loss of Ventilation; Dampers Close (recoverable)

Table 4-F-1
Unit 2 27' Switchgear Room (A311)
Fire Scenario Summary (Continued)

Scenario	Fire Scenario Description	Equipment Damaged
17	480V SWGR 21B Cubicle Fire	Loss of Individual Cubicle or Load Center; Loss of Ventilation; Dampers Close (recoverable)
18	480V SWGR 21B Bus Fire	Loss of Bus 21B; Loss of Ventilation; Dampers Close (recoverable); 2A0243, 2A1454, 2A5296, 2A0452, 2A0453, 2A0454, 2A5074, 2A0591, 2A0242, 2A5076, 2A0457, 2A0972, 2AF17, 2AF18, 2AB09, 2AB08
19	480V SWGR 22A Bus Fire	Loss of Bus 22A; Loss of Ventilation; Dampers Close (recoverable); 2A0451, 2A0153, 2A0450, 2A0449, 2A0152, 2A0447, 2A051, 2AF19, 2AB07, 2AB58
20	480V SWGR 22B Bus Fire	Loss of Bus 22B; Loss of Ventilation; Dampers Close (recoverable); 2A0446, 2A0150, 2A1478, 2A5009, 2A0441, 2A1400, 2A0442, 2A0154, 2A0443, 2A1304, 2AF19, 2AB07, 2AB58
21	Disconnect (6)	Loss of Disconnect Panel (no other impact); Loss of Ventilation; Dampers Close (recoverable)
24	4KV Bus 21 Transient Fire	Loss of Bus 21
26	480VAC Bus 21A Transient Fire	Loss of Bus 21A
27	480VAC Bus 21B Transient Fire	Loss of Bus 21B
28	480VAC Buses 22A and 22B Transient Fire	Loss of Bus 22A and 22B (Frequency represents both Buses; individual Bus would have 1/2 frequency)

Table 4-F-2
Unit 2 27' Switchgear Room (A311)
Fire Analysis Results

Initiating Event	Fire Scenario	Frequency	Ignition Source	Functional Impact	CDF
A311F1	2,4	4.62E-5	4KV Bus 21 or 22 Bus fire (does not include breaker failures that do not impact the bus)	Y2, QD, AC, Y3, Y4	6.01E-8
A311F2	8,10,12,15, 19,20	6.97E-4	Severe Transformer 21A, 21B or 22A, fire or 480VAC SWGR 21A, 22A or 22B Bus fires	QD, AC, Y3, Y4	1.49E-7
A311F3	1,7,9,11,13, 14,16,17,18, 24,26,27,	9.18E-4	4KV Bus 21 Cubicle (load losses only) or Minor Transformer 21A, 21B, 22A or 22B fire or Severe Transformer 22B fire, or 480VAC SWGR 21A, 21B, cubicles fires or 21B bus fire or Transient fires	AC, Y3	1.32E-8

Unit 1 27' Switchgear Room (A317)

Twenty-eight fire scenarios were identified for A317. Twenty-three are the result of fixed ignition sources and five are due to transient ignition sources. These scenarios identified in Table 4-F-3 are represented by twelve fire initiating events shown in Table 4-F-4b. The consolidation of fire scenarios is based on an assessment of the functional impact and ignition frequency of each scenario. The frequency of each initiator is the sum of the frequencies of all the fire scenarios it represents.

Table 4-F-3
Unit 1 27' switchgear Room (A317)
Fire Scenarios

Scenario	Fire Scenario Description	Equipment Damaged
01	4.16kV Switchgear Bus 11 Cubicle Fire (14 cubicles) impacting bus or load	Loss of Bus 11 or Individual Cubicle or Load Center; Loss of Ventilation; Dampers Close (recoverable). Note that this Bus 11 loss does not impact cable trays.
02	4.16kV Switchgear Bus 11 Bus Fire with cable tray impact	Loss of Bus 11; Loss of Ventilation; Dampers Close (recoverable); 1AD67, 1AD68, 1AK27, 1AK48, 1AK49, 1AK50, 1AK95, 1AB15, 1AB16, 1AB17, 1AB18, 1AB27, 1AB28, 1AB29
03	4.16kV NSR Switchgear Bus 12 Cubicle Fire impacting bus or load OR (480V SWGR 12A OR 12B Cubicle Fire)	Loss of Bus 12 or Individual Cubicle or Load Center; Loss of Ventilation; Dampers Close (recoverable). Note that this Bus 12 loss does not impact cable trays.
04	4.16kV Switchgear Bus 12 Bus Fire with cable tray impact	Loss of Bus 12; Loss of Ventilation; Dampers Close (recoverable); 1AD67, 1AD68, 1AK51, 1AK52, 1AK53, 1AB91, 1AB13, 1AB14, 1AB90, 1AB25, 1AB26
05	RCP 11 Breaker Cubicle (2)	Loss of RCP 11 (no other impact); Loss of Ventilation; Dampers Close (recoverable); 1A0161, 1A0162
06	RCP 12 Breaker Cubicle (2)	Loss of RCP 12 (no other impact); Loss of Ventilation; Dampers Close (recoverable)
07	Transformer 11A (1XU-440-11A) - DRY	Loss of Bus 11A (Transformer only); Loss of Ventilation; Dampers Close (recoverable); 1A804, 1AK37, 1AB19
08	Severe Transformer 11A Liquid Fire	Not a plausible fire scenario for a dry transformer.
09	Minor Transformer 11B Liquid Fire	Loss of Bus 11B (Transformer only); Loss of Ventilation; Dampers Close (not recoverable); 1A0807, 1A0808, 1A0809
10	Severe Transformer 11B Liquid Fire	Loss of Bus 11B (Transformer and Switchgear cabinet); Loss of Ventilation; Dampers Close (not recoverable); 1A0807, 1A0808, 1A0809, 1AK38, 1AB20, 1AB32, 1AB44
11	Minor Transformer 12A Liquid Fire	Loss of Bus 12A (Transformer only); Loss of Ventilation; Dampers Close (not recoverable); 1A0807, 1A0808, 1A0809, 1AK39, 1AB21, 1AB33, 1AB45

Table 4-F-3
Unit 1 27' Switchgear Room (A317)
Fire Scenarios (Continued)

Scenario	Fire Scenario Description	Equipment Damaged
12	Severe Transformer 12A Liquid Fire	Loss of Bus 12A (Transformer and Switchgear cabinet); Loss of Ventilation; Dampers Close (not recoverable); 1A0807, 1A0808, 1A0809, 1AK39, 1AK40, 1AB21, 1AB22, 1AB33, 1AB45
13	Minor Transformer 12B Liquid Fire	Loss of Bus 12B (Transformer only); Loss of Ventilation; Dampers Close (not recoverable); 1A0801, 1A0802, 1A0803, 1AK41, 1AB23, 1AB35, 1AB47
14	Severe Transformer 12B Liquid Fire	Loss of Bus 12B (Transformer and Switchgear cabinet); Loss of Ventilation; Dampers Close (not recoverable); 1A0801, 1A0802, 1A0803, 1AK41, 1AB23, 1AB35, 1AB47, 1AK40, 1AB21, 1AB22
15	480V SWGR 11A Cubicle Fire	Loss of Individual Cubicle or Load Center; Loss of Ventilation; Dampers Close (recoverable)
16	480V SWGR 11A Bus Fire	Loss of Bus 11A; Loss of Ventilation; Dampers Close (recoverable); 1A0157, 1A0790, 1A0322, 1A0324, 1A0325, 1A0313, 1A0326, 1A0327, 1A0328, 1A0662, 1AK37, 1AK27, 1AB18, 1AB19
17	480V SWGR 11B Cubicle Fire	Loss of Individual Cubicle or Load Center; Loss of Ventilation; Dampers Close (recoverable)
18	480V SWGR 11B Bus Fire	Loss of Bus 11B; Loss of Ventilation; Dampers Close (recoverable); 1A0333, 1A5053, 1A0335, 1A5057, 1A0336, 1A0314, 1A5055, 1A0338, 1AB76, 1A0339, 1A0663, 1A2782, 1A0315, 1A2064, 1A2278, 1AK38, 1AB20
19	480V SWGR 12A Bus Fire	Loss of Bus 12A; Loss of Ventilation; Dampers Close (recoverable); 1A0155, 1A0344, 1A0345, 1A0158, 1A0346, 1A0450, 1A0348, 1AK39, 1AK40, 1AB21, 1AB22
20	480V SWGR 12B Bus Fire	Loss of Bus 12A; Loss of Ventilation; Dampers Close (recoverable); 1AK41, 1AK40, 1AB23, 1AB21, 1A0348, 1A1452, 1A0349, 1A0161, 1A5034, 1AB46, 1A0162, 1A5055, 1A1251, 1A2278, 1A0353, 1A0352, 1A0351, 1AB76, 1A0156, 1A0355, 1A0357, 1A0358
21	Disconnect (6)	Loss of Disconnect Panel (no other impact); Loss of Ventilation; Dampers Close (recoverable)
22	1PNL1C86	Loss of 1C86 (no other impact); Loss of Ventilation; Dampers Close (recoverable)

Table 4-F-3
Unit 1 27' Switchgear Room (A317)
Fire Scenarios (Continued)

Scenario	Fire Scenario Description	Equipment Damaged
23	RPS MG sets	Loss of MG (no other impact); Loss of Ventilation; Dampers Close (recoverable)
24	4KV Bus 11 Transient Fire	Loss of 4kV Bus 11
25	4KV Bus 12 Transient Fire	Loss of 4kV Bus 12
26	480VAC Bus 11A Transient Fire	Loss of 480VAC Bus 11A
27	480VAC Bus 11B Transient Fire	Loss of 480VAC Bus 11B
28	480VAC Buses 12A and 12B Transient Fire	Loss of Bus 12A and 12B (Frequency represents both Buses; individual Bus would have 1/2 frequency)

Table 4-F-4
Unit 1 27' Switchgear Room (A317)
Fire Analysis Results

Initiating Event	Fire Scenario	Frequency	Ignition Source	Functional Impact	CDF
A317F1	7,10	2.03E-4	Severe Transformer 11A(Dry) or 11B(Wet) fire	GE, AA, HS*, HF, AE, N1, N2, HH*, NR*, NS*, DM*, PG*, KX, KZ, RS*, VC, MN, LF, CV, WY*	6.52E-7
A317F2	1,9,14	3.46E-4	4KV Bus 11 Cubicle, Minor Transformer 11B fire, Severe Transformer 12B fire	AA, HF, AE	9.84E-7
A317F3	12,11,13	4.5E-5	Severe Transformer 12A fire, Minor Transformer 12A or 12B fires	GE, AA, HF, AE, HH*, NR*, NS*, DM*, PG*, KX, KZ, VC, RS*, RQ, MN, LF, CV, WY*, SR*	1.38E-7
A317F4	5,6,8,23,28	2.84E-3	RCP 11 or 12 Breaker Cubicle, Severe Transformer 11A fire, 1C86, 480VAC Bus 12A, 12B transient fires	HF	2.30E-7
A317F5	15,26	1.23E-4	480VAC SWGR 11A Cubicle or Transient fire	HF, N1	5.07E-8

Table 4-F-4
Unit 1 27' Switchgear Room (A317)
Fire Analysis Results (Continued)

Initiating Event	Fire Scenario	Frequency	Ignition Source	Functional Impact	CDF
A317F6	17,27	1.23E-4	480VAC SWGR 11B Cubicle or Transient fire	HF, N2	5.07E-8
A317F7	18,19,20	7.44E-4	480VAC SWGR 11B,12A, or 12B Bus fire	GE, HS*, HF, AE, N1, N2, HH*, NR*, NS*, DM*, PG*, KX, KZ, RS*, VC, RQ, MN, LF, CV, WY*, SR*	6.18E-7
A317F8	2,24	3.19E-5	4KV Bus 11 Bus or transient fire	Y1, QD, AA, HF, AE, Y4, RS*, MN, IA*, IB*, FT, LF, CV, V1, DL*, SG*, SH*, SR*, TH, K3	1.18E-7
A317F9	4,25	3.19E-5	4KV Bus 12 or transient fire	QD, QE, JA, GE, GJ*, AA HF, AE, N1, N2, Y3, QZ*, I1, DM*, PG*, VC, BV, SL, IA*, IB*, F7, LF, CV, HB, V5, CS, CT, SG* SH*	4.78E-7
A317FA	3	5.63E-4	4KV Bus 12 Cubicle fire	HF,AE,	1.31E-7
A317FB	16	2.54E-4	480VAC SWGR 11A	Y1, GE, HS*, HF, N1, HH*, NR*, DM*, PG*, KX, KZ, RS*, VC, BV, MN, FT, LF, CV, WY*, SG*, SH*, TH, K3	7.93E-7
A317FC	21,22	1.60E-4	Disconnect Panel or 1PNL1C86	HF, RS*, HB	3.82E-8

Unit 2 45' Switchgear Room (A407)

Twenty-nine fire scenarios were identified for A407. Twenty-four are the result of fixed ignition sources and five are due to transient ignition sources. Fourteen scenarios are screened due to low functional impact. The remaining scenarios identified in Table 4-F-5 are represented by four fire initiating events shown in Table 4-F-6.. The consolidation of fire scenarios is based on an assessment of the functional impact and ignition frequency of each scenario. The frequency of each initiator is the sum of the frequencies of all the fire scenarios it represents.

Table 4-F-5
Unit 2 45' Switchgear Room (A407)
Fire Scenario Summary

Scenario	Fire Scenario Description	Equipment Damaged
02	4.16kV Bus 23 Bus Fire with cable tray impact	Loss of Bus 23; Loss of Ventilation; Dampers Close (recoverable); 2AD71, 2AD72, 2AD70, 2AM51, 2AF02, 2AC40, 2AC45,
03	4.16kV Bus 24 Cubicle Fire impacting bus or load	Loss of Individual Cubicle or Load Center; Loss of Ventilation; Dampers Close (recoverable). Note that this Bus 23 failure does not impact cable trays.
04	4.16kV Bus 24 Bus Fire with cable tray impact	Loss of Bus 24; Loss of Ventilation; Dampers Close (recoverable); 2AD75, 2AD73, 2AD74, 2AF06, 2AC40, 2AC52
07	Minor Transformer 23A Liquid Fire	Loss of Bus 23A (Transformer only); Loss of Ventilation; Loss of Ventilation; Dampers Close (not recoverable); 2A0663, 2A0664, 2A0665
08	Severe Transformer 23A Liquid Fire	Loss of Bus 23A (Transformer and Switchgear cabinet); Loss of Ventilation; 2A0663, 2A0664, 2A0665, 2AF07, 2AC35, 2AC47, 2AC59
09	Minor Transformer 23B Liquid Fire	Loss of Bus 23B (Transformer only); Loss of Ventilation; Dampers Close (not recoverable); 2A0666, 2A0667, 2A00668
10	Severe Transformer 23B Liquid Fire	Loss of Bus 23B (Transformer and Switchgear cabinet; Loss of Ventilation; Dampers Close (not recoverable); 2A0666, 2A0667, 2A00668, 2AF08, 2AC36, 2AC48, 2AC60
11	Minor Transformer 24A Liquid Fire	Loss of Bus 24A (Transformer only); Loss of Ventilation; Dampers Close (not recoverable); 2A0669, 2A0670, 2A0671
12	Severe Transformer 24A Liquid Fire	Loss of Bus 24A (Transformer and Switchgear cabinet); Loss of Ventilation; Dampers Close (not recoverable); 2A0669, 2A0670, 2A0671, 2AF09, 2AC37, 2AC49, 2AC61
16	Transformer 24B (2XU-440-24B) - DRY	Loss of Bus 24B (Transformer only); Loss of Ventilation; Dampers Close (not recoverable); 2A0672
17	480V SWGR 23A Bus Fire	Loss of Bus 23A; Loss of Ventilation; Dampers Close (recoverable); 2A0473, 2A0472, 2A4123, 2A0471, 2A0470, 2A0469, 2A0468, 2A0474, 2A0475, 2A0476, 2AF07, 2AC35, 2AC59
20	480V SWGR 24A Bus Fire	Loss of Bus 24A; Loss of Ventilation; Dampers Close (recoverable); 2A0481, 1A0677, 2A0482, 2A1989, 2A0234, 2A0466, 2A0484, 2A1243, 2A0322, 2AF08, 2AF09, 2AC36, 2AC37, 2AC48, 2AC49

Table 4-F-5
Unit 2 45' Switchgear Room (A407)
Fire Scenario Summary (Continued)

Scenario	Fire Scenario Description	Equipment Damaged
22	480V SWGR 24B Bus Fire	Loss of Bus 24A; Loss of Ventilation; Dampers Close (recoverable); 2A0467, 2A2126, 2A0486, 2A5077, 2A0971, 2A5075, 1A0676, 2A0235, 2A0489, 2A0490, 2A0485, 2A2364, 2A0236, 2AF10, 2AF11, 2AC38, 2AC39
26	4KV Bus 22 Transient Fire	Loss of Bus 22
29	480VAC Buses 22A and 22B Transient Fire	Loss of Bus 22A and 22B (Frequency represents both Buses; individual Bus would have 1/2 frequency)

Table 4-F-6
Unit 2 27' Switchgear Room (A407)
Fire Analysis Results

Initiating Event	Fire Scenario	Frequency	Ignition Source	Functional Impact	CDF
A407F1	2,4	4.62E-5	4KV Bus 23 or 24 Bus Fire	Y2, QD, QF, QE, GF, GJ, AD, N7, N8, Y3, Y4, DM*, PG*, GW, U-2 Feedwater, F9	3.88E-8
A407F2	8,10,12,20	2.32E-4	480VAC 23A, 23B, 24A Severe Transformer Fire, or 480VAC 24A Bus fire	Y2, QD, QF, GF, GJ, AD, N7, N8, HH*, Y4, DM*, PG*, U-2 Feedwater, FO*	3.51E-8
A407F3	17,22	4.65E-4	480VAC SWGR 23A, 24B Bus Fire	GF, GJ, N7, N8, HH*, Y4, NR, PG*, U-2 Feedwater, FO*	3.77E-8
A407F4	3,7,9,11,16,26,29	5.93E-4	4KV Bus 23 Bus Fire, 480VAC Transformer 23A, 23B, 24A Liquid Fire Transformer 24B Dry Fire, 4KV Bus 12 Transient Fire or 480VAC Bus 12A, 12B Transient Fire	AD, Y4, U-2 Feedwater	7.43E-9

Unit 1 45' Switchgear Room (A430)

Twenty-seven fire scenarios were identified for A311. Twenty-two are the result of fixed ignition sources and five are due to transient ignition sources. These scenarios identified in Table 4-F-7 are represented by eight fire initiating events shown in Table 4-F-8. The consolidation of fire scenarios is based on an assessment of the functional impact and ignition frequency of each scenario. The frequency of each initiator is the sum of the frequencies of all the fire scenarios it represents.

Table 4-F-7
Unit 1 45' Switchgear Room (A430)
Fire Scenarios

Scenario	Fire Scenario Description	Equipment Damaged
01	4.16kV NSR Bus 13 Cubicle Fire impacting bus or load OR (480V SWGR 13A AND 13B Cubicle Fire)	Loss of Bus 13 or Individual Cubicle or Load Center; Loss of Ventilation; Dampers Close (recoverable). Note that this Bus 13 failure does not impact cable trays.
02	4.16kV Bus 13 Bus Fire with cable tray impact	Loss of Bus 13; Loss of Ventilation; Dampers Close (recoverable); 1AK57, 1AK64, 1AC37, 1AC35, 1AC38, 1AC34
03	4.16kV Bus 14 Cubicle Fire impacting bus or load	Loss of Bus 14, Individual Cubicle or Load Center; Loss of Ventilation; Dampers Close (recoverable). Note that this Bus 14 failure does not impact cable trays.
04	4.16kV Bus 14 Bus Fire with cable tray impact	Loss of Bus 14; Loss of Ventilation; Dampers Close (recoverable); 1AK65, 1AK59, 1AK60, 1AC33, 1AC40, 1AC41, 1AC45
05	RCP 13 Breaker Cubicle (2)	Loss of RCP 13 (no other impact); 1A0125, 1A0127
06	RCP 14 Breaker Cubicle (2)	Loss of RCP 14 (no other impact); 1A0126, 1A0128
07	Minor Transformer 13A Liquid Fire	Loss of Bus 13A (Transformer only); ; Loss of Ventilation; Dampers Close (not recoverable); 1A0804, 1A0813, 1A0814, 1A0815
08	Severe Transformer 13A Liquid Fire	Loss of Bus 13A (Transformer and Switchgear cabinet); Loss of Ventilation; Dampers Close (not recoverable); 1A0804, 1A0813, 1A0814, 1A0815, 1AK63, 1AC35
09	Minor Transformer 13B Liquid Fire	Loss of Bus 13B (Transformer only); Loss of Ventilation; Dampers Close (not recoverable); 1A0816, 1A0817, 1A0818
10	Severe Transformer 13B Liquid Fire	Loss of Bus 13B (Transformer and Switchgear cabinet); Loss of Ventilation; Dampers Close (not recoverable); 1A816, 1A0817, 1A0818, 1AK64, 1AC34, 1AC46, 1AC58
11	Minor Transformer 14A Liquid Fire	Loss of Bus 14A (Transformer only); Loss of Ventilation; Dampers Close (not recoverable); 1A0819, 1A0820, 1A0821, 1AK65, 1AC33, 1AC45, 1AC57

Table 4-F-7
Unit 1 45' Switchgear Room (A430)
Fire Scenarios (Continued)

Scenario	Fire Scenario Description	Equipment Damaged
12	Severe Transformer 14A Liquid Fire	Loss of Bus 14A (Transformer and Switchgear cabinet); Loss of Ventilation; Dampers Close (not recoverable); 1A0819, 1A0820, 1A0821, 1AK65, 1AC33, 1AC45, 1AC57, 1AK64, 1AC34, 1AC46
13	Minor Transformer 14B Liquid Fire	Loss of Bus 14B (Transformer only); Loss of Ventilation; Dampers Close (not recoverable); 1A0822, 1A0823, 1A1812, 1AK66, 1AC32, 1AC44, 1AC56
14	Severe Transformer 14B Liquid Fire	Loss of Bus 14B (Transformer and Switchgear cabinet); Loss of Ventilation; Dampers Close (not recoverable); 1A0822, 1A0823, 1A1812, 1AK66, 1AC32, 1AC44, 1AC56, 1AK67, 1AC31
15	480V SWGR 14A Cubicle Fire	Loss of Individual Cubicle or Load Center; Loss of Ventilation; Dampers Close (recoverable)
16	480V SWGR 13A Bus Fire	Loss of Bus 13A; Loss of Ventilation; Dampers Close (recoverable); 1A1593, 1A0452, 1A0453, 1A0454, 1A0455, 1A0456, 1A0864, 1A0450, 1A1253, 1A1252, 1A0458, 1A0459, 1A0650, 1A0460, 1AD77, 1AD78, 1AK56, 1AK63, 1AK57, 1AC30, 1AC47, 1AC59
17	480V SWGR 14B Cubicle Fire	Loss of Individual Cubicle or Load Center; Loss of Ventilation; Dampers Close (recoverable)
18	480V SWGR 13B Bus Fire	Loss of Bus 13B; Loss of Ventilation; Dampers Close (recoverable); 1A0466, 1A0465, 1A0463, 1A0464, 1A0478, 1A1594, 1A1246, 1A1250, 1A1252, 1A0461, 1AK63, 1AK64, 1AC34, 1AC35
19	480V SWGR 14A Bus Fire	Loss of Bus 14A; Loss of Ventilation; Dampers Close (recoverable); 1A0480, 1A0477, 1A0435, 1A5056, 1A2185, 1A0475, 1A0474, 1A0690, 1A2109, 1A0434
20	480V SWGR 14B Bus Fire	Loss of Bus 14B; Loss of Ventilation; Dampers Close (recoverable); 1A0480, 1A0477, 1A0435, 1A5056, 1A2185, 1A0475, 1A0474, 1A0690, 1A2109, 1A0434, 1A2109, 1A2782, 1AK66, 1AK67, 1AC31, 1AC32
21	Disconnects (6)	Loss of Disconnect Panel (no other impact); Loss of Ventilation; Dampers Close (recoverable)
22	RPS MG sets	Loss of MG (no other impact); Loss of Ventilation; Dampers Close (recoverable)
23	4KV Bus 11 Transient Fire	Loss of Bus 11

Table 4-F-7
Unit 1 45' Switchgear Room (A430)
Fire Scenarios (Continued)

Scenario	Fire Scenario Description	Equipment Damaged
24	4KV Bus 12 Transient Fire	Loss of Bus 12
25	480VAC Bus 11A Transient Fire	Loss of Bus 11A
26	480VAC Bus 11B Transient Fire	Loss of Bus 11B
27	480VAC Buses 12A and 12B Transient Fire	Loss of Bus 12A and 12B (Frequency represents both Buses; individual Bus would have 1/2 frequency)

Table 4-F-8
Unit 1 45' Switchgear Room (A430)
Fire Analysis Results

Initiating Event	Fire Scenario	Frequency	Ignition Source	Functional Impact	CDF
A430F1	2,4,8,25	5.80E-5	4KV Bus 13 Cubicle, 4KV Bus 14 Bus, Severe Transformer 13A liquid fire or 480VAC Bus 11A transient fire	QD, QF, JB, GG, GJ, AB, HZ, HG*, AF, N3, N4, M8, Y4, QQ, I2, PG*, KY, KH, KZ, MC, RQ, SL*, F9* CV, HB, HW	9.91E-8
A430F2	7,9,15,19	4.14E-4	Minor Transformer 13A, 13B Liquid Fire, 480V SWGR 14A Cubicle or Bus fire or 480V Bus 12A and 12B Transient fire	AB, HZ, HG*, Y4, KH, SL*	2.33E-7
A430F3	5,6	9.24E-5	RCP 13 or 14 Breaker Cubicle	HZ, HG*, Y3, Y4, KH, SL*	8.99E-10
A430F4	10,11,12,13, 14,20	3.03E-4	Severe Transformer 13B liquid fire, Severe or Minor Transformer 14A, 14B liquid fire or 480VAC SWGR 14B	QF, AB, HZ, HG*, AF, N3, N4, M1, M8, Y4, QQ, I2, PG*, KY, KH, KZ, RS*, MC, RQ, SL*, F9*, CV	5.45E-7
A430F5	1,23,26	5.77E-4	4KV Bus 13 Cubicle or transient fire, or Minor Transformer 13A or 13B liquid fire	HZ, HG*, AF,Y3, KH, SL*	3.53E-9
A430F6	3,17,24	4.55E-4	4KV Bus 14 Cubicle fire, 480VAC SWGR 14B Cubicle Fire or 4KV Bus 12 Transient Fire	AB, HZ, HG*,Y4, KH, SL*	2.77E-7
A430F7	16,18	4.96E-4	480VAC SWGR 13A or 13B Bus fire	QF, HZ, HG*, AF, N4, M8, QQ, NR*, NS*, PG*, KY, KH, KZ, RS*, MC, RQ, SL*, F9*, CV	2.01E-7
A430F8	21,22	2.89E-3	Disconnect Panel or CEDM Motor Generator Set	HZ, HG*, KH, RS*, SL*, HB	1.81E-7

Fire Ignition Frequency

Both fixed and transient ignition frequencies were determined for all four of the Switchgear Rooms.

Fixed Ignition Frequency

The fixed ignition frequency is determined by starting with the compartment fixed ignition frequency results of Section 4.3.2 and then developing a scenario specific ignition frequency as described in Section 4.3.4.3. The sources and parameters used for the switchgear rooms are:

Source	Generic Fire Frequency	Location Weighting	Source Weighting	Total Sources in Aux Bldg	Room Specific Frequency
Electrical Cabinet	1.5E-2	0.154	-	-	2.31E-3
Motor Generator Set	5.5E-3	2	1	4	2.75E-3
Transformer (Dry)(A311)	7.9E-3	2	2	79	4.00E-4
Transformer (Dry)(A317, A407, A430)	7.9E-3	2	1	79	2.00E-4
Transformer (Wet)(A317, A407)	2.5E-5*	n/a	3	n/a	7.50E-5
Transformer (Wet)(A311, A430)	2.5E-5*	n/a	4	n/a	1.0E-4

*Note: The fire frequency for 480V PCB Oil Filled Transformers was determined to be 2.5E-5. The derivation of this value is explained further in this section.

The table below, based on walkdowns, shows the total panel count for each switchgear room.

Switchgear Room Panel Counts

UNIT 1				UNIT 2			
A317		A430		A311		407	
Equipment	Panel Count	Equipment	Panel Count	Equipment	Panel Count	Equipment	Panel Count
Bus 11	15	Bus 13	15	Bus 21	15	Bus 23	16
Bus 12	15	Bus 14	15	Bus 22	14	Bus 24	14
RCP	4	RCP	4	RCP	4	RCP	4
Bus 11A	16	Bus 13A	16	Bus 21A	16	Bus 23A	16
Bus 11B	16	Bus 13B	16	Bus 21B	16	Bus 23B	16
Bus 12A	14	Bus 14A	14	Bus 22A	14	Bus 24A	14
Bus 12B	14	Bus 14B	14	Bus 22B	14	Bus 24B	14
Disconnects	6	Disconnects	6	Disconnects	6	Disconnects	6
1C86	1			2C86	1		
Total	**101**	**Total**	**100**	**Total**	**100**	**Total**	**100**

The EPRI Fire Events Database identifies 17 electrical cabinet fire events. The breakdown is as follows:

Event Type	No. of Events	Percentage
Bus/Transformer Connection	8	47%
Breaker	6	35%
Other Cubicle Component	3	18%

Given the above operating experience, the percentage of breaker cubicle fires that result in de-energizing the bus is calculated as 6/9 (67%). The percentage of breaker cubicle fires that result in isolating/failing the load is calculated as 3/9 (33%). The frequency of breaker cubicle fires resulting in failure of the breaker or load center is calculated as:

(2.31E-03 / total number of cubicles) x number of cubicles for Bus x 33% of incidents

The frequency of breaker cubicle fires that result in de-energizing the bus is calculated as:

(2.31E-03 / total number of cubicles) x number of cubicles for Bus x 67% of incidents

Transformers

The largest potential ignition source are the four 4KV to 480VAC transformers in each switchgear room. Based on a search of the EPRI Fire Events Database, ten indoor transformer fires have occurred, two of them in Auxiliary Building Switchgear Rooms. Although the impact of these fires appears to be insignificant, "dry" transformers are typically installed inside, while yard transformers are generally oil-filled.

Accidents with oil-filled (wet) transformers can occur as the result of an internal fault, resulting in a build up of flammable gases inside the transformer casing. Such a fault could lead to electrical arcing which, in turn, could ignite the flammable gases. The burning gases cause a pressure increase within the casing which results in either 1) the venting of transformer oil out through a relief valve, or, in a more severe scenario, 2) an explosion that dumps the burning oil to the surrounding area. In order to minimize the risk of a transformer fire that destroys the entire room, the transformer fluid can contain additives that decrease the flammability of the fluid, such as silicone or PCBs.

Polychlorinated biphenyl (PCB) was added to the transformer fluid as a fire-retarding agent. However, the properties that make PCB advantageous for fire protection also make PCB a hazard poison that does not readily break down. Therefore, while PCB will slow the spread and limit the impact of the fire, it will impede recovery and cleanup after the fire extinguished.

ASKAREL is a generic name for a class of fire-resistant synthetic chlorinated hydrocarbons and mixtures used as dielectric fluids that commonly contain between 30% and 70% PCBs. INERTEEN™ is a trade name for dielectric fluid marketed by Westinghouse Electric and used at CCNPP. The fluid used in the transformers in the Switchgear Rooms contain 60 percent PCBs and is essentially nonflammable.

The fire ignition frequency for the 480VAC transformers is based on an EPRI analysis titled, "EPRI Economic Risk Management Models for Electrical Equipment Containing PCBs." The following ignition frequencies are used:

Oil-Filled Transformer Ignition Frequencies

Event Type	Frequency per reactor year
Smoke/No Fire	2E-5
Large Fire	5E-6
Total	2.5E-5

These values credit the following factors associated with the 480VAC transformers:

- The transformers are highly visible yet have limited access
- The transformers are regularly inspected for leaks and maintained
- The transformers have both high and low side electrical fault protection

Note: the following values are used in the below calculations:

0.20 = severity factor where 20% of all transformer fires are severe transformer fires

0.80 = severity factor where 80% of all transformer fires are minor transformer fires

0.67 = frequency of breaker cubicle fires that result in de-energizing the bus

0.33 = frequency of breaker cubicle fires resulting in failure of the breaker or load center

Any scenarios not included in the below calculations were screened due to low functional impact.

$$A311F1 = A311F1_{\text{Fix Ignition}} = F_2 + F_4$$

$$= (2.31E\text{-}3/100 \text{ cabinets in room}) + (2.31E\text{-}3/100 \text{ cabinets in room})$$

$$\boxed{A311F1 = 4.62E\text{-}5}$$

$$A311F2 = A311F2_{\text{Fix Ignition}} = F_8 + F_{10} + F_{12} + F_{15} + F_{19} + F_{20}$$

= [(1.0E-4/4 transformers in room)*(0.2 severity factor) +
(1.0E-4/4 transformers in room)*(0.2 severity factor) +
(1.0E-4/4 transformers in room)*(0.2 severity factor) +
(2.31E-3/100 cabinets in room)*(16 cubicles in panel)*(.67) +
(2.31E-3/100 cabinets in room)*(14 cubicles in panel)*(.67) +
((2.31E-3/100 cabinets in room)*(14 cubicles in panel)*(.67)]

A311F2 = 6.97E-4

$$A311F3_{\text{Fix Ignition}} = F_1 + F_7 + F_9 + F_{11} + F_{13} + F_{14} + F_{16} + F_{17} + F_{18}$$

= [(2.31E-3/100 cabinets in room)*(14 cubicles in panel)*(.67) +
(2.31E-3/100 cabinets in room)*(14 cubicles in panel)*(.33) +
(1.0E-4/4 transformers in room)*(0.8 severity factor) +
(1.0E-4/4 transformers in room)*(0.8 severity factor) +
(1.0E-4/4 transformers in room)*(0.8 severity factor) +
(1.0E-4/4 transformers in room)*(0.8 severity factor) +
(1.0E-4/4 transformers in room)*(0.2 severity factor) +
(2.31E-3/100 transformers in room)*(16 cubicles in panel)*(0.33) +
(2.31E-3/100 transformers in room)*(16 cubicles in panel)*(0.33) +
(2.31E-3/100 transformers in room)*(16 cubicles in panel)*(0.67)

$A311F3_{\text{Fix Ignition}}$ = 9.00E-4

Initiating Event frequencies for rooms A317, A407, and A430 were performed in a similar manner.

Transient Ignition Frequency

The transient ignition frequency is determined by starting with the compartment transient ignition frequency results of Section 4.3.2 and then developing a scenario-specific ignition frequency as described in Section 4.3.4.3. The sources and parameters used for the switchgear rooms are:

Source	Generic Fire Frequency	Location Weighting	Source Weighting	Total Sources in Aux Bldg	Room Specific Frequency
Transient - other	1.3E-3	2	7	232	7.84E-5

The critical height for the Target-In-Plume case is approximately four feet. There are no cable trays lower than ten feet, therefore, no overhead transient targets are evaluated.

The formula for calculating the damage frequency due to transient combustibles is contained in Section 4.3.4.3.2. The floor area of the 27' level Switchgear Room is 3,162 square feet. Floor mounted panels, including Switchgear Cabinets and Transformers, occupy approximately 989 square feet of the area. This leaves approximately 2,173 square feet of floor area available for combustibles to be placed.

Based on the In-Plume Exposure worksheets for transient fires in this room, the Critical Damage Distance within which damage to cable trays is expected is four feet. Given that there are no cable trays lower than approximately ten feet, A_s is zero.

The floor area around targets with the critical radial separation distance, was determined from the Radiant Exposure worksheet. For the 27' level Switchgear Room, the Radiant Exposure Critical Damage Distance is a radius of approximately 1.5 feet. A_{sr} is, therefore, approximately 958 square feet of area around equipment where it is possible for a transient fire to occur.

Therefore, for the Switchgear Room, the following values are calculated for Transformers and Switchgear Cabinets (4.16 kV and 480 VAC):

1) Transformers and 480 VAC Switchgear (12A and 12B)

$$u = (0.0 + 120\ ft^2)/2{,}173\ ft^2$$

$$= 5.52E\text{-}02$$

2) Transformers and 480 VAC Switchgear (11A or 11B)

$$u = (0.0 + 60\ ft^2)/2{,}173\ ft^2$$

$$= 2.76E\text{-}02$$

3) 4.16 kV Switchgear

$$u = (0.0 + 250\ ft^2)/2{,}173\ ft^2$$

$$= 1.15E\text{-}01$$

4) RCP Breaker (11 or 12)

$$u = (0.0 + 230\ ft^2)/2{,}173\ ft^2$$

$$= 1.06E\text{-}01$$

5) MG Set

$$u = (0.0 + 30\ ft^2)/2{,}173\ ft^2$$

$$= 1.38E\text{-}02$$

Using the value of 1.15E-1 for "Transformers and 480 VAC Switchgear (11A or 11B)", the ignition frequency for A311F3 can then be calculated as follows:

$$
\begin{aligned}
A311F3 &= A311F3_{\text{Fix Ignition}} + A311F3_{\text{Transient}} \\
&= 9.00E\text{-}4 + [\,F_{24} + F_{26} + F_{27}\,] \\
&= 9.00E\text{-}4 + [(7.84E\text{-}5)*(1.15E\text{-}1) + (7.84E\text{-}5)*(2.76E\text{-}2) + (7.84E\text{-}5)*(2.76E\text{-}2)] \\
&= 9.18E\text{-}4
\end{aligned}
$$

A311F3 = 9.18E-4

The above remaining values for transient combustible in vicinity of targets (*u*) are used in a similar manner for rooms A317, A407, and A430.

Fire Suppression

Each room contains detectors that alarm in the presence of smoke. All eight of the detectors are located in the ceiling. The room is equipped with a total flooding halon suppression system. Nine cylinders of Halon 1301 are arranged on the west wall of the 45' level of the SWGR such that, if the smoke detectors in the 27' level SWGR actuate the system, five cylinders of halon will discharge into the room. Should the smoke detectors in the 45' level actuate the system, then all nine halon cylinders will discharge. A single smoke detector actuation will only initiate a fire signal. Halon suppression is released only when a smoke detector monitoring a second zone is actuated. Smoke detector actuations provide a signal to 1C24B, resulting in an audible and visible alarm in the Control Room.

Fire Suppression Induced Equipment Failures

Halon 1301 (bromotrifluormethane) is a chemical compound that is effective against flammable liquid surface fires, most solid combustible fires, and electrical fires. Only a low concentration of halon is needed to stop the combustion process and prevent further flame propagation. Halon is non-conductive, and leaves no residue, and does not create electrical short circuits and grounds or cause corrosive damage to equipment. The Halon 1301 used at CCNPP complies with the requirements of NFPA 12A. No equipment damage is assumed to result from actuation of the halon system.

A315	**Unit 1 Main Steam Isolation Valve Room**	**Location:**	45' Auxiliary Building
		Fire Area:	11
		CDF:	Screened - Low Fire Ignition Frequency

The total length of this room is forty-eight feet, the width is nominally twenty-eight feet, and the resulting area is 1,344 square feet. The ceiling is sixteen and a half feet high for a total volume of 21,504 cubic feet. There is also a pipe tunnel between this room and the Turbine Building that is credited as a fire barrier. The tunnel is thirty-two feet long by fourteen feet wide with a $6^1/_2$ foot ceiling. The distance from this room to the Turbine Building is approximately thirty-five feet.

There are wet pipe suppression devices and smoke detectors installed throughout the room. The tunnel contains various smoke detectors and a minimal amount of combustibles.

Fire Analysis Results

Four fire scenarios are identified for this room. One for the Main Steam Penetration Room HVAC Unit and three transient scenarios which impact the HVAC Unit, 11 MSIV and 12 MSIV. The impact of each of these scenarios are limited to the equipment itself. The plant impact due to fire frequency is bounded by the individual component failure rates used in the internal events CCPRA, and as a result, this room has no additional equipment impact due to fire. This room is screened due to low ignition frequency.

Fixed Ignition Sources

Pressurized oil in fluid power systems presents a fire hazard, particularly where ignition sources are present, including welding. Hydraulic fluids are generally petroleum based, non-corrosive, compatible with a variety of seals, and have good lubricating properties. Flashpoints range from 300 to 600 degrees Fahrenheit and auto-ignition temperatures from 500 to 750 degrees Fahrenheit. High-pressure pipe with welded and screwed joints, steel tubing, and metal-reinforced rubber hose are used to conduct oil at pressures up to 10,000 psi. Typically, failure of piping at threaded sections, failure of valves and gaskets or fittings, and rupture of flexible hose are the principal causes of oil release from fluid power systems. Lack of adequate supports to prevent vibration contributes to the failure. Repeated flexing and abrasion creates weak spots that eventually fail.

When oil under pressure is released through equipment failure, the result is usually an atomized spray of mist or oil droplets, which depending upon the pressure, may encompass large areas. The oil spray is easily ignited and results in a fire that is torch-like with a very high rate of heat release. To protect the Main Steam System against such fires, FYRQUEL 220, a fire resistant hydraulic fluid, is used in the MSIVs. Each of the MSIVs contains approximately nine gallons of FYRQUEL.

FYRQUEL 220 is not flammable or combustible. It will, however, decompose in a fire situations and release toxic materials including phosphorus oxides and flammable organic substituents. FYRQUEL is self-extinguishing once the source of ignition is removed. Effective fire suppression media is water spray, carbon dioxide, foam, or dry chemical or agents. FYRQUEL has a flashpoint of 475 degrees Fahrenheit but no auto-ignition temperature. FYRQUEL is stable at ambient temperatures and atmospheric pressures, and is not self-reactive or sensitive to static discharge.

The MSIVs are, therefore, not considered to be a plausible ignition source.

Transient Ignition Sources

The likelihood a hot work induced transient fire is very small since a Hot Work Permit is required for these rooms. This means that there is continual fire watch during any hot work activities and for at least a period of thirty minutes after work is complete. Therefore, for the purpose of analyzing transient ignition sources in this room, the transient combustible is assumed to be maintenance refuse that comes in contact for a significant length of time with the hot piping (assumed to be in excess of 225 degrees Fahrenheit) to cause combustion.

The EPRI Fire PRA Implementation Guide indicates that fixed ignition sources do not play a significant role in the creation of transient combustible fires and that transient combustible fires are most often ignited by transient ignition sources. It is assumed that the transient fire occurs on the floor. This is consistent with FIVE Methodology as representative of the most likely scenario plausible for this room. It is unlikely that a trash can would be left in this room for a length of time sufficient to become an ignition target.

Suppression Systems

These rooms are equipped with wet pipe suppression devices and smoke detectors throughout the room.

Fire Suppression Induced Equipment Failures

Based on the approach described in Section 4.3.4.4.4, equipment failure due to the inadvertent actuation of the automatic fire suppression system is assumed not to occur. Cable and conduit, valves, piping and other PRA equipment in the room are not considered to be susceptible to water damage.

A318	**Unit 1 Purge Air Supply Fan Room**	**Location:**	27' Auxiliary Building
		Fire Area:	19A
		CDF:	1.15E-9

This compartment houses the containment purge fan, hot water piping, and a hot water circulating pump to heat the purge air. It also contains conduit, junction boxes, and cabling associated with 13kV and 4kV electrical protection as well as auxiliary feedwater controls. The room also houses repeater transmitters for the plant radio communications and a small chilled water air handler to cool the room.

The room is approximately twenty-one feet long and thirty feet wide for 630 square feet of area. The ceiling is approximately sixteen feet high for a room volume of 10,080 cubic feet. The compartment has concrete floors, walls, and ceilings.

Fire Analysis Results

Seven fire scenarios were identified for A318. Five are the result of combined fixed and transient ignition sources, and two are due to transient ignition sources only. Five scenarios are screened due to low functional impact. The screening is based on the same criteria described in Section 4.3.1.3. The remaining two scenarios are combined into one fire initiating event. This consolidation is based on an assessment of the functional impact and ignition frequency of each scenario. The frequency of each initiator is the sum of the frequencies of all the fire scenarios it represents.

Table 4-H-1
A318 Transient Fire Scenarios Summary

Scenario	Fire Scenario Description	Trays and Panels Damaged by Fire
T6	Transient Damage	1J005A
T7	Transient Damage	1J005B

Table 4-H-2
A318 Fire Analysis Results

Initiating Event	Fire Scenario	Frequency	Ignition Source	Functional Impact	CDF
A318F8	T6, T7	6.28E-6	Transient damage, 1J005A or 1J005B	Y1, QD	1.15E-9

0

A318 Fire Ignition Frequency

Both fixed and transient ignition frequencies were determined for the Purge Air Supply Fan Room.

Fixed Ignition Frequency

The fixed ignition frequency is determined by starting with the compartment fixed ignition frequency results of Section 4.3.2 and then developing a scenario-specific ignition frequency as described in Section 4.3.4.3. The sources and parameters used for A318 are:

Source	Generic Fire Frequency	Location Weighting	Source Weighting	Total Sources in Aux Building	Room Specific Frequency
Ventilation Subsystem	9.5E-3	2	2	331	1.15E-4
Electrical Cabinet	1.9E-2	2	4	135	1.13E-3

Neither ventilation unit had either plant or fire impact, so there are no fire modeling scenarios described here. Similarly, the electrical cabinet room frequency is apportioned among the four electrical panels (transmitters and their power supplies) in the room. These too, had no plant or fire modeling impact. Their fire modeling is not described.

Transient Ignition Frequency

The transient ignition frequency is determined by starting with the compartment transient ignition frequency results of Section 4.3.2 and then developing a scenario specific ignition frequency as described in Section 4.3.4.3. The sources and parameters used for A318 are:

Source	Generic Fire Frequency	Location Weighting	Source Weighting	Total Sources in Aux Building	Room Specific Frequency
Cable fires - welding	5.1E-3	2	1	232	4.40E-5
Transient fires - welding	3.1E-2	2	1	232	2.67E-4
Transient - other	1.3E-3	2	7	232	7.84E-5

For a description of transients and transient fire analysis see Section 4.3.4.3. The transient fire is modeled as a maintenance refuse fire and therefore uses the "Transient - other" as the ignition frequency, F_{it} = 7.84E-05. The Unit 1 Purge Air Supply Room has only, therefore P_{fs} = 1.0.

The room floor area is approximately 630 square feet. Floor mounted equipment and interferences occupy approximately 255 square feet of floor area, therefore:

$$A_F = 630ft^2 - 255ft^2 = 375ft^2$$

$$u = (A_s + A_{sr}) / A_F = (A_s + A_{sr}) / 375ft^2$$

$$P_{fs} = 1$$

$$\text{And}: \quad F_t = F_{it} * u * P_{fs} = F_{it} * [(A_s + A_{sr}) / A_F] * 1$$

$$= 7.84E\text{-}5 * [(A_s + A_{sr}) / 375ft^2]$$

JUNCTION BOXES

The refuse fire In-Plume worksheets show the corner configuration plume screening distance is 8.6 feet: Wall configuration plume distance is 6.6 feet and room center plume distance is five feet. The postulated transient fire is three feet high.

The lowest overhead targets are conduits approximately ten feet from the floor. Based on the transient fire worksheets for this room, the worst case in-plume Critical Damage Distance is 8.6 feet, at corner locations. Since the fire burns at a height of three feet, damage will occur up to twelve feet high, but there are no conduit targets in the corner area. Against the wall, the Critical Damage Distance is less than seven feet, so damage stops below ten feet. (In addition, the targets are conduit which provides additional protection to the wires/cables inside.)

The lowest Purge Air Supply Room cable trays are thirteen feet overhead, well beyond the transient damage.

The large junction boxes, overhead targets, each present fifteen square feet of floor area within the open plume. Therefore:

$$\text{J-Box}_{\text{Transient}} = 7.84\text{E-}5 * [(A_s + A_{sr}) / 375\text{ft}^2] = 7.84\text{E-}5 * [(15\text{ft}^2 + 0\text{ft}^2) / 375\text{ft}^2] = 3.14\text{E-}6$$

Total Ignition Frequency

The total ignition frequency is the sum of the appropriate fixed and transient scenarios. In this case, the sole scenario is the combined junction box transient damage.

$$\text{A318F8} = \text{J-Box}_{\text{Transient}} * 2 = 3.14\text{E-}6 * 2 = 6.28\text{E-}6$$

Suppression Systems

The Purge Air Supply Fan Room is equipped with smoke detection only.

Fire Suppression Induced Equipment Failures

Based on the approach described in Section 4.3.4.4.4, equipment failure due to the inadvertent actuation of the automatic fire suppression system is assumed not to occur since there is no automatic suppression systems installed.

A405	Main Control Room (MCR)	Location:	45' Auxiliary Building
		Fire Area:	24
		CDF:	2.53E-5

The Control Room, which is accessible from both the Auxiliary and Turbine Buildings, houses benchboard control boards and miscellaneous vertical control boards for both Unit 1 and Unit 2. The room contains numerous vertical floor mounted electrical cabinets aligned in rows. The majority of the cabinets are open steel cabinets forming a horseshoe along the West wall. Some of the panels are enclosed and are at a height to accommodate handswitches and other operator controls. Cables transverse horizontally in cable trays between panels. Office equipment, including CRTs, copier and fax machine do not have sufficiently large motors to offer a significant ignition source.

The Control Room is an enclosed rectangular structure, approximately fifty-five feet x ninety feet x twenty-two feet for a total volume of over 100,000 ft^3, having a three-hour minimum fire rating. Fire detection consists of ionization detectors strategically located directly above the main control board. One foot beneath the ceiling are seismically qualified metal egg crate panels that serve to protect the operators from falling glass and debris from broken light fixtures.

Fire Analysis Results

Ninety-nine panels are located in the Control Room. These panels are grouped and represented by 29 fire initiating events shown in Table 4-I-1.

Table 4-I-1
Control Room Panels

Initiator (Frequency)	Panels	Description	Panel Propagation Left	Panel Propagation Right	Functional Impact	CDF
FI1C03 (7.88E-05)	1C03	Condensate and Feedwater Control Board	SW2	SW2	QQ, T1, MC, RI, BV, DW, DV, BS, MN, FT, MS, FN*, HX*, UQ*, HU, TG, F9, MH, F3*, LF, OT	1.18E-6
FI1C04 (1.18E-04)	1C04	Auxiliary Feedwater and Computer Control Board	SW2	Y	QZ*, QQ, MC, RR, RI, PS, PV*	3.23E-6
	1C05	Reactivity Control Board	Y	Y	FT*, FN, FH, F7*, HX*, TF, TG, F9, MT, MH*, F3*, LF, OT	
FI1C06 (1.18E-04)	1C06	Reactor Coolant Control Board	Y	SW2	QQ, KX RS (DSS fails), RR, RI	1.42E-7
	1C05	Reactivity Control Board	Y	Y	PS, PV*, AQ, SL*, CV, OT, SA	

Table 4-I-1
Control Room Panels (continued)

Initiator (Frequency)	Panels	Description	Panel Propagation Left	Panel Propagation Right	Functional Impact	CDF
FI1C07	1C07	Chemical and Volume Control Board	SW2	Y	QQ, TB, AQ, SL*, MS*, CV, OT, RH, HA, HB,	7.22E-7
(1.58E-04)	1C08	Engineering Safeguards	Y	SW2	HW, DL*, CS, SR*	
FI1C09 (7.88E-05)	1C09	Engineering Safeguards	SW2	SW2	AA*, AB*, KI, OT, MV, RH, HA, HB, HW, EA, WY*, CT, SR*	2.96E-7
FI1C10 (7.88E-05)	1C10	Engineering Safeguards	SW2	End	HZ, KJ, KL, OT, VM, HA, HB, V5, HW, EB, SG*, WJ, SH*, SR*, SI	2.95E-7
FI1C13 (7.88E-05)	1C13	Salt Water, Service Water and Component Cooling Water Control Board	DW7	End	GG, HS*, NR*, I1, I2, S1, S2, KX, KY, KZ, VC, FC, FO, WY	9.99E-7
FI1C17 (7.88E-05)	1C17	4KV and 480VAC System Normal Control Board	End	SW2	H5, AE, AF, DM*, VC, RQ, LF	2.94E-7
FI1C18 (7.88E-05)	1C18	13KV and 4KV System Essential Control Board	SW2	SW2	QC, QE, GE, H5, AA, QQ, DM*, VC	2.44E-6
FIC18A (3.94E-05)	1C18A	EDG 1A 4KV and 480VAC Control Board	SW2	End	GE, GJ, QQ	4.02E-7
FIC18B (3.94E-05)	1C18B	EDG 1B 4KV and 480VAC Control Board	SW2	SW2	GG, GJ, QQ	3.4E-7
FI1C19 (7.88E-05)	1C19	13KV and 4KV System Essential Control Board	SW2	SW2	GF, GG, H5, GJ, AB, AD	3.34E-6
FIC19C (3.94E-05)	1C19C	EDG 0C Control Board	SW2	SW2	GJ, QQ	1.29E-7
FI1C20 (7.88E-05)	1C20	13KV and 4KV Essential Control Board	SW2	SW2	QD, QF, GH, H5, GJ, AC, QQ, DM*	5.93E-7
F1C20A (3.94E-05)	1C20A	2A DG Control Board	End	SW2	GF, GJ, QQ	4.02E-7
F1C20B (3.94E-05)	1C20B	2B DG Control Board	SW2	SW2	GG, GJ, QQ	3.40E-7
FI1C34 (3.94E-05)	1C34	HVAC System Control Board	End	SW2	HH, QQ, SL*, LF, VM, V1, V2, V5	1.29E-7
FI2C05 (2.36E-04)	2C04	Auxiliary Feedwater and Computer Control Board	SW2	Y	F9	7.70E-7
	2C05	Reactivity Control Board	Y	Y		
	2C06	Reactor Coolant Control Board	Y	SW2		

Table 4-I-1
Control Room Panels (continued)

Initiator (Frequency)	Panels	Description	Panel Propagation Left	Panel Propagation Right	Functional Impact	CDF
FI2C09 (7.88E-05)	2C09	Engineering Safeguards	SW2	SW2	AD*, AC*	2.95E-7
FI2C13 (7.88E-05)	2C13	SRW/Misc Services Control Panel	End	SW2	GF, GH, NS, F9	2.94E-7
FI2C17 (7.88E-05)	2C17	4kV/480VAC System Normalizer	SW2	End	Unit 2 MFW lost, DM*	2.94E-7
FIC24A (7.88E-05)	2C24A	DC Power Control Boards	End	SW2	S1, S2, GW, GZ	4.79E-7
A405F1 (1.97E-04)	1C01	Main Generator and Switchyard Control Board	SW2	Y	OP	1.18E-6
	1C02	Turbine Control Board	Y	SW2		
	1C29	CB Gen Metering & Misc	SW2	SW2		
A405F2 (1.58E-04)	1C15A	Reactor Protection System Channel A	End	SW2	QZ*, RS*, PV*, IA*, IB*, SA*, SB*	5.14E-7
	1C15B	Reactor Protection System Channel B	SW2	SW2		
	1C15C	Reactor Protection System Channel C	SW2	SW2		
	1C15D	Reactor Protection System Channel D	SW2	SW2		
	1C25A	RPS Channel A Power Supply Cabinet	SW2	SW2		
	1C25B	RPS Channel B Power Supply Cabinet	SW2	SW2		
	1C25C	RPS Channel C Power Supply Cabinet	SW2	SW2		
	1C25D	RPS Channel D Power Supply Cabinet	SW2	SW2		
A405F3 (3.94E-05)	1C35	Feedwater Regulation Control System 11 Cabinet	DW7	SW2	MP, MN, LF	1.46E-7
	1C36	Feedwater Regulation Control System 12 Cabinet	SW2	DW7		
A405F4 (1.18E-04)	1C24B	Fire Protection Control Board	SW2	SW2	HH	4.39E-7
	1C39	Miscellaneous Station Recorder Panel	SW2	SW2		
	2C39	Miscellaneous Station Recorder Panel	SW2	SW2		
A405F5 (5.91E-05)	1C28	Technical Support Center Isolation Panel	Y	Y	QZ*, RR, IA*,IB*	2.19E-7
	1C31	Reactor Regulation System Channel X Control Board	Y	SW2		
	1C32	Reactor Regulation System Channel Y Control Board	SW2	Y		
A405FM (3.55E-04)	2C03	Condensate and Feedwater Control Board	SW2	SW2	Unit 2 MFW lost	1.16E-6
	2C07	Chemical and Volume Control Board	SW2	Y		
	2C08	Engineering Safeguards	Y	SW2		

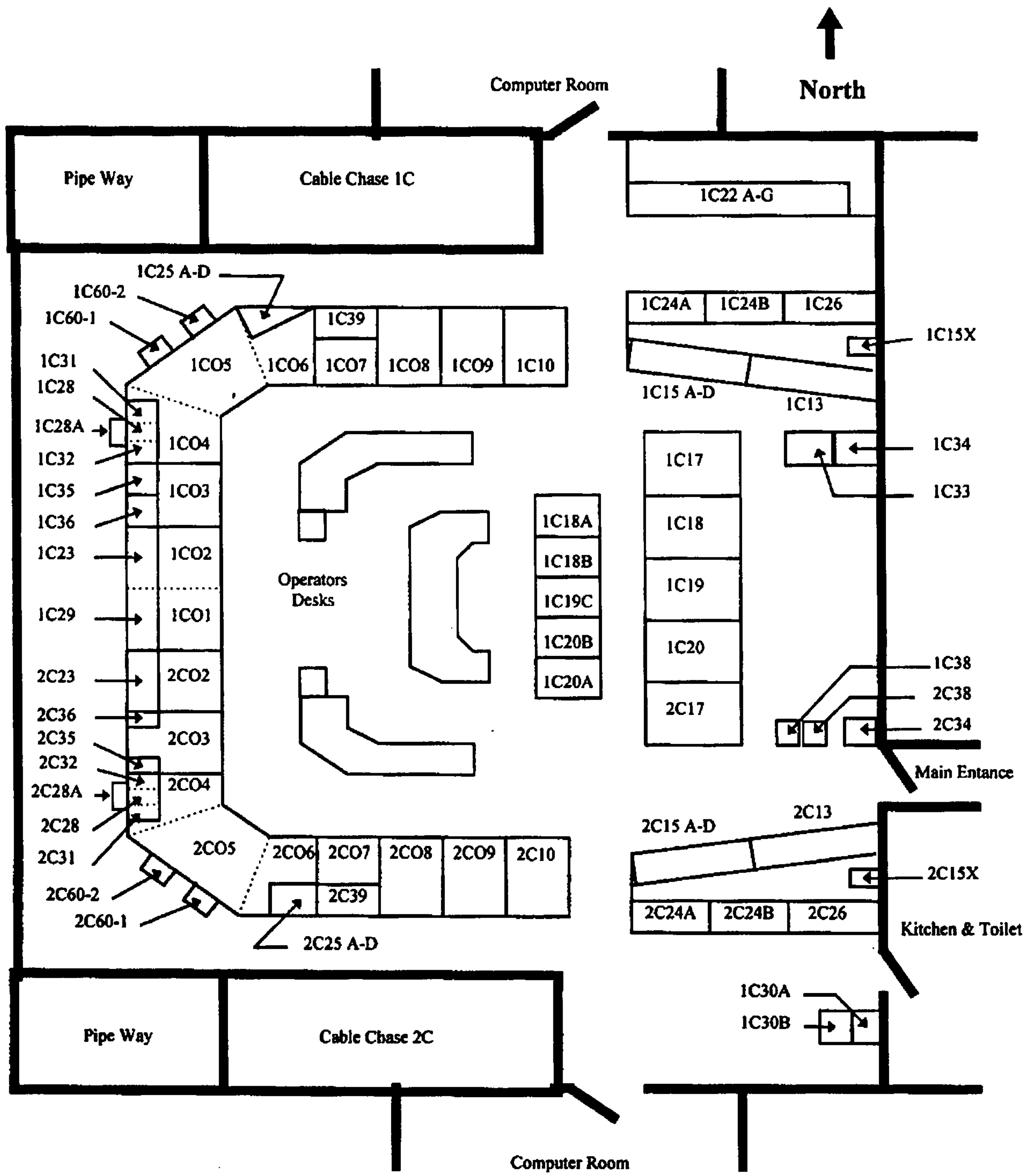
Table 4-I-1
Control Room Panels (continued)

Initiator (Frequency)	Panels	Description	Propagation Left	Propagation Right	Functional Impacts	CDF
	2C35	Feedwater RCS 21	DW7	SW2		
	2C36	Feedwater RCS 22 Control Board	SW2	DW7		
	2C38	Unit 2 Feed Pump Turbine Control Board				
A405FN (1.34E-03)	1C05A-D	RRS Reactivity Control	SW2	End	NONE	4.28E-6
	1C15X	Reactor Protection System Isolation Cabinet	DW7	DW7		
	1C22A-G	Radiation Monitoring and Meteorologic Control Board	End	End		
	1C23	Plant Oscillograph	SW2	DW7		
	1C24A	DC Power Control Board	SW2	End		
	1C26	Vibration Monitor Panel Control Board	SW2	SW2		
	1C28A	TSCC Computer Multiplexer	DW5	End		
	1C30A	Unit 1 Loose Parts Monitoring System	End	DW1		
	1C30B	Unit 2 Loose Parts Monitoring System	DW1	End		
	1C33	WPS Panel	End	SW2		
	1C38	Steam Generator Feed Water Pump 11 & 12 Signal Processor Cabinet	SW2	SW2		
	1C60-1	Plant Computer Cabinet	SW2	SW2 DW5 (back)		
	1C60-2	Plant Computer Cabinet	SW2	SW2 DW5 (back)		
	2C02	Turbine Control Board	Y	SW2		
	2C10	Engineering Safeguards	End	SW2		
	2C15A-D	Reactor Protection System Channel A/B/C/D	SW2	SW2		
	2C15X	Reactor Protection System Isolation Cabinet	DW7	DW7		
	2C23	Vibration Monitor	SW2	DW7		
	2C24B	DC Power Control Board	SW2	SW2		
	2C25A-D	RPS Channel A/B/C/D Power Supply Cabinet	SW2	SW2		
	2C26	Vibration Monitor Panel Control Board	SW2	SW2		
	2C28	Technical Support Center Isolation Panel	Y	Y		
	2C28A	TSCC Computer Multiplexer	DW5	N/A		

Table 4-I-1
Control Room Panels (continued)

Initiator (Frequency)	Panels	Description	Propagation		Functional Impacts	CDF
			Left	Right		
	2C31	Reactor Regulation System Channel X Control Board	Y	SW2		
	2C32	Reactor Regulation System Channel Y Control Board	SW2	Y		
	2C34	HVAC System Control Panel	SW2	N/A		
	2C60-1	Plant Computer Cabinet	SW2	SW2 DW5 (back)		
	2C60-2	Plant Computer Cabinet	SW2	SW2 DW5 (back)		

Figure 4-I-1
Main Control Room



Approach Overview

The Main Control Room fire risk is assessed by modeling the various consequences associated with a fire in each Control Room panel. This assessment is coded explicitly into the CCFPRA and is achieved by:

- Evaluating and grouping the Control Room panels in accordance with their physical and functional attributes into fire initiating events.
- Determining the ignition frequency for each fire initiating event. Two types of fires are evaluated for each panel: minor (67%) and severe (33%).
- Determining the likelihood of Control Room evacuation. Fifteen minutes is used as the time available for manual suppression. Failure to suppress results in Control Room evacuation.
- Determining the likelihood of propagation between panels (or panel groups) given the fire has not been suppressed within the first fifteen minutes. Propagation between panels is assumed to occur on failure of the Fire Brigade to suppress the fire within fifteen minutes after the Control Room has been evacuated.
- Evaluating the functional impact for the loss of a panel (or panel group), the evacuation of the Control Room with the loss of a panel (or panel group), and the evacuation of the Control Room with the loss of multiple panels. The risk-assessment associated with the evacuation of the Control Room includes the evaluation of the impact of implementing the Appendix R safe shutdown procedure for Control Room evacuation (AOP-9A).

Fire Ignition Frequency

Adjacent Rooms

There are areas adjacent to the Main Control Room whose fire barriers are not credited for separation. These areas are all contained within Fire Area 24 and are listed below:

Room	Description
A400	Control Room Vestibule
A401	Operations Shift Office
A402	Control Room Toilet
A403	Janitor Storage
A404	Kitchen - Control Room
A405	Main Control Room
A406	Unit 2 DAS Computer Room
A415	Area behind SW panel
A431	Unit 1 DAS Computer Room
A432	Tech Support 45'
A434	Passage
A436	Tech Support Center
A437	Tech Support Annex
A438	Shift Supervisor's Office
A442	Reserve Battery Room
A443	Passage
A444	Central Alarm Station

Smoke is not anticipated to migrate into the Control Room from fires in adjacent or outside rooms. Smoke from a fire in either Unit's Battery Rooms or the Reserve Battery Room will not migrate into the Control Room because there are separate supply and exhaust duct systems and fans, which are isolated from the Control Room by a separate duct system with an exhaust fan. The Control Room/Cable Spreading Room HVAC system does not provide any ventilation to these rooms. All Battery Rooms are under a negative pressure and have normally closed and locked doors which would further prevent smoke migration.

Rooms A401, A402 and A404 share a common supply from the Control Room HVAC system. They do not share the CR HVAC return but instead have a separate exhaust duct/fan to atmosphere. The supply does not have a smoke damper, however, should a fire occur the separate exhaust will prevent smoke spread into the Control Room. Supply and return ducting is common to the CR/CSR HVAC system for the Tech Support Center Rooms (A432 and A436) and the Central Alarm Station (A444). The TSC Annex (A437) does not have common return duct. The return air for A437 is pulled into the CAS via an air transfer grill in the common wall between room A437 and the CAS. All rooms have smoke detection which will activate Electro Thermal Link controlled dampers. These dampers will close and isolate the rooms from the rest of the HVAC system and prevent smoke migration into the Control Room.

An evaluation of onsite combustibles installed in the CR Vestibule under FCR 89-166 indicated that the combustible loading in this and other areas adjacent to the CR do not challenge the available fire protection of the plant. The barriers in the Vestibule, Shift Supervisor's Office, and Tech Support Center are made of fire retardent gypsum wallboard which conforms to ASTM C36-66 and has a one hour resistive rating. The Toilet area walls are constructed of ceramic tile. The walls in the remaining areas are constructed of concrete and masonry, which provide for a two-hour rated barrier.

These areas do not contain sufficient fixed ignition sources or combustibles that they pose a concern for propagation into the Control Room. These areas do contain combustible materials with low flame spread ratings, such as carpeting and office furniture. However, there are insignificant or no sources of ignition in these rooms. It is assumed that a fire in these rooms will not propagate into any adjacent compartments.

<u>*Control Room Panel Fires*</u>

The fire damage scenarios evaluated for the Main Control Room initiate as a panel fire from a single panel or from a panel group (panels without adequate isolation) within the Control Room. Ninety-nine physical panels for both Unit 1 and Unit 2 are evaluated. Based on the configuration of the panels, fifty-three fire scenarios are identified. The panels are consolidated based on an assessment of the functional impact and ignition frequency. Twenty-nine fire initiators are identified for the Control Room.

Based on a review of the EPRI Fire Events Database, there are a total of twelve events attributed to Control Room fires. Further evaluation revealed that one actually occurred outside the Control Room, one was a re-occurring event, and one was a Control Room kitchen fire due to an oven grease fire. These three fires are screened as unrelated to Control Room electrical fires. Of the remaining nine fires, two were self extinguished and five were manually extinguished. One of the manually suppressed fires was considered severe, while the remaining were considered minor. No additional information is available regarding the two remaining fires; however, they were considered to be benign. For this analysis, it is assumed that one of these was self-extinguished and the other is conservatively assumed to be severe. A total of six fires are considered to be manually extinguished. Of these, two were severe fires, while four were minor.

The panel fire ignition frequency was originally determined to be 1.93E-02, through the method described in Section 4.3.2. Excluding the self-extinguished fires, the compartment ignition frequency is determined to be that portion associated with manually extinguished fires.

6 (manually extinguished) / 9 (total CR fires) * 1.93E-02 = 1.29E-02

Given the above, two categories of panel fires are considered: minor and severe.

Minor Panel Fires

Minor fires are those that are quickly extinguished by the operating crew. Since the data for the minor fires includes manual suppression, it is possible for these fires to challenge the Control Room if left unchecked. Therefore, in CCFPRA, these minor fires are conservatively evaluated as resulting in Control Room evacuation if manual suppression is unsuccessful. When manual suppression is successful, the panel damage is assumed to be the loss of a single function and to be bounded by the random equipment failure rates used in the internal events PRA. Therefore, when manual suppression is successful, the minor panel fires are screened. Based on a review of the EPRI Fire Events Database, minor panel fires are assumed to be 67% of the ignition frequency.

Severe Panel Fires

These fires are considered to result in the loss of all functions within a single panel or panel group. As with the minor panel fire above, failure to suppress the fire will result in the evacuation of the Control Room. On Control evacuation, all functions in the ignition source panel are assumed to be lost and the actions associated with AOP-9A, Control Room Evacuation, are evaluated. Based on a review of the EPRI Fire Events Database, severe panel fires are assumed to be 33% of the ignition frequency.

Panel Weighting Factor

A cabinet frequency is determined by distributing the compartment frequency of 1.29E-02/yr by weighting the relative size of the electrical cabinets in the Control Room. The weighting factor given in Table 4-I-2 is based on the relative size of the panel. Due to this weighting factor, the 99 physical panels are reduced to 54. The frequency of a single large panel such as 1C03 is 1/54 x 1.29E-02 = 7.889E-5/year. This approach raises the frequency of the larger panels and lowers the smaller ones. It assumes that the combustible loading associated with a panel is proportional to the size and design of the panel.

Table 4-I-2
Control Room Panel Weighting Factor

Panel Type	Weighting Factor
Vertical electrical panel (main)	1.00
Benchboard	0.50
Small sub-panels	0.25
Enclosed standalone panels	0.50

Panel Function Determination

The functions for each panel are determined through a review of the cable routing and through the performance of walkdowns. Cable routing identifies panels as a to or from location. Therefore, by using the cable database developed for the PRA, the applicable functions can be identified.

Manual Fire Suppression Time

Cabinets and consoles are subject to damage from two distinct fire hazards:

1. Fire originating within the cabinet; and

2. Exposure fires involving combustibles in the room area.

The only ignition sources within an electrical cabinet are those associated with electrical faults. Given experience with fires reported in the EPRI Fire Events Database, damage can be confined to the site of the overload, and the impact is bounded by the random failure of the component itself (which has already been accounted for in the PRA model).

The likelihood of deteotion and suppression is dependent upon whether or not cabinets are fitted with in-cabinet detection. Heat and smoke detectors are installed in the Control Room but not within the individual cabinets. The Sandia cabinet fire tests indicate a five-minute time lapse between an in-cabinet fire detector detecting smoke and the time that actual flames were observed. These tests used vertical and benchboard cabinets loaded with unqualified cables ignited using an electrical ignition source. No credit is taken for detection and suppression during this phase since there is no cabinet detection.

The potential for significant damage due to a cabinet fire is very small prior to flame ignition. Ignition may be prevented by de-energizing the faulted component and using manual fire extinguishers. Controls in the cabinets are generally not temperature sensitive and temperature sensitive instruments do not typically control safety-related equipment. Fires within instrumentation cabinets are assumed to result in the loss of the entire cabinet. This assumption is consistent with the guidance in the EPRI Fire PRA Implementation and with experimental test results conducted at Sandia. All components operated by the cabinet in which the fire originates are assumed to fail (unless there is an alternate control means), given that the fire is not suppressed within fifteen minutes.

The Sandia test results indicated that cabinet fires were self-sustaining, and produced sufficient quantities of smoke to cause visual impairment with purge rates as high as fourteen room changes per hour. All of the actual Control Room fires in the EPRI Fire Database were small; however, this may have been the result of early extinguishing. Since there are insufficient tools available for assessing smoke production and historical fire data is not conclusive, it is assumed that any fire is capable of producing sufficient smoke to require evacuation of the Control Room, given it is allowed to continue burning for a sufficient length of time.

Eleven Sandia tests are available to provide information on smoke build up. Six tests were conducted for small enclosures (11,016 ft^3) with ventilation rates of about 14 room changes per hour. Only one of these, however, was electrically initiated and indicated visual obscuring within thirteen minutes, where time zero is the point at which smoke was first observed from the cabinet. Five tests were performed in larger enclosures (48,000 ft^3) in which two fires were electrically initiated. In both of these cases, the main control board was obscured with 15.5 to 19.5 minutes following observation of smoke. For large enclosure fires, the ventilation system does not appear to substantially affect the rate of smoke build up. The ventilation rate in one case was one room change per hour, and in the other case eight room changes per hour.

Based on the above discussion, it is concluded that the rate of smoke build up in the Control Room will be marginally slower than observed for the large test enclosure. It is determined that smoke obscuration of the control board will not occur for at least twenty minutes after ignition. Allowing three to five minutes for activation of the area ionization smoke detectors, fifteen to seventeen minutes would be available to extinguish a fire in a cabinet with no in-cabinet detection , prior to the need for evacuation of the Control Room. This effectively assumes that our operators would detect a fire in three to five minutes, this is believed conservative. Actual detection time may be in the two to three minute range.

The guidance provided in NSAC-181 suggests that approximately 15 minutes is available for a typical nuclear power plant Control Room before fire induced degradation of the Control Room environment requires evacuation. This fifteen-minute time period is consistent with the guidance in Appendix H of the EPRI Fire PRA Implementation Guide for fire propagation to adjacent control panels.

Appendix M of the EPRI Fire PRA Implementation Guide suggests that as much as fifteen to twenty minutes may be available for manual fire suppression in the Control Room depending upon a number of factors, including interpretation of the test data, volume of the Control Room, and volume changes per hour of the Control Room HVAC. The Implementation Guide recommends that the size of the room and the volume changes per hour, be investigated to verify that the SNL test conclusions reasonably apply to the specific site. The CCNPP Main Control Room is over twice the size of the SNL test facility (approximately 100,000 ft^3) with an HVAC system rated at 20 room changes per hour in full outside air mode.

Based on the HCR model derivation in the Fire PRA Implementation Guide, the probability of non-suppression in 15 minutes with no in-cabinet detection is 3.40E-3.

Fire Propagation Between Panels

The propagation between panels evaluation is based on the guidance provided in Appendices H and M of the EPRI Fire PRA Implementation Guide.

A walkdown was conducted to determine whether the panel features satisfy the EPRI document criteria for no propagation. In general, the criteria requires that a panel be fully enclosed and separated from an adjacent panel by an air gap. This configuration is referred to 'double wall with air gap' in the EPRI document. An air gap is defined in this study as an identifiable separation between panels such that there is no significant conductive heat transfer mechanism. An additional requirement is that the adjacent panel not contain 'sensitive' electronic equipment such as solid state devices.

The consequences of a postulated fire in panels which satisfy this criterion would be limited to only those plant systems whose wiring are present within the panel enclosure. If the adjacent panel contains 'sensitive' electronics, damage to the adjacent panel is assumed to occur at ten minutes unless cooling is provided, or the adjacent panel is not fully enclosed.

Many of the Control Room panels do not satisfy the 'double wall with air gap' criterion. No panels with sensitive electronics are identified. The majority of the Control Room panels are grouped into two categories:

1. Fully enclosed and separated from an adjacent panel by a single wall without an air gap

2. Open back and separated from an adjacent panel by a partial wall without an air gap

The following table summarizes the treatment for potential fire propagation to an adjacent panel.

Table 4-I-3
Treatment of Potential Control Room Panel Fire Propagation

BOUNDARY TYPE	EXPOSING PANEL CONFIGURATION	EXPOSED PANEL CONFIGURATION	SENSITIVE ELECTRONICS	PROPAGATION TIME	CODE
Double Wall with Air Gap	Enclosed	Enclosed	Y	10 minutes	DW1
			N	No propagation	DW2
		Open/Ventilated	Y	15 minutes	DW3
			N	No propagation	DW4
	Open/Ventilated	Enclosed	Y	15 minutes	DW5
			N	No propagation	DW6
		Open/Ventilated	Y	No propagation	DW7
			N	No propagation	DW8
Single Wall or no Air Gap	Enclosed	Enclosed	Y	0 minutes	SW1
	All other cases	All other cases	All other cases	15 minutes	SW2
Unsealed Penetrations	Enclosed	Enclosed	Y	0 minutes	UP1
	All other cases	All other cases	All other cases	Case specific	UP2

For those cases where a partial boundary separates adjacent panels, the likelihood of hot gas layer formation, spread of hot gases to an adjacent panel, arrangement of potential ignition sources and combustible material within the exposing and exposed panels, and the location of potential propagation pathways are qualitatively evaluated. Appendix H of the EPRI document states that fire spread to an adjacent panel is delayed by fifteen minutes even when there is not internal barrier. This suggests that damage to an adjacent panel may occur, but damage to the next further panel will not occur for at least fifteen minutes.

The following guidelines were used when performing a case specific assessment:

- Fires tend to result in damage to targets above the fire. Lateral fire spread via cable insulation is unlikely due to IEEE-383 'equivalent' cable qualification.

- Wide spread damage requires a notable time for the development of a hot gas layer. Such formation is unlikely in an open enclosure or other 'significantly' ventilated enclosure before the 15 minute suppression time required for Control Room habitability concerns.

- Cable fires typically involve the generation of smoke and other combustion byproducts and a relatively low heat rate.

- Openings in barriers near the bottom of a panel are an unlikely propagation pathway unless an ignition source is present as well as combustibles on both sides of the barrier.

- Openings near the top of a boundary between adjacent panels is a potential pathway for hot gases to propagate to an adjacent panel.

- The open backs of the main control panels creates a radiant exposure case for the panels on the opposite side of the interior passageway. However, a heat rate of over 282 Btu/s would be needed to yield a critical flux distance of three feet. The expected heat rate is only 65 Btu/s and the width of the passageway is over three feet. Therefore, damage due to radiant exposure is not credible.

As a result of the panel barrier analysis, those cabinets in which there was a total absence of any physical barrier are combined together. If two adjacent cabinets have a solid boundary which divides them, then they were treated as two separate panels. If the barrier has a limited number of openings, or does not extend into the panel apron area, the two panels were treated as two separate cabinets. Two cabinets are combined if the barrier has substantial open areas, has a significant wiring volume passing through it, or does not extend to the top of the panel.

For one important group of panels (1C04, 1C05 and 1C06), an overlapping zone approach is used. This approach divides this group into two zones, each represents by a fire initiating event. Initiating Event FI1C04 represents 1C04 and 1C05, and Initiating Event FIC06 represents 1C05 and 1C06. One half of the frequency for 1C05 is included in each initiating event. This approach allows the functional impact to be more realistically modeled by initially only impacting the functions of two panels on a severe fire.

Fire Suppression

There is no automatic fire suppression system installed in this room.

Fire Suppression Induced Equipment Failure

Suppression of fires is expected to be accomplished manually through utilization of CO_2 fire extinguishers. CO_2 is non-corrosive in the short-term and will not result in any equipment damage.

A419	**Cask and Equipment Loading Area-Truck Bay**	**Location:**	45' Auxiliary Building
		Fire Area:	11
		CDF:	1.86E-7/yr
A408	**Unit 2 45' Piping Area**	**Location:**	45' Auxiliary Building
		Fire Area:	11
A410	**45' Passageway**	**Location:**	45' Auxiliary Building
		Fire Area:	11
A413	**Unit 2 Sampling Room**	**Location:**	45' Auxiliary Building
		Fire Area:	11
A424	**Unit 1 Sampling Room**	**Location:**	45' Auxiliary Building
		Fire Area:	11
A426	**Passage**	**Location:**	45' Auxiliary Building
		Fire Area:	11
A428	**Unit 1 45' Piping Area**	**Location:**	45' Auxiliary Building
		Fire Area:	11
		CDF:	

The area addressed in this assessment covers seven adjoining areas (Unit 1, Unit 2, and common) of the 45' elevation of the Auxiliary Building.

Fire Analysis Results

Seven fire scenarios are identified for A419. All the scenarios are due to transient ignition sources. Three are conservatively assumed to result in core damage due to their low ignition frequency and large functional impact. The remaining four are represented by four initiating events.

Table 4-J-1
A419 Fire Analysis Results

Initiating Event	Fire Scenario	Frequency	Ignition Source	Functional Impact	CDF
A419F1	T1	3.92E-5	Compartment A408 Transient Refuse Fire	Failure of Unit 2 MFWS	1.12E-10
A419F2	T2	3.92E-5	Compartment A428 Transient Refuse Fire	Spurious MSIV Closure	8.41E-10
A419F3	T3	1.92E-7	Compartment A426 Unit 1 Passageway Transient Oil Fire with suppression	HR, QZ*, I2, RS*, RR, PS, PV*, SL, MN, HX*, UQ*, SG*, FC*, TF, TG, LF, CV, OT, FT*, RW, V2, MV, HB, V5, DL*, TW, Spurious SGIS, Various ESFAS Tops go to 1/3 Logic	1.52E-9
A419F4	T4	3.92E-9	Compartment A426 Unit 1 Passageway Transient Oil Fire without suppression	Assumed Core Damage	3.92E-9
A419F5	T5	1.92E-7	Compartment A426 Unit 2 Passageway Transient Oil Fire with suppression	GF, HH*, HL, NR*, NS, FO*, SH*	4.94E-13
A419F6	T6	3.92E-9	Compartment A426 Unit 2 Passageway Transient Oil Fire without suppression	Assumed Core Damage	3.92E-09
A419F7	T7	1.76E-7	Large Truck Fire	Assumed Core Damage	1.76E-07

*Note: The asterisk (*) indicates those top events which are impacted but not failed.

Fire Ignition Frequency

Fixed Ignition Frequency

The fixed ignition frequency is determined by starting with the compartment fixed ignition frequency results of Section 4.3.2 and then developing a scenario-specific ignition frequency as described in Section 4.3.4.3. The combined compartment areas contain motors, ventilation subsystems, transformers, and electrical cabinets.

The ventilation units are small fan motors. These include a window mounted air conditioner, sampling hood blowers, and hot water unit heater fans. These items were visually investigated and determined not to be fire scenarios of concern. These ignition sources will not damage any other adjacent equipment or cabling. As such, these ventilation units are appropriately dismissed as fire-induced core damage scenarios.

The transformers are small air-cooled units used for lighting and receptacles. The transformers were also visually investigated. These units also are not located close enough to combustibles or other equipment to damage them.

Seven motors were also investigated. Two of the motors are blowdown sampling pumps, one in each Piping Areas. The remaining motors, in the A419 Truck Bay and A426 Passage, are used with roll-up doors and hoist motors. No motors are located close enough to combustible or other significant equipment to damage them.

There are no fixed ignition scenarios which impact PRA functions in the combined area: Therefore, the fixed ignition sources and parameters are omitted in this analysis.

Transient Ignition Frequency

The transient ignition frequency is determined by starting with the compartment transient ignition frequency results of Section 4.3.2 and then developing a scenario-specific ignition frequency as described in Section 4.3.4.3. The sources and parameters used for A419 area are:

Compartment	Hot Work-Induced Cable Fires	Hot Work-Induced Transient Fires	Other Transients
A408	4.40E-5	2.67E-4	7.84E-5
A410	4.40E-5	2.67E-4	7.84E-5
A413	0	2.67E-4	7.84E-5
A419	4.40E-5	2.67E-4	7.84E-5
A424	0	2.67E-4	7.84E-5
A426	4.40E-5	2.67E-4	7.84E-5
A428	4.40E-5	2.67E-4	7.84E-5

Due to plant controls, the hot work-induced ignitions are dismissed from further analysis as legitimate fire growth and damage scenarios.

Because the transient fire characterization is not the same for each of the seven subcompartments, the scenario damage and frequency assessment for each is addressed separately below.

COMPARTMENT A410

This area is a corridor used for transit to other areas. Although the standard maintenance refuse fire is applicable to this area, other more conservative transient fires may also postulated, such as an oil spill (e.g., the Appendix R standard five-gallon oil transient). However, regardless of whether the fire is the less severe maintenance refuse fire or whether the fire is the five-gallon oil spill fire, there is no equipment in this corridor modeled in the PRA. Therefore, this corridor is dismissed from further analysis. Note that an oil spill fire that is close enough to the ends of the corridor to cause flames or very hot plume gases to shoot upward into the A426 passages, is subsumed by the transient oil fire assessment for the A426 passage.

Therefore, fire scenarios originating in the A410 corridor are dismissed from further evaluation as fire-induced core damage scenarios.

COMPARTMENT A413

The contents of this sampling compartment include typical sampling and chemistry items: steel benches, ventilation hoods, sink, beakers, etc. There are no other significant combustibles or targets in the compartments. The use of the standard refuse fire is judged applicable to characterize the transient fire for this compartment.

However, there is no equipment in this compartment that is modeled in the PRA. A postulated fire that damages everything in this compartment has no impact on the plant core damage frequency. In addition, a fire in this sampling compartment would not be large enough to damage equipment outside the compartment. There are no large volumes of combustible liquids stored in this compartment; a large fire that would propagate out of the compartment and damage cabling outside the compartment is judged not credible.

Therefore, fire scenarios originating in this compartment are dismissed from further evaluation as fire-induced core damage scenarios.

COMPARTMENT A424

The above assessment (for Compartment A413) applies to this sampling compartment as well. Fire scenarios originating in this compartment are dismissed from further evaluation as fire-induced core damage scenarios.

COMPARTMENT A408

The Unit 2 Piping Area is triangular shaped with generally no ignition sources or fire-induced damage targets of interest. Other than a small electric motor on the floor, all other ignition sources in these areas are insignificant (e.g., light bulbs) or contained within steel enclosures (e.g., sealed electrical cabinets).

This area is not an area that would be used for transit (it is a dead end). There is no major equipment in this area. As such, a transient oil spill fire is not applicable to this area. To bound the transient combustible fire scenario, the standard maintenance refuse fire is conservatively postulated in the compartment. This fire is characterized by a duration of fifteen minutes and a peak fire intensity of 100 btu/s.

The In-Plume Worksheet calculations show that the critical damage height to a target above the fire is 4.5 feet if the fire is in the open, six feet if the fire is up against a wall, and 7.9 feet if the fire occurs in a corner. The Radiant Exposure Worksheet calculation shows that the critical radial damage distance is less than two feet even when using a conservative 1.0 btu/s/ft^2 value for critical radiant flux. Given these damage distances and the configuration of equipment in the area, only a single "component" could be postulated to be damaged by the modeled fire.

Rather than calculating u, it is conservatively assumed here that $u = 0.5$. Effort is not expended here to refine this value as the impact of a transient fire in this compartment is not severe. Fire suppression is not credited in this analysis, therefore $P_{fs} = 1.0$, and:

$$A419F1 = F_t = 7.84E\text{-}5 * 0.5 * 1 = 3.92E\text{-}5$$

Therefore, transient fires for this area are modeled as a maintenance refuse fire. The only potential PRA target was 2C161, a main steam isolation valve control panel.

COMPARTMENT A428

The Unit 1 Piping Area is essentially a mirror image of A408 (above). The transient ignition frequency for this area is calculated in the same manner as area A408 (where $u = 0.5$ and $P_{fs} = 1.0$).

$$A419F2 = F_t = 7.84E\text{-}5 * 0.5 * 1 = 3.92E\text{-}5$$

Transient fires for this area are also modeled as a maintenance refuse fire. The only potential PRA target is 1C161, a main steam isolation valve control panel.

COMPARTMENT A426

The A426 Passage is a U-shaped area that surrounds the walls of the spent fuel pools and communicates with A419 Truck Bay and the piping areas (A408 and A428). The "legs" of the passage pass between the two containment walls and the spent fuel pool walls and connect to the Piping Areas. These legs contain cable trays passing horizontally in the ceiling.

This area is used for transit to other areas. Although the standard maintenance refuse fire is applicable to this area, other more conservative transient fires may also be postulated, such as an oil spill (the Appendix R standard five-gallon oil transient). Therefore, to conservatively bound the spectrum of postulated transient fires in the compartment, both the standard maintenance refuse fire and a severe five-gallon oil spill fire are addressed.

In-Plume Worksheet calculations show that the critical damage height to a target above the maintenance refuse fire is 4.5 feet if the fire is in the open, six feet if the fire is up against a wall, and 7.9 feet if the fire occurs in a corner. The Radiant Exposure Worksheet calculation shows that the critical radial damage distance is less than two feet even when using a conservative 1.0 btu/s/ft^2 value for critical radiant flux. Given these damage distances, the modeled maintenance refuse fire will not damage the overhead cable trays in this area. Therefore, this fire scenario is dismissed from further analysis as a fire-induced core damage scenario and the scenario frequency is not assessed here.

The modeled lube oil spill is associated with the transport of a lube oil container through the area. The storage of lube oil in the area in approved containers, though not allowed in this area (i.e., signs exist in the area that state not to store combustibles in the passage ways), should be dismissed from further consideration as a credible fire scenario per the FIVE Methodology. To acknowledge the transit/transport characteristic of this area and the large heat content of a transient combustible necessary to cause damage to overhead cable trays, an oil spill is postulated.

The oil spill is treated as a confined oil spill that lasts at least three minutes. The oil spill worksheet calculation results in a oil pool area of twenty-six square feet with a peak fire intensity of 2,984 btu/s. The oil spill is postulated to occur in either of the two "legs" of the A426 area between the containment wall and the spent fuel pool walls. The transient fire is postulated in these locations because cable trays are closest to the floor in these locations. In other locations in the A426 area, the damage to cable trays from the oil spill fire would either be less or no damage would be indicated.

The In-Plume and Radiant Exposure worksheet calculations for the postulated five-gallon oil spill fire show that cable trays approximately 18-1/2 feet above the floor will be damaged if the sprinkler system in the area does not actuate. Accounting for the sprinkler system and the transient thermal response of the cable insulation and the fusible links of the heads, approximately twenty-one feet above the floor in the "legs" of the A426 area, the Transient Analysis Worksheet calculation shows that if the sprinkler heads actuate they will prevent damage to cable trays above approximately thirteen feet above the floor. Therefore, if the sprinkler system does not actuate then cable trays up to 18-1/2 feet above the floor will be damaged. If the sprinkler system does actuate then cable trays up to only thirteen feet above the floor will be damaged.

The frequency of a five-gallon lube oil fire is discussed below (the remainder of the compartment transient ignition frequency is implicitly apportioned to the other less severe fires, including the standard maintenance refuse fire, which do not damage equipment modeled in the PRA). Again, rather than calculating u explicitly, it is conservatively assumed here that $u = 0.5$.

However, the FIVE Methodology and the EPRI Fire PRA Implementation Guide offer no guidance as to the estimation of the frequency of lube oil spill fires. To have a transient lube oil fire the transient ignition ("other" transient ignitions) must occur in proximity to and during the time when an oil spill exists (i.e., it has yet to be cleaned up). Such an unlikely scenario may be postulated as an activity on-going in the area at the time an oil spill occurs and results in ignition of the spilled oil (welding activity procedural controls should preclude the occurrence of such an event). Another unlikely scenario that may be postulated is that the lube oil spill occurs and comes into contact with a non-hot work transient ignition (e.g., hot pipe, extension cord short). Efforts may be made to calculate frequencies for a number of such postulated scenarios. However, it is judged more appropriate to simply assign a percentage of the transient ignition frequency that involves the modeled oil spill. This approach is consistent with the EPRI Fire PRA Implementation Guide which states:

> "...fire risk from transient ignition sources is almost always low because the frequency of a fire is low and compensatory measures, e.g., fire watch during welding, are often very effective. . . . [*transient fire risk modeling*] assumptions could be more easily made that might avoid significant evaluation time solely to evaluate unlikely sources of risk."

Therefore, it is assumed that one percent of the transient ignitions for this area involve ignition of an oil spill. This assumption is judged conservative given the fact that no incidents exist in the EPRI Fire Events Database for transient ignition of a significant quantity of oil, let alone an oil spill of multiple gallons.

The final variable to consider is fire suppression failure. The A426 area, as most locations in the Truck Bay and adjoining areas, is equipped with sprinkler heads in the ceiling. The sprinkler heads are approximately one foot below the ceiling in the "legs" of the A426 area (at a height of approximately 21 feet above the floor). Analysis shows that if the sprinkler heads actuate, damage to overhead cable trays will be limited to those trays approximately 13 feet above the floor and lower. If the sprinkler heads do not actuate, damage to cable trays approximately 18-1/2 feet above the floor and lower will occur.

The failure probability for the wet pipe sprinkler system in the area is taken from the FIVE Methodology and is estimated at 0.02.

Note that the A426 area contains two passageway "legs." Two scenarios may be postulated in either passageway leg with corresponding different damage states (i.e., cable trays damaged from the postulated fire). Therefore, the frequency of each scenario is divided in half. The unsuccessful suppression scenario frequency is:

$$F_t = (7.84\text{E-5} \ * \ 1\%) \ * \ 0.5 \ * \ 0.02 = 7.84\text{E-9}$$

And the oil spill fire without suppression frequency (damage limited to cable trays in the legs of the A426 area up to 18-1/2 feet above the floor) is:

A419F4 = A419F6 = 7.84E-9 ÷ 2 = 3.92E-9

And where and the suppression succeeds, the scenario frequency is:

$$F_t = (7.84\text{E-5} \ * \ 1\%) \ * \ 0.5 \ * \ (1-0.02) = 3.84\text{E-7}$$

Oil spill fire with suppression frequency (damage limited to cable trays in the legs of the A426 area up to thirteen feet above the floor) is:

A419F3 = A419F5 = 3.84E-7 ÷ 2 = 1.92E-7

COMPARTMENT A419

The A419 Truck Bay is a large, open, airy T-shaped area. The area has open communication with the A426 Passage. A steel roll-up door on the west wall opens to the outside of the building. The compartment is configured as one would expect a truck bay: Large open rectangular concrete floor area inside the roll-up door, tall ceiling, hoist in the ceiling, and bins of materials around the perimeter of the main floor area.

Due to the nature of the Truck Bay, numerous transient combustible materials are located in the area. A few approved trash cans (noted during the walkdown) are beneath overhead cable trays. However, this is not a fire concern because the trays are approximately twenty feet above.

Numerous bins in the main area of the Truck Bay generally contain Class A material (e.g., green waste, yellow waste, mop heads, rags, etc.). Although bin tops are indiscriminately open (i.e., some lids were noted open and some were noted closed), a postulated bin fire does not appear to be able to easily ignite adjacent bins. However, a transient ignition of one or more of these bins could be postulated and would be more severe than the standard maintenance refuse fire.

A truck fire may be postulated in this area as trucks are not prohibited from backing completely into the area. The most severe postulated fire is a large diesel fuel fire. Drains exist in the floor with local area sloping towards the drain that would aid in minimizing the spill area. In addition, the area is equipped with a wet pipe sprinkler system.

So, while the maintenance refuse transient fire is applicable to this area, other more severe transient fires may be postulated for this area. As such, the transient fire assessment for this area is bounded by modeling 1) a fire of a bin of Class A material, and 2) large diesel fuel fire associated with a transportation truck. The Class A bin fire is modeled as 1000 btu/s (approximately three times the intensity of the large trash can fire used in the FIVE Methodology, and ten times the intensity of the standard maintenance refuse fire) that lasts for 15 minutes. No attempt is made here to characterize the heat rate of the large truck fire; this assessment simply assumes that all equipment in the area is damaged if the sprinkler system fails to actuate.

The In-Plume Worksheet calculations show that the critical damage height to a cable target above the Class A bin fire is thirteen feet. However, there is no important equipment close enough to such a fire to be damaged. Overhead is concrete ceiling and conduits (i.e., no cable trays other intervening combustibles to transport a fire). Even if such a fire were to be postulated outside of the main floor area of the truck bay and placed under the cable trays in the ceiling of the "T" section of the area, damage to these trays would not occur as they are well above thirteen feet above the floor. These assessments do not take into account that the sprinkler system in the area would actuate and suppress the fire if the fire is severe enough. Therefore, this fire scenario is dismissed from further analysis as a fire-induced core damage scenario and the frequency of this scenario is not assessed here.

As stated before, no attempt is made here to specify the heat rate of the large truck fire. This analysis conservatively assumes that all equipment in the Truck Bay and adjoining areas is damaged if the sprinkler system fails to actuate. The frequency of the large truck fire (T_f) is discussed below.

$$T_f = [(\text{number of truck fires/year}) \div (\text{registered trucks/year})] * [\text{percent of year trucks are in area}]$$

Historical data from the National Fire Protection Association (NFPA) and the U.S. Department of Transportation (USDOT) is used here to estimate the frequency of large truck fires. The NFPA data provides annual averages for truck fire incidents. The USDOT data provides estimates of the number of registered trucks per year. To normalize this data to the exposure time in the Truck Bay, the number of hours per day that trucks are in the Truck Bay is estimated.

The NFPA data is categorized in a number of ways (e.g., number of fires, number of deaths, estimated property damage, type of ignition, area of fire origin). The fires of interest are the very large ones (i.e., those that involve a large amount of the engine fuel supply). Fires in the engine cavity or the cab are not of interest; the impact of such fires would be bounded by the effects of the Class A large bin fire, already shown not to be important in this area. Therefore, the NFPA data is scrutinized to determine which of the tabulated data refers to these types of large fires. In a tabularized summary by "Area of Origin," the NFPA provides the following annual averages:

Fuel Tank or Fuel Line Area	870 fires (2.0%)
Unclassified Areas	1,300 fires (3.0%)

In a separate tabularized summary by "Ignition Factor," the NFPA provides, in addition to others, the following annual averages:

Fuel Spilled or Released	1,010 fires (2.3%)
Unclassified Factors	790 fires (1.8%)

These categories are used to estimate the number of large truck fires. The above breakdowns indicate that the annual percentage of truck fires that are severe fires involving the fuel supply may be estimated at approximately 4-5% (about 1,800 to 2,300 such fires each year). The Unclassified totals are conservatively included here. Therefore, a value of 2,300 is used here to estimate the annual average of severe truck fires that would be capable of creating the damage distances of concern in this analysis (i.e., 30 feet or more).

The USDOT data shows that the annual average number of registered trucks over the years 1989-1993 (the same data period as the NFPA fire incident data above) is approximately 43,500,000 registered trucks/year.

The likelihood of the severe truck fire occurring in the Truck Bay is obtained by normalizing the above data to the percentage of time over the course of a year that a truck would be inside the Truck Bay. This exposure time is estimated by assuming that a truck is inside the Truck Bay an average of four hours per day ($^4/_{24} = {}^1/_6$) over the course of a year. As an annual average, $^1/_6$ is believed to be a conservative value.

Given the above information, the frequency of a truck fire large enough to create the extreme damage distances of concern in this area is estimated as follows:

T_f = [(number of truck fires/year) ÷ (registered trucks/year)] * [average hours/day trucks are in area]

$$= [(2300) \div (43{,}500{,}000)] * {}^1/_6 = 8.81\text{E-}6$$

The final variable to consider is fire suppression failure. The Truck Bay is equipped with sprinkler heads in the ceiling. Considering that the targets of interest are not directly above the floor area where a truck would be and that drains exist in the floor to minimize the spread of a pool fire, the sprinkler system would realistically extinguish or suppress the large truck fire, such that damage to important targets would not occur. The failure probability for the wet pipe sprinkler system in the area is taken from the FIVE Methodology and is estimated at 0.02.

Therefore, as the modeled Class A large bin fires (and other smaller fires such as trash fires, truck engine fires, truck cab fires, etc.) do not damage equipment modeled in the PRA, the postulated transient fire scenario for this area is an unsupervised large truck fuel fire that disables all PRA functions in the Truck Bay and adjoining areas with an occurrence frequency:

$$\text{A419F7} = F_t = 8.81\text{E-}6 * 0.02 = 1.76\text{E-}7$$

Fire Suppression

The fire detection and suppression configuration for the seven subcompartments in this area is as follows:

A408	wet pipe sprinklers and smoke detection
A410	partial wet pipe sprinkler coverage and smoke detection
A413	no suppression or detection
A419	wet pipe sprinklers and smoke detection
A424	no suppression or detection
A426	wet pipe sprinklers and smoke detection
A428	wet pipe sprinklers and smoke detection

Fire Suppression Induced Equipment Failures

Based on the approach described in Section 4.3.4.4.4, equipment failure due to the inadvertent actuation of the automatic fire suppression system is assumed not to occur. Cable and conduit and other PRA equipment in the room are not considered to be susceptible to water damage.

This general area is equipped with floor drains which are often centered in the downward-sloped cement floor areas. Flooding is very unlikely. There are few pieces of equipment in this area that are of PRA concern, and those that are (such as the local MSIV control panels, ADV controls) are inside-sealed enclosures or of a drip-proof design.

A423	**Unit 1 West Electrical Penetration Room**	**Location:**	45' Auxiliary Building
		Fire Area:	32
		CDF:	6.78E-8/yr

The Unit 1 45' West Electrical Penetration Room is located in the Auxiliary Building. The compartment is a rectangular area cut on two sides by the circular containment wall. The approximate compartment dimensions are fifty-nine feet long and a nominal width of twenty feet for 1,025 square feet of area. The ceiling height is approximately twenty-two feet for a compartment volume of 22,550 cubic feet. The compartment has a concrete floor, concrete walls, a concrete ceiling, and has both fire suppression and detection devices.

This compartment contains a motor control center (MCC104), two containment cooler fan starter cabinets, one hydrogen recombiner controller, and four PROTECTOWIRE fire cabinets. There are also miscellaneous wall mounted junction boxes, emergency lights, annunciation panels, and a panel mounted recorder. The compartment also serves as a connection location for numerous electrical penetrations into the containment.

A423 Fire Analysis Results

Eighty fire scenarios were identified for A423. Thirty-two are the result of fixed ignition sources and forty-eight are due to transient ignition sources. These scenarios are represented by four fire initiating events. The consolidation of fire scenarios is based on an assessment of the functional impact and ignition frequency of each scenario. The frequency of each initiator is the sum of the frequencies of all the fire scenarios it represents.

Table 4-K-1
A423 Fixed Ignition Fire Scenario Summary

Scenario	Fire Scenario Description	Trays and Panels Damaged by Fire
C1-30	MCC104 compartments and bus	No other effects
C31	Containment Air Cooler 12 Starter	No other effects
C32	Containment Air Cooler 14 Starter	No other effects

Table 4-K-2
A423 Transient Fire Scenario Summary

Scenario	Fire Scenario Description	Trays and Panels Damaged by Fire
T1-39	Transient Induced MCC104R Fires	No other effects
T40	Transient Induced CAC 12 Fires	No other effects
T41	Transient Induced CAC 14 Fires	No other effects
T47-48	Transient Fire Impacting Cable Trays	1AE70, 1AE72, 1AE76, 1AE77

Table 4-K-3
A423 Fire Analysis Results

Initiating Event	Fire Scenario	Frequency	Ignition Source	Functional Impact	CDF
A423F1	C1-30, T1-39	2.11E-3	Motor Control Center 104R and Transient Induced MCC104R Fires	M1, H6, XW	6.70E-8
A423F2	C31, T40	7.12E-5	Containment Air Cooler 12 Starter and Transient Induced CAC 12 Fires	XW, H6, WY*	2.48E-10
A423F3	C32, T41	7.12E-5	Containment Air Cooler 14 Starter and Transient Induced CAC 14 Fires	XW, H6, WY*	2.50E-10
A423F4	T47, T48	5.92E-6	Transient Fire Impacting Cable Trays	XW, QZ*, H6, RS*, PV*, IA*, IB*, SA*, SB*, EA*, EB*, RA*, RB*, PA*, PB*	2.49E-10

A423 Fire Ignition Frequency

Both fixed and transient ignition frequencies were determined for the Unit 1 West Electrical Penetration Room.

Fixed Ignition Frequency

The fixed ignition frequency is determined by starting with the compartment fixed ignition frequency results of Section 4.3.2 and then developing a scenario specific ignition frequency as described in Section 4.3.4.3. The sources and parameters used for A423 are:

Source	Generic Fire Frequency	Location Weighting	Source Weighting	Total Sources in Aux Building	Compartment Specific Frequency
Electrical Cabinets	1.5E-2	0.154	-	-	2.31E-3
Fire Protection Panels	2.4E-3	2	4	35	5.94E-4

The Electric Cabinet fire frequency is apportioned based on the total number of electrical cabinets in the compartment. Note: The hydrogen recombiner and fire protection panels are not considered for PRA modeling. However, these items are discussed later.

The electric cabinet fire frequency is apportioned based on the total number of electrical cabinets in the compartment.

A423 Panel Count

Equipment	Bus	$N_{cubicles}$	Count
MCC 104R	1	29	30
Containment Fan	-	-	2
H_2 Recombiner Controller	-	-	1
		Total	33

MOTOR CONTROL CENTER 104R

The MCC is an assembly of joined sectional cabinets with no ventilation openings, grills, or louvers. Each section contains smaller cubicles (except for the distribution panel section) in various configurations and sizes: Each cubicle is unique to an individual load. The typical cubicle houses a breaker, control transformer, load starter, ground sensor, current transformer, indication lights, and wiring. Below the cubicle, in a separate doored compartment, a terminal block connects the load power leads to the MCC. The cubicles are sealed from the external environment and any ignition within the cubicle would be contained.

The back of each vertical section forms a common (undivided) area with terminal blocks. These blocks connect field control wiring to the cubicle components. The wiring and blocks do not represent a typical ignition source, but any fires in this compartment would be contained within it.

Vertical bus drops are sandwiched between the front cubicles and rear compartment. This area is enclosed. The vertical bus connects to the horizontal main bus-work at the top of the section. The horizontal bus-work is closed in with sheet metal panels further covered by an access plate (both front and rear). Here too, the bus-work is sealed from the external environment and any ignition would be contained.

All wiring enters the MCC via conduit. The usual entry is a short (approximately twelve inch) nipple where the wire leaves an overhead cable tray.

Because of the sealed design, the MCC bus-work and individual cubicles are not considered a source of propagation. As an ignition source the MCC is analyzed at a 65 Btu heat release rate. No cable trays or equipment are in the damage range.

MCC ignition frequencies are developed by apportioning the compartment electrical cabinet ignition frequency and apportioning it among the MCCs as follows:

1. For each MCC, count the number of non-spare (in service) cubicles.
2. For each MCC, add one to that MCCs cubicle count in Step 1 to account for the MCC bus itself.
3. Sum the individual counts for the MCC from step 2 and other non-MCC panels (recombiner controller and containment fan starters)
4. Compute the individual panel ignition frequency or MCC bus ignition frequency by dividing the compartment electrical cabinet ignition frequency by the sum from step 3.
5. Eighty percent of individual cubicle ignitions are assumed to fail only the initiating cubicle load.
6. Twenty percent of individual cubicle ignitions are assumed to fail the entire MCC by propagating into the buswork, as would a buswork ignition.

Individual non-spare cubicle ignition frequency or MCC bus ignition frequency:

$$F_{\text{electrical panel}} = 2.31E\text{-}3 \div 33 = 7.00E\text{-}5$$

Individual cubicle ignitions that fail only the initiating cubicle load:

$$F_{\text{single cubicle}} = 80\% * F_{\text{electrical panel}} = 80\% * 7.00E\text{-}5 = 5.60E\text{-}5$$

Individual cubicle ignitions that fail the entire MCC and the MCC buswork ignition:

$$F_{\text{mcc}} = F_{\text{electrical panel}} + (N_{\text{cubicles}} * 20\% * F_{\text{electrical panel}})$$
$$= F_{\text{electrical panel}} * [1 + (N_{\text{cubicles}} * 20\%)]$$

For MCC 104, ignitions that fail the entire MCC equals:

$$F_{mcc104} = 7.00E\text{-}5 * [1 + (29 * 20\%)] = 4.76E\text{-}4$$

However, for modeling purposes all individual and MCC ignition scenarios are conservatively grouped together and fail the entire MCC. Thus:

$$A423F1_{Fix} = \Sigma\,(F_{single\ cubicle}) + F_{mcc104}$$
$$= (N_{cubicles} * F_{single\ cubicle}) + F_{mcc104} = (29 * 5.60E\text{-}5) + 4.76E\text{-}4$$
$$= 1.62E\text{-}3 + 4.76E\text{-}4 = 2.10E\text{-}3$$

CONTAINMENT COOLER FANS

Each cabinet houses various control devices (transformer, fuses, relays, indication), interconnecting wiring, three large motor starters, and connection blocks, all mounted on an internal metal panel. The cabinet is a sealed enclosure. As an ignition source, sealed enclosures have a 65 BTU heat release rate. The analyzed damage distance shows no other components or cable are within range.

Each cooler is a single panel, so the fixed ignition frequency is:

$$A423F2_{Fix} = A423F3_{Fix} = F_{electrical\ panel} = 2.31E\text{-}3 \div 33 = 7.00E\text{-}5$$

HYDROGEN RECOMBINER CONTROLLER

This is a metal enclosed cabinet with a large screened ventilation opening located about four feet up from the floor. Internally, the controller contains a transformer, circuit boards, and several capacitors. In addition, heat sinks, small capacitors, and various other small electronic components and terminal blocks are used.

The controller is analyzed as an open panel with an estimated 135 Btu/s heat release rate. The calculated in-plume damage distance (a wall location) is seven feet. The sole overhead cable tray (1AB84) is approximately twelve feet above the recombiner top. In addition, the radiant exposure screening distance at 1.0 Btu/s/ft^2 is approximately two feet. There are no targets within this distance.

The recombiner is not modeled as PRA equipment.

FIRE PROTECTION PANELS

The four "PROTECTOWIRE" fire protection panels are sealed enclosures with conduit entries. Each panel has a separate enclosure that houses a back-up battery. Because of the small internal components and sealed enclosure, these panels may be dismissed for fire modeling purposes. In addition, the fire panels are not modeled as PRA equipment.

Transient Ignition Frequency

The transient ignition frequency is determined by starting with the compartment transient ignition frequency results of Section 4.3.2 and then developing a scenario specific ignition frequency as described in Section 4.3.4.3. The sources and parameters used for A423 are:

Source	Generic Fire Frequency	Location Weighting	Source Weighting	Total Sources in Aux Building	Compartment Specific Frequency
Cable fires - welding	5.1E-3	2	1	232	4.40E-5
Transient fires - welding	3.1E-2	2	1	232	2.67E-4
Transient - other	1.3E-3	2	7	232	7.84E-5

* Calculations use the more accurate value 7.845E-5.

For a description of transients and transient fire analysis see Section 4.3.4.3. The transient fire is modeled as a maintenance refuse fire and therefore uses the "Transient - other" as the ignition frequency. The U-1 West Electrical Penetration Room has detection and water suppression: These are not credited in this analysis, therefore $P_{fs} = 1.0$.

The refuse fire In-Plume worksheets show the corner configuration plume screening distance is 7.9 feet: Wall configuration plume distance is six feet and compartment center plume distance is 4.6 feet. The postulated transient fire is three feet high. For the various compartment tray configurations there are no overhead targets within the transient plume and overhead cable damage is unlikely and A_s = zero.

The U-1 West Electrical Penetration Room floor area is 1,025 square feet. The area displaced by floor mounted cabinets and equipment is approximately 167 square feet. Therefore, the floor area where combustibles could be placed is:

$$A_F = 1{,}025\text{ft}^2 - 167\text{ft}^2 = 858\text{ft}^2$$

$$u = (A_s + A_{st}) / A_F$$

$$\text{so:} \quad u = (A_s + A_{st}) / 858\text{ft}^2$$

$$P_{fs} = 1$$

$$\text{And:} \quad F_t = F_{it} * u * P_{fs} = F_{it} * [(A_s + A_{st}) / A_F] * 1$$

$$= 7.845\text{E-}5 * [(A_s + A_{st}) / 858\text{ft}^2]$$

MOTOR CONTROL CENTER (MCC) 104 TRANSIENT

The "L" shaped MCC 104, mounted on a cement riser, sits in the center of the compartment. The MCC is constructed of multiple vertical sections with up to four individual compartments per section. For this analysis the transient fire, maintenance refuse, is placed within 1.8 feet of the MCC 104R. Motor control center transient induced damage is conservatively modeled assuming:

1. Entire vertical section(s) are damaged by the transient, and
2. The transient fire will overlap up to three vertical sections.

The vertical section modeling provides load specific transient ignition frequencies and resulted in thirty-nine transient scenarios. However, in the plant model, these scenarios were binned such that any transient effecting MCC104 failed the entire MCC. Therefore, the total floor area around the exposed MCC perimeter is used to compute the transient induced damage frequency.

The A_{sr} for MCC104 is estimated by summing a 1.8 foot width times the length of the adjacent perimeter for the number of exposed sides. For MCC104, $A_{sr} = 142ft^2$, $A_s = 0$, and:

$$F_{t(MCC104)} = 7.845E\text{-}5 * [(A_s + A_{sr}) / 858ft^2] = 7.845E\text{-}5 * [(0ft^2 + 142ft^2) / 858ft^2] = 1.30E\text{-}5$$

$$A423F1_{trans} = F_{t(MCC104)} = 1.30E\text{-}5$$

CONTAINMENT COOLERS

By extending a 1.8 foot path around exposed perimeter sides, each containment cooler cabinet has approximately 12.6 square feet of floor area from which the transient could cause damage. Again, there are no overhead targets. Therefore, for each cooler cabinet:

$$F_{t(Cooler)} = 7.845E\text{-}5 * [(A_s + A_{sr}) / 858ft^2] = 7.845E\text{-}5 * [(0ft^2 + 12.6ft^2) / 858ft^2] = 1.15E\text{-}6$$

$$A423F2_{trans} = A423F3_{trans} = F_{t(Cooler)} = 1.15E\text{-}6$$

FLOOR LEVEL CABLE TRAYS

The containment wall forms a large portion of the penetration room. The wall encases five tiers of penetration canisters. Each tier has a horizontal cable tray to route field wires to the canisters. The bottom tray is about three inches off the floor with a top cover of marinite. The bottom of the next tray up is covered in marinite. Vertical trays join to the horizontal trays at about the midpoint of their arc. The vertical trays are covered with marinite. Each tray is considered a transient target with 32.4 square feet (=A_{sr}) of exposing floor area.

$$F_{t(tray)} = 7.845E\text{-}5 * [(A_s + A_{sr}) / 858ft^2] = 7.845E\text{-}5 * [(32.4ft^2 + 0ft^2) / 858ft^2] = 2.96E\text{-}6$$

Both trays are binned into one scenario, therefore:

$$A423F4_{Transient\ Ignition} = 2 * F_{t(tray)} = 2 * 2.96E\text{-}6 = 5.92E\text{-}6$$

Total Ignition Frequency

The total ignition frequency is the sum of the appropriate fixed and transient scenarios.

$$A423F1 = A423F1_{fix} + A423F1_{trans} = 2.10E\text{-}3 + 1.30E\text{-}5 = 2.11E\text{-}3$$

$$A423F1 = A423F2_{fix} + A423F2_{trans} = 7.00E\text{-}5 + 1.15E\text{-}6 = 7.12E\text{-}5$$

$$A423F1 = A423F3_{fix} + A423F3_{trans} = 7.00E\text{-}5 + 1.15E\text{-}6 = 7.12E\text{-}5$$

$$A423F4 = A423F4_{\text{Transient Ignition}} = 5.92E\text{-}6$$

Suppression Systems

The 45' West Penetration Room is equipped with fire detection and water suppression. The compartment modeling did not require an analysis of the suppression response, and takes no credit for fire suppression.

Fire Suppression Induced Equipment Failures

Based on the approach described in Section 4.3.4.4.4, equipment failure due to the inadvertent actuation of the automatic fire suppression system is assumed not to occur. Cable and conduit, pumps and other PRA equipment in the room are not considered to be susceptible to water damage. In the event of a suppression actuation, cables in the compartment are not affected by water. The penetration canisters and the cable connections to the canisters are environmentally qualified.

The MCC, containment cooling fan cabinets, and hydrogen recombiner are mounted approximately four inches up from the floor which has two drains, therefore, flooding is not likely. Water intrusion into the MCC is unlikely because the conduit entries are sealed with putty. Similarly, the cooling fan cabinets use full length conduit (not nipple entries). The various wall mounted equipment and transmitters are also well sealed. The hydrogen recombiner, with its large screened side openings, is subject to suppression failure, however this is not PRA modeled equipment.

A429	Unit 1 East Electrical Penetration Room	Location:	45' Auxiliary Building
		Fire Area:	33
		CDF:	7.10E-8/yr

The Unit 1 45' East Electrical Penetration Room is located in the Auxiliary Building. The compartment is a irregular area with five sides cut by the circular containment wall. The nominal compartment dimensions are sixty feet long and a nominal width of eighteen feet and 1,080 square feet of area. The ceiling height is approximately twenty-two feet for a compartment volume of 24,300 cubic feet. The compartment has a concrete floor, concrete walls, a concrete ceiling, and has both fire suppression and detection devices.

This compartment contains a pressurizer heater controller with a small motor control center (MCC12P), two containment cooler fan starter cabinets, one hydrogen recombiner controller, and four PROTECTOWIRE fire cabinets. There are also miscellaneous wall mounted junction boxes, emergency lights, annunciation panels, a panel with CRT video displays, and a panel mounted recorder. The compartment serves primarily as a connection point for numerous electrical penetrations into the containment.

A429 Fire Analysis Results

Sixteen fire scenarios were identified for A429. Nine are the result of fixed ignition sources and seven are due to transient ignition sources. All but two scenarios are screened due to low functional impact. The screening is based on the same criteria described in Section 4.3.1.3. The remaining two scenarios are represented by one initiating event. The consolidation is based on an assessment of the functional impact and ignition frequency of each scenario. The frequency of the initiator is the sum of the frequencies of the two fire scenarios it represents.

Table 4-L-1
A429 Fixed Ignition Fire Scenario Summary

Scenario	Fire Scenario Description	Trays Damaged by Fire
C1	12 Hydrogen Recombiner Controller	1AE83, 1AE82, 1AE07, 1AE08

Table 4-L-2
A429 Transient Fire Scenario Summary

Scenario	Fire Scenario Description	Trays Damaged by Fire
T6	Transient Fire Impacting Cable Trays	1AE82, 1AE83

Table 4-L-3
A429 Fire Analysis Results

Initiating Event	Fire Scenario	Frequency	Ignition Source	Functional Impact	CDF
A429F1	C1, T6	2.84E-4	H2 Recombiner Control Cabinet Fire with cable tray impact Transient Fire Impacting Cable Trays 1AE82, 1AE83	QZ*, RS*, PV*, IA*, IB*, HX*, UQ*, SA*, SB*, EA*, EB*, WY*, SG*, PA*, PB*	7.10E-8

A429 Fire Ignition Frequency

Both fixed and transient ignition frequencies were determined for the Unit 1 East Electrical Penetration Room.

Fixed Ignition Frequency

The fixed ignition frequency is determined by starting with the compartment fixed ignition frequency results of Section 4.3.2 and then developing a scenario specific ignition frequency as described in Section 4.3.4.3. The sources and parameters used for A429 are:

Source	Generic Fire Frequency	Location Weighting	Source Weighting	Total Sources in Aux Building	Compartment Specific Frequency
Electrical Cabinets	1.9E-2	2	7	135	1.97E-3
Fire Protection Panels	2.4E-3	2	4	35	5.94E-4

ITEMS NOT CONSIDERED

The pressurizer heater controller, pressurizer heater MCC, containment cooling fan cabinets, annunciator and video panel are all apportioned an ignition frequency. None of these items damaged other targets within the compartment. Except for the containment coolers, these items are not PRA modeled equipment: The containment cooler ignition frequencies are low enough to screen for plant impact.

The "PROTECTOWIRE" fire protection panels are sealed enclosures with conduit entries. Each panel has a separate enclosure that houses a back-up battery. Because of the small internal components and sealed enclosure, these panels are dismissed for fire modeling purposes.

HYDROGEN RECOMBINER CONTROLLER

A hydrogen recombiner controller, mounted to a four inch cement riser, also sits against the wall. The controller is approximately three feet by two feet and five feet tall. The south panel has louvers and screen vents; this panel is essentially open. The west panel has a single screen vent. About five feet overhead and two feet south of the recombiner is a cable tray stack.

This is a metal enclosed cabinet with a large screened ventilation opening located about four feet up from the floor. Internally, the controller contains a transformer, circuit boards, and several capacitors. In addition, heat sinks, small capacitors, and various other small electronic components and terminal blocks are used.

The controller is analyzed as an open panel with an estimated 135 Btu/s heat release rate. The calculated in-plume damage distance (a wall location) is seven feet. The sole overhead cable tray (1AB84) is approximately twelve feet above the recombiner top. In addition, the radiant exposure screening distance at 1.0 Btu/s/ft^2 is approximately two feet.

The recombiner is not modeled as PRA equipment; however in the A429 compartment configuration, cable tray targets travel overhead of the recombiner in the plume region.

An ignition frequency is developed for the H_2 Recombiner Controller by apportioning the compartment electrical cabinet ignition frequency to the total number of cabinets in the compartment.

A429 Panel Count

Equipment	Count
PZR Heater MCC	1
PZR Heater Controller	1
Containment Fan	2
H_2 Recombiner Controller	1
Annunciator	1
Video Panel	1
Total	**7**

$$A429F1_{fix} = F_{electrical\ panel} = 1.97E\text{-}3 \div 7 = 2.81E\text{-}4$$

<u>Transient Ignition Frequency</u>

The transient ignition frequency is determined by starting with the compartment transient ignition frequency results of Section 4.3.2 and then developing a scenario specific ignition frequency as described in Section 4.3.4.3. The sources and parameters used for A429 are:

Source	Generic Fire Frequency	Location Weighting	Source Weighting	Total Sources in Aux Building	Compartment Specific Frequency
Cable fires - welding	5.1E-3	2	1	232	4.40E-5
Transient fires - welding	3.1E-2	2	1	232	2.67E-4
Transient - other	1.3E-3	2	7	232	7.84E-5

* Calculations use the more accurate value 7.845E-5

For a description of transients and transient fire analysis see Section 4.3.4.3. The transient fire is modeled as a maintenance refuse fire and therefore uses the "Transient - other" as the ignition frequency. The Unit 1 East Electrical Penetration Room has detection and water suppression: These are not credited in this analysis, therefore $P_{fs} = 1.0$.

The Unit 1 East Electrical Penetration Room floor area is 1,080 square feet. The area displaced by floor mounted cabinets and equipment is approximately 112 square feet. Therefore, the floor area where combustibles could be placed is:

$$A_F = 1{,}080\text{ft}^2 - 112\text{ft}^2 = 968\text{ft}^2$$

$$u = (A_s + A_{sr}) / A_F$$

$$\text{so: } u = (A_s + A_{sr}) / 968\text{ft}^2$$

$$P_{fs} = 1$$

$$\text{And: } F_t = F_{it} * u * P_{fs} = F_{it} * [(A_s + A_{sr}) / A_F] * 1$$

$$= 7.845\text{E-}4 * [(A_s + A_{sr}) / 968\text{ft}^2]$$

CABLE TRAYS

The refuse fire In-Plume worksheets show the corner configuration plume screening distance is eight feet: Wall configuration plume distance is 6.1 feet and compartment center plume distance is 4.6 feet. The postulated transient fire is three feet high. For the various compartment cable tray configurations there are no overhead targets within the transient plume and overhead cable damage is unlikely and A_s = zero.

The containment wall forms a large portion of the penetration compartment. The wall encases five tiers of penetration canisters. Each tier has a horizontal cable tray to route field wires to the canisters. The bottom tray is about three inches off the floor with a top cover of marinite. The bottom of the next tray up is covered in marinite. Vertical trays join to the horizontal trays at about the midpoint of their arc. The vertical trays are covered with marinite. Each tray is considered a transient target with 32.4 square feet (=A_{sr}) of exposing floor area.

$$\boxed{A429F1_{\text{Transient}} = 7.845\text{E-}4 * [(A_s + A_{sr}) / 968\text{ft}^2] = 7.845\text{E-}4 * [(0\text{ft}^2 + 32.4\text{ft}^2) / 968\text{ft}^2] = 2.63\text{E-}6}$$

<u>Total Ignition Frequency</u>

The total ignition frequency is the sum of the appropriate fixed and transient scenarios. In this case, the sole scenario involves cable damage.

$$\boxed{A429F1 = A429F1_{\text{fix}} + A429F1_{\text{trans}} = 2.81\text{E-}4 + 2.63\text{E-}6 = 2.84\text{E-}4}$$

Suppression Systems

The 45' East Penetration Room is equipped with fire detection and water suppression. The compartment modeling did not require an analysis of the suppression response, and takes no credit for fire suppression.

Fire Suppression Induced Equipment Failures

Based on the approach described in Section 4.3.4.4.4, equipment failure due to the inadvertent actuation of the automatic fire suppression system is assumed not to occur. Cable and conduit, pumps and other PRA equipment in the room are not considered to be susceptible to water damage. In the event of a suppression actuation, cables in the compartment are not affected by water. The penetration canisters and the cable connections to the canisters are environmentally qualified.

The MCC, containment cooling fan cabinets, and hydrogen recombiner are mounted approximately four inches up from the floor, therefore, flooding damage is not likely. Water intrusion into any of the compartment components (except for the hydrogen recombiner) is unlikely because of the sealed or louvered design and conduit entries. However, because of the large open grills, the hydrogen recombiner is subject to suppression failure.

A512	**Control Room HVAC Room**	**Location:** **Fire Area:** **CDF:**	69' Auxiliary Building 11 1.91E-9
A520	**Spent Fuel Vent Room**	**Location:** **Fire Area:** **CDF:**	69' Auxiliary Building 11 Screened - Low Fire Frequency
A524	**Unit 1 Main Vent Fan Room**	**Location:** **Fire Area:** **CDF:**	69' Auxiliary Building 11 2.37E-8
A525	**Unit 1 Containment Access Area**	**Location:** **Fire Area:** **CDF:**	69' Auxiliary Building 11 Screened - Low Fire Frequency

Control Room HVAC Room (A512)

The Control Room HVAC Room is located on the 69' level of the Auxiliary Building. The Control Room HVAC Room is approximately 47.5 feet long and thirty-nine feet wide for an area of 1,852 square feet. The height of the compartment is twenty-two feet for a total compartment volume of 40,755 cubic feet. The compartment has concrete floor, walls and ceiling.

The compartment contains area-wide smoke detection devices, but there is no fire suppression system.

Spent Fuel Vent Room (A520)

The Spent Fuel Vent Room is located on the 69' level of the Auxiliary Building. The compartment is thirty-one feet long and thirty-one wide for an area of 961 square feet. The ceiling is twenty-two feet high for a total compartment volume of 21,142 cubic feet. The primary ignition sources contained in this compartment are the Fuel Pool Exhaust Fan Motors 11 and 12. The other major component, which occupies most of the compartment, is the Spent Fuel Pool Exhaust Filter Unit 11. This compartment is designed to provide a supply and exhaust of tempered air during normal operations and to filter radioactive iodine from the air.

The compartment contains smoke detection devices, located directly over the Exhaust Filters, but there is no fire suppression system. There is also no overhead cable in the compartment to serve as potential fire target.

Plant Exhaust and Equipment Room (A524)

The Main Vent Fan Room is located on the 69' level of the Auxiliary Building. This compartment is relatively crowded but most of the items in the compartment have little or no fire impact. The compartment is seventy-six feet long and forty-six feet wide with a total compartment area of approximately 3,127 square feet (after subtracting the area for Cable Chase 1A). The compartment height is twenty-two feet for a total compartment volume of 70,357 cubic feet. The floor is concrete with a smooth surface. The north-east area is divided approximately halfway up by an open grate platform that supports HVAC equipment.

Containment Access (A525)

The Containment Access area is approximately thirty-five feet long and thirty-seven feet wide with an area of 1,305 square feet. The ceiling height is twenty-two feet high for a total compartment volume of 22,710 cubic feet. (This compartment volume does not include the 6000 cubic feet volume of the horizontal cable chase.)

The compartment contains smoke detection devices, but there is no fire suppression system.

A512 Area Fire Analysis Results

Twenty fire scenarios are identified for A512 and twenty-three for A524. Compartments A520 and A525 are screened due to low fire frequency.

For A512, eighteen scenarios are the result of fixed ignition sources and two are due to transient ignition sources. Seven scenarios are screened due to low functional impact. The screening is based on the criteria described in Section 4.3.1.3. The remaining thirteen scenarios are represented by two fire initiating events. The consolidation of fire scenarios is based on an assessment of the functional impact and ignition frequency of each scenario. The frequency of each initiator is the sum of the frequencies of all the fire scenarios it represents.

For A524, twenty-one scenarios are the result of fixed ignition sources and one is due to transient ignition sources. Thirteen scenarios are screened due to low functional impact. The screening is also based on the criteria described in Section 4.3.1.3. The remaining ten scenarios are represented by three fire initiating events.

Table 4-M-1
A512 Fire Scenario Summary

Scenario	Fire Scenario Description	Equipment, Trays and Conduit Damaged
1	CR A/C Unit 11	0HXHVACCR11, 1M1435, 0FANHVACCRS11
2	CR A/C Unit 12	0HXHVACCR12, 2M0410, 0FANHVACCRS12
3	A/C Compressor 11 Small Oil Fire	0COMPCRA/CCOMPR11, 1MB108, 1PNL1NB108, 1A2334, 1A2329, 1AK02, 1A2329, 1A2328, 1A1740, 1A0447, 1AK93, 1A5194, 1A5193, 1A2335, 1A5192, 1A3368
4	A/C Compressor 11 Large Oil Fire	0COMPCRA/CCOMPR11, 1MB108, 1PNL1NB108, 1A2334, 1A2329, 1AK02, 1A2329, 1A2328, 1A1740, 1A0447, 1AK93, 1A5194, 1A5193, 1A2335, 1A5192, 1A3368, 1A1699, 1A1691, 1AK01, 1M1435, 0HXHVACCR11, 1M1435

Table 4-M-1
A512 Fire Scenario Summary (continued)

Scenario	Fire Scenario Description	Equipment, Trays and Conduit Damaged
5	A/C Compressor 11 Motor Fire	0COMPCRA/CCOMPR11, 1MB108, 1PNL1NB108
6	A/C Compressor 12 Small	0COMPCRA/CCOMPR12, 2MB408, 2PNL2NB408, 2A1738, 2A5139, 2A2870, 2A1741, 2A5140, 2A5141, 2A1740, 1A1741, 1A9033, 1A2910, 2A1739
7	A/C Compressor 12 Large Fire	0COMPCRA/CCOMPR12, 2MB408, 2PNL2NB408, 2A1738, 2A5139, 2A2870, 2A1741, 2A5140, 2A5141, 2A1740, 1A1741, 1A9033, 1A2910, 2A1739, 1A1741, 2A1332, 0HXHVACCR12, 2M0410
8	A/C Compressor 12 Motor Fire	0COMPCRA/CCOMPR12, 2MB408, 2PNL2NB408
9	Return Air Fan 11	0FANHVACCRC11, 1M1465
10	Return Air Fan 12	0FANHVACCRC12, 2M0433
11	Post LOCI Filter Fan 11	0FANHVACPL11, 1M1447
12	Post LOCI Filter Fan 12	0FANHVACPL12, 2M0447
13	Access Control A/C 14	0FAN7350, 1CHL1P74
14	Access Control HVAC 11	1M0207
15	CR HVAC Equipment Room HVAC 11	1PNL1N0227, 1M0227, 0FAN7330
16	Smoke Removal Fan	0FAN5346, 0DAMP5361, 1M0223
17	Equipment Room Supply Fan	1M0224
18	Area Heater	1UP1532
19	A512 Transient Fire (Welding)	NO IMPACT - Hot Work Permit and Fire Watch
20	A512 Transient Fire (Maintenance Refuse)	1PNL1NB108, 0COMPCRA/CCOMPR11

Table 4-M-2
A524 Fire Scenario Summary

Scenario	Fire Scenario Description	Equipment, Trays and Conduit Damaged
1	Small Oil Spill for SWGR RM AC Compressor 11	1 HVAC/A SWGR RM A/C COMP, HVAC/A SWGR RM A/C Compressor, SWGR RM AC COMPR 11 CNTL
2	Large Oil Spill for SWGR RM AC Compressor 11	All Components in Block and 1COMP5431, 1MB410, 1PNL1NP0428, 1M1436, 1M0436; also damages 12 SRW Head Tank Level Instruments 1LT1565 & 1LS1565
3	Motor Fire for SWGR RM AC Compressor 11	1 HVAC/A SWGR RM A/C Compressor, HVAC/A SWGR RM A/C Compressor, SWGR RM AC Compressor 11 Controller
4	SWGR RM HVAC Unit 11 Fan	No Other Effect
5	– N/A –	< This scenario is not used >
6	Small Oil Spill for SWGR RM AC Compressor 12	1 HVAC/A SWGR RM A/C Compressor, HVAC/A SWGR RM A/C Compressor, SWGR RM AC Compressor 12 Controller
7	Large Oil Spill for SWGR RM AC Compressor 12	All Components in Block and 1COMP5427, 1MB110, 1PNL1NP1428, 1M0436, 1M1436; 11 AND 12 SRW Head Tank Level Instruments 1LT1565, 1LS1565, 1LT1579, AND 1LS1579
8	Motor Fire for SWGR RM AC Compressor 12	1 HVAC/A SWGR RM A/C Compressor, HVAC/A SWGR RM A/C Compressor, SWGR RM AC Compressor 12 Controller
9	SWGR Room HVAC Unit 12 Fan	No Other Effect
10	44UH AUXILIARY BLDG	No Other Effect
11	45UH AUXILIARY BLDG	No Other Effect
12	1 HVAC/A Main Exhaust Fan 11	No Other Effect
13	1 HVAC/A Main Exhaust Fan 12	No Other Effect
14	0 HVAC/C Battery Room Supply Fan	No Other Effect
15	Radioactive Gas Monitor 1N0528	No Other Effect
16	Radioactive Gas Monitor 1N0529	No Other Effect
17	Radioactive Gas Monitor 1N0530	No Other Effect
18	Radioactive Gas Monitor 1N0531	No Other Effect
19	Main Vent Gas Monitor	No Other Effect

Table 4-M-2
A524 Fire Scenario Summary (Continued)

20	Control Room Vent Rad Monitor/Radioactive Gas Monitor	No Other Effect
21	Particulate and Iodine Mid/High Range Sample Flow	No Other Effect
22	MID/HIGH RANGE SAMPLE FLO	No Other Effect
23	Transient	Worst Case Fire Damages HVAC Control Panel Failing 1COMP5431, 1MB410, 1PNL1NP0428

Table 4-M-3
A512/A524 Fire Analysis Results

Initiating Event	Fire Scenario	Frequency	Ignition Source	Functional Impact	CDF
A512F1	1, 3, 4, 5, 9, 20	1.93E-4	CR A/C Unit 11: fan, compressor oil fire, compressor motor, Return Fan 11, Supply Fan 11 and transient induced fire	HH*, M2*, N1*	1.02E-9
A512F2	2, 6, 7, 8, 10	1.72E-4	CR A/C Unit 12: fan, compressor oil fire, compressor motor, Return Fan 12, and Supply Fan 12	HH*, M3*, N7*	8.88E-10
A524F1	1, 3, 4	1.10E-4	SWGR HVAC 11: compressor small oil, compressor motor, fan	HS*, CB, CA, GW*, GZ*	9.11E-9
A524F2	2, 7	1.03E-5	SWGR HVAC 11 or 12: compressor large oil	HS, CA, CB, GW*	4.60E-9
A524F3	6, 8, 9, 23	1.19E-4	SWGR HVAC 12: compressor small oil, compressor motor, fan and transient induced fire	HS*, CB, CA, GW*	9.95E-9

Fixed Ignition Frequency

Ignition sources in compartments A512 and A524 include the main vent fans, switchgear HVAC equipment, transient combustible fires, and other miscellaneous items. The development of fire scenarios utilized the below information.

Sealed Cabinets

Totally enclosed cabinets with no ventilation openings, grills, or louvers are classified as sealed. For these cabinets, a fire will not propagate beyond the contents of the cabinet itself because combustion products suppress fire growth. The switchgear HVAC local controller and main vent RMS package are analyzed as sealed cabinets. None of these components cause collateral damage. Therefore, all fire scenarios with sealed cabinets will fail the cabinet only and are screened.

Motors

These include motors used with the following equipment: Radiation monitoring skids, the battery compartment ventilation unit, the Switchgear HVAC blowers, the Switchgear and Control Room HVAC refrigeration compressors, main vent fans, and unit heaters. Each motor has a 65 Btu release rate and is treated as a radiant source. None of these components cause collateral damage.

HVAC Compressors

These are a source of oil. The site "Controlled Oil Log" shows the Control Room and Switchgear HVAC contain four gallons of oil. This analysis uses the largest plausible oil spill as a large spill and uses twenty-five percent of the large spill volume as a representative small spill volume. The proposed large spill contains three gallons of oil (one gallon is spread throughout the system or in the overhead separation tank) at or near the compressor. The large spill height is assumed to be three feet because a pressurized oil leak might spray onto other equipment at this height. The spill is confined to 100 square feet because of the floor's porosity and irregularities.

The small spill is three quarts. The spill could occur near the compressor or anywhere along the freon piping. The spill is arbitrarily confined to four square feet as a realistic representation.

The fixed ignition frequency is determined by starting with the compartment fixed ignition frequency results of Section 4.3.2 and then developing a scenario specific ignition frequency as described in Section 4.3.4.3. The sources and parameters used for A512 and A524 are:

Source	Generic Fire Frequency	Location Weighting	Source Weighting	Total Sources in Auxiliary Bldg	Compartment Specific Frequency
A512 Ventilation Subsystems	9.5E-3	2	14	331	8.04E-4
A524 Ventilation Subsystems	9.5E-3	2	9	331	5.17E-4

By apportioning the Compartment Specific Frequency among the ventilation subsystems (VSS), each subsystem ignition frequency is:

$$A512F_{vss} = 8.04E\text{-}4 \div 14 = A524F_{vss} = 5.17E\text{-}4 \div 9 = 5.74E\text{-}5$$

The fire ignition frequency for devices with oil (compressors and pumps) does not indicate that each ignition results in a large fully developed fire if not suppressed. Review of the EPRI Fire Events Database shows that approximately fifty percent of pump fire events involve lubricating oil fires and fifty percent represent motor winding fires or other miscellaneous ignitions. The Fire PRA Implementation Guide indicates that approximately eighty-two percent of oil spill fires are "small" spills and eighteen percent are "large" spills. This information is used to calculate scenario frequencies.

CR A/C Unit 11: Compressor oil fire, compressor motor, Return Fan 11, Supply Fan 11

$$A512F1_{Fix} = [5.74E\text{-}5 * 50\% * (82\% + 18\%)] + (5.74E\text{-}5 * 50\%) + 5.74E\text{-}5 + 5.74E\text{-}5 = 1.72E\text{-}4$$

CR A/C Unit 12: Compressor oil fire, compressor motor, Return Fan 12, and Supply Fan 12

$$A512F2 = [5.74E\text{-}5 * 50\% * (82\% + 18\%)] + (5.74E\text{-}5 * 50\%) + 5.74E\text{-}5 + 5.74E\text{-}5 = 1.72E\text{-}4$$

SWGR HVAC 11: Compressor small oil, compressor motor, fan

$$A524F1 = (5.74E\text{-}5 * 50\% * 82\%) + (5.74E\text{-}5 * 50\%) + 5.74E\text{-}5 = 1.10E\text{-}4$$

SWGR HVAC 11 or 12: Compressor large oil

$$A524F2 = (5.74E\text{-}5 * 50\% * 18\%) + (5.74E\text{-}5 * 50\% * 18\%) = 1.03E\text{-}5$$

SWGR HVAC 12: Compressor small oil, compressor motor, and fan

$$A524F3_{\text{Fix Ignition}} = (5.74E\text{-}5 * 50\% * 82\%) + (5.74E\text{-}5 * 50\%) + 5.74E\text{-}5 = 1.10E\text{-}4$$

Transient Ignition Frequency

The transient ignition frequency is determined by starting with the compartment transient ignition frequency results of Section 4.3.2 and then developing a scenario-specific ignition frequency as described in Section 4.3.4.3. The sources and parameters used for A512 and 524 are:

Source	Generic Fire Frequency	Location Weighting	Source Weighting	Total Sources in Auxiliary Bldg	Compartment Specific Frequency
A512 Transient fires - welding	3.1E-2	2	1	232	2.67E-4
A512 Transient - other	1.3E-3	2	7	232	7.84E-5
A524 Transient fires - welding	3.1E-2	2	1	232	2.67E-4
A524 Transient - other	1.3E-3	2	7	232	7.84E-5

The transient fire is modeled as a maintenance refuse fire and therefore uses the "Transient - other" as the ignition frequency. Fire suppression is not credited in this analysis. Therefore $P_{fs} = 1.0$.

CR HVAC Compressor 11 Transient

The floor area of the Control Room HVAC Room is approximately 1,835 square feet. Floor mounted equipment and interferences occupy approximately 1,017 square feet of floor area. Therefore:

$$A_F = 1{,}852\text{ft}^2 - 1{,}017\text{ft}^2 = 835\text{ft}^2$$

The formula for calculating the damage frequency due to transient combustible fires is contained in Section 4.3.4.3.2. Based on the In-Plume exposure worksheets for transient fires in this compartment, the critical damage distance is 4.5 feet. The control panels at the compressors would be damaged by a transient fire, thus failing the refrigeration units. There are no cable trays in the compartment and no conduits low enough to become targets. Therefore, A_s equals four square feet.

The critical damage distance (approximately two feet) for radiant exposure is extended around the CR HVAC unit to calculate the floor area where a transient fire could cause damage. A_{sr} is approximately 215 square feet for the accessible areas around the HVAC motor-compressor units and blower units. So:

$$u = (A_s + A_{sr})/A_F = A_{CR\ HVAC}/A_F$$
$$= (4ft^2 + 215ft^2) / 835ft^2 = 0.2623$$

$$A512F1_{Transient} = F_t = 7.84E\text{-}05 * 0.2623 * 1 = 2.06E\text{-}05$$

SWGR HVAC Unit 12 Transient

The floor area of the Main Vent Fan Room is approximately 3,127 square feet. Floor mounted equipment and interferences occupy approximately 315 square feet of floor area. Therefore:

$$A_F = 3{,}127ft^2 - 315ft^2 = 2{,}812ft^2$$

Based on the In-Plume Exposure worksheets for transient fires in this compartment, the Critical Damage Distance is 4.5 feet. The control panels at the switchgear HVAC refrigeration units would be damaged by a transient fire, thus failing the refrigeration (not the fan however). There are no cable trays in the compartment, and no conduits low enough to become targets. Therefore, A_s is four square feet.

The critical damage distance (approximately two feet) for radiant exposure is extended around targets to calculate the floor area where a transient fire could cause damage. A_{sr} is conservatively 315 square feet for the accessible areas around the switchgear HVAC motor-compressor units and blower units. For this analysis the Service Water Head Tanks are not damaged by the fire. Therefore:

$$u = (A_s + A_{sr}) / A_F = A_{SGR\ HVAC}/A_F$$
$$u = (4\ ft^2 + 315ft^2) / 2{,}812\ ft^2 = 0.1134$$

$$A524F3_{Transient} = F_t = 7.84E\text{-}05 * 0.1134 * 1.0 = 8.89E\text{-}06$$

Total Ignition Frequency

The total ignition frequency is the sum of the appropriate fixed and transient scenarios.

$$A512F1 = A512F1_{Fix\ Ignition} + A512F1_{Transient} = 1.72E\text{-}4 + 2.06E\text{-}5 = 1.93E\text{-}4$$

$$A524F3 = A524F3_{\text{Fix Ignition}} + A524F3_{\text{Transient}} = 1.10E\text{-}4 + 8.89E\text{-}6 = 1.19E\text{-}4$$

A520 and A525 **Screening**

There are no significant ignition sources in the Spent Fuel Vent Room (A520) or the Unit 1 Containment Access (A525). Propagation of a fire from these compartments to other compartments in this grouping will not occur. Therefore, the ignition frequency of A520 and A525 is determined to be zero.

Fire Suppression

The Control Room HVAC contains area wide smoke detection devices, but there is no fire suppression system. The Spent Fuel Vent Room contains smoke detection devices, located directly over the Exhaust Filters, but there is no fire suppression system. There is also no overhead cable in the compartment to serve as potential fire target. The Containment Access and Plant Exhaust areas contains smoke detection devices, but there is no fire suppression system.

Fire Suppression Induced Equipment Failures

Based on the approach described in Section 4.3.4.4.4, equipment failure due to the inadvertent actuation of the automatic fire suppression system is assumed not to occur. Cable and conduit and other PRA equipment in the room are not considered to be susceptible to water damage. In the event of a suppression actuation, the motors are of a design that prevents water spray intrusion and their connections and wiring also resist water intrusion. In addition, cabling in the room is not affected by water spray.

A529 Unit 1 69' West Electrical Room

Location: 69' Auxiliary Building
Fire Area: 37
CDF: 9.18E-8/yr

The Unit 1 West Electrical Room is located on the 69' elevation of the Auxiliary Building. The room is roughly rectangular with north and east walls cut by the circular containment wall. The nominal room dimensions are sixty-five feet long and twenty-eight feet wide with 1,848 square feet of area. The ceiling height is approximately twenty-two feet which yields a room volume of 40,656 cubic feet. The compartment has a concrete floor, concrete walls and a concrete ceiling and is equipped with fire detection.

This room houses four motor control centers (MCCs 102, 105, 114, and 115). There are also miscellaneous wall-mounted junction boxes, emergency lights, a control station, and panels. The room also serves as a junction node where 1A and 0C Diesel Generator cables leave the Auxiliary Building to traverse the roof.

A529 Fire Analysis Results

Two hundred and forty-four fire scenarios are identified for A529. One hundred forty-four are screened due to low functional impact. The screening is based on the same criteria described in Section 4.3.1.3. The remaining 100 scenarios are represented by one initiating event shown in Table 4-N-3. The consolidation is based on an assessment of the functional impact and ignition frequency of each scenario. The frequency of the initiator is the sum of the frequencies of the fire scenarios it represents.

Table 4-N-1
A529 Fixed Ignition Fire Scenario Summary

Scenario	Fire Scenario Description	Trays and Panels Damaged by Fire
C1-58	MCC114 compartments and bus	< No other effects >

Table 4-N-2
A529 Transient Fire Scenario Summary

Scenario	Fire Scenario Description	Trays and Panels Damaged by Fire
T1-42	Transient damage to MCC114	< No other effects >

Table 4-N-3
A529 Fire Analysis Results

Initiating Event	Fire Scenario	Frequency	Ignition Source	Functional Impact	CDF
A529F1	C1-58, T1-42	9.91E-4	MCC 114R bus-work, breaker or transient induced fire	M2, H6, XW*, PH*	9.18E-8

A529 <u>Fire Ignition Frequency</u>

Both fixed and transient ignition frequencies are determined for the Unit 1 West Electrical Room.

<u>Fixed Ignition Frequency</u>

The fixed ignition frequency is determined by starting with the compartment fixed ignition frequency results of Section 4.3.2 and then developing a scenario-specific ignition frequency as described in Section 4.3.4.3. The sources and parameters used for A529 are:

Source	Generic Fire Frequency	Location Weighting	Source Weighting	Total Sources in Aux Building	Room Specific Frequency
Electrical Cabinets	1.5E-2	0.154	-	-	2.31E-3

MOTOR CONTROL CENTER 114R

The MCC is an assembly of joined sectional cabinets with no ventilation openings, grills or louvers. Each section contains smaller cubicles (except for the distribution panel section) in various configurations and sizes. Each cubicle is unique to an individual load. The typical cubicle houses a breaker, control transformer, load starter, ground sensor, current transformer, indication lights, and wiring. Below the cubicle, in a separate doored compartment, a terminal block connects the load power leads to the MCC. The cubicles are sealed from the external environment and any ignition within the cubicle would be contained.

The back of each vertical section forms a common (undivided) area with terminal blocks. These blocks connect field control wiring to the cubicle components. The wiring and blocks do not represent a typical ignition source, but any fires in this compartment would be contained within it.

Vertical bus drops are sandwiched between the front cubicles and rear compartment. This area is enclosed. The vertical bus connects to the horizontal main bus-work at the top of the section. The horizontal bus-work is closed in with sheet metal panels further covered by an access plate (both front and rear). Here also, the bus-work is sealed from the external environment and any ignition would be contained.

All wiring enters the MCC via conduit. The usual entry is a short (approximately 12-inch) nipple where the wire leaves an overhead cable tray.

Because of the sealed design, the MCC bus-work and individual cubicles are not considered a source of propagation. As an ignition source, the MCC is analyzed at a 65 Btu heat release rate. No cable trays or equipment are in the damage range.

MCC ignition frequencies are developed by apportioning the room's electrical cabinet ignition frequency and apportioning it among the MCCs as follows:

1. For each MCC, count the number of non-spare (in service) cubicles.
2. For each MCC, add one to that MCC's cubicle count in Step 1 to account for the MCC bus itself.
3. Sum the individual counts for each MCC from Step 2.
4. Compute the individual non-spare cubicle ignition frequency or MCC bus ignition frequency by dividing the room electrical cabinet ignition frequency by the sum from Step 3.
5. Eighty percent of individual cubicle ignitions are assumed to fail only the initiating cubicle load.
6. Twenty percent of individual cubicle ignitions are assumed to fail the entire MCC by propagating into the bus-work, as would a bus-work ignition.

The Electric Cabinet fire frequency is apportioned based on the total number of electrical cabinets in the room.

A529 Panel Count

Equipment	Bus	$N_{cubicles}$	Count
MCC 114R	1	57	58
MCC 102F	1	28	29
MCC 105R	1	29	30
MCC 115R	1	18	19
		Total	**136**

Individual non-spare cubicle ignition frequency or MCC bus ignition frequency:

$$F_{\text{cubicle or bus}} = 2.31E\text{-}3 \div 136 = 1.699E\text{-}5$$

Individual cubicle ignitions that fail only the initiating cubicle load:

$$F_{\text{individual}} = 80\% * F_{\text{cubicle or bus}} = 80\% * 1.699E\text{-}5 = 1.359E\text{-}5$$

Individual cubicle ignitions that fail the entire MCC and the MCC bus-work ignition:

$$F_{mcc} = F_{cubicle\ or\ bus} + (N_{cubicles} * 20\% * F_{cubicle\ or\ bus})$$
$$= F_{cubicle\ or\ bus} * [1 + (N_{cubicles} * 20\%)]$$

For MCC 114, ignitions that fail the entire MCC:

$$F_{mcc114} = 1.699E\text{-}5 * [1 + (57 * 20\%)] = 2.106E\text{-}4$$

However, for modeling purposes, all individual and MCC ignition scenarios are conservatively grouped together and fail the entire MCC. Thus:

$$A529F1_{Fix} = \Sigma (F_{individual}) + F_{mcc114}$$
$$= (N_{cubicles} * F_{individual}) + F_{mcc114} = (57 * 1.359E\text{-}5) + 2.106E\text{-}4$$
$$= 7.75E\text{-}4 + 2.106E\text{-}4 = 9.85E\text{-}4$$

Transient Ignition Frequency

The transient ignition frequency is determined by starting with the compartment transient ignition frequency results of Section 4.3.2 and then developing a scenario specific ignition frequency as described in Section 4.3.4.3. The sources and parameters used for A529 are:

Source	Generic Fire Frequency	Location Weighting	Source Weighting	Total Sources in Aux Bldg	Room Specific Frequency
Cable fires - welding	5.1E-3	2	1	232	4.40E-5
Transient fires - welding	3.1E-2	2	1	232	2.67E-4
Transient - other	1.3E-3	2	7	232	7.84E-5

For a description of transients and transient fire analysis see Section 4.3.4.3. The transient fire is modeled as a maintenance refuse fire and therefore uses the "Transient - other" as the ignition frequency, F_{it} = 7.84E-05. The 69' West Electrical Room has no fire suppression, therefore P_{fs} = 1.0.

The refuse fire In-Plume worksheets show the corner configuration plume screening distance is 7.9 feet. Wall configuration plume distance is six feet and room center plume distance is 4.6 feet. The postulated transient fire is three feet high. For the various room tray configurations there are no overhead targets within the transient plume and overhead cable damage is unlikely. Therefore, A_s = zero.

The 69' Unit 1 Electrical Room floor area is 1,848 square feet. The area displaced by floor mounted cabinets and equipment is approximately 133 square feet. Therefore, the floor area where combustibles could be placed is:

$$A_F = 1{,}848ft^2 - 133ft^2 = 1{,}715ft^2$$

$$u = (A_s + A_{sr}) / A_F = (A_s + A_{sr}) / 1{,}715ft^2$$

$$P_{fs} = 1$$

$$\text{So:} \quad F_t = F_{it} * u * P_{fs} = F_{it} * [(A_s + A_{sr}) / A_F] * 1$$

$$= 7.84E\text{-}4 * [(A_s + A_{sr}) / 1{,}715ft^2]$$

MOTOR CONTROL CENTER (MCC) 114 TRANSIENT

For this analysis, the transient fire, maintenance refuse is placed within two feet of the MCC 114R. Motor control center transient-induced damage is conservatively modeled assuming:

1. Entire vertical section(s) are damaged by the transient, and
2. The transient fire will overlap up to three vertical sections.

The vertical section modeling provides load-specific transient ignition frequencies and resulted in 42 transient scenarios. However, in the plant model, these scenarios are binned such that any transient affecting MCC114 fails the entire MCC. Therefore, the total floor area around the exposed MCC perimeter is used to compute the transient induced damage frequency.

The A_{sr} for MCC114 is by multiplying a 1.8 foot margin times the available floor area around the target component. For MCC114, $A_{sr} = 139.8ft^2$, $A_s = 0$, and:

$$A529F1_{trans} = F_{t(MCC114)} = 7.84E\text{-}4 * [(A_s + A_{sr}) / 1{,}715ft^2] = 7.84E\text{-}4 * [(0ft^2 + 139.8ft^2) / 1{,}715ft^2] = 6.39E\text{-}6$$

Total Ignition Frequency

The total ignition frequency is the sum of fixed and transient scenarios.

$$A529F1 = A529F1_{trans} + A529F1_{fix} = 6.39E\text{-}6 + 9.85E\text{-}4 = 9.91E\text{-}4$$

Suppression Systems

The 69' West Electrical Room is equipped only with fire detection, therefore suppression-induced failures are not considered.

Fire Suppression Induced Equipment Failures

Based on the approach described in Section 4.3.4.4.4, equipment failure due to the inadvertent actuation of the automatic fire suppression system is assumed not to occur due to the absence of automatic suppression equipment in the room.

AB	**Auxiliary Building Stairtowers and Elevator Shaft**	**Location:**	Auxiliary Building
		Fire Area:	AB-1,3,4,5,E
		CDF:	Screened - Low Fire Frequency

Access between the various levels of the Auxiliary Building is available via the Stairtowers through Class B fire doors. Each of the Stairtowers is approximately eighteen feet in length and nine feet in width for a room area of 162 square feet. The ceiling in each room is approximately 96 feet from the floor for a total room volume of 15,552 cubic feet.

Each of the Stairtowers are constructed of concrete block Appendix R walls. The steps and floors are poured concrete. The Stairtowers do not contain fire suppression or detection.

AB Fire Analysis Results

There are no fixed ignition sources in these rooms. The only potential ignition sources are transient combustible fires. The transient fire ignition sources are minimal and likely exist only during transport. There are also minimal targets within the stairtowers. These stairtowers are screened due to both low functional impact and low fire frequency.

1. Stairtower AB-1

   Access to this Stairtower is through a locked security door either on the 69' elevation or -15' elevation. Access is not available to any other level from this Stairtower. The Stairtower is located near the Rad Con area on the 69' elevation. There are no fixed combustibles in the stairwell. Conduit runs along the ceiling on the 69' elevation.

2. Stairtower AB-3

   Access to the Cable Spreading and Battery Rooms from the Control Room is available via this Stairtower through door D-301. There are no fixed combustibles or targets within the stairwell.

3. Stairtower AB-4

   Access to the Unit 2 outer Auxiliary Building rooms, including the MSIV and Piping Penetration Rooms, is available via this Stairtower. At the 69' level access is through door D-524. The stairwell down to the 45' level contains plant emergency page equipment. Access to the 45' level is via door D-412. The stairwell down to the 27' level contains a junction box mounted to the wall. Access the east side of the 27' level is via door D-310. At all levels there are no fixed combustibles within ten feet of the doorway.

4. Stairtower AB-5

Access to the Unit 1 outer Auxiliary Building rooms, including the MSIV and Piping Penetration Rooms, is available via this Stairtower. At the 69' level access is through door D-535. The stairwell down to the 45' level contains plant emergency page equipment. Access to the 45' level is via door D-428. The stairwell down to the 27' level contains a junction box mounted to the wall. Access the east side of the 27' level is via door D-315. At all levels there are no fixed combustibles within ten feet of the doorway.

5. Elevator AB-E

Access to all levels of the inner Auxiliary Building rooms is available via this elevator. The elevator is located near access to AB-2 from the 69' elevation via the Vestibule (A528). The elevator shaft walls are concrete Appendix R barriers. The door is a Type M fire rated sliding door.

Fire Suppression

This area does not contain detection or suppression.

Fire Suppression Induced Equipment Failures

Based on the approach described in Section 4.3.4.4.4, equipment failure due to the inadvertent actuation of the automatic fire suppression system is assumed not to be plausible for these areas due to the lack of suppression.

CC-A&B	**Cable Chases 1A, 1B, 2A, 2B A518 and A517**	**Location:**	45' to 69' Auxiliary Building 83' Auxiliary Building	
		Fire Area:	20	1A
			21	1B
			22	2A
			23	2B
			35	A517
			36	A518
		CDF:	9.67E-07	

Cable Chase 1A, 1B, 2A, and 2B

Cable Chase 1A is a compartment 14 feet in length by 14 feet in width for an area of 196 square feet. The compartment height is 55.5 feet, giving the compartment a volume of 10,878 cubic feet. It is equipped with a smoke detection and wet pipe suppression system. Access to this compartment at the 45' level is from Switchgear Room (A430) through a normally locked three-hour fire door. Secondary entry is from the Personnel Access Area (A525) into the main Plant Exhaust Equipment Room (A524), also through a normally locked three-hour fire door.

Although the previous discussion specifically addressed Unit 1 Cable Chase 1A, the general observations and the conclusions are applicable to Cable Chases 1B, 2A, and 2B for the following reasons:

- All four chases are bounded by acceptable two-hour barriers.
- All four chases are accessed through normally locked three-hour fire doors.
- All four chases are devoid of transient combustibles.
- All four chases are devoid of any significant ignition sources, as referenced in Section B below.
- All four chases are equipped with smoke detection and wet pipe sprinklers.

Horizontal Cable Chases (A518 and A517)

Unit 1 Cable Chase HCCU1 (A518) runs horizontally east to west in the Auxiliary Building from the top of vertical Cable Chase CC1A to the Unit 1 West Electrical Penetration Room (A529). The floor elevation of A518 is 83 feet. The chase is accessed through a normally locked three-hour rated fire door on the west end of the chase, via a ladder from A529 on the 69' elevation.

The chase is a concrete compartment approximately nine feet wide and 118 feet long. The height of the compartment is seven feet floor-to-ceiling. Stacked cable trays run along both the north and south walls. Both banks of trays penetrate the west end of the compartment. The trays along the south wall span approximately 100 feet from the west end, at which point the trays turn and pass through the south wall and into vertical Cable Chase 1C. The trays along the north wall span the entire length of the wall and then pass through the east wall and into vertical Cable Chase 1A. The cable trays running along the south wall are enclosed for fire separation.

With the cable trays running along the walls, approximately three-and-a-half feet remain between the trays for access. The concrete floor is bare and uncluttered. The room is devoid of any material and equipment other than the cable trays. Two ventilation openings exist in the ceiling. These openings do not contain fire dampers or ventilation fans.

Although the previous discussions are specific to Unit 1 horizontal Cable Chase (A518), the general observations and the conclusions are applicable to Unit 2 horizontal Cable Chase (A517) for the following reasons:

- Both chases are bounded by acceptable two-hour barriers.
- Both chases are accessed through normally locked three-hour fire doors.
- Both chases are devoid of transient combustibles.
- Both chases are devoid of any significant ignition sources (refer to Section B).
- Both chases are equipped with smoke detectors in the ceiling.

Fire Analysis Results

One fire scenario is identified for each cable chase. The ignition frequencies for these scenarios have been evaluated beyond the compartment ignition frequency process shown in Section 4.3.3. This closer evaluation is done to account for the unique features associated with the cable chases. Table 4-P-1 list the fire scenarios considered for the cable chases. Table 4-P-2 shows the resulting core damage frequency associated with each scenario.

Table 4-P-1
Cable Chase Fire Scenario Summary

Scenario	Fire Scenario Description	Equipment Damaged
1	Unknown Transient Fire CC-1A	It is conservatively assumed that cable damage occurs.
2	Unknown Transient Fire CC-1B	It is conservatively assumed that cable damage occurs.
3	Unknown Transient Fire CC-2A	It is conservatively assumed that cable damage occurs.
4	Unknown Transient Fire CC-2B	It is conservatively assumed that cable damage occurs.
5	Unknown Transient Fire A518	It is conservatively assumed that cable damage occurs.
6	Unknown Transient Fire A517	It is conservatively assumed that cable damage occurs.

Table 4-P-2
Cable Chase Fire Analysis Results

Initiating Event	Fire Scenario	Frequency	Ignition Source	Functional Impact	CDF
Unassigned	1	3.21E-07	Cable Fire CC-1A	CCDP of 0.5 is assumed	1.61E-07
Unassigned	2	3.21E-07	Cable Fire CC-1B	CCDP of 0.5 is assumed	1.61E-07
Unassigned	3	3.21E-07	Cable Fire CC-2A	CCDP of 0.5 is assumed	1.61E-07
Unassigned	4	3.21E-07	Cable Fire CC-2B	CCDP of 0.5 is assumed	1.61E-07
Unassigned	5	3.21E-07	Cable Fire A518	CCDP of 0.5 is assumed	1.61E-07
Unassigned	6	3.21E-07	Cable Fire A517	CCDP of 0.5 is assumed	1.61E-07

Fire Ignition Frequency

There are no fixed ignition sources in the cable chases. The cable cases are also devoid of any transient combustibles and their configurations will not allow storage of any significant volume of transient combustibles.

The initial calculation of transient ignitions in fire compartments (addressed in Section 4.3.4.3) allows for the inclusion of various types of transient ignition contributions without rigorous assessment as to whether certain transient ignitions are indeed applicable to the fire compartment in question. This approach is appropriate for initial fire risk screening as it results in conservative ignition frequencies.

However, one of the first steps in detailed fire damage modeling is the assessment of the appropriateness of ignition sources. The transient fire ignition contribution to fire compartment ignition frequency is divided into the following contributors:

- Hot work induced cable fires
- Hot work induced transient fires
- Other transient ignitions

The first contributor defines hot work activities that accidentally result in the ignition of cable insulation. The second contributor defines hot work activities that accidentally result in the ignition of transient combustibles in the area. The third category covers the spectrum of other transient ignitions (e.g., hot pipe ignites nearby piping insulation, pinched electrical cord results in burned cord insulation). These examples are actual events from the EPRI Fire Events Database. The detailed fire modeling of a compartment needs to assess whether each of the above categories of transient ignitions is appropriate for further consideration.

To address the first and second contributors, it is necessary to first consider whether or not hot work activities are allowed in the compartment while at power. Review of CCNPP procedure SA-1-100, Fire Prevention, shows that hot work activities are not explicitly precluded while at-power. However, SA-1-100 does indicate the following areas where hot work activities are extremely restricted (e.g., review by

POSRC, authorization by Superintendent - Nuclear Operations):

- Control Room
- Cable Spreading Rooms
- Switchgear Rooms
- DAS Computer Rooms

Based on procedural guidance and discussions with Fire Protection Engineers, hot work in all areas of the plant are tightly controlled, however, a procedure change has been requested to explicitly include Cable Chases in the list of restricted areas in SA-1-100. See Improvement 7.6.

The second question to consider is whether transient combustibles can realistically exist in the area. The 1A, 1B, 2A, and 2B Cable Chases are concrete vertical chases with extremely limited open floor space, and are not compartments designed for ease of access. These Cable Chases require scaffolding to ascend. The likelihood of transient combustibles existing in such a compartment is judged negligible. The Combustible Loading Analysis conservatively assumed five gallons of lube oil in each of these compartments, which is not a realistic characterization of the compartments. The hot-work-induced transient ignition contributor is dismissed from further analysis as a legitimate fire scenario.

The third consideration is to assess the adequacy of the other transient ignitions in the compartment. The compartment ignition frequency calculation considers the following other transient ignitions:

- heater
- open flame
- extension cord

The first two do not realistically apply to this compartment. The third is determined not to generate sufficient heat intensity to damage IEEE 383 cable.

<u>Adjusted Compartment Ignition Frequency</u>

Based on the above assessment, the initial Compartment Ignition Frequency is adjusted to take advantage of more recent data and CCNPP specific compartment configurations. Original frequencies were calculated based on data from the FIVE methodology and the EPRI Fire Events Database. The EPRI data is based on events that occurred from 1965 through 1988. A recent study commissioned by the NRC presented updated data through 1996.

The original EPRI data included four events of cable fires caused by welding and cutting activities. However, the updated information points out that since Appendix R improvements have been implemented, this number has been reduced to zero. Also, none of the fires occurred in cable chases (Note that this excludes the candle initiated fire at Browns Ferry. The use of candles is not allowed in accordance with SA-1-100). For the purpose of this analysis, the generic frequency for welding-induced cable fires is assumed to be one per 1666.7 reactor years, with an industry wide capacity factor of 0.62, or 3.72E-04/reactor year (rather than the original 5.13E-3). This number is converted to a site specific number using the methodology outlined in Section 4.3.2.

3.72E-04 * 2 (Location Weighting Factor) * (1/232 (Ignition Source Weighting Factor)) = 3.21E-06

The hot work-induced transient combustible contribution is reduced by a factor of 10 since the areas will be administratively restrictive control hot work activities (this credits the implementation of Improvement 6). This approach is consistent with NSAC-181, "Fire PRA Requantification Studies", and the EPRI Fire PRA Implementation Guide. Therefore, the revised ignition frequency per Cable Chase per reactor year is:

3.21E-06 * 0.10 (reduction factor for administrative controls) = 3.21E-07

Given that there are six Cable Chases with similar configuration, the total frequency for all Cable Chases is 1.92E-06.

Fire Detection/Suppression

The 1A, 1B, 2A and 2B Cable Chases contain both smoke detectors, and a wet pipe sprinkler system, located at various levels in each cable chase. The wet pipe sprinkler heads are rated at 212° F and the spray nozzles at 250° F.

Cable Chases A517 and A518 have smoke detection, but no automatic suppression.

Fire Suppression Induced Equipment Failures

Based on the approach described in Section 4.3.4.4.4, equipment failure due to the inadvertent actuation of the automatic fire suppression system is assumed not to occur. Cable and conduit, pumps and other PRA equipment in the room are not considered to be susceptible to water damage. Fire suppression induced equipment failures in the Horizontal Cable Chases is not plausible due to the lack of suppression.

Risk Assessment Analysis

According to the Interactive Cable Analysis, a fire in the Vertical Cable Chases (1A and 1B) requires the Unit to be placed in cold shutdown within 72 hours. The affect of a fire in the cable chases has been analyzed and alternate shutdown means identified. The alternate actions to be taken in the event of a fire in these areas is captured in AOP-9L (Cable Chase 1A) and AOP-9M (Cable Chase 1B).

Control cables for the Saltwater, Component Cooling, and Service Water Pumps are routed through these Cable Chases. In the event of a fire in a chase, the pumps will not receive power from the Diesel. Isolation switches permit the operation of the pumps from their respective switchgear room in the event of loss of control cable. Inventory control is available through operation of manual valves independent of this area.. Damage to the control cables associated with the PORVs (1ERV-402 and 1ERV-404) can be mitigated from the Control Room.

There are no specific AOPs associated with the Horizontal Cable Chases. These chases contain power and control cables for valves that are either not required for safe shutdown, sufficient redundancy exists independent of this area, or the valve can be manually operated locally. Instrument indication that may be lost through a fire in this area can be overridden in the Control Room.

Due to the low fire ignition frequency associated with the cable chases, the development of a detailed model which includes the appropriate Appendix R recovery actions was not believed to be cost-effective. Therefore, given the availability of the Appendix R recovery actions, a CCDP of 0.5 is conservatively assumed.

CC-C	**Cable Chases 1C & 2C**	**Location:**	45' to 69' Auxiliary Building
		Fire Area:	16 CC1C 17 CC2C
		CDF:	Screened - Low Fire Ignition Frequency

Unit 1 Cable Chase CC1C runs vertically in the Auxiliary Building from the ceiling of the Cable Spreading Room up through the 45' and 69' elevations. The ceiling of the chase is the roof at elevation 91'-6". The bottom of the chase is open to the Unit 1 Cable Spreading Room below. The walls of the chase are all rated concrete barriers.

The chase is a concrete compartment approximately seven feet wide and 21.5 feet long for the portion of the chase up to the floor of the 69' elevation. Beginning at the 69' elevation, the length of the chase lessens to approximately 16 feet long. The total height of the vertical chase is approximately 43 feet. Numerous cable trays and electrical conduits run up through the chase.

Access to the cable chase is via the opening in the Cable Spreading Room ceiling or a locked rated steel fire door on the 69' elevation (via A520). Although possible, the numerous cable trays and conduits in the Cable Spreading Room ceiling make access into the chase from the Cable Spreading Room ceiling difficult. The door on the 69' elevation opens to a small platform. Similar to the 1(2)A and B vertical chases, ascending or descending into this chase requires climbing and/or the placement of scaffolding.

Although the previous discussions are specific to Unit 1 Cable Chase CC1C, the general observations and the conclusions are applicable to Unit 2 Cable Chase CC2C for the following reasons:

- Both chases are bounded by acceptable two-hour barriers.
- Both chases are accessed through normally locked steel hatches or from below via the associated Cable Spreading Room ceiling.
- Both chases are devoid of transient combustibles.
- Both chases are devoid of any significant ignition sources.
- Both chases are equipped with detectors in the ceiling and automatic halon suppression.

Fire Analysis Results

No accident sequence quantification are necessary for Chases CC1C and CC2C. The fire-induced core damage frequency contribution from fires initiated in these compartments was determined to be negligible. Transient fires postulated for this area will not result in cable damage due to the height of the cable above the postulated transient maintenance refuse fire.

Fire Ignition Frequency

A review of the base ignition frequency calculation for this area shows that the contributors to the fire ignition frequency of this area include:

- Junction box in Q cable: 1.39E-5/yr.
- Transient sources: 4.26E-4/yr.

In the calculation of ignition frequencies for various plant areas it is appropriate to exclude various minor fixed components as credible fire scenarios that would result in core damage end states. Such items in CC1C include miscellaneous electrical junction and pull boxes. Ignitions associated with these items would result in minor, if any, fires with low heat intensities which would be confined within the boundaries of the item itself. None of these minor items can result in a credible fire-induced core damage scenario.

Hot work-induced cable ignitions are dismissed as legitimate fire growth and damage scenarios due to the fact that the number of hot work-induced cable fires is very small and the incidents are minor in nature. The plant is also equipped with IEEE 383 cable which does not support fire propagation. This approach is consistent with NSAC-181, "Fire PRA Requantification Studies", and the EPRI Fire PRA Implementation Guide.

To address hot work-induced transient fires, consideration is first given to the frequency with which hot work activities are permitted in the compartment while at power. The likelihood of hot work occurring in the chase while at power is negligible. The next consideration is whether or not transient combustibles can realistically exist in the compartment. The configuration and access of the compartment realistically precludes the possibility of significant transient combustible material existing in the compartment. Given these two considerations, the hot work-induced transient combustible contributor is dismissed from further analysis as a legitimate fire scenario.

The remaining ignition sources applied to the ignition frequency calculation of this compartment are the following "other" transient ignitions:

- open flame
- extension cord
- heater

The first contributor does not realistically apply to this compartment. The second and third (hand-held heaters for testing of the installed halon heat detectors) are judged not to generate sufficient heat intensity to damage IEEE 383 cable.

As there are no credible ignition sources in the chase, scenarios involving ignitions in the chase causing smoke-induced actuation of the halon system and subsequent isolation of the Cable Spreading Room ventilation system, are assessed to be negligible frequency. Such scenarios initiated by ignitions in the Cable Spreading Room are appropriately discussed in the fire damage assessment for the Cable Spreading Room.

The above discussion and conclusions for Chase CC1C also apply to Unit 2 Chase CC2C due to the similarities among the chases.

In summary, there are no significant fixed ignition sources in these compartments. In addition, the likelihood of a transient induced fire damage scenario is assessed to be negligible. Therefore, there are no credible fire damage scenarios assessed for Chases CC1C and CC2C.

Fire Detection/Suppression

The chases contain both fire detection and halon suppression. Two smoke detectors are located in the chases for alarm purposes. A heat detector is also located in the chase for automatic initiation of the Halon system.

Fire Suppression Induced Equipment Failures

Based on the approach described in Section 4.3.4.4.4, equipment failure due to the inadvertent actuation of the automatic fire suppression system is assumed not to occur. Cable and conduit, and other PRA equipment in the room are not considered in the short term to be susceptible to damage from halon.

INTAKE Unit 1 and Unit 2 Intake Structure

Location: East of the Turbine Building
Fire Area: IS
CDF: 1.22E-8

The Intake Structure is located to the east of the Turbine Building, separated from the Turbine Building by the North Service Building. The majority of the floor area of the Intake Structure is at the three-foot elevation. Access is through a water tight door in the east wall toward the Unit 2 end (i.e., south end) via the twelve-foot elevation of the North Service Building or a water tight door in the north wall via the yard.

The Intake Structure is a large spacious compartment. The compartment is approximately 372 feet in length and thirty-four feet in width for a total floor area of approximately 12,648 square feet. The nominal height of the compartment is twenty-four feet.

Six Unit 1 Circulating Water pumps and six Unit 2 Circulating Water pumps are spaced equally down the center line of the length of the compartment. Three Unit 1 Salt Water pumps and three Unit 2 Salt Water pumps are arranged generally equidistant in pits (elevation - eight feet) along the east wall. The Unit 1 and Unit 2 portions of the Intake Structure are divided by a short flood wall at the center line of the compartment.

Each Salt Water pump motor is located approximately twenty-seven feet above the pit floor at an elevation of approximately nineteen feet. The closest equipment to the motors are the six Intake Structure supply fans, one installed overhead of each Salt Water pump motor. Each fan motor is approximately seven feet from the nearby pump motor; the fan electrical feeds are at least five feet away.

Fire Analysis Results

Eighteen fire scenarios were identified for the Intake Structure. Twelve are the result of fixed ignition sources and six are due to transient ignition sources. These eighteen scenarios are identified in Table 4-R-1, and are represented by five initiating events identified in Table 4-R-2. The consolidation is based on an assessment of the functional impact and ignition frequency of each scenario. The frequency of the initiator is the sum of the frequencies of the fire scenarios it represents.

Table 4-R-1
INTAKE Fire Scenarios Summary

Scenario	Fire Scenario Description	Equipment Damaged
1	SW Pump 11, Large Oil Spill Outside Pit	SW Pump 11, CW Pump 11, CW Pump 12
2	SW Pump 12, Large Oil Spill Outside Pit	SW Pump 12, CW Pump 13, CW Pump 14
3	SW Pump 13, Large Oil Spill Outside Pit	SW Pump 13, CW Pump 15, CW Pump 16
4	SW Pump 21, Large Oil Spill Outside Pit	SW Pump 21, CW Pump 21, CW Pump 22
5	SW Pump 22, Large Oil Spill Outside Pit	SW Pump 22, CW Pump 23, CW Pump 24
6	SW Pump 23, Large Oil Spill Outside Pit	SW Pump 23, CW Pump 25, CW Pump 26

Table 4-R-1
INTAKE Fire Scenarios Summary (Continued)

Scenario	Fire Scenario Description	Equipment Damaged
7	CW Pump 11, Large Oil Spill	CW Pump 11, SW Pump 11
8	CW Pump 13, Large Oil Spill	CW Pump 13, SW Pump 12
9	CW Pump 15, Large Oil Spill	CW Pump 15, SW Pump 13
10	CW Pump 22, Large Oil Spill	CW Pump 22, SW Pump 21
11	CW Pump 23, Large Oil Spill	CW Pump 23, SW Pump 22
12	CW Pump 25, Large Oil Spill	CW Pump 25, SW Pump 23
13	Intake Structure Transient Fire	SW Pump 11, CCW Pump 11
14	Intake Structure Transient Fire	SW Pump 12, CCW Pump 13
15	Intake Structure Transient Fire	SW Pump 13, CCW Pump 15
16	Intake Structure Transient Fire	SW Pump 21, CCW Pump 22
17	Intake Structure Transient Fire	SW Pump 22, CCW Pump 23
18	Intake Structure Transient Fire	SW Pump 23, CCW Pump 25

Table 4-R-2
INTAKE Fire Analysis Results

Initiating Event	Fire Scenario	Frequency	Ignition Source	Functional Impact	CDF
INTKF1	1,7,13	8.10E-5	SW Pump 11 large oil fire, CW Pump 11 large oil fire, Transient induced fire	Y3, S1, VC	2.77E-9
INTKF2	2,8,14	8.10E-5	SW Pump 12 large oil fire, CW Pump 13 large oil fire, Transient induced fire	Y4, S2, VC	6.41E-9
INTKF3	3, 6, 9, 12, 15, 18	1.62E-4	SW Pump 13 large oil fire, SW Pump 23 large oil fire, CW Pump 15 large oil fire, CW Pump 25 large oil fire, Transient induced fire	Y3, Y4, VC, U-2 MFWS	2.40E-9
INTKF4	4,10,16	8.10E-5	SW Pump 21 large oil fire, CW Pump 22 large oil fire, Transient induced fire	Y3, GZ, U-2 MFWS	4.05E-10
INTKF5	5,11,17	8.10E-5	SW Pump 22 large oil fire, CW Pump 23 large oil fire, Transient induced fire	Y4, GW, U-2 MFWS	2.52E-10

Fire Ignition Frequency

Both fixed and transient ignition frequencies were determined for the Intake Structure.

Fixed Ignition Frequency

The fixed ignition frequency is determined by starting with the compartment fixed ignition frequency results of Section 4.3.2 and then developing a scenario specific ignition frequency as described in Section 4.3.4.3. The primary contributors to the Compartment Ignition Frequency for this room were identified as the following components:

- Various electrical cabinets: 4.80E-3/yr
- Various junction boxes: 8.52E-5/yr
- Twelve (12) ventilation units: 7.00E-4/yr
- Four (4) transformers: 7.44E-4/yr
- Eighteen (18) major pumps: 6.40E-3/yr
- Transient sources: 3.13E-3/yr

Excluded Ignition Sources

In the calculation of ignition frequencies for various plant areas it is appropriate to exclude various minor fixed components as credible fire scenarios which would not result in core damage end states. Such items in the Intake Structure include battery-operated emergency lights, instrument racks, and miscellaneous small electrical boxes. Ignitions associated with these items would result in minor, if any, fires with low heat intensities and which would be confined within the boundaries of the item itself. Appropriately, none of these minor items that exist in the compartment are included in the quantification of the compartment fire ignition frequency. In addition, the various electrical cabinets and junction boxes were investigated and determined to be such that any ignitions of these items would be confined within the component and would not damage other equipment.

The ventilation units are ventilation fans mounted in the roof. There are no combustibles or targets near enough to each of these fans to either be damaged or ignited (a standard 65 Btu/s small motor fire generates a critical damage distance of less than 2 feet). As such, these six fans are appropriately dismissed as fire induced core damage scenarios. The transformers are not located close enough to combustibles or other equipment to damage them; therefore, they were dismissed as a plausible ignition source.

The remaining credible ignition sources are the Salt Water and Circulating Water pumps, and transient ignition sources.

The sources and parameters used for the Intake Structure are:

Source	Generic Fire Frequency	Location Weighting	Source Weighting	Total Sources in Intake Structure	Room Specific Frequency
1INTK	3.2E-3	2	1	-	6.40E-3

The total number of major pumps in the Intake Structure is shown below:

Intake Structure
Pump Count

Equipment	Count
Unit 1 SW Pumps	3
Unit 1 CW Pumps	6
Unit 2 SW Pumps	3
Unit 2 CW Pumps	6
Total	**18**

Salt Water Pump Fire

A pump fire may include an electrical motor windings fire or a lubricating oil spill fire. As there is no equipment closer than approximately five feet to each SW pump motor other than its own electrical feed, a motor fire would not damage any other equipment. The most conservative fire is an oil spill. As such, the SW pumps are modeled as postulated oil spill and ignition fires.

Review of the EPRI Fire Events Database shows that approximately 50% of pump fire events involve lubricating oil fires and 50% represent motor fires and other miscellaneous ignitions. The Fire PRA Implementation guide indicates that approximately 88% of oil spill fires are "small" spills and 12% are "large spills".

One complicating configuration issue is that the lubricating oil is contained in the Salt Water pump motor which is located approximately twenty-seven feet above the Salt Water pit floor and approximately sixteen feet above the three-foot elevation. An oil spill could be postulated to be located at the pump itself, on the three-foot elevation floor, or the pit floor. A lubricating oil fire that occurs at or near the pump motor could be a "large" volume oil spill fire. A large volume of oil would travel down to the floor below. If a lubricating oil spill fire occurs at the pump motor it is judged to most likely be a fire feeding off a small oil leak or spill. Therefore, the small oil spill fire is modeled as occurring at the pump motor. Due to the cross-sectional area of the pit (twelve feet x ten feet), a large oil spill would likely fall into the pit below. Although a spill fire can be postulated to occur outside the pit, it is judged less likely. So that both scenarios can be modeled, it is assumed that 90% of the large oil spill fires occur within the pit and 10% occur outside the pit. Note that this assumption is very conservative and it is believed that all the oil will spill within the pit due in part to the configuration of ventilation baffles near the pump motor.

Therefore the scenario frequency for each large SW pump oil fire is:

$$Pump_{SW} = (6.4E\text{-}3)*(1/18 \text{ pumps}) * (0.5 \text{ oil fires/pump fire}) *$$
$$(0.12 \text{ large oil fires/pump oil fire}) * (10\% \text{ outside pit}) = 2.13E\text{-}6$$

Circulating Pump Fire

Circulating pump fires are determined similarly to that of the Salt Water Pumps. The only difference is the all the oil will be contained in the conical well associated with each pump. Therefore:

$$Pump_{CCW} = 6.4E\text{-}3 * 1/18 \text{ pumps}) * (0.5 \text{ oil fires per pump fire}) * (0.12 \text{ large oil fire/pump oil fire}) = 2.13E\text{-}5$$

Combined Fixed Ignition Frequency

For a single pump bay, the fixed ignition frequency is calculated as follows:

$$INTK1,3,4,5_{\text{Fixed Ignition}} = Pump_{SW} + Pump_{CCW} = 2.13E\text{-}6 + 2.13E\text{-}5 = 2.34E\text{-}5$$

The fixed ignition frequency for two pump bays is as follows:

$$INTK2_{\text{Fixed Ignition}} = 2 * (Pump_{SW} + Pump_{CCW}) = 2 * (2.13E\text{-}6 + 2.13E\text{-}5) = 4.69E\text{-}5$$

Transient Ignition Frequency

The transient ignition frequency is determined by starting with the compartment transient ignition frequency results of Section 4.3.2 and then developing a scenario specific ignition frequency as described in Section 4.3.4.3. The sources and parameters used for the Intake Structure are:

Source	Generic Fire Frequency	Location Weighting	Source Weighting	Total Sources in Intake Structure	Room Specific Frequency
Transient fires - welding	3.1E-2	2	1	232	2.67E-4
Transient - other	1.3E-3	2	7	232	7.84E-5

Although the standard maintenance refuse fire is applicable to this compartment, other more severe transient fire are also possible. Therefore, to conservatively bound the spectrum of postulated transient fires, a 5 gallon oil spill is assumed.

The damage is assumed to include a circulating pump and salt water pump combination.

The transient fire is modeled as a maintenance refuse fire and therefore uses the "Transient - other" as the ignition frequency, F_{it} = 7.84E-05. Fire suppression is not credited in this analysis. Therefore, conservatively, P_{fs} = 1.0.

Transient Combustible in Range of Targets (u) and Resulting Frequency

For this analysis, u is assumed to be 0.5. And since there are no overhead targets damaged:

$$INTK1,3,4,5_{Transient\ Ignition} = F_t = F_{lt} * u * P_{fs} = 7.84E\text{-}05 * 0.5 * 1 = 3.92E\text{-}5$$

In the case of the initiator representing two pump bays:

$$INTK2_{Transient\ Ignition} = 2 * 3.92E\text{-}5 = 7.84E\text{-}5$$

Fire Detection/Suppression

The Intake Structure contains smoke detectors for alarm purposes. No automatic suppression is installed in the compartment. Hose stations for manual fire fighting are located in this compartment.

Fire Suppression Induced Equipment Failure

Based on the approach described in Section 4.3.4.4.4, equipment failure due to the inadvertent actuation of the automatic fire suppression system is assumed not to occur due to the lack of installed suppression.

T603 Unit 1 AFW Pump Room

Location: 12' Turbine Building
Fire Area: 42
CDF: 4.76E-7

The Auxiliary Feedwater (AFW) Pump Room is accessed from the 12' Turbine Building. The approximate compartment dimensions are nineteen feet wide by thirty-one feet long by 14.5 feet high for a total compartment area of 589 square feet. The room height is fourteen feet for a room volume of 8,158 cubic feet. Two watertight hatches provide access to the Turbine Building: A locked-closed double-wide door permits large equipment moves and a single-wide card reader door provides personnel access.

T603 **Fire Analysis Results**

Six fire scenarios are identified for T603. Five are the result of fixed ignition sources and one is due to transient ignition sources. The six scenarios shown in Tables 4-S-1 and 4-S-2 are represented by the six fire initiating events shown in Table 4-S-3.

Table 4-S-1
Fixed Ignition Fire Scenarios Summary

Scenario	Fire Scenario Description	Trays and Panels Damaged by Fire
1	AFW Pump 11 Small Spill	AFW Pump 11
2	AFW Pump 11 Large Spill	AFW Pump 12, AFW Pump 11, 1A0890, 1A0891, 1A0073
3	AFW Pump 12 Small Spill	AFW Pump 12, 1A0893, 1A0892, 1A0048, 1A0894, 1A08951
4	AFW Pump 12 Large Spill	AFW Pump 11, AFW Pump 12, 1A0890, 1A0891, 1A0073, 1A0893, 1A0892, 1A0048, 1A0894, 1A0895, 1A0047
6	AFW Pump Room A/C Unit	Loss of HVAC

Table 4-S-2
T603 Transient Fire Scenarios Summary

Scenario	Fire Scenario Description	Trays and Panels Damaged by Fire
5	Maintenance Refuse	AFW Pump 12, 1A0893, 1A0892, 1A0048

Table 4-S-3
T603 Fire Analysis Results

Initiating Event	Fire Scenario	Frequency	Ignition Source	Major Impact	CDF
T603F1	1	8.83E-5	AFW Pump 11 Small Spill	TF	6.95E-9
T603F2	2	1.94E-5	AFW Pump 11 Large Spill	Y3, S1, TF, TG	1.54E-8
T603F3	3	8.83E-5	AFW Pump 12 Small Spill	TF*, TG	2.24E-7
T603F4	4	1.94E-5	AFW Pump 12 Large Spill	Y3, S1, TF, TG	1.82E-7
T603F5	5	2.17E-5	Maintenance Refuse	TF*, TG	4.73E-8
T603F6	6	5.74E-5	AFW Pump Room A/C Unit	FC*	2.92E-10

T603 **Fire Ignition Frequency**

Both fixed and transient ignition frequencies were determined for the Unit 1 AFW Pump Room.

Fixed Ignition Frequency

The fixed ignition frequency is determined by starting with the compartment fixed ignition frequency results of Section 4.3.2 and then developing a scenario-specific ignition frequency as described in Section 4.3.4.3. The sources and parameters used for T603 are:

Source	Generic Fire Frequency	Location Weighting	Source Weighting	Total Sources in Aux Building	Compartment Specific Frequency
Other Pumps	6.3E-3	2	2	117	2.15E-4
Ventilation Subsystem	9.5E-3	2	1	331	5.74E-5

AFW PUMPS

Each AFW pump is mounted on a pedestal base above the floor level. The base perimeter forms a gutter that collects water or oil leakage and directs the leakage into a floor drain. 12 AFW Pump sits near a wall with three sets of conduits and junction boxes associated with the saltwater pumps. These conduits also traverse the compartment in the overhead and pass over top of one or both AFW pumps.

The AFW turbine/pump has no windings, wire, or cable to evaluate as electrical ignition sources. The pump uses oil bath anti-friction bearings inboard and outboard. The turbine uses oil bath journal bearing inboard and outboard. In addition, a small oil-filled gearbox drives the governor from the turbine shaft and the governor itself is filled with oil. This analysis uses the largest single source of oil (the governor) as a large spill. FIVE does not provide specific guidance on the small spill quantities, so for this analysis, a spill that is 25% percent of the large spill volume is judged reasonable.

Since there are two turbine-driven AFW pumps in this compartment, each pump is assigned 50% of the compartment specific pump frequency. The fire frequency for these pumps is further subdivided into large and small oil spills. Per FIVE, 82% of oil spills are classified as small and the remaining 18% are large spill. Therefore, for each AFW pump small spill frequency is:

T603F1 or T603F3 = 2.15E-4 * 50% * 82% = 8.82E-5

The small oil spill worksheet calculated a 459 btu/s heat release rate based on a four square foot confined spill. This spill burns for approximately 24 seconds. The In-Plume worksheet calculation shows that these conduits are outside the damage zone.

The radiant exposure screening distance is 2.7 feet using the critical flux value 2 btu/s/ft^2. Equipment within this distance was considered damaged. This included wall mounted junction boxes near 12 AFW Pump.

For the large spill:

T603F2 or T603F4 = 2.15E-4 * 50% * 18% = 1.94E-5

The large spill modeled heat release rate is 2,869 btu/s based on the 25 square feet pump pedestal area. The fire lasts approximately 15 seconds. This fire exposes the overhead conduit to temperatures in excess of 700°F. These conduits are, conservatively, considered damaged in this analysis. Ignition evaluation is not required because the cables are enclosed and propagation is not possible

An Outside Plume analysis shows that there is no outside plume damage at the 11 feet target height. However, the conduits turn up at the west wall. Another outside plume analysis showed the longitudinal screening distance for damage near ceiling height. A longitudinal distance of five feet represents the critical damage distance. The conduits at the west wall are all greater than five feet from the pump pedestal, and thus the conduits are undamaged.

A Radiant Exposure Worksheet shows the critical flux distance is 6.8 feet using 2 btu/s/ft^2 as the critical flux value. Wall mounted conduit and junction boxes near 12 AFW Pump are considered damaged within this range. In addition, the radiant field damages the adjacent AFW pump.

Analysis of the compartment sprinkler system shows the sprinkler head response time exceeds damage time, so fire suppression is not credited in this analysis. Specific raceway and equipment damage are listed in the Fixed Ignition Fire Scenario Table.

VENTILATION SYSTEM

The compartment air conditioner is package type refrigeration system. The compressor, blower motor, and controls are inside a cabinet. The air conditioner is not considered a source of propagation or other damage. The ignition frequency value of the ventilation subsystem is used directly.

T603F6 = 5.74E-5

Transient Ignition Frequency

The transient ignition frequency is determined by starting with the compartment transient ignition frequency results of Section 4.3.2 and then developing a scenario-specific ignition frequency as described in Section 4.3.4.3. The sources and parameters used for A429 are:

Source	Generic Fire Frequency	Location Weighting	Source Weighting	Total Sources in Aux Building	Compartment Frequency
Cable fires - welding	5.1E-03	2	1	232	4.40E-05
Transient fires - welding	3.1E-02	2	1	232	2.67E-04
Transients-other	1.3E-3	2	10	232	1.12E-04

For a description of transients and transient fire analysis, see Section 4.3.4.3. The transient fire is modeled as a maintenance refuse fire and therefore uses the "Transient - other" as the ignition frequency, F_{it} = 7.84E-05. The water suppression is not credited in this analysis, therefore $P_{fs} = 1.0$.

The postulated transient fire is three feet high and has a 100 btu/s heat release rate. The In-Plume worksheet calculations show the wall configuration plume screening distance is seven feet, the corner location screening distance is 8.5 feet, and compartment center distance is 4.9 feet. The radiant screening distance is 1.3 feet using a 2.0 btu/s/ft^2 critical flux value.

For the compartment configuration there are no ceiling targets within a transient plume range. Transient susceptible targets include the AFW pump governors and trip solenoids. These are not calculated here because the transient induced damage frequency would be subsumed by normal equipment failure rate and the fixed ignition frequency.

The remaining targets of concern are the wall-mounted junction boxes and conduits associated with saltwater pumps. The most damaging transient placement is along the south wall near the junction box 1J051A. This location places multiple raceway elements within plume and/or radiant exposure in addition to the 12 AFW pump governor.

Floor mounted equipment, piping interference and the pump pedestals occupy approximately 145 square feet of floor area. The compartment floor area is 589 square feet, so the remaining area available is 444 square feet (= A_F).

The critical distance (approximately two feet) for radiant exposure is extended around targets at the compartment south wall to calculate the floor area where a transient fire could cause damage. The exposed floor area is conservatively 86 square feet. Therefore:

$$u = (A_s + A_{sr}) / A_F = (0ft^2 + 86ft^2) / 444ft^2 = 0.194$$

and: $$F_{lt} * u * P_{fs} = 1.12E\text{-}04 * 0.194 * 1 = 2.17E\text{-}5$$

$$\boxed{T603F5 = F_{it} = 2.17E\text{-}5}$$

Fire Suppression

The AFW Pump Room is equipped with a wet pipe automatic sprinkler fire suppression.

Fire Suppression Induced Equipment Failures

Based on the approach described in Section 4.3.4.4.4, equipment failure due to the inadvertent actuation of the automatic fire suppression system is assumed not to occur. Cable and conduit, pumps and other PRA equipment in the room are not considered to be susceptible to water damage. The AFW pumps and turbines would be unaffected by water spray as these are closed mechanical housings. In addition, the pump pedestals are about one foot above the floor. The governor, trip solenoids and instruments are a drip-proof or closed design which would prevent water intrusion. Also, the compartment is equipped with a large sump cavity with two pumps. Since cabling is not damaged by water, suppression-induced equipment failure is judged to be unlikely.

TB **Unit 1 and 2 Turbine Building**

Location: East of Auxiliary Building
Fire Area: TB
CDF: 1.66E-05

The Turbine Building is oriented parallel and adjacent to the shoreline of the Chesapeake Bay with the twin Containment Structures and the Auxiliary Building located on the west, or landward, side of the Turbine Building. The service building and the intake and discharge structures are on the east, or bay side, of the Turbine Building. The service building, below the 45' level, is considered to be an extension of the Turbine Building for fire analysis purposes.

The Turbine Building houses the two turbine generators, condensers, feedwater heaters, condensate and feed pumps, turbine auxiliaries and certain switchgear assemblies The service building houses the water treatment portions of the Main Condensate System for Unit 1 and Unit 2.

Fire Analysis Results

Three fire scenarios are identified for the Turbine Building. These scenarios are represented by three initiating events as shown below. These three scenarios are chosen as the only likely scenarios which will have significant impact on plant operations.

Table 4-T-1
Turbine Building Fire Analysis Results

Initiating Event	Fire Scenario	Frequency	Ignition Source	Functional Impact	CDF
TBALLB	1	2.91E-4	Main Turbines and Generators	OP, GJ, HZ, HF*, HG*, HH, Y3, Y4, NR, IH, NS, CA, CB, MF, MG, GW*, GZ*, CD*, KH, TX, TT, T1, DW, DV, MP, SL*, FT, FN,FG, FH*, FC*, TG, FO, MH*, F1*, FJ, FW, F9, F3*, TH	1.26E-5
TBMFW1	2	3.92E-3	Unit 1 - Steam Generator Feedwater Turbine and Pumps	Y1, H5, HZ, HH, Y3, NR, NS, DM, PG*, TX, TT, T1, DW, DV, MP, SL*, FT, FH*,FC*, FW, TG	1.57E-6
TBMFW2	3	3.92E-3	Unit 2 - Steam Generator Feedwater Turbine and Pumps	OP, Y3, Y4, NR, NS, TT, T1, FT, F9	2.43E-6

* 80.2% of this initiating event frequency is due to Yard Transformer fires which have an equivalent impact. See "Yard" section for a detailed description of these initiators. Actual Steam Generator Feedwater Turbine and Pumps is 7.76E-4.

Fire Ignition Frequency

The standard approach outlined in Section 4.3.2 was initially used to determine frequency of fires in the Turbine Building. This methodology resulted in a high number of fires expected in the Turbine Building. However, a review of industry data was performed and it was found that virtually all turbine building fires noted in the EPRI Fire Events Database were considered to have relatively insignificant impact. Some caused plant trips, but were typically of a short duration and the exceptions to this did not impact more than one component. Component failure data, used to develop component failure rates, includes failures from these small fires. Also, the general transient module plant trip initiator includes historical data and encompasses the trips caused by fires. Due to the generally short durations and limited scope of these fires, the impact on human actions is determined to be negligible. To address the probability that large significant fires could occur, the three categories of Unit 1 SGFP, Unit 2 SGFP, and Large Turbine Building fires were developed.

TBMFW1 Unit 1 Significant Steam Generator Feedwater Turbine or Pump fires (SGFP)

The area around the SGFPs is important because it is adjacent to the Unit 1 Auxiliary Feedwater Pump Room (T603). Fires in this area will impair the operator actions associated with AFW turbine pump operation. Also impacted are the actions associated with the 27' Switchgear Room (A317). The 27' level is impacted because the floor of the 27' level in the Turbine Building is mostly grating. A large fire in this category could fill the Unit 1 12' and 27' levels, particularly on the west side, with significant amounts of smoke. The Unit 2 Turbine Building human actions will not be impacted due to the fire wall between the Unit 1 and Unit 2 Turbine Buildings on the 12' and 27' levels.

The frequency for this scenario is based on data from NSAC 178L, "Fire Events Database for U.S. Nuclear Power Plants" (Ref. 19, Section 3.3.9, Page 3-22). Per NSAC-178L there have been 10 Main Feedwater Pump fires in 2576.54 main feedwater pump years. Seven of these fires did not cause a plant trip and are screened. None of the three remaining fires which caused plant trips could be described as a large fire. The three fires included two bearing fires and one small insulation fire. All lasted approximately 10 minutes. Therefore, these fires would only impact the individual Main Feedwater Pump and would have negligible impact on human actions. Even if sufficient smoke were generated by these fires to impede access to the area, the fires are out within 10 minutes and none of the human actions in question occur during the first 10 minutes after a trip. Therefore, all 10 fires noted would be screened from the category of "large" Main Feedwater Pump fires. To develop the frequency of TBMFW1 (and TBMFW2) it is assumed that the next Main Feedwater Pump fire would meet the criteria of a "large" fire. Thus the frequency is 1/2576.54 years or 3.88E-04 large fires per main feedwater pump years. As there are two Steam Generator Feed Water Pumps per Unit at CCNPP, the frequency per unit is 7.76E-04 per year.

TBMFW2 Unit 2 Significant Steam Generator Feedwater Turbine or Pump fires (SGFP)

The frequency for this scenario was developed per the methodology in TBMFW1. The impacts are essentially the same except that the human actions associated with the Unit 2 AFW Pump Room (T605) and 27' Switchgear Room (A311) are impacted.

TBALLB Turbine Building Large Fire which impacts the Auxiliary Building Ventilation

This fire scenario addresses the potential of smoke migration of a large turbine building fire into the Auxiliary Building ventilation. Such migration could result in the loss of the Control Room and in the actuation of the Halon systems in the Switchgear Rooms and in the Cable Spreading Rooms. The probability that the large Turbine Building fire causes smoke to enter the Control Room and Switchgear Room ventilation systems is determined using the methodology presented in the Attachment 4-U. The probability of smoke entering the Control Room ventilation is 4.23E-02 given a large Turbine Building fire. Probability of smoke entering the Switchgear Room ventilation is 4.02 E-01 given a large Turbine Building fire. Another impact of this fire is that all human actions which require access to the Turbine Building are assumed to fail. All hardware and cables in the Turbine Building are also assumed to fail. This scenario, therefore, defines the frequency of turbine building fires which could result in this challenge to Control Room and Switchgear ventilation and to the turbine building hardware.

To evaluate the frequency of occurrence, operating data and events of large turbine building fires on all nuclear power plants, worldwide, was collected and reviewed.

Worldwide industry operating hours, to the end of 1995, is obtained from a IAEA press release. The table below lists, by country, the amount of operating experience through 1995.

Country	Number of Operating Units	Total Operating Experience to End of 1995	
		Years	Months
Argentina	2	34	7
[redacted]			
Belgium	7	135	7
Brazil	1	13	9
Bulgaria	6	83	1
Canada	21	348	9
China	3	8	5
Czech Republic	4	38	8
Finland	4	67	4
France	56	878	10
Germany	20	510	7
Hungary	4	42	2
India	10	129	1
Japan	51	704	5
[redacted]			
Korea RP	11	100	10
[redacted]			
Mexico	2	7	11
Netherlands	2	49	9
Pakistan	1	24	3
[redacted]			
S. Africa	2	22	3
Slovak R.	4	61	5
Slovenia	1	14	3
Spain	9	147	2
Sweden	12	219	2
Switzerland	5	103	10
UK	35	1063	4
[redacted]			
USA	109	2028	8

Because of the lack of confidence that all large Turbine Building fires were reported by what was the USSR, the operating time for the plants located within the borders of the USSR are not included. Therefore, the operating time for the plants in Russia, Ukraine, Kazakhstan, Armenia and Lithuania are excluded in the total. The total number of operating years used in determining the frequency of a large Turbine Building fire is, therefore, 6860.58 years.

Large Turbine Fire Events

Reviewing operating experience data from the NRC, WANO and from the various U. S. Utilities, it was determined that two large Turbine Building fires have occurred. The two fires identified are:

- Fire which occurred on March 31, 1993 at the Narora Atomic Power Station, Unit 1 (NAPS-1). This fire was fueled by hydrogen, from the generator, and turbine lubricating oil. Due to heavy smoke, the Control Room was evacuated. The Operators were able to trip the reactor, and maintain adequate core cooling by using an alternate source of water, not normally used.

- A second major turbine/generator occurred on October 19, 1989, at the Vandellos Nuclear Power Plant, near Madrid Spain. Specific details are not readily available on this event. This event is included as a large turbine building fire.

These two fires are used to determine the frequency of a Large Turbine Fire. The frequency of a Large Turbine Building Fire is 2/6860.58 years or 2.91 E-4 per year.

Yard	**Location:**	Yard
	Fire Area:	Yard
	CDF:	3.53E-6

The Yard area consist of various transformers, tanks and several independent structures including the Emergency Diesel Generator 1A and 0C buildings, the Unit 1 and 2 13KV Switchgear Houses and the Fire Pump House. See Figure 4.6.2.6.

Transformers

Deluge systems are used to protect the yard transformers. The deluge automatic water spray system has two important features which distinguish it from the basic sprinkler system. First, the system is equipped with open nozzles and second, because there are no fusible sprinkler heads, a separate heat detection system is provided. The heat detection system supplies, at a predetermined temperature, an electrical signal which activities the deluge valve.

Emergency Diesel Generator Buildings

Emergency Diesel Generator (EDG) 1A and 0C buildings are treated as separate CCFPRA initiating events. The barriers between these buildings has been added to a surveillance program. See Section 7. In addition, the impact of a fire on the Auxiliary Building Roof is combined with 0C EDG building due to similar functional impact.

Fire Pump House

The Fire Pump House is also treated as a separate fire area due to its physical separation from other structures in the Yard.

Fire Analysis Results

Fifty fire scenarios are identified for the Yard. These scenarios identified in Table 4-U-1 are represented by thirteen fire initiating events shown in Table 4-U-2. The consolidation of fire scenarios is based on an assessment of the functional impact and ignition frequency of each scenario. The frequency of each initiator is the sum of the frequencies of all the fire scenarios it represents.

Table 4-U-1
Yard Fire Scenario Summary

Scenario	Transformer	Smoke Present at the intake for :	Propagate into TB	Frequency (events/yr)
1	U-25000-11	Any or all of: U-1 SWGR , CR HVAC 11, CR HVAC 12, U-2 SWGR	Yes	1.63E-06
2	U-25000-11	Any or all of: U-1 SWGR , CR HVAC 11, CR HVAC 12, U-2 SWGR	No	6.52E-06
3	U-25000-11	None	Yes	1.57E-04
4	U-25000-11	None	No	1.16E-03
5	U-25000-12	U-1 SWGR	Yes	1.75E-06
6	U-25000-12	U-1 SWGR	No	6.99E-06
7	U-25000-12	Any or all of: U-1 SWGR , CR HVAC 11, CR HVAC 12, U-2 SWGR	Yes	1.63E-06
8	U-25000-12	Any or all of: U-1 SWGR , CR HVAC 11, CR HVAC 12, U-2 SWGR	No	6.52E-06
9	U-25000-12	None	Yes	1.55E-04
10	U-25000-12	None	No	1.15E-03
11	U-4000-11	U-1 SWGR	No	2.04E-05
12	U-4000-11	Any or all of: U-1 SWGR , CR HVAC 11, CR HVAC 12, U-2 SWGR	No	8.15E-06
13	U-4000-11	None	No	1.29E-03
14	U-4000-21	U-1 SWGR	No	2.04E-05
15	U-4000-21	Any or all of: U-1 SWGR , CR HVAC 11, CR HVAC 12, U-2 SWGR	No	8.15E-06
16	U-4000-21	None	No	1.29E-03
17	U-22000-22	Any or all of: U-2 SWGR , CR HVAC 12, CR HVAC 11, U-1 SWGR	Yes	1.63E-06
18	U-22000-22	Any or all of: U-2 SWGR , CR HVAC 12, CR HVAC 11, U-1 SWGR	No	6.52E-06
19	U-22000-22	None	Yes	1.57E-04
20	U-22000-22	None	No	1.16E-03

Table 4-U-1
Yard Fire Scenarios Summary (Continued)

Scenario	Transformer	Smoke Present at the intake for :	Propagate into TB	Frequency (events/yr)
21	U-22000-21	U-2 SWGR	Yes	1.65E-06
22	U-22000-21	U-2 SWGR	No	6.60E-06
23	U-22000-21	Any or all of: U-2 SWGR , CR HVAC 12, CR HVAC 11, U-1 SWGR	Yes	1.63E-06
24	U-22000-21	Any or all of: U-2 SWGR , CR HVAC 12, CR HVAC 11, U-1 SWGR	No	6.52E-06
25	U-22000-21	None	Yes	1.55E-04
26	U-22000-21	None	No	1.15E-03
27	U-4000-12	U-2 SWGR	No	1.60E-05
28	U-4000-12	U-2 SWGR, CR HVAC 12, CR HVAC 11, U-1 SWGR	No	8.15E-06
29	U-4000-12	None	No	1.30E-03
30	U-4000-22	U-2 SWGR	No	1.60E-05
31	U-4000-22	U-2 SWGR, CR HVAC 12, CR HVAC 11, U-1 SWGR	No	8.15E-06
32	U-4000-22	None	No	1.30E-03
33	P-13000-11	None	No	1.32E-03
34	P-13000-21	None	No	1.32E-03
35	U-4000-13	None	No	1.32E-03
36	U-4000-23	None	No	1.32E-03
37	13KV11/12	Unit 1 13 KV Service Bus 11 and 12	No	2.47E-03
38	13KV21/22	Unit 1 13 KV Service Bus 21 and 22	No	2.75E-03
39	13KVSGR1	Unit 1 13KV Switchgear Isolation and Bypass Switches	No	8.70E-4
40	13KVSGR2	Unit 2 13KV Switchgear Isolation and Bypass Switches	No	8.68E-4

Table 4-U-1
Yard Fire Scenarios Summary (Continued)

Scenario	Transformer	Smoke Present at the Intake for :	Propagate into TB	Frequency (events/yr)
41	1H1101	Unit 1 Voltage Regulator	No	1.95E-03
42	1H1102	Unit 1 Voltage Regulator	No	1.95E-03
43	1H1103	Unit 1 Voltage Regulator	No	1.95E-03
44	2H2101	Unit 2 Voltage Regulator	No	1.95E-03
45	2H2102	Unit 2 Voltage Regulator	No	1.95E-03
46	2H2103	Unit 2 Voltage Regulator	No	1.95E-03
47	WWPPHS	Well Water Pump House	No	9.72E-4
48	0C EDG/AB Roof	0C Emergency Diesel Generator & Auxiliary Building Roof	No	2.50E-2
49	1A EDG	1A Emergency Diesel Generator	No	2.63E-2
50	FFPPHS	Fire Pump House	No	1.81E-2
* The fire that propagates into the Turbine Building is assumed to fail feedwater.				

Table 4-U-2
Yard Fire Analysis Results

Initiating Event	Fire Scenario	Frequency	Ignition Source	Functional Impact	CDF
FCYRD1	1, 7, 17, 23	6.52E-6	U-25000-11, 12, 21, or 22 where smoke present at any or all of U-1/U-2 SWGR, 11/12 CRHVAC and fire propagates to Turbine Bldg.	CR*, HF*, HG*, AD*, AC*	6.57E-8
FCYRD2	2, 8, 12, 15,18, 24, 28, 31	5.87E-5	U-25000-11, 12, 21, or 22 where smoke present at any or all of U-1/U-2 SWGR, 11/12 CRHVAC and fire does not propagate to Turbine Bldg.	CR*, OP, HF*, HG*, AD*, AC*	2.50E-7

Table 4-U-2
Yard Fire Analysis Results

Initiating Event	Fire Scenario	Frequency	Ignition Source	Functional Impact	CDF
FCYRD3	5	1.75E-6	U-25000-12, smoke present in U-1 SWGR	HF*, HG*	4.76E-9
FCYRD4	6	4.78E-5	U-25000-11, 12, 21 smoke present in U-1 SWGR	QC, QD, HF*, HG*	9.23E-9
FCYRD5	21	1.65E-6	U-22000-21, smoke present in U-2 SWGR	AD, AC	1.70E-8
FCYRD6	22, 27, 30	3.86E-5	U-22000-12, 21, 22 smoke present in U-2 SWGR	QE, QF, AD, AC	1.37E-9
FCYRDA	13, 16, 33, 34, 37, 39, 42, 43	1.12E-2	13KV Service Bus 11/12, 13KVSGR1 Unit 1 13KV SWGR Isolation & Bypass Switches, 1H1102 and 1H1103 Voltage Regulators, P13000-1 and U400011 & 12 Service Transformers, No smoke impact or propagation	OP, Y3, Y4	1.45E-6
FCYRDB	29, 32, 34, 38, 40, 45, 46	1.14E-2	13KV Service Bus 21/22, 13KVSGR2 Unit 1 13KV SWGR Isolation & Bypass Switches, 2H2102 and 2H2103 Voltage Regulators, P13000-2 and U400021 & 22 Service Transformers, No smoke impact or propagation	H5, Y3, Y4, DM PG*, T1, VC, LF	1.09E-6
FCYRDC	35, 41	3.27E-3	1H1101 Voltage Regulator and U4000-13 Service Transfomer	Y2, H5, Y3, Y4, QQ, DM, PG*, FN	1.05E-7
FCYRDD	36, 44, 47	4.24E-3	2H2101 Voltage Regulator, U4000-23 Service Transfomer, Well Water Pumps House	Y4, DM	1.97E-8
F0CEDG	48	2.50E-2	0C EDG and Auxiliary Building Roof	GJ, M3*, M1*, M2*, Y3, Y4	1.54E-7
F1AEDG	49	2.63E-2	1A EDG	GE, Y3	2.89E-7
FFPPHS	50	1.81E-2	Fire Pump House	M8, DM, PG	7.47E-8

Since Yard Fire Scenarios 3 and 9 propagate to the turbine building and are assumed to fail Unit 1 Main Feedwater, they are binned with the turbine building Initiator Event TBMFW1. Yard Fire Scenarios 19 and 25 which propagate to the turbine building and fail Unit 2 Main Feedwater are binned with turbine building Initiating Event TBMFW2. See Attachment 4-T for the core damage frequency contribition.

The only impact applicable to Fire Scenarios 4 and 10 is a Unit 1 plant trip. The frequencies of these two scenarios are much lower then the general transient plant trip initiator and are assumed to be included in this initiating frequency. Thus these scenarios are screened from further evaluation.

Fire Scenarios 20 and 26 cause a Unit 2 plant trip. This has no significant impact on the Unit 1 plant model and are screened from further evaluation.

Transformer Ignition Frequencies

Using Table 4.3.2, Yard Transformers are divided into three categories:

Category	Frequency per Year per Reactor Unit
Transformer Fires Propagating to Turbine Building	4.0E-3
Yard Transformer Fires with a Loss of Off-Site Power	1.6E-3
Other Yard Transformer Fires	1.5E-2

The EPRI database methodology adds several weighting factors to these frequencies to determine a fire frequency for the yard. The Location Weighting Factor considers a transformer in a switchyard with multiple units. The weighting factor is not appropriate for the main transformer and the 13KV/4KV transformer, since each transformer is servicing only one unit.

The Location Weighting Factor is not applied to transformers P-13000-11 and P-13000-21 since these transformers are separate from the switchyard and are considered for modeling purposes as two separate switchyards, one for Unit 1 and one for Unit 2.

The other significant weight factor is the Transient Source Fires. Since transient source fires were included in the data to determine the fire frequency per year per reactor unit, it is not necessary to include a separate contribution due to transient fires.

The first fire category is "Transformer Fires Propagating to the Turbine Building". The CCNPP transformers that are included in this category are:

1. U-25000-11, Unit 1 Main Transformer
2. U-25000-12, Unit 1 Main Transformer
3. U-22000-21, Unit 2 Main Transformer
4. U-22000-22, Unit 2 Main Transformer

The second transformer fire category is "Yard Transformer Fires with a Loss of Off-Site Power". The CCNPP transformers that are included in this category are:

1. P-13000-1, 13KV Service Transformer 11
2. P-13000-2, 13KV Service Transformer 12

The third transformer fire category is "Other Yard Transformer Fires". All remaining outside transformers in the power distribution system are included in this category.

Transformer Fires Propagating to the Turbine Building

Eight fire scenarios are developed for the four CCNPP transformers which can propagate to the Turbine Building. These scenarios include fire propagation into the Turbine Building and smoke from these fires entering the fresh air inlet ducts for the following system:

1. U-1 SWGR HVAC: Unit 1 Switchgear Room Heating, Ventilation and Air Conditioning System
2. U-2 SWGR HVAC: Unit 2 Switchgear Room Heating, Ventilation and Air Conditioning System
3. CR HVAC 11: The Control Room Heating, Ventilation and Air Conditioning, Train 11
4. CRHVAC 12: The Control Room Heating, Ventilation and Air Conditioning, Train 12

The impact of smoke entering either Switchgear room is the inadvertent actuation of the Halon system, isolating ventilation to the switchgear room. The impact of a large quantity of smoke entering the Control Room is that is the potential evacuation of the Control Room.

Because the HVAC fresh air ducts are aligned downwind of the transformers, separate fire scenarios are considered for the smoke entering each of the ducts consecutively.

The fire scenarios for U25000-11 or U-25000-12 are:

1. Transformer fire propagating to the Turbine Building and the transformer smoke entering the fresh air ducts for U-1 SWGR, CRHVAC 11, CR HVAC 12, and U-2 SWGR
2. Transformer fire not propagating to the Turbine Building and the transformer smoke entering the fresh air ducts for U-1 SWGR, CRHVAC 11, CR HVAC 12, and U-2 SWGR
3. Transformer fire propagating to the Turbine Building and the smoke having no impact
4. Transformer fire not propagating to the Turbine Building and the smoke having no impact

The fire scenarios for U22000-21 and U-22000-22 are identical except the order of the ducts is reversed.

The category "Transformer Fires Propagating to Turbine Building" includes five industry events. Each event was reviewed to determine if the event was of sufficient duration to generate smoke that could impact the HVAC fresh air ducts. The events were also reviewed to determine the number of events where the fire did spread to the Turbine Building.

All five industry events were of a duration to ensure sufficient smoke generation. No events are screened. For the five event, only two resulted in a fire impact on the Turbine building. The ignition frequency for "Transformer Fires Propagating to Turbine Building" is 4.0E-3 and given that a transformer fire occurs, the probability that the Turbine Building is impacted by the fire is 4.0E-1.

Transformer that Could Cause a Loss of Off Site Power

Smoke from the two transformers in this category, transformers P-13000-11 and P-13000-21, are located a significant distance the HVAC fresh air inlet vents. The rise in the Diesel Generator Building and Auxiliary Building roof provides a barrier to wind blown smoke. Also, because of these transformers are located a significant distance from the Turbine Building, a fire at either of these transformers can not propagate to the Turbine Building. Therefore, the only fire scenario for these transformers is the transformer burns with no smoke or fire impact to other structures.

For the category " Yard Transformers with Loss of Offsite Power", the ignition frequency is 1.63E-3. Since there is only one CCNPP fire scenario, no screening of industry events for this category was done.

Other Yard Transformer Fires

Two of the yard transformers, U4000-13 and U-4000-23, result in no smoke or fire impact to other structures. Like the P-13000 transformers, these transformers are located far from the Turbine Building and a fire, originating at either of these transformers can not propagate into the turbine building. Also, because of the location of these transformers, the turbine building provides a barrier to smoke, preventing the smoke from entering the HVAC fresh air ducts.

The other four transformers, U-4000-11, U-4000-21, U-4000-12 and U-4000-22, are located in the same general vicinity of the main transformers and are separated from the Turbine Building by a fire barrier. However, because of their location, smoke from these transformers, could impact the HVAC fresh air vents.

The fire scenarios for the U-4000-11 or U-4000-21 transformers are:

1. Transformer fire not propagating to the Turbine Building and the associated smoke entering the fresh air ducts for U-1 SWGR.
2. Transformer fire not propagating to the Turbine Building and the associated smoke entering the fresh air ducts for U-1 SWGR, CRHVAC 11, CR HVAC 12, and U-2 SWGR
3. Transformer fire not propagating to the Turbine Building and the smoke having no impact

The fire scenarios for the U4000-12 or U-4000-22 are identical except in Case 1, U-2 SWGR fresh air intake is impacted as opposed U-1 SWGR.

Eighteen events have occurred in the category "Other Yard Transformer Fire". For the transformers, U-4000-11, U-4000-21, U4000-12 and U-4000-22, ten events were screened. Four events were screened because of the short duration of the fires. These events would not have generated sufficient smoke to be of concern. One event was screened because no fire ignited. A trip of the deluge system on high transformer pressure occurred. Four events were screened because of the fires did not involved oil and therefore would have not generated enough smoke to be of concern. One event screened because it did not involved a yard transformer. It involved a transformer located with a structure, in this case the Diesel Generator Building.

Determination of Smoke Impact Frequency

Burning one of the six transformer located near the fresh air inlet for the HVAC does not guarantee that smoke will be present at the intake. Because of the distance from the transformer to the intakes, the wind must blow in the correct direction and at the correct speed to force the smoke plume to the inlet duct.

A simple model was developed to determine the frequency of smoke in the vicinity of the HVAC fresh air intakes. The smoke from the fire is treated as a one dimensional object with a vertical velocity. The vertical speed of the smoke plume was estimated using equations from the SFPE Handbook of Fire Protection Engineering (Reference 4-16). These equations yield a variety of results for different ambient conditions, size of fire, vaporization rate of fuel, and other input variables. For the analysis, a conservative estimate of 6 feet/sec was used for the plume speed.

The vertical and horizontal distance that the smoke is required to travel be at the duct is determined. Table 4.6.2.6.Yard.c summarizes the distances from the transformers to the ducts.

Table 4-U-3
Transformer/Intake Distance Determination
(Distance in Feet)

		U-1 HVAC	CR HVAC 11	CR HVAC 12	U-2 SWGR
U-25000-11	x distance	229	308	323	461
	y distance	30	30	30	30
U-25000-12	x distance	160	239	254	392
	y distance	30	30	30	30
U-4000-11	x distance	122	201	216	354
	y distance	30	30	30	30
U-4000-21	x distance	122	201	216	354
	y distance	30	30	30	30
U-22000-22	x distance	461	323	308	229
	y distance	30	30	30	30
U-22000-21	x distance	392	254	239	160
	y distance	30	30	30	30
U-4000-12	x distance	354	216	201	122
	y distance	30	30	30	30
U-4000-22	x distance	354	216	201	122
	y distance	30	30	30	30

Knowing the vertical speed of the plume and the vertical distance to the ventilation intakes, the time required for the plume to reach the required height is calculated. Knowing the elapsed time and the horizontal distance the plume must cover to reach the duct, a horizontal velocity of the plume is determined. If it is assumed that the horizontal vector of the plume is due entirely to wind, a wind velocity and required direction can be determined.

Independent Spent Fuel Storage Installation Updated Environmental Report (Reference 17) provides three years of historical meteorological data. This data was used to determine the frequency of wind speed and directions. A summary of wind data is given in Table 4.6.2.6.Yard.d

Table 4-U-4
Historical Wind Speed and Direction
(Wind Speed in MPH)

	1-3	4-7	8-12	13-18	19-24	>24
N	1.00%	3.39%	4.20%	0.84%	0.04%	0.02%
NNE	0.86%	3.64%	3.14%	0.56%	0.01%	0.01%
NE	0.80%	2.33%	1.82%	0.41%	0.01%	0.01%
ENE	0.95%	2.14%	0.42%	0.03%	0.01%	0.01%
E	1.21%	1.40%	0.23%	0.01%	0.01%	0.01%
ESE	1.04%	1.19%	0.20%	0.03%	0.01%	0.01%
SE	1.04%	1.51%	0.46%	0.03%	0.01%	0.01%
SSE	1.06%	2.50%	1.80%	0.24%	0.01%	0.01%
S	1.38%	3.19%	1.84%	0.26%	0.02%	0.01%
SSW	1.75%	3.94%	3.95%	1.00%	0.06%	0.02%
SW	1.61%	4.85%	5.16%	1.14%	0.04%	0.01%
WSW	1.44%	2.80%	1.63%	0.22%	0.02%	0.01%
W	1.31%	1.75%	0.89%	0.22%	0.03%	0.01%
WNW	1.13%	1.93%	1.47%	0.51%	0.03%	0.01%
NW	0.90%	2.90%	2.33%	0.73%	0.05%	0.01%
NNW	0.90%	2.35%	2.34%	0.32%	0.01%	0.01%
Calm:	1.01%					

It should be noted that where the ISFSI data indicated 0.00%, a value of 0.005% probability was assigned.

For the transformers next to Unit 1's Turbine Building and Containment (U25000-11, U-25000-12, U4000-11, U4000-21), the wind directions of concern are ESE, SE and SSE. For the transformers next to Unit 2's Turbine Building and Containment (U22000-21, U-22000-22, U-4000-12, U-4000-22), the wind directions of concern are WNW, NW, and NNW.

Table 4-U-5
Probability of Smoke Impact due to Wind

Transformer	Direction		U-1 SWGR	CR HVAC 11	CR HVAC 12	U-2 SWGR
U-25000-11	SE	Speed	31	42	44	63
		Probability	0.02%	0.02%	0.02%	0.02%
U-25000-12	SE	Speed	22	33	35	54
		Probability	0.09%	0.02%	0.02%	0.02%
U-22000-21	NW	Speed	54	35	33	22
		Probability	0.03%	0.03%	0.03%	0.03%
U-22000-22	NW	Speed	63	44	42	31
		Probability	0.02%	0.02%	0.02%	0.02%
U-4000-11,21	SE	Speed	17	27	29	48
		Probability	1.57%	0.02%	0.02%	0.02%
U-4000-12,22	NW	Speed	48	29	27	17
		Probability	0.02%	0.02%	0.02%	1.0%

Reviewing the ISFSI USAR (Reference 18), it was determined that a temperature inversion occurs approximately 6% of the time. As part of the analysis, it is assumed, during a temperature inversion and calm conditions, that smoke from any of the six transformers will enter the duct. Calm conditions occur about 16.9% of the time during temperature inversions. Therefore, in addition to the wind impact, an estimated 1% likelihood exist that smoke from the transformers will infiltrate the ventilation intakes due to a calm temperature inversion.

Determination of Transient Fire Frequency

The preceding evaluation determined the frequency, given a transformer fire occurs, where smoke would impact the HVAC system(s) for the Control Room, Cable Spreading Room and Switchgear Room(s). The following evaluation determines a Transient Fire Frequency which could propagate to a transformer causing damage to the transformer.

Field walkdowns showed that there are no transient combustibles or ignition sources within the vicinity of any transformer which could cause a fire. Additionally, the closest transient combustibles will not sustain a fire of significant duration to cause transformer damage. Procedure controls prohibit the storage of transient combustibles within 20 feet of safety related components. Hot work is strictly controlled to preclude the possibility of a fire developing near a transformer.

EPRI Fire Events Data Base showed that all transformer related fires during power operation are a result of transformer failures or faults associated with the electrical distribution system. At CCNPP, there have been no occurrences of fires started from a transient ignition source leading to damage of a transformer or related equipment.

Based on field walkdowns, procedure controls, industry fire events (EPRI Fire Events Data Base), and plant history, it can be shown that a transient induced fire causing damage to a transformer is not likely to occur. Therefore, the Transient Fire Frequency for all Yard transformers is 0.00E+00.

Other Fire Area Ignition Frequencies

In addition to the transformer ignition frequencies discussed above, there are five other Yard Area categories evaluated below.

1. Unit 1 & Unit 2 Diesel Fuel Oil Storage Tanks 11 & 21
2. Unit 1 & Unit 2 Demin Water, Pretreat Water Refueling Water and Condensate Water Storage Tanks
3. Bulk Hydrogen and Nitrogen Storage Tanks
4. Well Water Pump House and Switch Yard

Unit 1 & Unit 2 Diesel Fuel Oil Storage Tanks 11 & 21

The possible transient ignition source is considered to be Hot Work (i.e., welding, cutting, grinding) for these plant locations. Based on established controls for Hot Work, and industry history data for transient induced fires associated with diesel fuel storage tanks, it is not likely that fire could damage either Diesel Fuel Storage Tank 11 or 21 that would lead to an increase in CDF. There are no other credible in-situ ignition sources for these plant locations. Therefore, the ignition frequency for these plant locations is set to 0.00E+0.

Unit 1 & Unit 2 Water Storage Tanks

The Unit 1 Water Storage Tanks consist of CDSTK 11 & 12; PTWSTK 11; DEMNTK 11 and RFWTK 11. The possible transient ignition source is considered to be Hot Work (i.e., welding, cutting, grinding) for these plant locations. Based on established controls for Hot Work, and industry history data for transient induced fires associated with water storage tanks, it is not likely that fire could damage the above mentioned tanks that would lead to an increase in CCDF. Therefore, the transient ignition frequency for these plant locations are set to 0.00E+0. The associated room (A439) for the RFWTK was evaluated separately and screened. There are in-situ ignition sources for RFWTK 11 which are associated with electrical cabinets for a frequency of 2.81E-04.

The Unit 2 Water Storage Tanks consist of CDSTK 21; PTWSTK 12; and RFWTK 21. The possible transient ignition source is considered to be Hot Work (i.e., welding, cutting, grinding) for these plant locations. Based on established controls for Hot Work, and industry history data for transient induced fires associated with water storage tanks, it is not likely that fire could damage the above mentioned tanks that would lead to an increase in CDF. Therefore, the ignition frequency for these plant locations are set to 0.00E+0. The associated compartment (A440) for the RFWTK was evaluated separately and screened. There are in-situ ignition sources for RFWTK 21, associated with electrical cabinets; and PTWSTK 12 which are associated with electrical cabinets, and equipment related to water treatment trailer for fire frequencies of 2.81E-04 and 3.26E-03 respectively.

Bulk Hydrogen Storage Vessels and Nitrogen Storage Tank

The Hydrogen Storage Vessel supplies the H_2 for both units generator for cooling. The Nitrogen Tank supplies N_2 for both units main generators for purging. The possible transient ignition source is considered to be Hot Work (i.e., welding, cutting, grinding) for these plant locations. Based on established controls for Hot Work, and industry history data for transient induced fires associated with hydrogen storage vessels or the nitrogen storage tank, it is not likely that a transient induced fire could damage either the Hydrogen Storage Vessels or the Nitrogen Storage Tank would contribute to the CDF. There are no other credible in-situ ignition sources for these locations. Therefore, this ignition frequency for these plant locations are set to 0.00E+0.

Well Water Pump House and Switch Yard

The Well Water Pump House is located at the north west corner of the plant. This building does not have early warning detection or suppression. Because it is possible for transient ignition in this building, and there are several pumps within close proximity of each other in addition to stored combustibles in this building, a transient fire frequency of 3.45E-4 was assigned to this location. The total fire frequency includes the in-situ ignition sources (6.26E-4) and the transient ignition sources for a total of 9.72E-4.

The Switchyard is remotely located from the plant to the west. The frequency of fires in the Switchyard is determined to be 1.96E-02 fires per year. The hardware impact of a Switchyard fire is conservatively assumed to be a loss of offsite power. The human actions impact is relatively insignificant. Although the fire may cause some additional confusion in the Control Room, due to communication with the fire brigade, it will not be significantly different then any other loss of offsite power. The outside operator would not be directly impacted. The switchyard is relatively remote from the protected area and it is extremely unlikely that smoke from the switchyard could obscure his access to any areas. The loss of offsite power initiating event frequency for the general transient module is developed from industry data which includes fire failures. Thus the switchyard fire impact is addressed in the general transient module and is not further evaluated.

Subdividing of the Appendix R Yard Fire Area

The Appendix R Yard Fire Area consist of all outside storage tanks, Fire Pump House, Well Water Pump House, 1A EDG and 0C EDG buildings, and transformer areas for both units. This is divided into smaller areas to take credit for the spatial separation of these various areas. This spatial separation allows for determining the likelihood of fire propagation and failure probabilities which are inherently more realistic than the methodology used for the Appendix R analysis, which assumes a total loss of all equipment and components in the Yard.

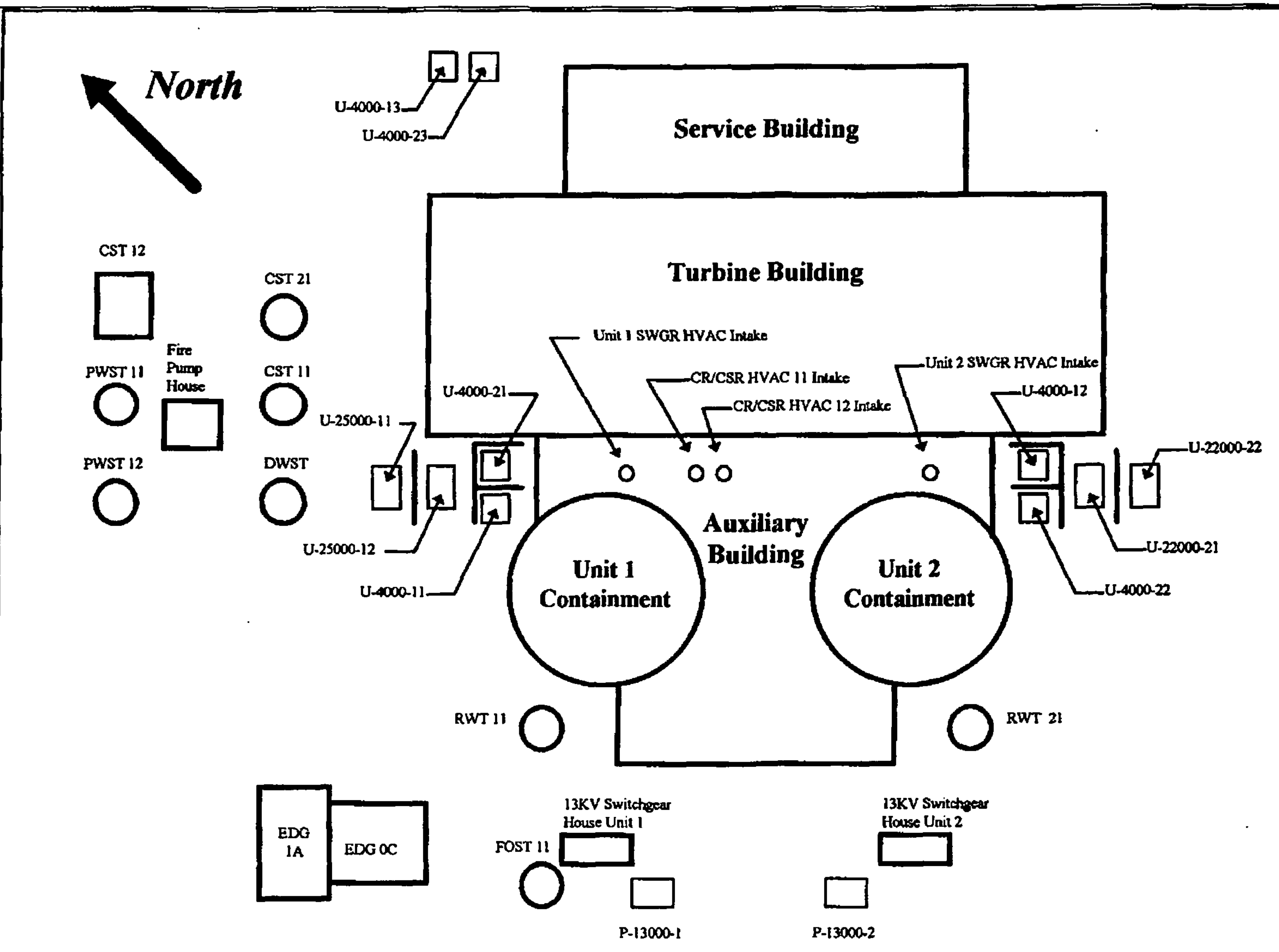


FIGURE 4-U-1 Plant Layout (Transformer Locations)
Not to Scale

SECTION 5

HIGH WINDS, FLOODS, AND OTHER EXTERNAL EVENTS

5.0 INTRODUCTION

The Individual Plant Examination of External Events (IPEEE) requires an evaluation of the impact on the plant of hazards that are external to it. The hazards are classified into seismic, fire, and other. In NUREG-1407 (Ref. 5-1), concluded that, of the other external events, only the following needed to be considered on a plant specific basis: high winds, external floods, and transportation and nearby facility accidents. However, there is also a requirement to perform a review to confirm that there are no external hazards unique to the site, that would invalidate the conclusions of NUREG-1407. This includes, for example, assessing the possibility that lightning strikes are likely to cause more than just a loss of offsite power or reactor trip. This section of the report documents the analysis of other external hazards for the Calvert Cliffs Nuclear Power Plant (CCNPP).

The approach used follows the method described in NUREG/CR-4839 (Ref. 5-2) and NUREG-1407 by performing an initial screening analysis, followed by bounding or detailed analyses as necessary. Thus the process used basically follows the steps outlined in Figure 5.1 of NUREG-1407. In each specific analysis, whenever it is necessary, evaluation of meeting the 1975 Standard Review Plan (SRP) criteria and identification of significant changes since Operating License (OL) issuance is performed.

Section 5.1 provides a brief site description of CCNPP. Section 5.2 describes the initial screening analysis approach and presents the conclusions of that screening which confirms the conclusions of NUREG-1407. Sections 5.3, 5.4, 5.5, and 5.6 discuss high winds, floods, transportation and nearby facility accidents and turbine missiles respectively. Conclusions are given in Section 5.7 and references in Section 5.8.

5.1 Generic Plant Description

5.1.1 Site Description

The site is located in Calvert County, MD, approximately 10-1/2 miles southeast of Prince Frederick, MD, and on the west bank of the Chesapeake Bay. It covers an area of approximately 1,135 acres. Calvert County is a peninsula bounded on the east by the Chesapeake Bay and on the west by the Patuxent River. The metropolitan centers closest to the site are: Washington, DC, approximately 45 miles to the northwest; Baltimore, MD, approximately 60 miles to the north; Richmond, VA, approximately 80 miles to the southwest; and Norfolk, VA, approximately 110 miles to the south.

The cooling water for the plant is drawn from and returned to the Chesapeake Bay. The exclusion area around the plant has a minimum radius of 1,150 meters. The distance to the nearest permanent residence is approximately one mile. There are several communities within a three mile radius of the plant. In addition, there are three operating airports within 10 miles of the plant. There are no missile bases located within ten miles of the plant. The region surrounding the site is predominantly rural in character and is sparsely populated with the exception of localized areas along the coast of the Chesapeake Bay which attract many summer residents.

The topographical location of the site makes it susceptible to tornadoes, thunderstorms, and freezing precipitation. In addition, approximately one hurricane per year poses a threat to the area and about one hurricane every 10 years produces significant effects (Ref. 5-3). Northeasters, or extratropical storms, can also influence the area in terms of flooding of low-lying land. However, the detrimental effects of northeasters are considerably less than those postulated for hurricanes in the site area.

5.1.2 Identification of Structures, Systems and Components Susceptible to External Events

The first step in performing an external events analysis is the identification of structures, systems or components which are susceptible to damage and which, if damaged, could lead to a loss of capability to safely shut down the reactor. The Individual Plant Examination addresses the equipment necessary to achieve safe shutdown to hot standby. The majority of this equipment is safety-related and protected by major structures. The major Class I structures designed to withstand tornado impact are: reactor containment building; Auxiliary Building including the Control Room area, HVAC rooms, Switchgear Rooms, and fuel handling area; diesel generator building; intake structure; one CST tank concrete structure; and one diesel fuel oil storage tank. In addition, the new safety-related diesel generator, including its fuel oil storage tank, is protected by virtue of being enclosed in a tornado-proof structure. Also protected by a tornado missile shield is the cable raceway into the Auxiliary Building. All the Class I structures were designed to meet both earthquake and tornado design criteria. However, there are systems and components that are needed for safe shutdown following the types of initiating events that could arise from external events, typically transients, including loss of offsite power, which are not protected by these structures and are consequently vulnerable to these same external events. These include:

- Switchyard
- Main transformers
- Power conversion system which is housed in the Turbine Building (sheet metal sided steel structure designed to a lower capacity than the Class I structures)
- Condensate storage tanks (CST)11 and 21
- Fuel oil storage tank (FOST)11
- Refueling water storage tanks (RWT) 11 and 21
- Demineralized water storage tank (DWT)
- Pretreated water storage tanks (PWST) 11 and 12
- Service water head tanks, both Units
- Fire pump house
- Top structure on the traveling screens (located on the intake structure and enclosed within metal casings)
- Emergency diesel generator (EDG) combustion air intakes and exhaust assemblies for the three original EDGs
- New Station Blackout (SBO) diesel generator
- HVAC condenser units for the Control Room, and for the Switchgear Rooms of both units, located on the roof of the Auxiliary Building

The loss of the switchyard and/or transformers would cause a loss of offsite power to the plant. The loss of offsite power is covered by the CCNPP IPE (Ref. 5-4). Therefore, the effects of "other external events" on the switchyard and the various transformers is only specifically addressed in this analysis when other essential equipment is affected simultaneously. It is assumed that the contributions from external events that only cause a loss of offsite power are already included in the loss of offsite power with long recovery times.

In addition to potential damage to the above equipment, there is a concern as to whether there exists the possibility for external agents to impact the Class I structures sufficient to damage the equipment they contain. This impact may be a result of a direct impulsive force, or by ingress of harmful agents through penetrations in the structures. The most obvious of the latter are water due to flooding, or toxic gases leading to Control Room habitability problems. The effects of each possible external event on CCNPP structures/systems/components are discussed in the following sections.

5.2 Screening of External Hazards

5.2.1 Description of Approach

The objective of the screening analysis is to provide confirmation of the NUREG–1407 conclusion that there are no hazards unique to the site that require evaluation other than those posed by high winds, external floods, and transportation and nearby facility accidents.

The PRA Procedures Guide (Ref. 5-5) provides an exhaustive list of potential external hazards which provides the starting point for the screening. The screening is performed by reviewing the information on the site region and plant design to identify external events that are applicable using the screening criteria below. The data in the safety analysis report on the geologic, seismologic, hydrologic, and meteorological characteristics of the site region as well as present and projected industrial activities (i.e., the building of a reservoir, increases in the number of flights at an airport, construction of a road that carries explosive materials, etc.) in the vicinity of the plant are reviewed for this purpose. The set of screening criteria has been formulated to minimize the possibility of omitting significant risk contributors while reducing the amount of detailed analyses to manageable proportions. The following screening criteria have been adopted from those given in the PRA Procedures Guide and NUREG/CR-4839 (Ref. 5-2). The screening criteria are valid whether or not the plant meets the Standard Review Plan criteria.

An external event is excluded if:

1. The event is of equal or lesser damage potential than the events for which the plant has been designed. This requires an evaluation of plant design bases in order to estimate the resistance of plant structures and systems to a particular external event. For example, it is shown by Kennedy, Blejwas, and Bennett (Ref. 5-6) that safety-related structures designed for earthquake and tornado loadings in Uniform Building Code Zone 1 can safety withstand a 3.0 psi static pressure from explosions. Hence, if the PRA analyst demonstrates that the overpressure resulting from explosions at a source (e.g., railroad, highway or industrial facility) cannot exceed 3 psi, these postulated explosions need not be considered.

2. The event has a significantly lower mean frequency of occurrence than other events with similar uncertainties and could not result in worse consequences than those events. For example, the PRA analyst may exclude an event whose mean frequency of occurrence is less than some small fraction of those for other events. In this case, the uncertainty in the frequency estimate for the excluded event is judged by the PRA analyst as not significantly influencing the total risk.

3. The event cannot occur close enough to the plant to affect it. This is also a function of the magnitude of the event. Examples of such events are landslides, volcanic eruptions and earthquake fault ruptures.

4. The event is included in the definition of another event. For example, storm surges and seiches are included in external flooding; the release of toxic gases from sources external to the plant is included in the effects of either pipeline accidents, industrial or military facility accidents, or transportation accidents.

In addition to these, another criterion is added:

5. The event is slow in developing and there is sufficient time to eliminate the source of the threat, or to take precautionary measures to minimize the consequences.

5.2.2 Results of Screening Analysis

Each of the external hazards listed in the PRA Procedures Guide was reviewed. Based on information in the UFSAR and on the information gathered during the walkdown, it was determined that the conclusions of NUREG–1407 were valid for CCNPP, namely that there are no other known plant-unique external events that pose a significant threat of severe accidents within the context of the NUREG–1407 screening approach. The screening criteria invoked for each of the events is documented in Table 5.2.2. In the case of lightning, the plant history was examined to determine whether there have been any lightning strikes that have caused damage other than reactor trips (e.g., spurious actuations). None are identified and it is concluded that there is no evidence for concern about lightning and it can be screened per NUREG-1407 guidelines. The next three sections discuss the potential for significant impact of the three hazards identified by NUREG–1407 for plant-specific evaluation, namely, high winds, external floods, and transportation and nearby facility accidents. A detailed analysis of turbine missile impact is described in Section 5.6.

5.3 High Winds

The risk assessment of high winds includes the impact of hurricanes, tornados and associated missiles. In general the approach used considers: (1) the frequency and intensity of high winds which may strike CCNPP, (2) the vulnerability of plant structures and components to high winds, and (3) the impact on core damage frequency which is calculated using the updated IPE plant model. The details of the above approach are described in the following sections.

5.3.1 Design of Plant Structures for High Winds

All structures are designed for a wind velocity of 100 mph, 30 feet above ground, in accordance with Figure 1(a) of ASCE Paper No. 3269. Wind velocity is varied with height in accordance with Table 1(a) in the paper, and a gust factor of 1.1 is used (Ref. 5-3).

In addition, all structures housing critical equipment (required to assure safe shutdown of the reactor) are designed for tornado loading (not coincident with accident or earthquake) on the following basis:

(a) The velocity components are applied as wind on the structure, with a maximum tangential velocity of 300 mph, traveling at 60 mph. However, the maximum effective velocity on the structure is 300 mph.

(b) The tornado-induced pressure differential is applied as 3 psi positive (bursting) pressure occurring in three seconds (1 psi/sec), followed by a calm for two seconds, and a repressurization to atmospheric pressure at a rate of 1 psi/sec. The containment is the only structure that is designed to withstand the bursting pressure. All other structures are vented to maintain pressure equalization inside and outside.

(c) A missile impingement at any height is used equivalent to a 4 in. by 12 in. by 12 ft. long wood plank (108 lbs.) traveling end-on at 300 mph, or a passenger automobile (4,000 lbs.) flying through the air at 50 mph and not more than 25 feet above the ground with a contact area of 20 square feet.

While the design basis is not in accordance with Regulatory Guide 1.76, (Ref. 5-40) which specifies a maximum wind speed of 360 mph, a maximum translational speed of 70 mph, and a rate-of-pressure drop of 2 psi/second for a plant in Tornado Region I, the likelihood of a tornado with maximum windspeeds approaching 300 mph is considered to be vanishingly small. It should be noted that the structure housing the new safety-related diesel is designed to the higher design basis.

5.3.2 Identification of Structures and Components Susceptible to Wind Damage

The majority of the equipment necessary to achieve safe shutdown to hot standby is safety-related and, as such, is protected by being enclosed within the major structures, namely the Containment Building, Auxiliary Building (which houses the Control Room), diesel generator rooms for the three original diesel generators, and the Intake Structure. These are seismic Class I structures, and are also designed to withstand tornado impact, both from the direct effect of the winds and from impact by wind-generated missiles. In addition, at CCNPP, one of three condensate storage tanks (Tank 12), and one of two diesel generator fuel oil storage tanks (Tank 21) are protected against tornado missiles by being enclosed within concrete structures. One of the two new diesel generators is enclosed within a tornado proof structure and the cable raceway into the Auxiliary Building is enclosed by a tornado missile shield.

For the intake structure roof, it is at an elevation of 28 ft.-6 in. and made of 12 in. reinforced concrete slab and metal decking. Only large missiles can penetrate this kind of concrete slab. The only apparent sources of large missiles at an elevation higher than the intake are the wooded areas to the north and south of the intake. However, due to the acute angle required to strike the intake, it is unlikely the missiles generated from the woods can penetrate the intake structure roof. Further, per NUREG-5042 (Ref. 5-44), an F-4 intensity tornado is required to generate large missiles. Per NUREG-4461 (Ref. 5-45), tornadoes of F-4 intensity or greater are located in the central part of the U.S. considerably west of the Appalachian Mountains. Therefore, it is concluded that the intake structure roof is of no concern to the tornado missiles.

In addition, as discovered during the walkdown of the plant, there are doorways and air intake vents into the original diesel generator rooms, and doorways into the Auxiliary Building and intake structure which do not appear from the outside to be tornado missile-proof. In the case of the Auxiliary Building, however, there is no vital equipment on the 45 ft. level where the doorways are located. In the case of the diesel generator rooms, architectural drawing No. A-41 shows concrete barriers behind the doors which would prevent a direct strike from a missile on any equipment in the room. In the case of the intake structure, there is another steel door behind the outside door. This double door arrangement mitigates against a direct missile strike on the equipment inside the intake structure. In addition, the outside door is a very small target which is protected to some extent by its location near the retaining wall adjacent to the intake structure.

However, as identified in Section 5.1.2, there are components that are required for the safe shutdown of the units which are not protected by the structures discussed above and are, therefore, potentially vulnerable to external influences. The structures and components that are susceptible to the hurricanes and tornados are summarized below.

5.3.2.1 Hurricane Winds

Control and Switchgear Rooms HVAC

The air conditioning (AC) condenser units and air intake ducts for the Control Room, Cable Spreading Room, and the Switchgear Rooms for both units are located on the roof of the Auxiliary Building, adjacent to the Turbine Building. The Turbine Building, however, is a potential source of missiles during hurricanes and tornadoes, in that the sheet metal siding is designed to collapse when a differential pressure of 0.45 psi exists (for the Main Steam Line Break event). This differential pressure is assumed to occur with a hurricane wind speed of 100 mph. The collapsed metal siding is assumed to damage the AC condenser units and the air intake ducts (i.e., the striking probability is assumed to be 1.0), which in turn, would result in losing cooling capability of all the HVAC systems. The Switchgear Room HVAC can also be lost due to losing the ducts and dampers inside the main plant exhaust equipment rooms on the 69 ft. level for both units. These rooms are part of the Auxiliary Building and also have metal siding walls. The metal sidings are assumed to collapse at a wind speed of 100 mph and expose the ducts and dampers to damage from the hurricane winds.

However, the SWGR HVAC in the 45 ft. and 27 ft. levels can be recovered by pre-staging and starting the emergency fans and portable generator within eight hours of the anticipated hurricane arrival. This recovery action is discussed in more detail in Section 5.3.5.

Emergency Diesel Generators

The three original emergency diesel generators 1B, 2A and 2B are located in individual rooms attached to the west side of the Auxiliary Building and are thus protected from the direct wind effects and, for the most part, to the impact of missiles. However, the combustion air intake structures and exhaust piping are located on the roofs of the diesel generator rooms. The roofs are part of the Auxiliary Building. Engineering calculations demonstrate that these components can withstand the direct effect of design basis tornadic winds (and by inference hurricane winds). The remaining concern, therefore, is the potential for damage from wind-generated missiles. These components are, to a large extent, protected by a high wall on the east side, by the Auxiliary Building to the east and north (Unit 1), and east and south (Unit 2) and by the RWT at the northwest (Unit 1), or southwest (Unit 2) corners allowing only a small gap through which low flying missiles could strike the components. Metal siding missiles originating from the adjacent Auxiliary Building are judged not to have sufficient energy to block the diesel generator intakes or exhausts during hurricane events.

The new safety-related diesel generator (Diesel Generator 1A) is tornado-proof even to the extent of the cables into the Auxiliary Building being missile protected. Likewise they are also hurricane-proof. Also there is a protected fuel oil storage tank within the Diesel Generator 1A Building that can supply fuel oil for seven days. While this diesel has enough capacity to provide power to two units, it is configured to power Unit 1 only.

The new station blackout diesel (Diesel Generator OC) is enclosed in a building which is not protected against tornado or hurricane impact and is assumed to have failed because the collapsed Turbine Building metal sidings can cause damage to the EDG 0C cable trays located on the roof of the Auxiliary Building.

Unprotected Tanks in the Tank Farm

The unprotected tanks are: Condensate Storage Tanks 11 and 21, Demineralized Water Storage Tank 11, Pretreated Water Storage Tanks 11 and 12, Refueling Water Storage Tanks 12 and 21 and the Fuel Oil Storage Tank. These tanks are assumed to survive the hurricane winds.

Fire Pump House

The Fire Pump House is located within the Tank Farm area but is assumed not to survive the hurricane winds. Since the fire protection water provides the make-up for Service Water (SRW)/Component Cooling Water (CCW), SRW/CCW is assumed to have failed when the Fire Pump House fails.

5.3.2.2 Tornado Winds

Control and Switchgear Rooms HVAC

The Control and Switchgear Rooms HVAC is assumed to be lost due to losing the air conditioning condensor units and air intakes for both Units 1 & 2 on the Auxiliary Building roof because of the collapsed metal siding from the Turbine Building. The Switchgear Room HVAC can also be lost due to losing the ducts and dampers inside the main plant exhaust equipment rooms on the 69 ft. level for both Units. These rooms also have metal siding walls which are assumed to collapse and would allow the ducts and dampers to be exposed to the tornado winds and be damaged. The SWGR HVAC is assumed not recoverable from tornado winds due to the limited available recoverable time.

Emergency Diesel Generators

EDG 0C is assumed to have failed because the building itself is not designed for the tornado winds. Additionally, the cable trays for EDG 0C on the Auxiliary Building roof are assumed to be damaged. EDGs 1A, 2A and 2B are not damaged because the air intake structure and exhaust pipe located on the Auxiliary Building roof can survive the tornado winds. Also, the air intake structure and exhaust pipe are configured in such a way that the collapsed metal siding from the Auxiliary Building siding will not have sufficient energy to destroy or block them.

Unprotected Tanks in the Tank Farm

The unprotected tanks are assumed to survive the tornado winds.

Fire Pump House

The Fire Pump House is located within the Tank Farm area but is assumed not to survive the tornado winds. Since the fire protection water provides the make up for SRW/CCW, SRW/CCW is assumed to have failed when the Fire Pump House fails.

5.3.2.3 Tornado Missiles

Control and Switchgear Rooms HVAC

The air conditioning condenser units and air intakes located on the Auxiliary Building roof for the Control and Switchgear Rooms' HVAC for both units are somewhat protected by the containment structures. They are also high up so that the only missiles that can reach this equipment must either be elevated by more than 46 ft. or originate at that level. Potential sources of missiles in the proximity of the ventilation equipment include the stairways crossing the ducting on the roof, and other ventilation structures such as exhaust stacks. These are, however, well anchored, presumably to prevent their becoming missiles. Nonetheless, the roof-top AC equipment is not protected from the missiles generated from the Turbine Building metal siding and is assumed to be lost.

The Switchgear Room HVAC can also be lost due to losing the ducts and dampers inside the main plant exhaust equipment room on the 69 ft. level because of the metal siding walls. Tornado missiles can penetrate these metal siding walls and cause damage to the ducts and dampers. This event is assumed not recoverable because very limited recovery time is available from the tornado missiles.

Service Water Head Tanks

The SRW Head Tanks are located inside the main plant exhaust equipment rooms on the 69 ft. level for both units. These rooms also have metal siding walls and are not tornado-missile protected. It is assumed that the tornado missiles can penetrate the metal siding walls and puncture the tanks. Losing one tank due to leakage can result in losing the other tank due to the cross-connection in the return lines in the SRW pumps suction side. However, losing the SRW system, due to losing the head tanks, does not contribute significantly to the CDF.

Unprotected Tanks in the Tank Farm

The unprotected tanks can be damaged by tornado missiles. The damage probabilities for these tanks, taking into consideration the location dependencies, are discussed in Sections 5.3.4 and 5.3.6.1 below.

Emergency Diesel Generators

EDG 0C is assumed to have failed because the building itself is not designed for tornado missiles. Additionally, the cable trays on the Auxiliary Building roof are assumed to be lost. EDGs 1B, 2A and 2B are assumed to have failed due to losing their air intake structure and exhaust pipe located on the Auxiliary Building roof from tornado missiles.

Fire Pump House

The Fire Pump House is located within the Tank Farm area but is assumed not to survive the tornado missiles. Since the fire protection water provides the make up for SRW/CCW, SRW/CCW is assumed to fail when the Fire Pump House fails.

5.3.3 Impact of High Winds on the Plant

The most likely immediate impact given high winds at the site is a loss of offsite power, which could be prolonged as a result of structural damage to transmission lines or the switchyard. Given that this is the case, it is assumed to be a two-unit loss of power, that is not recoverable within 24 hours. Following any transient, such as that caused by a loss of offsite power, the principal requirement for safe shutdown is that there be a reliable means of accomplishing decay heat removal.

Decay heat removal is provided by the AFW system. The components of the AFW system are protected against high winds by being inside the Auxiliary Building or in seismic Class 1 rooms within the Turbine Building. In addition, one of the sources of water for the AFW system, CST 12, is protected against tornados and will provide enough cooling to maintain both units in hot standby for six hours. However, there is insufficient inventory to achieve shutdown cooling conditions for both units. Therefore, the remaining CSTs are important, but since they are not protected they must be considered vulnerable. If the AFW functions were lost, feed-and-bleed is an alternative, which in turn requires an intact RWT. However, the RWT is not protected and has a limited capacity and would require switchover to the containment recirculation in a few hours, which in turn requires the salt water and service water systems to be operable.

To support long term operability of AFW in the event of a prolonged loss of offsite power, it is necessary to maintain emergency AC power (one diesel generator for each unit) for an extended period of time, which could be well in excess of 24 hours as evidenced by the impact of Hurricane Andrew on Turkey Point. In addition to the diesel generators themselves, all the required supporting equipment, including the fuel oil transfer system, the service water system, the salt water system, and the traveling screens are required.

In addition to providing AC power, it is also essential for the operators to be able to maintain control of the equipment for the duration of the accident. An important function is provided by the Control Room and Switchgear Room HVAC systems.

The impact of high winds and missiles on the plant as discussed above and in Section 5.3.2 is quantified in terms of Core Damage Frequency using the updated IPE plant model. The quantification approach and results are discussed in the following sections.

5.3.4 Analysis of Risk from Tornadoes and Missiles

As discussed in the sections above, susceptible components may fail as a direct result of tornado wind effect or as a consequence of damage caused by tornado generated missiles. In this analysis it is assumed that whenever there is a tornado, missiles are always generated. Losing any single or combinations of these components could have some risk impact on the plant. The risk impact is analyzed using the approaches of point and area strike models and initiating events. These approaches are discussed below.

5.3.4.1 Point and Area Strike Models

The point strike model is used to calculate the frequency of a tornado strike directly over a target. When a target is impacted by a point strike, it is affected by high winds and has a likelihood of missile strike. The area strike model calculates the frequency that a tornado will be within a 2000 ft radius of the site. The striking radius is related to the probability of missile strike parameter (P_{ms} defined below). Two values, 2000 ft and one mile were considered. The probability of missile strike, P_{ms}, decreases as the striking radius is increased. The product of 2000 ft radius and its associated P_{ms} value results in a higher probability of target missile strike and is therefore used (to be conservative, see Ref. 5-7).

It is assumed that components which are not wind resistant (e.g., Control Room HVAC, Switchgear Room HVAC, and EDG 0C) would fail on a tornado point strike. Components which are wind resistant but not missile hardened have a likelihood of damage due to missiles which is related to the degree of coverage and exposed area (e.g., EDG 1B 2A and 2B intakes and exhausts). It is assumed that if a missile strikes a component, the component fails.

Based on the above, the tornado targets can have both point and area strikes. As discussed below there are two means for evaluating the missile strike probability and component failure probability using the point and area strike models. The most conservative (highest probability) of the two methods is used for each component. It turned out that the risk impact due to point strike is always higher than that due to area strike (Ref. 5-19). Thus the point strike model is used throughout the analysis.

<u>Area Strike - Missile Strike Probability/Component Failure Probability</u>

This is the probability that a missile will strike a component given a tornado impacts within 2000 feet of the plant site. It is assumed that a tornado can throw missiles up to this distance. The probability that a missile will strike a component is equal to:

$$P_{ms} = A * N_m * \Psi_{as}$$

Where A = Surface Area Exposed to a Missile
N_m = Number of Potential Missiles on Site
Ψ_{as} = The Missile Impact Parameter (area strike) - the probability per ft^2, per missile, that a missile will strike the component

Thus the overall failure probability for a component will be the area strike frequency times the probability that a component will be hit by a missile if an area strike occurs (P_{ms}).

Thus the component failure probability equation for area strikes is:

Failure probability = Tornado Frequency (area strike) * P_{ms} (area strike)

<u>Point Strike - Missile Strike Probability/Component Failure Probability</u>

This is the probability that if a tornado comes in direct contact with a component, essentially travels directly over a component, that a missile will strike the component. The probability of a missile striking a component, if there is a tornado point strike on the component, is calculated as follows:

$$P_{ms} = A * N_m * \Psi_{ps}$$

Where A = Surface Area Exposed to a Missile
N_m = Number of Potential Missiles on Site
Ψ_{ps} = The Missile Impact Parameter (point strike) - the probability per ft^2, per missile that a missile will strike the component

Thus, the overall failure probability for a component will be the point strike frequency times the probability that a component will be hit by a missile if point strike occurs (P_{ms}).

Thus the component failure probability equation for point strikes is:

$$\text{Failure probability} = \text{Tornado Frequency (point strike)} * P_{ms} \text{ (point strike)}$$

Dependent Failures due to a Tornado (Winds and/or Missiles) Striking Multiple Components

There is a certain length and width associated with each tornado. This is also defined as the footprint of a tornado. For CCNPP site, a length of 11,880 ft and a width of 317 ft were calculated (Ref. 5-7). Within the range of this footprint, when one component fails due to a point strike of a tornado, there is an increased potential that the tornado could strike the other susceptible components. A methodology was developed for determining the dependency between components based on the distance between the components and the width of the tornado footprint. The methodology assumes a footprint width of 317 ft. and an infinite length. For example, components which are within 200 ft. of another component which fails due to a point strike are assumed to also have a point strike. The probability decreases as the distance increases. Between 200 ft. and 249 ft. the dependency factor is 0.61, a factor of 0.46 for 250 ft. to 299 ft., a factor of 0.33 for 300 ft. to 349 ft., a factor of 0.3 for 350 ft. to 399 ft., a factor of 0.27 for 400 ft. to 449 ft., a factor of 0.23 for 450 ft. to 499 ft., and a factor of 0.21 for greater than 500 ft. (Ref. 5-19).

5.3.4.2 Initiating Events

Figure 5.3-1 shows the locations of the various susceptible components. Components which are close to each other are grouped together and this is defined as a node. For example, all the tanks in the tank farm are grouped as Node 1. The reason for grouping is that when a tornado (together with a missile-point strike) strikes one component in the group, it is very likely that all the other components in the same group would be struck at the same time. Thus, when a tornado strikes a node, the impact of losing all the components in that node is the same. Table 5.3.4.2 below shows the groupings and the corresponding nodes. Loss of the components in each group or node is defined as the tornado/missile initiating event as shown also in Table 5.3.4.2. Each initiating event is assigned a designator (i.e., LTOR1 for Node 1, etc.).

Table 5.3.4.2
Affected Components Group

Initiating Event	Node	Affected Components Group
LTOR1	1	Tanks in the Tank Farm
LTOR2	2	EDG0C, FOST, 11RWT, EGD1B&2A
LTOR3	3	21RWT, EDG2B
LTOR4	4	U2 SWGR HVAC, U2 SRW Head Tank
LTOR5	5	CR HVAC, U1 SWGR HVAC, U1 SRW Head Tank

The initiating event frequency for each node is the tornado point strike frequency which is assumed to be the same at each node. This is because each node has an equal chance of being struck by a tornado. The point strike frequency is 3.13E-05 per year (Ref. 5-7).

5.3.5 Analysis of Risk from Hurricanes

While most of the structures are designed to withstand 100 mph winds and would likely survive stronger winds, the one concern is the potential for the metal sidings to be sucked off because of the pressure reduction effect of the wind. This is because the metal siding at CCNPP is designed to blow out at 0.45 psi pressure differential and may not survive the external wind force equivalent to a 100 mph wind speed. This is particularly important because of the potential damage to the Control and Switchgear Rooms HVAC AC units, ducts and dampers from the metal siding missiles as discussed earlier.

The area in the vicinity of the plant is not susceptible to a direct hit from a high category hurricane. Most hurricanes turn to the east following the coastline or approach the site from overland with diminished wind speeds. Based on historical data for hurricane strikes on the coasts of Maryland, Virginia, and New Jersey, it is estimated that the annual frequency of a large storm with wind speeds in excess of 100 mph is on the order of 1.0E-03 (Ref. 5-34). This is the hurricane initiating event frequency and is designated as LHUR1.

Thus a prolonged loss of offsite power coupled with a failure of the Control Room and Switchgear Room HVAC could occur with as high a frequency as 1E-03/year. The Emergency Response Plan Implementation Procedure 3.0, Attachment 17, Section D.3 (Ref. 5-41) is being revised to require placing the plant in Hot Standby if a hurricane is predicted to strike within eight hours and pre-staging portable fans and generator which may be used to recover Switchgear Rooms HVAC in the event of a complete loss of the normal ventilation systems.

5.3.6. Plant Model Quantification of High Winds Impact

Top events and associated split fractions are developed to account for the component failures due to hurricane and tornado/missiles. The CCPRA plant model rules are revised to account for the impact of failed components and quantify the impact in terms of CDF.

5.3.6.1 Top Events and Split Fractions

<u>Tornado/Missile</u>

Each susceptible component listed in the 5 nodes (Table 5.3.4.2 above) with the addition of Fire Pump House is assigned a top event. Within each top event, split fractions are developed based on the failure of a specific component at a node given that a tornado strike has occurred at the nearby nodes. Table 5.3.6.1 is a list of the top events and the associated split fractions which include the ones that are unique to the hurricane and tornado events and those that appear in the quantification sequences (Section 5.3.6.2 below). The split fraction values are calculated using the point strike model (higher value than the area strike) described in Section 5.3.4.1 above including the distance dependencies.

Note that the hurricane initiating event LHUR1 is assumed to be successful in all tornado/missile split fractions to simplify modeling the hurricane and allow placing the impact directly into the general transient plant model tops.

Hurricane

Hurricane fails the following top events and associated split fractions.

Top Event PG - "Fire Protection makeup to the SRW and CCW Head Tanks"

Top Event GJ - "EDG 0C starts & provides power to a 4KV Bus"

Top Event HH - "Control Rm/Cable Spread Rm HVAC header operates"

Top Event HS - "SWGR HVAC header operates"

Top Event HZ - "Operator locally ventilates both Switchgear Rooms using temporary fans" has two split fractions which can be questioned on LHUR1. HZ8 is questioned when 4KV Bus 14 is available and HZ9 when 4KV Bus 14 is not available. These split fractions include human actions which include pre-staging of the equipment prior to a hurricane impact on the site.

Top Events AC and AD - "4KV Bus 21 energized and 4KV Bus 24 energized". These buses are conservatively assumed to fail since Unit 2 SWGR Ventilation is not explicitly modeled in the Unit 1 model.

Top Event F1 - Split Fraction F1T - "AFW delivers adequate flow, given SBO with S/G overfill, decreased decay heat due to early shutdown on hurricane imminent."

5.3.6.2 Quantification Results

The Core Damage Frequency due to high winds from hurricanes or tornadoes and associated missiles together with a LOOP is 4.35E-6 per year. The combined hurricane sequences account for approximately 65% of the cumulative importance. This is due to the higher initiating event frequency of the hurricane.

The first 500 sequences account for about 86% of the cumulative importance to CDF. As an example, Table 5.3.6.2 shows the first 50 sequences. As seen in this table, these first 50 sequences contribute nearly 50% of the cumulative importance to CDF. The split fractions in these sequences are defined in Table 5.3.6.1.

These sequences show that as the hurricane initiator automatically fails Switchgear Room HVAC together with a LOOP, the dominant hurricane sequences are generally related to the failure of the Switchgear Room HVAC recovery actions which fails electrical support systems. These failures are generally combined with failures of AFW. The AFW failures include flow control failures and failures of the turbine driven AFW Pumps. The flow control failure is due to the loss of support systems to the AFW system which is due to the Switchgear Room HVAC failure. Another dominant contributor is the failure of the Switchgear Room HVAC and the recovery action along with a failure which causes a Loss of Coolant Accident such as a stuck open PORV or a Seal LOCA. Without power, the Safety Injection system cannot makeup the loss of inventory in the Reactor Coolant system.

The majority of the tornado sequences which lead to core damage also result from failure of Switchgear Room HVAC combined with AFW failures.

5.3.7 Summary of High Winds

As the initiating event frequencies of the hurricanes and tornados are above the NUREG-1407 screening criteria of 1.0E-6/yr, detailed PRA analysis has been performed to assess the overall risk of hurricanes, tornados and associated missiles. The final Unit 1 CDF value for the combined high wind effect together with a LOOP is 4.35E-6/yr. This CDF includes an operator action to pre-stage the portable ventilation fans and generator within eight hours of predicted hurricane arrival on site. This operator action is required to provide emergency cooling for the Switchgear Rooms in the hurricane event in case of a complete failure of the normal ventilation system. The Emergency Response Plan Implementation Procedures 3.0, Attachment 17, Section D.3 is being revised to incorporate this operator action.

A Unit 2 CDF value is estimated to be slightly higher at 4.4E-6/yr. This is due to the emergency diesel generator configuration differences and the degradation of SRW make-up capability due to the loss of off-site power and the loss of the Fire Pump House. See Section 4.6.8.1.1 for a description of the diesel generator differences.

5.4 External Floods

Extreme floods present a potential threat to nuclear power plants due to high water at the site, or excessive ponding on the building roofs. Such flooding may have a number of effects including:

- Excessive hydrostatic loads on supporting walls
- Excessive dynamic forces due to wave run up
- Water intrusion via doorways or floor/equipment drains which may be connected to the storm drain system
- Excessive hydrostatic loads on building roofs caused by ponding
- Water intrusion via roof openings due to excessive accumulation

The buildings which contain equipment required for safe plant shutdown and therefore of concern in this analysis are:

- Turbine Building (interfaces directly with Auxiliary Building)
- Auxiliary Building
- Intake Structure
- Station Blackout (SBO) diesel
- Safety-related emergency diesel generator (EDG 1A)

Section 5.4.1 compares the external flood design basis with the Standard Review Plan (SRP). This is used as a basis for screening specific threats in accordance with the guidance in NUREG 1407 (Ref. 5-1). Section 5.4.2 evaluates the plant design against the latest Probable Maximum Precipitation Criteria (PMP) as proposed in Generic Letter 89-22. Section 5.4.3 discusses roof ponding and Section 5.4.4 provides a summary of the analysis.

5.4.1 Comparison of Plant Design Against 1975 SRP

This section provides a review of the Calvert Cliffs external flood design basis against the acceptance criteria associated with the issues addressed in the 1975 SRP (Ref. 5-8).

Section 2.4.2 of the SRP identifies the following types of flood-producing phenomena which must be considered in establishing the flood design basis of safety-related structures. Those phenomena are briefly summarized as:

- Stream Flooding
- Surges
- Seiches
- Tsunamis
- Dam Failures
- Land Slides
- Ice Loadings from Water Bodies

Stream Flooding

Section 2.5 of the UFSAR discusses the hydrology of the site. CCNPP is located on the western shore of the Chesapeake Bay at the mid point of its 195 mile length. At the site, the Bay is approximately six miles wide. The site is well drained by short intermittent streams. A drainage divide which is generally parallel to the coast line extends across the site. The area to the east of the divide is about 20% of the site and includes the plant area drained. This area drains to the Chesapeake Bay. The area to the west is drained by small tributaries of the Patuxent River, which is the closest major river, located approximately 10 miles to the south of the plant. The plant area has an elevation of 45 ft. (MSL) and has a storm drain system to handle water run off. (For reference, the Chesapeake Bay is at 0 ft. MSL)

The site occupies the head of water area of several small drainage basins and is not subject to flooding. It is possible that high intensity rain storms may cause water to back up in some valleys due to local constrictions in the stream beds, but this would be temporary.

Surges

Section 2.8.3 of the UFSAR describes studies to assess the effects of historic storms and tides on CCNPP. As a result, the controlling event for the design basis flood was determined to be maximum wave run-up during a Probable Maximum Hurricane (PMH). The wave run-up elevation is 27.5 (MSL). This is well below the plant grade at 45 ft. elevation. It was further concluded that the integrity of the intake structure, located at the 10 ft. elevation, is sufficiently protected with respect to water intrusion and excessive dynamic loading, to allow the safety-related salt water cooling pumps to continue to operate.

Recent hurricane inundation data, obtained from the U.S. Army Corps of Engineers and FEMA, shows that the maximum hurricane surge level for a Category 4 hurricane is 12.0 ft. at Cove Point. This is well below the "still water level" assumed in the UFSAR storm surge and wave run-up analysis. The UFSAR analysis is, therefore, still judged to be valid.

All other issues related to external flooding were addressed in Section 5.2, "Screening of External Hazards".

5.4.2 Effects of Local Intense Precipitation

5.4.2.1 PMP Hazard

Consideration of Probable Maximum Precipitation (PMP) criteria does not appear to have been part of the original design of the plant, although such criteria have since been addressed in the design of the new diesel generator buildings. NUREG-1407 requires consideration of PMP criteria over areas as small as one square mile and for durations as short as five minutes. Such criteria were the subject of NRC Generic Letter 89-22.

PMP is defined in HMR No. 52, dated August 1982 (Ref. 5-9), as "theoretically the greatest depth of precipitation for a given duration that is physically possible over a given storm area at a particular geographical location at a certain time of year and results from mutual agreement between the National Weather Service, the US Corps of Engineers and the bureau of reclamation."

5.4.2.2 Site Ponding

A detailed PMP analysis was performed by Bechtel for BGE to evaluate the drainage around the new diesel generator buildings. This analysis considered a drainage area of one square mile and included the power block as well as the surrounding embankment areas. Rainfall depths for various durations including five-minute, 15-minute, 30-minute, one-hour, six-hour, 12-hour and 24-hour were derived from HMR 51 (Ref. 5-11) and 52 (Ref. 5-9).

The analysis predicted a peak water surface elevation in front of the diesel generator building of 44.9 ft.. Since the floor elevation of the diesel generator building is at the 45.5 ft. elevation, no equipment damage would be sustained.

Water surface elevations at the power block were not specifically reported. However, the grade over the plant area which includes both the diesel generator building and the power block is reasonably level. Consequently the water surface elevation around the power block will be similar to that predicted for the diesel generator building, (i.e., 44.9 ft.).

Possible flood water intrusion pathways into the Auxiliary Building are via rolling equipment doors and personnel access doors located on the west side of the building at the 45 ft. elevation. Those doors which provide access to the Unit 1 and Unit 2 Diesel Generator rooms are protected by 6" curbs and consequently will not permit water intrusion given a PMP event. The remaining door is a 16 ft. wide rolling door which provides access for truck loading. The door is tight and would only permit minimal leakage in the event of site flooding. Any water intrusion would eventually drain to the Auxiliary Building basement. The critical flood volume for the Auxiliary Building basement is over 500,000 gallons, which, because of the limited duration of the PMP, will not be realized.

Water intrusion into other plant buildings, including the Service Building and Turbine Building is of no consequence in terms of mitigating system damage, and, at worst, would result in a flooding of the Turbine Building basement.

In conclusion, site ponding due to PMP events does not present a significant hazard at CCNPP.

5.4.2.3 Roof Ponding

Roof ponding can either cause roof failure and/or water intrusion by overflowing barriers. Each of the roofs of the safety-related structures are evaluated to determine whether there is a danger of sufficient ponding to cause collapse. Most of the roof ponding concerns are found to be dismissable based on limiting ponding depths set by geometrical features. With the drains inoperable, however, the decking on the Auxiliary Building at 118 ft. could fail. In addition, ponding on the Auxiliary Building roof at the 69 ft. elevation (over the emergency diesel generator rooms and the RWT pump rooms) could result in overflow into the EDG rooms if the roof drains were not operable. Hydraulic calculations are performed which demonstrates the adequacy of the roof drain system to protect the roof at the 118 ft. level (Auxiliary Building), and the prevention of overflow from the roof at the 69 ft. elevation.

In the event of the drains being blocked, it is calculated that structural failure of the Turbine Building roof could occur in less than twenty minutes. This would lead to accumulation of water in the basement of the Turbine Building. However, significant water intrusion into the adjoining Auxiliary Building via connecting doorways will not occur, since there are minimal door to frame/threshold clearances.

Based on the Plant Engineering Guideline for System Walkdowns (PEG-7) (Ref. 5-43), the Auxiliary Building roof areas are inspected monthly by both, the plant drain engineers and the plant area system engineers. If there is any blockage on the drains, an Issue Report is written to clear the blocked drains. In addition, if there is flooding on the Auxiliary Building roof, there will be some leakage between the Containment and the Auxiliary Building walls. This leakage can be noticed by the operator and an Issue Report is also written to correct the problem.

5.4.2.4 Summary of External Floods

The site ponding is not a concern at CCNPP. The roof drains have adequate capacity and are inspected monthly to ensure the integrity of the critical roofs. Therefore, it is concluded that the external flooding analyzed presents no risk to plant safety.

5.5 Transportation and Nearby Facilities Accidents

In this section, the risk associated with the following is analyzed.

Aircraft Hazard
Truck Transport of Hazardous Materials
Marine Transport of Hazardous Materials
Nearby Facility Hazards
Onsite Chemical Storage Hazard

5.5.1 Aircraft Hazard

An analysis of the crash frequency due to aircraft on the plant was performed. The contributions from airways and airports are considered separately. The approach used generally follows the guidelines outlined in Reg. Guide 1.7, Rev. 3, Sect. 3.5.1.6. The following is an example:

"An aircraft hazard analysis should be provided for each of the following:

- P_{FA} Federal airways or airport approaches passing within two miles of the nuclear facility
- P_A All airports located within 5 miles of the site
- P_O Airports with projected operations greater than 500 d^2 movements per year located within 10 miles of the site and greater than 1000 d^2 outside 10 miles, where d is the distance in miles from the site
- P_M Military installations or any airspace usage that might present a hazard to the site. For some uses such as practice bombing ranges, it may be necessary to evaluate uses as far as 20 miles from site."

The total probability of an aircraft crash is the sum of:

$$P_{FA} + P_A + P_O + P_M$$

5.5.1.1 Federal Airways/Aviation Corridors

The calculation of the probability of an aircraft crashing onto the plant (P_{FA}) for situations where federal airways or aviation corridors pass through the vicinity of the plant (vicinity = 2 mi, (Ref. 5-12)) is made using the guidelines of Section 3.5.1.6, III.2 of the Standard Review Plan (Ref. 5-13). For each airway or corridor, this probability is:

$$P_{FA} = C \times N \times A / W$$

where C = in-flight crash rate per mile for aircraft using the airway

W = width of airway (plus twice the distance from the airway edge to the site when the site is outside the airway) in miles

N = number of flights per year along the airway

A = effective area of plant in square miles

Thus, for the set of T corridors meeting the analysis criteria in the vicinity of the plant, the total probability is:

$$P_{FA} = \sum_{i=1}^{T} C_i N_i A / W_i$$

There are eight airways situated in the vicinity of the plant site. Four are high altitude airways intended for use by commercial aviation, and four are low altitude airways used by general and commercial aviation. Only two of these eight airways meet the requirements for analysis stated in SRP 3.5.1.6 (i.e., the plant site either lies within the airway or is located less than two miles from one of the airway's outer borders). The other six airways pass further than two miles from the site.

The FAA has provided information (Ref. 5-33) related to flights over the PXT VORTAC. There are five airways (out of eight near the site) that cover this point. The average number of flights that make use of the five airways is determined to be 276 flights per day or 100,740 per year. Of these, 84.4% are commercial, 12.6% are general aviation, and 3.0% are military aircraft. Since the plant does not lie in the path of any high-altitude airways, the contribution of these to the total number of flights that could potentially impact Calvert Cliffs can be discounted. Also, since only two of the low-altitude airways pass near the plant, the contribution from these airways can be slightly reduced. The following assumptions are used to conservatively estimate the number of flights on the two airways that directly affect this analysis:

- High altitude airways handle only commercial traffic.
- Commercial traffic was assumed to be split evenly between high and low altitude airways.
- All low-altitude airway traffic was assumed to use the two low-altitude airways of interest.
- Low-altitude air traffic was assumed to be split evenly among the two low-altitude airways, as there is no reason to assume any preferential routing on one or the other.

Using these assumptions, data on the number of flights, crash rates (for commercial, general aviation, and military aircraft) from standard references (Refs. 5-26, 27 and 28), and evaluations of the effective area, the total frequency of an aircraft crash from the airways is:

$$P_{FA} = 4.42 \times 10^{-7} \text{ cr/year.}$$

The effective area takes the following into account: shadow area, skid area, and roof area of the vital structures (containment, Auxiliary Building, intake structure and switchyard) as required by Ref. 5-13 and is calculated in Ref. 5-29.

5.5.1.2 Airports Within Five Miles of the Site

The probability (P_A) per year of an aircraft crashing into the site due to airports within five miles of the site can be calculated using the following equation (see Section 5.5.1.3 below for definition of terms).

$$P_A = \sum_{i=1}^{L} \sum_{j=1}^{M} C_j N_{ij} A_j$$

One of the airports within five miles of the site is the helipad located 1000 feet from the northern edge of the nearest vital structure. An average of seventeen corporate flights per year originating from BGE headquarters (about 50 miles north of the plant, near Baltimore, MD) make use of the helipad. There are no specific flight paths or exclusion areas for helicopter flights in the vicinity of the plant, so crash approaches from any direction must be considered. However, such flight paths are only used when weather conditions force the helicopter to take such a route to ensure a safe landing or takeoff, or if a flyover of the plant was requested by BGE personnel. This would be expected to be a small percentage of the total flights, and would (according to Helicopter Transport Services, Inc., the operator of the BGE corporate helicopter) certainly represent much less than one-half the number of flights. Thus, the vast majority of the helicopter landings and takeoffs will occur at least 1,000 feet from any vital structure which is shielded by dense woods. To maintain an adequate level of conservatism in this analysis, however, all helicopter flights were considered capable of damaging vital structures; the impact of this and other related conservatism are discussed in Section 5.5.1.6.

Only one other air strip is located within five miles of the plant site. This privately operated strip, Mears Creek, is only sporadically used for leisure purposes by its owner/operator. Two small single-engine aircraft are based there and are the only aircraft that are expected to use the field. For these reasons, the overall risk from nearby airfields will be dominated by the helicopter activities on site, and therefore, Mears Creek operations will not be pursued further in this analysis.

The overall rate for aircraft crashes from airports less than five miles away is represented by the rate for helicopter crashes adversely impacting vital structures at the Calvert Cliffs site (P_A). This can be estimated by multiplying the per mi^2 average crash rate (2.94×10^{-5} cr/mi^2) from available data (Ref. 5-30) by the area vulnerable to crashes (A_{hel}) (Ref. 5-29) and the number of flights per year.

$$C_{hel} = 2.94 \times 10^{-5} \text{ cr/mi}^2$$
$$N_{hel} = 17/\text{yr}$$
$$A_{hel} = 2.50 \times 10^{-3} \text{ mi}^2$$
$$P_A = C_{hel} N_{hel} A_{hel} = 1.25 \times 10^{-6} \text{ cr/yr}$$

5.5.1.3 Airports Further than Five Miles from CCNPP

The probability (P_O) per year of an aircraft crashing into the site due to airports further than five miles which meets the criteria of Reg. Guide 1.70, Rev. 3, item 3 (Ref. 5-12) is given by:

$$P_O = \sum_i^L \sum_j^M C_j N_{ij} A_j \text{ (Ref. 5-13)}$$

where:

- M = Number of different types of aircraft using airport
- L = Number of flight paths affecting the site
- C_j = Probability per square mile of a crash per aircraft movement for the j^{th} aircraft
- N_{ij} = Number (per year) of movements by the j^{th} aircraft along the i^{th} flight path
- A_j = Effective plant area (in square miles) for the j^{th} aircraft

Two airports (Chesapeake Ranch Airpark and St. Mary's County Airport) are located within ten miles of the plant. However, the frequency of operations at both airports are low enough that they can be screened from consideration using the criteria in Ref. 5-12.

A major naval air station (Patuxent River NAS) is located 11 miles from the plant site. There have been as many as 100,000 takeoffs and landings per year from this station. This is approximately equal to the number of flights that would be calculated as a screening criterion so it will be evaluated further even though the projection for the next several years is 50,000 to 60,000 per year.

Because of the large number of types of aircraft using the NAS and the similarity in operation among them, P_O was calculated in aggregate for all Patuxent River NAS operations. This is equivalent to performing the summation above over all aircraft and flight paths.

According to Patuxent River NAS Air Operations, pilots are generally sent on three-mile bypass loops around the plant site to avoid flyovers. However, there are three potential flight paths which are routed over Calvert Cliffs but are used only in rare circumstances. An actual Naval Facilities Engineering Command count of air traffic provided by Patuxent River NAS revealed that only 214 planes used these three routes in the past year, thus L = 214 flights/yr. Using this value, crash rate of 0.2 x 10-8 cr/na-mi2-fl (Ref. 5-31), and evaluations of effective area that take into account the shadow area, skid area, and roof area as required by Ref. 5-13 and calculated in Ref. 5-29, the total frequency of an aircraft crash from the aircraft using the Patuxent River NAS is:

$$P_O = 4.65 \times 10^{-9} \text{ /year.}$$

5.5.1.4 Military/Other Airspace Usage

Military usage of airspace in the vicinity of the plant site is covered by the activities at Patuxent River NAS and the military flights in local airways both of which have already been analyzed. No other types of airspace usage are anticipated. Thus, for this analysis, $P_M = 0$.

5.5.1.5 Aircraft Hazard for the Independent Spent Fuel Storage Installation (ISFSI)

A separate aircraft hazard analysis was performed for the ISFSI. The results show that the crash probability value meets the SRP screening criteria (at the lower 1.0×10^{-7} range). In addition, since the outer shell of the storage bunkers is composed of concrete of a minimum 3 ft. thickness, it is treated as being equivalent to the containment structure. Based on NUREG/CR-5042, the penetrating probability of a helicopter crash on a thick concrete structure is essentially zero. Thus, the aircraft hazard (including the helicopter) over the ISFSI is of no concern.

5.5.1.6 Summary of Aircraft Hazard

Based on the above, the total aircraft crash frequency on CCNPP is:

$$P_{TOT}=P_{FA}+P_A+P_O+P_M = 4.42\times10^{-7} + 1.25\times10^{-6} + 4.65\times10^{-9} + 0 = 1.70\times10^{-6} \text{cr/yr}$$

Approximately 74% of the total probability of an air crash impacting vital structures comes from helicopter operations which serve the CCNPP site. For this analysis, it is conservatively assumed that all 17 flights per year would fly over vital areas and could crash causing damage to safety-related equipment. In fact it was always the policy for helicopter pilots to avoid flights over security areas at all times unless it was absolutely necessary for reasons of safety or at BGE's specific request. This policy is now enforced by a restriction included in BGE's contract with the Helicopter Transport Services to accordingly limit the number of flights over the plants protected area (Ref. 5-32). With this restriction in place, there is now a high degree of confidence that the number of over flights will be significantly less than 6 per year, and as a consequence, the total annual aircraft crash frequency will be less than 1.0×10^{-6}.

5.5.2 Transportation Hazards

Transportation hazards considers truck and marine transportation. The risk analysis of each is discussed below.

5.5.2.1 Truck Transport of Hazardous Materials

Calvert County, MD, is a primarily rural area that has few facilities to/from which significant amounts of hazardous materials would be normally shipped. Furthermore, no major interstate or intercity highways run through the county; preferred routes for industrial transportation within and through the Baltimore/Washington metropolis include US Route 301, US Route 50, and Interstate 95, all of which pass well west and north of the plant site. The only significant roadway in the vicinity of the plant is Maryland Route 2/4, which, at its closest approach, passes slightly over one mile from plant structures.

Thus, it is assumed that no inter-city/interstate transport of hazardous materials is conducted past the plant site. The only chemicals assumed to be transported in the vicinity of the plant are those used or processed within the County.

5.5.2.1.1 Explosion Hazard

Regulatory Guide 1.91, Revision 1 (Ref. 5-14) describes a method for determining distances from critical plant structures to a railway, highway, or navigable waterway beyond which any explosion that might occur on these transportation routes is not likely to have an adverse effect on plant operation or to prevent a safe shutdown. The peak positive incident overpressure for these calculations is conservatively chosen in Reg. Guide 1.91 at 1 psi. Based on hemispherical charges of TNT, a safe distance can be conservatively defined by the relationship:

$$R = 45 \times W^{1/3} \qquad (Equation\ 1,\ Ref\ 5-14)$$

R = Distance in feet from an exploding charge
W = Size of charge in pounds of TNT

Conservatively, the most explosive substance potentially transported by road in the county will be used to judge safe standoff distances. Propane has the highest heat of combustion of the compressed gases in this category. The heat of combustion for propane is 21,670 btu/lb. The maximum truck shipment is usually assumed to be 9,500 gallons or, for propane, about 39,600 pounds (19.8 tons).

For vapor cloud explosions involving compressed flammable gases such as propane, Ref. 5-14 provides a TNT mass equivalence of 240 percent in establishing an upper bound for a safe stand-off distance. Using equation 1 of Ref. 5-14:

$$R = 45 \times (19.8 tons \times 2{,}000 lb / ton \times 2.4)^{1/3} = 2{,}054 ft$$

Since the closest roadway is over one mile (5,280 feet) away from critical structures, there is no adverse affect on the plant for an explosion involving truck transport of hazardous materials.

5.5.2.1.2 Toxic Hazard

Specific information on shipment sizes and frequencies for specific hazardous chemicals by road is neither required by state or federal agencies nor generally kept by facilities. Since the main highway through Calvert County, MD 2/4, is not an inter-city/interstate route as explained earlier, it is assumed here that hazardous material transportation on this road involves only those chemicals which are handled within the county.

The Calvert County Department of Public Safety, Emergency Management Division (EMD) keeps Tier 2 Reports, as required by law, listing chemicals stored at local facilities in significant quantities. Review of these reports demonstrates that only automotive gasoline and related products are stored at facilities other than CCNPP. Such products are neither considered volatile enough nor toxic enough in vapor form to pose a significant toxic gas threat to humans. Thus, truck transport of these materials can be screened out.

5.5.2.2 Marine Transport of Hazardous Materials

The Waterborne Commerce Statistics Center of the U.S. Army Corps of Engineers publishes annual summaries of commercial waterway traffic in the United States. Data from the period 1990-1993 was reviewed to determine the frequency and nature of hazardous materials transport in the Chesapeake Bay that could potentially pose a threat to CCNPP. It is assumed that traffic into and out of the Baltimore Harbor is representative of the marine traffic in the Chesapeake Bay area that would pass the plant site.

5.5.2.2.1 Explosive Hazards

Shipments in the Chesapeake Bay include a number of petroleum and other hydrocarbon products and byproducts which could (although are unlikely to) explode without other catalysts shipped nearby. The nearest edge of the shipping channel through the Chesapeake Bay is roughly parallel to the shore and lies approximately two miles away. Reg. Guide 1.91 (Ref. 5-14) defines the largest probable quantity of potentially explosive material transported by ship as approximately 10,000,000 lbs. This upper bound is used in the calculation of maximum safe standoff distance for one psi overpressure, although the great majority of shipments will be well below this quantity. Using a TNT equivalence of 5%, and the maximum probable quantity (10,000,000 lbs.) of material shipped, the <u>maximum</u> required safe standoff distance is given by:

$$R = 45x(10{,}000{,}000 lbs. x .05)^{1/3} = 3{,}572 ft.$$

While this shows that incidents involving explosions within the shipping lane are of no concern, the possibility of a ship straying from the shipping lane and moving closer to the plant sustaining a catastrophic explosion was also reviewed. It can be shown that an explosion of this type which jeopardizes the plant requires such an extraordinary set of circumstances that it can be neglected as not credible, even though its probability of occurrence cannot be directly quantified.

In theory, if a barge filled with the maximum amount of fuel oil spilled its entire contents (which then vaporized and exploded) after drifting over a mile out of its lane, there could be an adverse effect on the plant (i.e., overpressure greater than 1 psi). However, for this scenario to develop, <u>all</u> of the following things must happen:

- The ship must drift at least 1.3 miles out of its shipping lane toward the plant before releasing its cargo;
- The ship must be loaded with the full assumed amount (10,000,000 lbs.) of fuel oil;
- The entire contents must vaporize within a very short period of time, then explode.

The first is a possibility that must be considered, but is <u>highly</u> unlikely, particularly considering the decrease in water depth (and navigability) as the barge moves closer to shore. The weight limit is felt to be conservative; it is unlikely that a barge with anywhere near that amount of fuel oil will be present in the shipping channel. Finally, and most importantly, fuel oil is a heavy substance with a relatively high boiling point and relatively low vapor pressure. The chances of immediate and complete vaporization of its contents are essentially nil. It can be confidently speculated that these three events will not be in place for a single event and though there are safety-related structures within 3,572 feet of the shoreline, explosions which endanger these structures from fuel oil shipments in the Chesapeake Bay are not credible.

In the four-year period for which data was reviewed, there was one instance (1990, foreign imports) in which "explosives" are listed as a commodity shipped into Baltimore. No shipments of explosives have been recorded since that time so it can certainly be assumed that such shipments are rare. However, even if these were analyzed, it can be shown that they do not add any significant explosion hazard to those discussed above. One-thousand tons (2,000,000 pounds) were shown shipped in the 1990 survey and to conservatively estimate the danger presented by this material, it is assumed to all be transported in the same shipment. Conservatively assuming that the total maximum quantity can explode instantaneously and that the explosive potential of the material is equivalent to that of TNT, the safe standoff distance would be calculated as:

$$R = 45x(2{,}000{,}000lbs.x1.0)^{1/3} = 5{,}670ft.$$

Since the nearest edge of the shipping lane is about one mile further from the plant site than this, an explosion <u>within</u> the shipping lane is far enough away that no adverse effect at the plant site will be felt. Furthermore, using the same three arguments as above, the probability of a ship drifting close enough to the plant to endanger it is remote (even reducing the distance in the first argument from 1.3 to 1.0 miles).

5.5.2.2.2 Toxic Hazards

Section C.2 of Reg. Guide 1.78 (Ref. 5-15) states that "[if] hazardous chemicals... are known or projected to be frequently shipped by rail, water, or road routes within a five-mile radius of a nuclear power plant, estimates of these shipments should be considered in the evaluation of Control Room habitability. Shipments are defined as being frequent if there are...50 per year for barge traffic..." While the data does not provide shipment frequencies, it does provide cumulative commodity shipment amounts (given in thousands of tons) for all materials transported along the specified routes. While it is impossible to determine from this data whether or not each commodity shipped is above or below this screening criterion, it can be safely assumed (because of the large shipment capacity of most industrial shipping vessels) that those with a low cumulative shipment weight are shipped at a frequency well below this level.

A review of the chemicals that are shipped in significant cumulative annual quantity into and out of the Baltimore Harbor reveals no dangerously toxic chemicals that meet the frequency criterion. Those chemicals having toxic properties that are shipped with a frequency that would likely be classified as "frequent" by the standards of Reg. Guide 1.91 are either cumulative poisons (e.g., benzene) or solids or liquids with very low vapor pressures (e.g., sulfuric acid, sodium hydroxide) that will not produce vapors capable of traveling great distances.

5.5.2.3 Summary of Transportation Hazards

Three modes of transportation are used for hazardous chemical transportation in the vicinity of CCNPP. The third, which has not been addressed until now, is the transportation of liquefied natural gas (LNG) via pipeline. This has been thoroughly evaluated in Ref. 5-16 and determined to pose no significant hazard to the plant. The other modes of transportation are road and water. No significant shipments of dangerously toxic chemicals are made on the Chesapeake Bay near the plant. Any transport of explosive materials likely to be made by ship or by truck is performed at a distance far enough away from the plant site to preclude adverse effects. Thus, hazardous materials transportation in the vicinity of the plant is not expected to adversely impact the plant in any way and no additional design or procedural protection to mitigate their consequences are required.

5.5.3 Nearby Facility Hazards

Section C of Reg. Guide 1.78 (Ref. 5-15) states that "chemicals stored or situated at distances greater than five miles from the facility need not be considered because, if a release occurs at such a distance, atmospheric dispersion will dilute and disperse the incoming plume to such a degree that there should be sufficient time for Control Room operators to take appropriate action. In addition, the probability of a plume remaining within a given sector for a long period of time is quite small." Thus, only facilities within five miles of the plant will be evaluated.

An October 1995 hazards analysis conducted by the Calvert County Department of Public Safety, Emergency Management Division (EMD) and an assessment of the hazards presented by the presence of the Cove Point LNG plant were used as the basis for this evaluation. Eight facilities have been identified as those which "store, use, or produce significant amounts of hazardous materials" in the county. Other than the Calvert Cliffs plant site itself, the only facility within the five-mile radius specified that is included in that list is the Cove Point LNG plant which has been shown to present no significant hazard to Calvert Cliffs (Ref. 5-35). Thus, no nearby industrial or commercial facilities handle an amount of any hazardous material that can adversely impact the plant or its operators.

5.5.4 Onsite Chemical Storage

A separate hazard analysis is performed for the onsite chemical storage to evaluate the probability of a release which could result in a loss of Control Room habitability, incapacitation of operators, damage to vital equipment and subsequent off site exposure levels exceeding 10 CFR100 limits. About 3000 chemicals are considered in all (Ref. 5-36). Chemicals which meet one or more of the following criteria are screened from further consideration.

1. Common household/office products. Chemical assumed present only in commercial quantities and packaging, with essentially no danger presented to Control Room personnel from spills or leaks.

2. Chemicals not stored onsite in a form that presents an immediate hazard to Control Room operators. Generally, these are heavy liquids and solids with low volatility.

3. Eliminated based on chemical properties provided in Ref. 5-37 regarding vapor pressure, toxicity of constituent chemicals, etc.

4. Eliminated based on detailed information provided in MSDS sheets for compound or chemical. Screening factors that were taken and analyzed for these chemicals included hazard ratings, vapor densities, vapor pressures, required personnel protection, and judgments made based on qualitative descriptions of chemicals' effects provided in the MSDS text.

5. Eliminated based on minimum adjusted allowable weight calculation.

6. Eliminated based on full Reg. Guide 1.78 adjusted allowable weight calculation.

7. Chemical remains in BGE inventory data base, but is no longer on site.

8. Diffusion calculation based on material, quantity and location.

Any non-screened chemicals are evaluated for their toxicity and explosion hazards.

Based on the analysis results presented in Ref. 5-36, it is concluded that no chemicals currently stored onsite pose a significant hazard to Control Room operability or plant equipment.

5.6 Turbine Missiles

5.6.1 Turbine Description

The turbines for Unit 1 & 2 are located inside the Turbine Building and are essentially the same. They are 1800 rpm, two-stage reheat, tandem compound, six-flow exhaust machines with a last row blade of 38 in. and 40 in., respectively. The turbines are designed for 815 psia saturated steam inlet pressure and 2.0 in. Hg absolute exhaust pressure. There are six stages of feedwater heaters.

Dry saturated steam from the two steam generators enters the high pressure turbine through four sets of stop-throttle and control valves. The steam expands in the high pressure turbine and then flows to the moisture separator reheat units. After the moisture separator reheat unit, the steam flows through either combined intercept valves (Unit 1) or reheat stop and intercept valves (Unit 2) to the low pressure turbine where the steam expands and then exhausts to the condenser.

A turbine bypass system is provided in order to allow excess steam generator energy to be bypassed into the condenser whenever the turbine cannot accept all of the generated steam (e.g., during startup or a sudden change in load). Positive closing non-return valves are provided in all extraction lines, except those located in the condenser neck, to limit flow of stored energy, which could cause the turbine to overspeed, from the heaters to the turbine on a turbine trip. These valves are actuated by the Electro-Hydraulic Control (EHC) Trip System.

The speed governor will start to close the control, combined intercept (Unit 1) or reheat stop and intercept valves (Unit 2) at 101% of rated speed. At 110% of rated speed the mechanical overspeed trip will operate and close all steam valves. The backup overspeed trip system will actuate the master trip solenoid valve at 112% of rated speed. Two independent speed signals are used, permitting speed control with either one of the signals incapacitated.

5.6.2 Mechanism of Turbine Failures and Missile

5.6.2.1 Unit 1

The General Electric Company, manufacturer of the Unit 1 turbine generator, determined their product's most severe turbine missile by assuming the instantaneous loss of load from a full load operating condition. In addition, it was postulated that the normal speed governing system failed to close the emergency stop valves. The turbine can then accelerate from 150 to 170% of rated speed before severe generator damage (due to thrown windings and probable retaining ring failure) will decelerate the turbine. The last stage low pressure turbine wheel is postulated to fail when the turbine reaches 169% of rated speed resulting in disc fragments. High pressure turbine rotors are not expected to fail at destructive overspeed. Generator field and retaining ring parts are expected to be retained by the generator housing which, by its construction, is an ideal energy absorber.

5.6.2.2 Unit 2

The Westinghouse Electric Corporation, manufacturer of the Unit 2 turbine generator determined the most serious turbine missile by assuming both the stop-throttle and control valves fail to close following the opening of the main generator circuit breaker at full load even though control system reliability and equipment redundancy makes such a turbine runaway highly improbable.

Disc No. 2 on the low pressure element is the most highly stressed disc with a failure speed of 200% of rated speed. Upon failure, the disc fragments will damage the turbine to the extent that additional overspeed will not occur. Missiles from high pressure section were assumed not to develop and the generator field and retaining ring parts were expected to be retained by the generator housing.

5.6.3 Safety-related Equipment Susceptible to Damage from Low Trajectory

The Auxiliary Building is located on the west side of the two turbine generators. The Switchgear Room for Unit 1 and Unit 2 are located inside the Auxiliary Building adjacent to the Turbine Building and fall within the LTM strike zone. In addition, the Rams Head for Unit 2 also falls within the strike zone. The Rams Head is a collection of safety-related piping associated with the salt water system. The Unit 1 Rams Head is outside the strike zone.

The Control Room area at the 45 ft. elevation also falls within the strike zone. However, since the actual Control Room area is protected by the 3 ft. wide wall and a couple of 8 in. to 2 ft. wide walls, depending on the path the generated missiles would take, it was assumed that the missile would not reach the Control Room area.

5.6.4 Analysis of Risk from Turbine Missile Generation

In this analysis, a methodology for evaluating the probability of a turbine-generated missile striking safety-related systems or equipment that is consistent with the approach in Refs. 5-18 and 5-20 was used. The risk associated with the turbine missiles is defined by the "total missile damaging probability (P_4)" which consists of the following missile-related probabilities:

P_1 = missile generation probability, also defined as the initiating frequency

P_2 = striking probability with respect to the barrier between the turbine and the target

P_3 = probability of damaging the target

P_4 = $P_1 * P_2 * P_3$ per year

NRC Reg. Guide 1.115 (Ref. 5-17) states that an acceptable risk rate for the loss of an essential system from a single event due to low trajectory turbine missiles (P_4 value) is less than 10^{-7} per year. However Ref. 5-20 stipulates that the risk is also acceptable if the missile generation probability (or initiating event frequency) P_1 is less than 10^{-5} per year.

5.6.4.1 Unit 1 GE Low Pressure Turbine

GE has recalculated the missile generation probabilities (P_1) for CCNPP (Ref. 5-38) utilizing a revised wheel missile calculation procedure approved by the NRC which includes an updated failure rate data on the primary steam valves of GE nuclear units. Ref. 5-38 presents the P_1 results for Low Pressure Rotors A, B and C as a unit based on the inspection results conducted in April 1992 for LPA, April 1996 for LPB, and March 1994 for LPC. The highest value of the turbine unit missile generation probability (P_1) is 3.0E-5 per year which corresponds to the quarterly testing on the stop/control/combined intercept valves and a service time of 15.88 years (years of operation for the turbine rotor spinning time up to the latest inspection date of LPB in April 1996) plus additional eight years. This highest value is chosen for the risk analysis because it is the value corresponding to the maximum service time provided by GE and is treated as the worst case value.

Obviously, the P_1 value of 3.0E-5 does not meet the NRC's screening criteria. A detailed analysis was then performed (Ref. 5-39) to determine if a P_4 value would meet the NRC's screening criteria of 1.0E-7. The analysis takes into consideration the geometry between the turbine and the striking targets. The results are summarized below:

> The only target within the strike zone is the Auxiliary Building. Although the k-line wall between the Turbine Building and Auxiliary Building is sufficiently thick (3 ft.) to prevent missile penetration, damage may still occur to the Unit 1 45 ft. Switchgear Room due to the presence of an unprotected roll-up door and two personnel access doors. Using geometric considerations, the probability a missile will strike all the doors (P_2) is calculated to be 2.33×10^{-3} (Ref. 3-39). Assuming the probability of damaging the safety-related equipment is 1.0 given a strike, the frequency of damage is:
>
> $P_4 = P_1 * P_2 * P_3 = 3.0E-5 \times 2.33E-3 \times 1.0 = 6.99E-8$ per year

This P_4 value meets the NRC acceptance criteria of 1.0E-7 per year provided that the turbine valves are to be tested quarterly.

5.6.4.2 Unit 2 Westinghouse LP Turbines

For Westinghouse turbines, the annual frequency of missile ejection (destructive overspeed probability, P_1 value) is a function of the following:

- Failure of one governor valve and one throttle valve to close
- Frequency of system separation which is defined as the opening of the generator breaker before the turbine trips. Loss of Offsite Power would be an example
- Turbine valves test interval
- Allowance for missile ejection at the design and intermediate overspeed

WCAP-14732 (Ref. 5-21) presents the latest calculated P_1 value for the Westinghouse Turbine Mini Group (Owner's Group) of which CCNPP is a member. For a quarterly turbine valves test interval, the P_1 value reported in Table 7-1 of Ref. 5-21 is 8.8E-7. Since this value is smaller than the NRC's acceptance criteria of 1.0E-5 for P_1, it is concluded that the turbine missile risk for the CCNPP Unit 2 Westinghouse turbine is acceptable providing that the turbine valves are to be tested quarterly.

5.6.4.3 Turbine Missile Event at Salem Unit 2

Salem 2 experienced turbine missile due to blade failure in 1992 as the result of on-line testing of the trip protective solenoid valves (Ref. 5-22). The failed LP blade did penetrate the LP casing. NUREG-1275 (Ref. 5-22) is a NRC evaluation of the Salem 2 event. The key evaluation results seem to point out that the methodology used by the turbine vendors to calculate the missile generation probability is non-conservative and that the vendor-recommended valve test procedures and the control system maintenance procedures are not strictly followed by the utilities. Sections 6 and 7 of Ref. 5-22 provide the summary of evaluation and the recommended corrective actions to prevent turbine missile from recurring.

As for CCNPP, since the Salem 2 event, preventive measures including procedure changes and design modifications have been implemented. For example, in the case of Unit 2 turbine, Westinghouse's Availability Improvement Bulletin (AIB) 9301 (Ref. 5-24) provides a list of 35 improvements in the areas of overspeed protection control system, emergency trip solenoid valves, valve test frequencies, maintenance, operator actions during testing, etc., with the goal to achieve system redundancies and on-line testing capabilities. Ref. 5-25 provides CCNPP's responses to AIB 9301 indicating that all the improvements have been completed with three exceptions, two are not applicable and one is on the on-line testing of the mechanical trip solenoid valves. CCNPP took exception to this because: (1) these valves will be tested each refueling outage in response to CAL 92-02 (Ref. 5-42), (2) there is redundancy in the overspeed trip system (i.e., the trip logic contains an additional electrical overspeed trip), (3) Unit 2 turbine has never failed to trip on demand, and (4) there is increased risk of tripping the unit.

In the case of the Unit 1 turbine, the following lists the key issues associated with GE's guidance and CCNPP's responses:

(1) Periodically testing the turbine trip system -- CCNPP is in full compliance with the requirements of GEK 17812E, GET-8039.1 and TIL 1165-3 [Refs. 5-23, 46 and 47].

(2) Investigate failures that occur during testing and remedy failures diligently--Failures of these tests would require an Issue Report to be written which would result in effective issue resolution.

(3) Sequentially tripping the Generator--The current system and Operating Procedure (OI-43A) are adequate to ensure that overspeed condition would not be reached.

(4) Reduce the likelihood of spurious scrams during automatic overspeed testing--Two trip reduction programs have been issued by GE (i.e., TILs, 969-3R1 and 1212-2 (Refs. 5-48 and 49)). Of the 21 recommendations, 15 have either been determined not applicable or installed. The remaining six are under review.

(5) Section 6 of NUREG-1275, Vol. 11 lists eight areas that contributed to the Salem 2 overspeed event. These items were reviewed and no additional action is recommended for the Unit 1 turbine.

Based on the above, from the procedure and hardware modifications point of view, the concerns addressed in NUREG-1275, Vol. 11 presents no additional turbine missile risk to CCNPP. However, there is still a concern that P_1 values provided by GE and Westinghouse do not account for the loss of overspeed protection redundancy or the increased human error potential associated with valve testing. This concern may impact the P_1 values and was not addressed by this assessment.

5.6.5 Summary of Turbine Missiles

The risk associated with the turbine missiles for both units meets the NRC acceptance criteria provided that the turbine valves are tested quarterly.

5.7 Overall Conclusions

An initial screening analysis performed according to NUREG-1407 guidelines demonstrates there are no unique external events which present a significant hazard at CCNPP. Detailed analyses of high winds, external flooding, transportation and nearby facility accidents and turbines missiles were performed. In the case of high winds, the final CDF value for the combined effect of hurricane and tornado/missile together with a LOOP is 4.35E-6/yr. The hurricane sequences account for approximately 65% of the total. During the hurricane event, an operator action is required to prestage the portable ventilation fans and generator within eight hours of predicted hurricane arrival to provide emergency cooling for the Switchgear Rooms in case of a complete loss of the normal ventilation system. The Emergency Response Plan Implementation Procedures 3.0, Attachment 17, Section D.3, will be revised to incorporate this operator action.

External flooding hazards, including site and roof ponding are determined to be within the capacity of the plants design and no damage to safety-related equipment was postulated. Hazardous material associated with local transportation, nearby facilities or onsite storage are not expected to adversely impact the plant in any way, and no additional design or procedural protection to mitigate their consequences is required. An aircraft crash analysis demonstrated helicopter flights to be a potentially significant hazard. However, this was resolved by restricting flights over protected areas of the plant, which subsequently reduces the overall crash frequency to less than 1.0E-6 per year. A detailed turbine missile analysis demonstrated that frequency of damage to safety-related equipment due to such events is less than 1.0E-7 per year.

5.8 References

5-1 NUREG–1407, "Procedural and Submittal Guidance for the Individual Plant Examination of External Events (IPEEE) for Severe Accident Vulnerabilities," June 1991

5-2 Ravindra M.K. and Bannon, H., "Methods for External Event Screening Quantification: Risk Methods Integration and Evaluation Program (RMIEP) Methods Development," NUREG/CR–4839, July 1992

5-3 BGE, Calvert Cliffs Nuclear Power Plant Updated Final Safety Analysis Report, Chapter 2, Section 2.6

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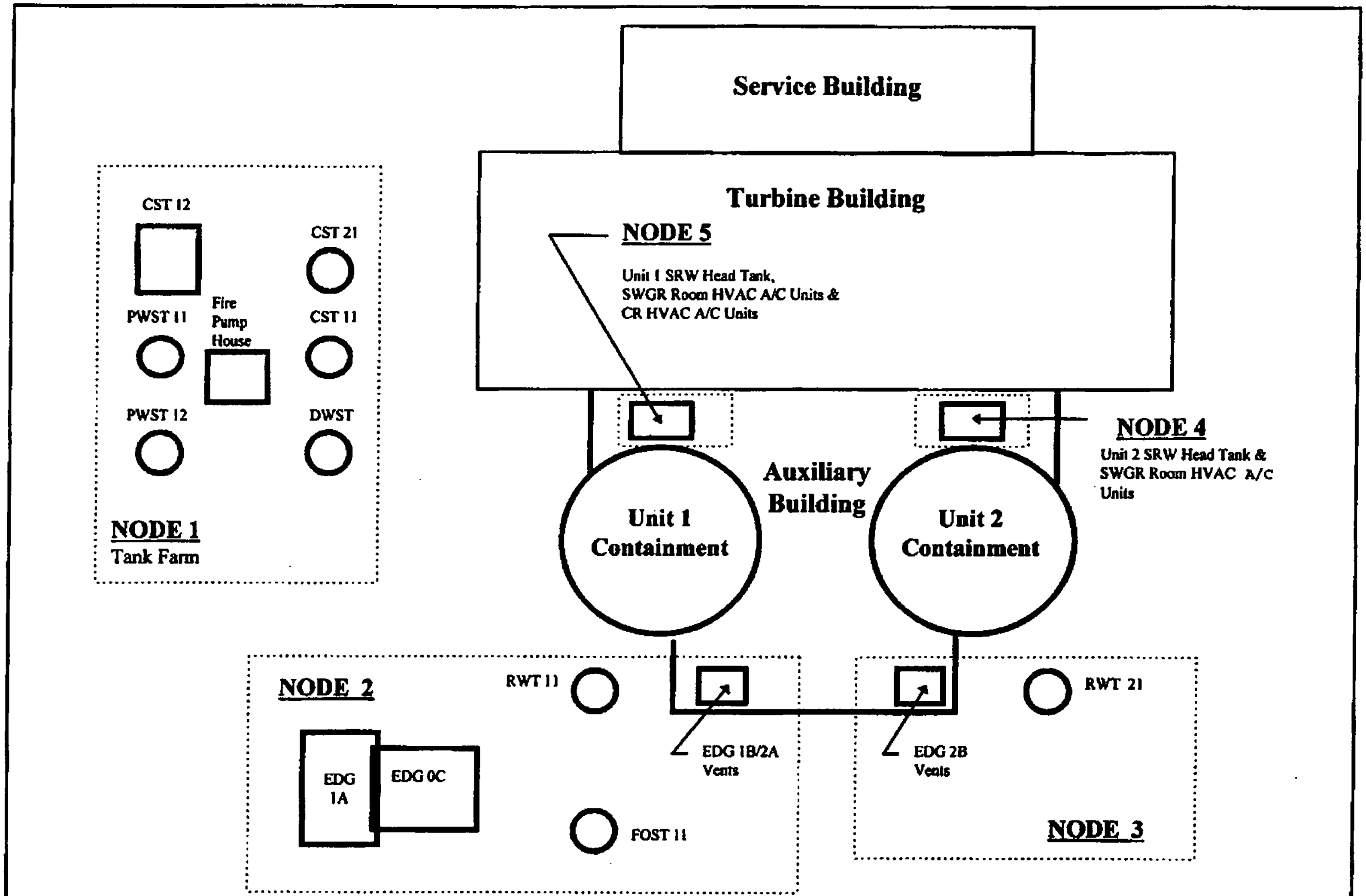


FIGURE 5.3 - 1
Plant layout (Tornado Nodes Assignment)
Not to Scale

Table 5.2.2
External Events Considered

Event	Screening Criteria	Remarks
Aircraft Impact	-	Further analysis required per NUREG-1407. See Section 5.5.1.
Avalanche	1	The topography of the site is such that no avalanche is possible.
Coastal Erosion	5	This is a slow progressing condition, therefore there will be sufficient time to shut the plant down in case of significant coastal erosion. Moreover, approximately 3700 lineal ft. of shore protection has been placed in front of the plant area. The shore protection consists of onsite material placed in front of the cliffs and faced with filter cloth and layered riprap (CC UFSAR, Rev. 15, pg. 2.4-9).
Drought	3,5	The principal source of water for the plant is the Chesapeake Bay. The possibility of water shortage is unlikely.
External Flooding	-	Further analysis required per NUREG-1407. See Section 5.4.
Extreme Winds and Tornadoes	-	Further analysis required per NUREG-1407. See Section 5.3.
Forest Fire	1	Fire is unlikely to propagate to the site, because the site is cleared for several hundred feet.
Fog	2	Fog can effect the frequency of occurrence of transportation accidents and aircraft impacts. The transportation accident frequencies and the frequency of the aircraft crashes implicitly include the impact of fog.
Frost	3	Loads induced on structures due to frost are much lower than snow and ice loads.
Hail	3	Less damaging than tornado generated missiles. Thus, hail is not considered for the scoping analysis.
High Tide, High River Stage	2	Included in the effects of external flooding.

Table 5.2.2 (cont.) External Events Considered		
High Summer Temperature	2,3,5	A slowly developing condition within the plant. The failure of the ultimate heat sink is inconceivable, as the source of water is the Chesapeake Bay.
Hurricane	-	Further analysis required per NUREG-1407. See Section 5.3.
Ice Cover	3	Considered in the plant design. Icing which results in an obstruction at the intake canal is a slowly developing phenomenon, and there is sufficient time to reduce power and shut the unit down in a controlled fashion.
Industrial or Military Facility Accident	-	Further analysis required per NUREG-407. See Section 5.5.
Internal Flooding	-	Independently analyzed as part of the IPE.
Landslide	1	Impossible because CCNPP is built on flat land.
Lightning	1,2	CCNPP operating history indicates no evidence of detrimental effects seen as a result of lightning other than loss of offsite power and reactor trip.
Low River Water Level	3,5	Not applicable.
Low Winter Temperature	3,5	Thermal stresses and embrittlements are insignificant and are covered by design codes and standards for plant design. Generally, there is adequate warning of icing so that remedial action could be taken.
Meteorite	4	This event has a very small likelihood of occurrence.
Pipeline Accident	-	Further analysis required per NUREG-1407. See Section 5.5.
Intense Precipitation	-	Further analysis required per NUREG-1407. See Section 5.4.
Release of Chemicals in Onsite Storage	-	Further analysis required per NUREG-1407. See Section 5.5
River Diversion	1	Not applicable.

Table 5.2.2 (cont.)
External Events Considered

Sandstorms	1	This is not relevant for this region; the plant is sufficiently situated from the sand sources.
Seiche	1	This is a great lake phenomenon.
Seismic	-	This event is addressed in a separate analysis.
Snow	2,3	Snow melt causing flooding is included under external flooding. Snow load is bounded by PMP loading which will be evaluated as part of intense precipitation.
Soil Shrink-swell Consolidation	3	Site suitability evaluation and site development for the plant were designed to preclude the effects of this hazard (CCNPP UFSAR, Rev. 15, Section 2.4).
Storm Surge	2	Included in the effects of external flooding.
Transportation Accidents	-	Further analysis required per NUREG-1407. See Section 5.5.
Tsunami	4	The occurrence of tsunamis is infrequent in the Atlantic Ocean. It is believed that the site will not be subjected to a significant tsunami effect. The maximum expected tsunami would result in only minor wave action. The effect is bounded by the hurricane tidal effects, which was a more critical factor in the design (CCNPP UFSAR, Rev. 20, pg. 2.6-19).
Toxic Gas	-	Further analysis required per NUREG-1407. Included in the effect of release of offsite or onsite chemicals. See Sections 5.5.
Volcanic Activity	1	This is not a concern for east coast sites.
Wave	-	Included in the effects of external flooding.
Turbine Generated Missiles	-	As per NUREG/CR-4839, the frequency of turbine generated missile damage will be negligible if appropriate maintenance and inservice inspections are in place. A more detailed analysis has shown that turbine generated missiles are not a risk significant issue at CCNPP. See Section 5.6.

Table 5.3.6.1
Split Fraction Definition and Values

Split Fraction	Description	Failure Probability
DC	**125VDC Bus 21 (2D01) remains energized (short term)**	
DC1	125VDC Bus 21 energized for 4 hrs, LOOP and all supports available	5.75E-04
F1	**AFW delivers adequate feedwater flow**	
F1R	AFW delivers adequate flow, given SBO with S/G overfill	3.60E-01
F1T	AFW delivers adequate flow, given SBO with S/G overfill, decreased decay heat due to early shutdown on hurricane imminent	1.76E-01
GE	**EDG 1A starts and runs providing power to 4KV Bus 11**	
GE5	EDG 1A starts & provides power to 4KV Bus 11, LOOP>11 hours, all support systems available (ASA)	7.74E-02
GF	**EDG 2B starts and runs providing power to 4KV Bus 24**	
GF5	EDG 2B starts & provides power to 4KV Bus 24, LOOP>11 hours, ASA	1.10E-01
GG	**EDG 1B starts and runs providing power to 4KV Bus 14**	
GG5	EDG 1B starts & provides power to 4KV Bus 14, LOOP>11 hours, EDG 2B succeeds, ASA	1.05E-01
GH	**EDG 2A starts and runs providing power to 4KV Bus 21**	
GH5	EDG 2A starts & provides power to 4KV Bus 21, LOOP>11 hours, EDG 1B & 2B not questioned or succeed, ASA	1.11E-01
GHA	EDG 2A starts & provides power to 4KV Bus 21, LOOP>11 hours, either EDG 1B & 2B fails - second EDG succeeds or is not questioned, ASA	1.41E-01
GW	**SW Hdr 22 & SRW Hdr 22 operate**	
GW1	SW & SRW HDR 22 operate, given Unit 1 SW & SRW HDRs succeed, or not asked, top DW available	3.62E-03
HX	**Operator controls AFW flow**	
HX3	Operator controls AFW flow, locally due to Control Room AFW Flow control support unavailable for either flow path where flow exists, and no S/G level indication available, procedure EOP-08 applies	5.08E-01
HZ	**OP locally ventilates both SWGR rms using temporary fans**	
HZ2	Operator (OP) locally ventilates both SWGR Rooms using temporary fans, LOOP, 4KV Bus 14 available	3.63E-02
HZ8	OP locally ventilates both SWGR Rooms using temporary fans, LOOP, 4KV Bus 14 available - loss of HVAC due to hurricane weather conditions	2.82E-03

HZ9	OP locally ventilates both SWGR Rooms using temporary fans, LOOPS, 4KV Bus 14 failed - loss of HVAC due to hurricane weather conditions	1.59E-02
MH	**OP recovers a failed steam admission line (BHEF1I)**	
MH5	Operator recovers failed steam admission line to AFW turbine driven pumps, no control air or operating or indicating power available - local operation of valves	4.45E-02
PH	**Both PORV flow paths are isolated as required (long term action (1 HR))**	
PH5	Operator isolates Stuck Open PORV(s) given one PORV opens during an SSSA where U-4000-21 Service Transformer failed and one-of-two MCCs fail over the course of the mission or an SSSA occurring as a result of an MCC as an initiator	1.87E-02
PH6	Operator isolates Stuck Open PORV(s) given both PORVs open during an SSSA with U-4000-21 Service Transformer failed (thus PV not questioned) and one-of-two MCCs fail over the course of the mission	3.74E-02
TF	**AFW pump 11 operates**	
TF1	AFW Turbine Pump 11 provides adequate flow given Motor Pump 13 successful or not questioned, OP actions successful	3.17E-02
TG	**AFW pump 12 operates**	
TGA	AFW Turb, Pump 12 works, AFW Turb Pump 11 fails and 13 fails, Op actions succeed	2.71E-01
TGQ	AFW Turbine Pump 12 works, given AFW Turb Pump 11 succeeded, Operator Global actions succeed, flow control fails (HX=F) causing SG overfill human action BHEF1Y not questioned) - evaluated in recovery from overfill human action)	5.29E-02
UQ	**OPs do not underfeed the S/Gs when flow is lost**	
UQ3	OPs do not underfill S/Gs when AFW flow control is lost, given remote flow control failed and no S/G indication, EOP-08	5.00E-02
XW	**OP supplies a 120VAC Vital Panel from 208/120VAC Instrument Bus**	
XW2	OP supplies a 120VAC Vital Panel from 208/120 VAC Instrument Bus, given MCC 104R is available	5.48E-02
Z1	**Demineralized Water Storage Tank**	
Z11	DWST fails due to tornado missile, given ZH=F	9.21E-02
Z1S	DWST fails due to tornado missile, given ZH=S	0.00E+00
Z2	**Condensate Storage Tank #21**	
Z21	CST #21 fails due to tornado missile, given ZH=F	9.21E-02
Z2S	CST #21 fails due to tornado missile, given ZH=S	0.00E+00
Z3	**Condensate Storage Tank #11**	
Z31	CST #11 fails due to tornado missile, given (ZH=F * (Z1=S + Z2=S)	9.21E-02
Z32	CST #11 fails due to tornado missile, given Z1=F*Z2=F	4.60E-01
Z3S	CST #11 fails due to tornado missile, given ZH=S	0.00E+00

Z4	**#11 Pretreated Water Storage Tank**	
Z41	#11 PTWST fails due to a tornado missile, given ZH=F * ((Z1=F * Z2=S * Z3=S) + (Z1=S * Z2=F * Z3=S) + (Z1=S * Z2=S * Z3=F) + (Z1=S * Z2=S * Z3=S))	1.07E-01
Z42	#11 PTWST fails due to a tornado missile, given two of three (Z1-Z3) have failed	4.60E-01
Z4F	#11 PTWST fails due to a tornado missile, given Z1=F * Z2=F * Z3=F	1.00E+00
Z4S	#11 PTWST fails due to a tornado missile, given ZH=S	0.00E+00
Z5	**#12 Pretreated Water Storage Tank**	
Z51	#12 PTWST fails due to a tornado missile, given ZH=F * (none or one of Z1 thru Z4 fails)	2.25E-02
Z52	#12 PTWST fails due to a tornado missile, given two of four (Z1-Z4) tanks have failed	4.60E-01
Z5F	#12 PTWST fails due to a tornado missile, given three or four previous tanks (Z1-Z4) fail	1.00E+00
Z5S	#12 PTWST fails due to a tornado missile, given ZH=S	0.00E+00
Z6	**EDG 0C (note that EDG 0C is guaranteed to fail on a tornado point strike**	
Z61	EDG 0C fails due to tornado missile, (LTOR4 * ZH=S)	2.70E-01
Z62	EDG 0C fails due to tornado missile, given (LTOR1) + (LTOR3 * ZH=S)	3.00E-01
Z63	EDG 0C fails due to tornado missile, given (LTOR5) + ((LTOR3 + LTOR4) * ZH=F)	4.60E-01
Z6F	EDG 0C fails due to tornado missile, LTOR2	1.00E+00
Z7	**EDG 2A**	
Z71	EDG 2A fails due to a tornado missile, Z6=F	3.79E-03
Z7S	EDG 2A fails due to a tornado missile, Z6=S	0.00E+00
Z8	**EDG 1B**	
Z81	EDG 1B fails due to a tornado missile, Z6=F	3.79E-03
Z8S	EDG 1B fails due to a tornado missile, Z6=S	0.00E+00
Z9	**#11 Fuel Oil Storage Tank**	
Z91	#11 FOST fails due to a tornado missile, Z6=F	8.95E-02
Z9S	#11 FOST fails due to a tornado missile, Z6=S	0.00E+00
ZA	**#11 Refueling Water Tank**	
ZA1	#11 RWT fails due to a tornado missile, Z6=F	1.95E-01
ZAS	#11 RWT fails due to a tornado missile, Z6=S	0.00E+00
ZB	**Control Room HVAC and Unit 1 Switchgear Room HVAC (guaranteed Failure on a point strike)**	
ZB1	Control Room HVAC and U1 Swgr Room HVAC fail due to tornado, given LTOR1	3.00E-01
ZB2	Control Room HVAC and U1 Swgr Room HVAC fail due to tornado, given (LTOR3) * (ZH=S)	3.30E-01
ZB3	Control Room HVAC and U1 Swgr Room HVAC fail due to tornado, given (LTOR2), OR (LTOR4 * ZH=S) *See Note 1	4.60E-01
ZBF	Control Room HVAC and U1 Swgr Room HVAC fail due to tornado, given LTOR5 or ((LTOR3 +LTOR4) * ZH=F)	1.00E+00

ZC	**Unit 1 Service Water Head Tanks (If ZB has failed guaranteed point strike. Also if ZB=S, then ZC is guaranteed to succeed.)**	
ZC1	Unit 1 Service Water Head Tanks fail due to a tornado missile, ZB=F	1.41E-02
ZCS	Unit 1 Service Water Head Tanks fail due to a tornado missile, ZB=S	0.00E+00
ZD	**#21 Refueling Water Tank**	
ZD1	#21 RWT fails due to a tornado missile, given (LTOR1 * Z6=S * ZB=S)	4.10E-02
ZD2	#21 RWT fails due to a tornado missile, given (LTOR1 * Z6=F) OR (LTOR2)	5.85E-02
ZD3	#21 RWT fails due to a tornado missile, given (LTOR1 * ZB=F) OR (LTOR5)	6.44E-02
ZD4	#21 RWT fails due to a tornado missile, given LTOR3 OR LTOR4	1.95E-01
ZE	**EDG 2B**	
ZE1	EDG 2B fails due to a tornado missile, given (LTOR1 * Z6=S * ZB=S * ZD=S)	7.97E-04
ZE2	EDG 2B fails due to a tornado missile, given (LTOR1 * Z6=F * ZD=S), OR (LTOR2 * ZD=S)	1.14E-03
ZE3	EDG 2B fails due to a tornado missile, given (LTOR1 * ZB=F * ZD=S), or (LTOR5 * ZD=S)	1.25E-03
ZE4	EDG 2B fails due to tornado missile, given (LTOR3) or (LTOR4) or (ZD=F)	3.79E-03
ZF	**Unit 2 Switchgear Room HVAC (guaranteed failure on point strike)**	
ZF1	Unit 2 Switchgear HVAC fails due to a tornado, given (LTOR1 * ZB=S)	2.10E-01
ZF2	Unit 2 Switchgear HVAC fails due to a tornado, given (LTOR2 * ZB=S)	2.70E-01
ZF3	Unit 2 Switchgear HVAC fails due to a tornado, given ((LTOR1 + LTOR2) * ZB=F), OR (LTOR5)	4.60E-01
ZFF	Unit 2 Switchgear HVAC fails due to a tornado, given (LTOR3) + (LTOR4) + (ZD=F) + (ZE=F)	1.00E+00
ZG	**Unit 2 Service Water Head Tanks**	
ZG1	Unit 2 Service Water Head Tanks fail due to tornado missile, given ZF=F	1.41E-02
ZGS	Unit 2 Service Water Head Tanks fail due to tornado missile, given ZF=S	0.00E+00
ZH	**Tank Farm receives a Tornado point strike and the Fire Pump House fails**	
ZH1	Tank Farm receives a tornado point strike and the Fire Protection Building fails, given LTOR3 or LTOR4	2.10E-01
ZH2	Tank Farm receives a tornado point strike and the Fire Protection Building fails, given LTOR2 or LTOR5	3.00E-01
ZHF	Tank Farm receives a tornado point strike and the Fire Protection Building fails, given LTOR1	1.00E+00

Note 1 *If ZH=F a point strike occurred on the tank farm, if the initiating event started at Node 3 or Node 4 then the tornado is*
also assumed to strike Node 5 (also increase probability that Node 2 struck).

Table 5.3.6.2
High Wind Sequences

	Initiator	Sequence Probability		Sequence	Cumulative Importance	Sequence Importance
1	LHUR1	2.56E-07	HRIF	HZ8*HX3*F1T	5.86%	5.86%
2	LHUR1	1.52E-07	HRIF	GG5*HZ9*HX3*F1T	9.34%	3.48%
3	LHUR1	1.05E-07	MRIF	HZ8*PH6	11.75%	2.41%
4	LTOR5	9.67E-08	HRIF	ZF3*HZ2*HX3*F1R	13.97%	2.22%
5	LTOR4	9.67E-08	HRIF	ZB3*HZ2*HX3*F1R	16.19%	2.22%
6	LHUR1	7.67E-08	HRIF	HZ8*HX3*TGQ	17.95%	1.76%
7	LHUR1	7.27E-08	HHIP	GE5*GF5*TF1*TGA	19.62%	1.67%
8	LHUR1	7.25E-08	HRIF	HZ8*HX3*UQ3	21.28%	1.66%
9	LTOR3	6.94E-08	HRIF	ZB2*HZ2*HX3*F1R	22.87%	1.59%
10	LHUR1	6.45E-08	HRIF	HZ8*HX3*MH5	24.35%	1.48%
11	LHUR1	5.22E-08	HRIF	DC1*HX3*F1T	25.55%	1.20%
12	LHUR1	4.59E-08	HRIF	HZ8*HX3*TF1	26.60%	1.05%
13	LHUR1	4.55E-08	HRIF	GG5*HZ9*HX3*TGQ	27.64%	1.04%
14	LTOR2	4.45E-08	HRIF	ZB3*ZF3*HZ2*HX3*F1R	28.66%	1.02%
15	LTOR4	4.42E-08	HRIF	ZH1*HZ2*HX3*F1R	29.68%	1.02%
16	LTOR3	4.42E-08	HRIF	ZH1*HZ2*HX3*F1R	30.69%	1.01%
17	LHUR1	4.30E-08	HRIF	GG5*HZ9*HX3*UQ3	31.68%	0.99%
18	LHUR1	4.26E-08	HRIF	GE5*GH5*XW2*HX3*F1T	32.65%	0.97%
19	LHUR1	4.09E-08	MRIF	GG5*HZ9*PH5	33.59%	0.94%
20	LHUR1	4.02E-08	HRIF	GE5*GG5*GHA*GWI*HX3*F1T	34.51%	0.92%
21	LHUR1	3.91E-08	HHIP	GE5*TF1*TGA*F9E	35.41%	0.90%
22	LHUR1	3.82E-08	HRIF	GG5*HZ9*HX3*MH5	36.28%	0.87%
23	LHUR1	3.38E-08	HRIF	GE5*GG5*GWI*TF1*TGC	37.06%	0.78%
24	LHUR1	3.04E-08	HRIF	HZ8*TF1*TGC	37.76%	0.70%
25	LHUR1	2.98E-08	HRIF	R41*HZ9*HX3*F1T	38.44%	0.68%
26	LTOR1	2.90E-08	HRIF	ZB1*ZF3*HZ2*HX3*F1R	39.11%	0.67%
27	LHUR1	2.72E-08	HRIF	GG5*HZ9*HX3*TF1	39.73%	0.62%
28	LHUR1	2.65E-08	HHIP	W41*GE5*TF1*TGA	40.34%	0.61%
29	LHUR1	2.61E-08	MRIF	GE5*GG5*AU1*PV7	40.94%	0.60%
30	LHUR1	2.41E-08	HRIF	HZ8*F1G	41.49%	0.55%
31	LHUR1	2.40E-08	HRIF	DB1*HZ7*HX3*F1T	42.04%	0.55%
32	LHUR1	2.29E-08	HRIF	GE5*GF5*GGA*GHG*HX3*F1T	42.56%	0.52%
33	LHUR1	2.29E-08	HHIP	GE5*TF1*TGA*FO2	43.09%	0.53%
34	LHUR1	2.28E-08	HRIF	GF5*GGA*HZ9*HX3*F1T	43.61%	0.52%
35	LHUR1	2.13E-08	MRIF	GF5*GGA*XW2*PH5	44.10%	0.49%
36	LTOR5	1.95E-08	MRIF	ZF3*HZ2*PH6	44.55%	0.45%
37	LTOR4	1.95E-08	MRIF	ZB3*HZ2*PH6	44.99%	0.44%
38	LTOR1	1.95E-08	MGIP	Z11*Z21*Z32*K5J	45.44%	0.45%
39	LHUR1	1.80E-08	HRIF	GG5*HZ9*TF1*TGC	45.85%	0.41%
40	LHUR1	1.75E-08	HHIP	GE5*MG1*TF1*TGA	46.26%	0.41%
41	LHUR1	1.72E-08	HRIF	GE5*GG5*GWI*MH2	46.65%	0.39%
42	LHUR1	1.63E-08	HHIP	GE5*IH1*TF1*TGA	47.03%	0.38%
43	LHUR1	1.57E-08	MRIF	GE5*GG5*SLL	47.39%	0.36%
44	LHUR1	1.57E-08	HRIF	DC1*HX3*TGQ	47.75%	0.36%
45	LHUR1	1.55E-08	HRIF	HZ8*MH2	48.10%	0.35%
46	LHUR1	1.54E-08	HRIF	W31*GE5*XW2*HX3*F1T	48.45%	0.35%
47	LHUR1	1.48E-08	HRIF	DC1*HX3*UQ3	48.79%	0.34%
48	LHUR1	1.43E-08	HRIF	GG5*HZ9*F1G	49.12%	0.33%
49	LTOR5	1.42E-08	HRIF	ZF3*HZ2*HX3*TGQ	49.45%	0.33%
50	LTOR4	1.42E-08	HRIF	ZB3*HZ2*HX3*TGQ	49.77%	0.32%

SECTION 6

LICENSEE PARTICIPATION AND INTERNAL REVIEW TEAM

6.1 IPEEE Program Organization

Calvert Cliffs Nuclear Power Plant's IPEEE was developed by a project team consisting primarily of a project manager and eight BGE PRA engineers. For each major task (seismic, fire, and other external events) there was a lead project engineer. Several of the project team members were actively involved in developing the CCNPP IPE and therefore had previous knowledge of PRA methodology.

Contractors with special expertise assisted in the seismic, fire and other external events analyses:

EQE International Incorporated, together with IPEEE project team members, performed the seismic plant walkdowns for the initial component screening, component HCLPF and fragility calculations. Stevenson & Associates performed the soil liquefaction, soil-structural interaction and structural fragility analyses.

The majority of the fire PRA was performed by the project team. ERIN Engineering and Research provided technical support for the Fire PRA in terms of technology transfer, general technical insights, development of detailed fire modeling and barrier assessment methods.

SCIENTECH (formerly NUS) performed the risk analysis of other external events with the exception of High Winds and Turbine Missiles.

6.2 Composition of Independent Review Teams

The IPEEE analysis received four levels of review:

1. Selective independent peer reviews of assumptions, methodology and technical calculations by other IPEEE team members who were not involved in the original analysis.
2. All technical work performed by outside contractors was reviewed by the IPEEE team members to ensure technical knowledge transfer and accuracy of results.
3. High level peer reviews by recognized experts.
4. Selective independent peer reviews by discipline engineers who are familiar with the subject.

The following discussion provides a general description of how these four levels of review were applied for the external events initiators.

6.2.1 Seismic

The component fragility calculations were reviewed by EQE and were owner acceptance reviewed by a BGE IPEEE project team member. EQE's internal review was performed in accordance with their QA procedures in order to meet BGE's safety-related purchase order requirements. Similarly, Stevenson & Associates conducted an internal review of soil liquefaction, SSI, and structural fragility analyses. A BGE IPEEE team member provided a owner acceptance review on these structure related analyses. A BGE senior civil-structural engineer who was involved in the seismic A-46 project also provided a limited review on the analyses performed by EQE and Stevenson & Associates. In addition, ERIN Engineering and Research provided an overall peer review on all seismic related analyses. The peer review comments and their resolutions are included in Table 6.3.

6.2.2 Other External Events

This analysis was primarily performed and reviewed by SCIENTECH in accordance with their QA procedures in order to meet the requirements of BGE's safety-related purchase order. These analyses were owner acceptance reviewed by a BGE IPEEE project team member. Some sections of the analyses which were performed by BGE IPEEE team members (High Winds and Turbine Missiles) were independently reviewed by another team member. BGE's owner acceptance comments and their resolutions are included in Table 6.3.

6.2.3 Fire

The majority of the fire PRA was performed by the BGE IPEEE project team with technical assistance from ERIN. BGE senior fire protection engineers provided technical input to many aspects of the analysis and also provided comments on the early draft of the summary report. In addition, an peer review for the entire fire PRA was performed by SCIENTECH. The peer review comments and their resolutions are included in Table 6.3.

6.3 Areas of Review and Major Comments

Table 6.3 provides a matrix of subject, comments and resolutions that resulted from the internal and external IPEEE analysis peer review either by the project team or contractors.

6.4 Resolution of Comments

See Table 6.3.

Table 6.3 IPEEE Peer Review Comments and Resolution

Subject	Comment	Resolution
Fire	In the Control Room, the cabinet fire frequency is distributed equally among all sixty cabinets. This is not consistent with the Fire PRA Implementation Guide which describes a method of apportionment based on the nature of specific fire sources contained within the individual cabinets. While this may be the recommended approach, other methods may have been used in previous fire PRAs and IPEEE submittals which do not require this level of information. A commonly used approach is to distribute the fire frequency based on the relative size (occupied floor area) of each cabinet. Whatever method is used, the analysis should include some form of justification. NOTE: If the CCNPP cabinets are of roughly equal size, the approach already used may be perfectly adequate. In that case, all that would be required is a statement of the basis for the frequency distribution.	There are 99 physical panels for both Unit 1 and Unit 2 in the Main Control Room. The following weighting factor is assigned to panels based on the size of the panel. It is assumed that the combustible loading associated with a panel is proportional to the size and design of the panel. Vertical electrical panel (main) 1.00 Benchboard 0.50 Small sub-panels 0.25 Enclosed standalone panels 0.50 Consistent with the Implementation Guide, the panel fire ignition frequencies are determined by dividing the number of critical panels (54 "weighted" panels) into the total room fire ignition frequency. This approach is conservative as it apportions the entire room ignition frequency to the most important panels. This approach precludes the need to perform a transient combustible fire calculation as the transient combustible ignition frequency contribution is subsumed in the more severe critical panel fire scenarios. Given that the fire is suppressed (self-extinguished fires are excluded), it is assumed that 33% of the fires are severe, causing damage to wiring or adjacent cabinets prior to being extinguished. For the 67% of non-severe fires, it is assumed that only the function of concern associated with the cabinet is lost, and that such a failure has already been captured within the component failure rate.
Fire	The Control Room scenario fire event tree appears to credit manual suppression twice. That is the severity factor (0.2) already includes credit for the early suppression of six of the ten cabinet fires in the database [EPRI Fire Events Database]. Manual suppression is then credited again through the use of the HCR suppression model prescribed Fire PRA Implementation Guide. Obviously there is a dependence between these events which must be accounted for. It is recommended that the severity factor be increased to (0.8) which only accounts for the two fires in the data base which appear to have self extinguished, and is independent of the manual suppression model used later.	The Control Room analysis has been updated to incorporate this comment.

Table 6.3 IPEEE Peer Review Comments and Resolution (Continued)

Subject	Comment	Resolution
Fire	Control Room fire analyses presented in some IPEEEs to date have been criticized for not properly accounting for the fire initiation, development, detection and damage time line. The current documentation for the CCNPP analysis also suffers from lack of such information which is particularly important given the use of the HCR model (which is time based) for determining the probability of non-suppression. An analysis of the Sandia Cabinet fire test data has been provided separately, which permits the cabinet fire development to be divided and analyzed in three separate phases: Phase I Pre-Ignition Phase II Pre- Growth Phase III Pre-Control Room Evacuation The timing of each phase, including detection is addressed and can be used to develop a more defendable model.	The Control Room analysis has been updated to incorporate this comment.
Fire	The assessment of compartment boundaries utilizes the criteria provided in the FIVE methodology fire compartment interaction analyses (FCIA). However, a key step in that analysis appears to have been omitted from the discussion provided to date. That is, compartment boundaries (which are not rated barriers) must be walked down to ensure there are no combustible concentrations, or combustible continuity, at the compartment interfaces which might otherwise compromise the integrity of the barriers. Such walkdowns need to be performed and documented.	Walkdowns of fire areas have been conducted and documented for the PRA Compartment Zoning Analysis. The results of these walkdowns, and review of the Appendix R drawings, are largely the basis for establishing compartment boundaries and creating composite multi-room compartments. A detailed fire modeling is performed on a room-by-room basis. The concentration of combustibles is considered in the development of fire scenarios, including transients.
Fire	The transient fire assessment method, described in Attachment C, incorrectly applies the FIVE approach. The CCCNP analysis states that the core damage frequency (F_t) due to transient fires in a given room is: $F_t = F_{it} * u * p * w * P_{ft} * CCDP$ where F_{it} is the transient fire frequency (other parameters are defined in FIVE) which is non-conservative in that the transient fire frequency (F_{it}) already accounts, to some extent, for the likelihood that transient material is located in the room and is exposed. Parameters p and w are therefore not independent of F_{it}, as implied by the equation. The FIVE and CCNPP equations are similar, with the exception that, in the former, F_{it} is replaced by F_1, the total room fire frequency (see FIVE page 6-38). That is the FIVE equation, which it must be emphasized as a screening (cont)	The alternative approach is now used to calculate the frequency of transient fires for all rooms.

Table 6.3 IPEEE Peer Review Comments and Resolution (Continued)

Subject	Comment	Resolution
	methodology, determines the likelihood of having exposed transient material in the room when any ignition source is present. An alternative approach is to evaluate the potential for transient fires to damage particular targets or groups of targets and evaluate the associated core damage frequency F_{ti}. $F_{ti} = F_{it} * u_i * P_{fs} * CCDP_i$ where u_i is the critical area ratio for the specific target (or group of targets) (l) and $CCDP_i$ is the CDF given damage is limited to that target or group of targets. The analysis is then repeated for each set of targets in the room.	
Fire	The analysis dismisses the fire frequency contribution (8.7E-03) Radwaste Buildings/Areas as identified in FIVE and the Fire PRA Implementation Guide on the basis that there are no separate Radwaste Buildings/Areas at CCNPP. Excluding this contribution is non-conservative, unless it can be demonstrated that the locations being considered in the CCNPP analysis are completely devoid of Radwaste System related components (i.e. the Radwaste components typically found in other plants do not exist at CCNPP or are located outside the rooms being addressed). Note the Fire Events Data Base indicates Radwaste fire sources were pumps, MCCs, heaters and a temporary cable, and the Radwaste system fires were initiated in the Auxiliary Building, Reactor Buildings (BWR), a Chemical Cleaning Building and two other unknown buildings. It is recommended that this issue be revisited with a view to adding a contribution from Radwaste System related component fires, or providing further justification for their exclusion.	Changed Compartment A418 to "Radwaste Area" as this is the only compartment which reflects the Fire Events Data Base description for Radwaste Area. Developed a new Radwaste Fire Frequency Worksheet according to the sample in FIVE and used the Miscellaneous Components Generic Fire Frequency of 8.7E-03. As a result of this change, the Fire Frequency for this compartment changed from 1.97E-02 to 3.52E-02.
Fire	In most cases the ignition source weighting factor calculations should include a total count of all components (of a particular type), or compartments which are situated within the Generic Locations identified in FIVE Reference Table 1.1. It appears that some of the compartments which do not include any PRA related components may have been excluded. This approach leads to somewhat conservative weighting factors.	All remaining non-PRA component compartments are evaluated and associated ignition sources are included in the total weighting factor for each General (Plant) Location. Therefore, the ignition weighting factor is now more realistic.

Table 6.3 IPEEE Peer Review Comments and Resolution (Continued)

Subject	Comment	Resolution
Fire	The CCNPP evaluation of Control Room fires includes a contribution from welding and other transient combustible fires. This is inconsistent with the Fire Events Data Base (Section 3.3.6) which indicates that plant wide sources should not be included in the Control Room fire frequency. In addition, the current CCNPP Control Room analysis only addresses cabinet fires.	The initial evaluation conservatively considered all possible ignition sources. The screening process for the Control Room screens out welding and adjusted the other potential ignition frequencies.
Fire	The CCNPP Switchgear Room transformers are oil filled and are therefore unusual when compared to indoor transformers at other nuclear power plant sites within the U.S. This being the case, the associated fire frequency derived for the CCNPP switchgear transformers (1.72 E-04/transformer year) is not applicable. The actual frequency (and associated severity factor) may be significantly greater based on a review of oil filled yard transformer fires performed by SCIENTECH in support of the Davis Besse analysis (3.95E-03/transformer year).	Based on a review of EPRI's "Economic Risk Management Analysis Models for Electrical Equipment Containing PCBs", an assumed fire frequency of 2.5E-05 for PCB oil filled transformers is used for the Unit 1 and Unit 2 Switchgear Rooms. This is based on a case study of like transformers, including the location and application in a power plant situation. There are a total of fifteen PCB oil filled transformers in all Switchgear Areas for which this value applies.
Fire	For some reason one of the FCIA criteria appears to have been relaxed for the CCNPP application: FIVE < 20000 btu/ft^2 and auto detection installed versus CCNPP < 20000btu/ft^2 The rationale for this relaxation is not clear.	This criteria is no longer used.
Fire	The Control Room analysis will probably need to address Control Room re-entry, and use of equipment which is not controllable from outside the Control Room. Such recovery actions should be addressed when the initial analysis has been completed.	Control Room re-entry was considered in the analysis of a fire in the Control Room and found not to be required.
Fire	Section 4.3.1.4 Screened Rooms: Low Fire Hazard This section gives the impression that an area or compartment may have been screened out on the basis of a low assessed fire hazard even though the potential functional impact may be quite serious. This would be an acceptable approach unless it could be demonstrated quantitatively that the fire frequency is exceptionally low (e.g. less than 1E-06/yr). Experience shows this is very difficult to achieve. The actual approach employed, as documented in the last paragraph of this section, is difficult to understand. A better explanation of the approach and if necessary, additional analyses, are required to justify this method of screening.	This section has been revised to clarify that rooms may be screened for "Low Fire Ignition Frequency" when rooms are both low fire hazard and low functional impact. Additional analysis has been incorporated to quantitatively demonstrate that the fire frequency in these rooms is >1E-06/yr. (i.e., Cable Chases) or, through fire modeling techniques (i.e., Aux Building Stairtowers), that no hazard exists.

Table 6.3 IPEEE Peer Review Comments and Resolution (Continued)

Subject	Comment	Resolution
Fire	Section 4.3.1.5 Containment The current screening implementation for CCNPP containment fires is not complete. In order to satisfy the FIVE approach, a qualitative review of any CCNPP containment fires is required, together with an evaluation of the RCP fire suppression systems and potential for any fires to cause damage to redundant shutdown trains via direct exposure or hot gas layer formation. (An example from the Perry NPP analysis was provided separately)	The FIVE Methodology does not include ignition source information for the containment location because of the small number of fire events and the conclusion, by previous fire PRAs, that such fires were not risk significant. However FIVE recommends a qualitative assessment to determine if a further, more detailed, analysis of the containment is needed. FIVE cites two issues that should prompt a detailed analysis: 1. Plant experience indicating that containment fires have occurred on a recurring basis. 2. The potential for redundant trains of critical equipment within containment to be exposed to the same fire plume or be in a confined space subject to damage by a hot gas layer. A qualitative review of the CCNPP containment fires concluded that: 1. There is no plant specific history of, or known susceptibility to fires inside containment. 2. The separation of redundant critical equipment and cable trains is such that no single fire (plume or hot gas layer) will damage both trains. When necessary, shielding or other measures prevent flame spread across the redundant trains. Also, the RCP lube oil system is encapsulated and uses an oil collection system to divert and control any leakage.
Fire	Page 4-160 Main Control Room. The control room evacuation scenario appears to credit manual suppression twice. the severity factor (0.2) already includes credit for the early suppression of six of the cabinet fires in the database (see Implementation Guide page D-7). Manual suppression is then credited again through the use of the HCR suppression model prescribed in Fire PRA Implementation Guide. Obviously there is a dependence between these events which must be accounted for. It is recommended that a cabinet fire severity factor of 0.67, which is independent of suppression, would be more justified for use in this case. The basis is as follows: In total there are twelve fires assigned to the control room location in the data base.	This comment has been incorporated, i.e., 0.67 is used (see Control Room Panel Fires, Attachment 4-I).

Table 6.3 IPEEE Peer Review Comments and Resolution (Continued)

Subject	Comment	Resolution
	However, one actually occurred outside the control room, one was a re-occurring event, and one was a kitchen fire (not associated with an electrical cabinet). There are therefore actually nine fires which can be considered as applicable to control room cabinets. Of these, two fires self extinguished, five were suppressed by manual extinguisher and no information was available for the remaining two events. Presumably the two fires for which no information exists were fairly benign, so let's assume one of the two self extinguished. The severity factor (independent of suppression) is therefore 6/9 or 0.67.	
Fire	Page 4-166 Main Control Room. This page presents information on times to reach three stages of fire growth and the associated probabilities of non suppression. In fact the current CR analysis only utilizes this data for the last stage of growth (Control Room Obscured). The data provided for the second growth stage (significant fast growing fire leading to inter cabinet propagation occurs within seven minutes with a PNS of 4.9E-02) contradicts the values actually used in the analysis which are based on an alternate approach (PNS=0.2). While both approaches are valid, the later (as currently used) approach is more conservative. In any event presentation of both approaches is misleading for the reader.	The information on this page has been changed to read as follows: Based on the HCR model derivation in the Fire PRA Implementation Guide, the probability of non-suppression in 15 minutes with no in-cabinet detection is 3.40E-3.
Fire	Page 4-239 Yard. At the time of reviewing the Yard Transformer fires analysis BG&E were in the process of trying to correlate the fire frequency data, as categorized in the FIVE methodology, with the actual CCNPP configuration and plant specific fire damage scenarios. Guidance was provided for re-evaluating the generic data to derive individual transformer fire frequencies and associated severity factors together with a event tree approach for assessing the various possible combinations of fire and smoke related damages. An example from the H.B. Robinson study was provided. BG&E's implementation of the approach will be reviewed upon completion.	Fire analysis results for Yard has been completed including the Yard Transformer Fires and associated frequencies. See Tab Yard in Section 4.0 of this submittal report.

Table 6.3 IPEEE Peer Review Comments and Resolution (Continued)

Subject	Comment	Resolution
Fire	The Turbine Building analysis is appropriate given the potential for a severe fire to impact various critical ventilation systems. However the analysis should address the likelihood of an unfavorable wind direction/speed in the same manner as the Yard Transformer fire analysis.	An unfavorable wind direction / speed is now considered.
Fire	General All fire sequence analyses need to address the impact of operators implementing post fire shutdown procedures which may disable specific equipment even when there is no related fire damage. For example, procedures for control room evacuation apparently require termination of main feedwater which is not considered in the current analysis.	The fire risk assessment associated with the evacuation of the Control Room includes the evaluation of the impact of implementing the Appendix R safe shutdown procedure for Control Room evacuation (AOP-9A).
External Floods	Leakage from the Turbine Building to the Auxiliary Building is said to be insignificant. Can this be referenced to a walkdown or to some other document?	Due to open stairways, gratings, etc., in the Turbine Building floors, there will be no significant water hold up. Water will flow unrestricted to the Turbine Building basement, where there are no interconnecting doorways or other flood pathways to the Auxiliary Building. Walkdowns performed as part of the internal flooding analysis confirmed this, and an appropriate reference has been added to the text.
External Floods	Ponding on the Auxiliary Building roof at 69' could cause flooding problems in the EDG rooms if the drains are not working on the roof. This should be brought out clearly in the conclusions. Presumably procedures are in place or should be in place to assure that these drains will be inspected to assure functionality. This needs to be confirmed. Common cause failure of the EDG's and the RWST pumps during a severe storm which could cause LOOP would perhaps be significant. This also applies to the Auxiliary Building Decking at 118'.	Roof drains are inspected by a system engineer monthly based on Plant Engineering Section Guide, PEG-7. If the drains are blocked, an Issue Report is written to clear the blocked drains.
External Floods	Section 3.2.3, in calculating the allowable live load for the concrete slab, a safety factor of 0.6 is used as a divisor. In other places, a safety factor of 1.7 is used as a multiplier. Is there a difference between these two?	Design calculations for concrete slabs and metal decking apply safety factors in different ways. The determination of ultimate strength performed in this analysis reflect this. Realistic value is based on total working stress (166 psf), not live load (100 psf). Thus, the analysis is correct. The report has been modified to make this clear.

Table 6.3 IPEEE Peer Review Comments and Resolution (Continued)

Subject	Comment	Resolution
External Floods	The BGE reviewer's comments are generally on providing references and/or calculations to clarify some of the values used.	The Tier 2 report has been revised to incorporate the comments.
High Winds	In assessing the impact on the plant due to high winds following a LOOP, should address provision for maintaining decay heat removal after CST source runs out after six hours (i.e., other methods such as RHR or alternate water sources).	If water sources for CST including all makeups are gone, OTCC mode of long term decay heat removal would be used. This has been added to the Tier 2 report.
High Winds	The BGE reviewer's comments are generally on providing references and/or calculations to clarify some of the values used.	The Tier 2 report has been revised to incorporate the comments.
High Winds	The BGE IPEEE project team has reanalyzed the high wind risk impact on the plant. The team has also revised the Tier 2 report to incorporate the new analysis and results.	The new analysis and results have been incorporated by SCIENTECH.
High Winds	Internal review comments on the analysis performed by the BGE project team are generally editorial and on providing additional references for clarification.	Comments have been incorporated.
Aircraft Hazards	Add risk analysis to assess the impact of aircraft hazard on the Independent Spent Fuel Storage Installation (ISFSI) facility.	A separate aircraft hazard analysis has been performed for the ISFSI facility and the results were documented in the Tier 2 report.
Aircraft Hazards	Airways used in the analysis: There are four low altitude airways within approximately eleven miles of the site. The one not accounted for is V20-33.	This route neither meets the requirement for the analysis stated in SRP-3.5.1.6 (less than two miles) nor coincides with the PXT VORTAC, thus it plays no role in the analysis.
	Military flights should also be included in the analysis for both high and low structures since both are used almost equally.	Split of military flights could be done reducing the final accident frequency slightly, but it is more conservative to force all military flights to be counted in the low altitude given the lack of specific information.
	Airways are eight NM wide, four NM was used in the analysis.	Four NM was intended to be conservative since the lower the width, the higher the probability of accident.
Aircraft Hazards	Mears Creek is a private field close to the plant that was not considered (approximately four NM).	A visit on the actual site indicated that this privately operated strip is only sporadically used for leisure purposes by its owner/operator. Two small single-engine aircraft are based there and are the only aircraft expected to use the field. It is thus concluded that there is no risk concern for this airfield.

Table 6.3 IPEEE Peer Review Comments and Resolution (Continued)

Subject	Comment	Resolution
Aircraft Hazards	Helicopter crash risk contributes to 74% of the total aircraft hazard and exceeds the NUREG-1407 screening criteria of 1.0E-6/yr.	A memo was issued from BGE to Helicopter Transportation Services to limit the number of flights over the protected areas to an acceptable level of crash risk. This is considered as a vulnerability.
Aircraft Hazards	Other BGE reviewer's comments in general are on providing references and/or calculations to clarify some of the values used.	The Tier 2 report has been revised to incorporate the comments. The calculation on the intake structure roof area was also corrected. It did not change the overall crash probability however.
Transportation and Nearby Facilities Hazards	The map (Attachment 1) shows two facilities, Bowen Gas and Appliances and Clay's Pest Control, which are within five miles of the site. How does Attachment 2 (Calvert County Hazardous Materials Incident Plan) eliminate these as risk?	A further review of Attachment 2 shows that the county has determined additionally that the quantities of the hazardous materials on the Bowen's and Clay's sites will not affect Calvert Cliffs' site in any way.
Transportation and Nearby Facilities Hazards	Other BGE reviewer's comments are generally editorial and on providing additional references for clarification.	Comments have been incorporated in the Tier 2 report.
On-site Chemical storage	The screening criteria on the chemicals required for Control Room habitability assessment was misinterpreted from Reg. Guide 1.78. Both limits, toxicity and weight should be used.	This comment resulted in a separate hazard analysis for the on-site chemicals storage. About 3000 chemicals were considered. This also resulted in a procedure change on Controlled Materials Management for screening the hazardous chemicals at CCNPP.
Turbine Missiles	Internal review comments on the analysis performed by the BGE project team are generally editorial and on providing additional references for clarification.	Comments have been incorporated.
Seismic	Top Event LK is a surrogate top event included as the last top event in the seismic event tree to model seismic failure of the secondary systems. The intent of Top Event LK is to screen/eliminate sequences from the model that are of low intensity, have no associated seismic failures, and are therefore not initiating events (i.e., do not cause a plant trip). The current use of Top Event LK to achieve this will be difficult to defend. Top Event LK should be eliminated from the model and top events that model secondary systems should be assumed failed for all seismic events.	The Top Event LK fragility is based on the judgment that the seismic capacity of the modeled systems is likely to be as good as or comparable to the relatively low-capacity switchyard fragility. The switchyard (offsite power) fragility is the most limiting seismic top event, (Top Event LJ). The median capacity for LK is conservatively set to one half of that for LJ. The randomness (Br) and uncertainty (Bu) are conservatively assigned a value of twice that for LJ. In addition, LK is only used for seismic initiating events at or below the Operational Basis Earthquake (OBE =0.08G). Top Event LK is set to guaranteed failure for all seismic initiators above this level.

Table 6.3 IPEEE Peer Review Comments and Resolution (Continued)

Subject	Comment	Resolution
	The split fraction value for Top Event LK is based on a fragility family with a median acceleration of approximately 0.1. There is no basis for this median acceleration.	The intent of Top Event LK could have been achieved by truncating the low end of the fragility curves at a higher level, such as the HCLPF value. This has been done in some PRA's and was suggested for Calvert Cliffs. The HCLPF corresponds to about 1% failure fraction. The fragility curves in the Calvert Cliffs PRA are truncated 0.005%. Truncation at the HCLPF was considered but no basis could be found. Top Event LK achieves a result similar to truncation at the HCLPF, although not as dramatic. Truncation at the HCLPF would mean that all seismic top events would be guaranteed success for seismic initiators at or below 0.08pga. We believe that the use of Top Event LK with the current fragility curve truncation provides more realistic modeling than truncation at the HCLPF. Also, most of the functions LK represents are already modeled in the internal events model by initiating events such as Loss of Main Feedwater, Loss of Condenser Vacuum, or Loss of Instrument Air. These initiating events have comparable or greater frequencies than the three lowest-g seismic initiating events used with LK. See Section 3.1.5.2 for additional details on the use of LK.
Seismic	Equipment judged to have a HCLPF peak ground acceleration of 0.3g or greater were screened from the analysis. Surrogate Top Event LA was used to consider all components screened out based on this criteria. Top Event LA represents five different systems, each with a HCLPF of 0.3g, and the condensate storage tank. The failure of Top Event LA is calculated as an "OR" gate with inputs from each of the six system failures. Failure of Top Event LA is mapped directly to core damage. There are non-conservative issues associated with this treatment such as seismic induced fires and floods, relay chatter, LOCA initiators, and containment bypass sequences. Options to consider: 1) remove the surrogate top event from the presentation of realistic risk contributions of seismic events and add it back in as a sensitivity analysis but use two top events instead of one, 2) one for equipment needed to protect the core, and another to prevent a large early release.	The suggested options were incorporated into the analyses. The Seismic event trees were quantified both with and without surrogate Top Event LA. The quantification without LA is used for evaluating sequences and may be used for relative risk contributions. Also, the containment isolation function was broken out of Top Event LA and added back in as a second surrogate top event.

Table 6.3 IPEEE Peer Review Comments and Resolution (Continued)

Subject	Comment	Resolution
Seismic	It is recommended that the final report sections on both seismic induced fires and floods draw specific conclusions on the existence or non-existence of unscreened components whose failure has the potential to create fires or floods that could cause additional damage and dependent failure effects beyond the direct consequences of the earthquake.	This recommendation was incorporated. Seismic-fire interaction, Section 3.1.3.3, was expanded considerably and describes the risk associated with the unscreened components.
Seismic	The generic dismissal of RCS components and piping and associated penetrations may have bypassed adequate consideration of the potential for large early containment isolation failures and bypass events.	See response below
Seismic	Piping and penetration induced bypasses appear to have been generically dismissed. Hence, the potential for plant specific vulnerabilities has not been addressed. The NUREG-CR-4551 analyses for Surry is one example where plant specific vulnerabilities were identified.	See response below
Seismic	The exclusion of seismic induced LOCAs is based on a generic methodology document provided by EQE and not a plant specific evaluation. These components were excluded from the IPEEE walkdown list (see page A67 of the walkdown evaluation sheets; i.e., Ref.3-3. Is this justified to dismiss these events generically?	Most of the piping penetrations were walked down by the SRT and are screened at 0.5 g. However, the large NSSS components (Rx vessel, pressurizer, RCP's, steam generators) are generically screened based on the screening criteria in Table 2-4 of EPRI NP 6041-SL (Ref. 3-4). These guidelines are based on NUREG/CR-4334. Reference 3-4 states that they are based on "a general industry and regulatory consensus that, in fact, there are wide classes of elements in nuclear power plants which have demonstrated a substantial seismic ruggedness either because of their performance in past earthquakes, available generic ruggedness or fragility data, or because generally accepted seismic margin capacity evaluations have been performed on like elements in previous seismic margin or SPRA elements". Appendix A of Reference 3-4 outlines the basis for the screening guidelines. The criteria in Table 2-4 of EPRI NP 6041-SL is that evaluation of NSSS supports is not required if supports are designed for combined loading determined by dynamic SSE and pipe break analysis. Calvert Cliffs meets this requirement. Per the Calvert Cliffs Updated Final Safety Analysis Report, Reference 3-23, the RCS is designated a seismic Class 1 system for seismic design and is designed for three categories of load

Table 6.3 IPEEE Peer Review Comments and Resolution (Continued)

Subject	Comment	Resolution
		combinations and stress. Category three is Normal Operating loadings + pipe rupture + Safe Shutdown earthquake. Under this loading condition, deflection of the NSSS supports is limited to maintain supported equipment within limits (given in Reference 3-23). One of the peer reviewer concerns was that the potential for large-early-release failures caused by failure of the NSSS components may not have been adequately addressed. The current modeling uses containment isolation top event LK which is mapped to a large break in containment. Results show about 81% of seismic CDF results in late containment failure, and about 14 % of seismic CDF is shared between large and small early release (see section 3.1.6.1). So although NSSS component failure is not explicitly modeled, Top Event LK serves to represent the impact of this failure type. This peer review comment was also discussed with the lead SRT expert. He feels that the screening was appropriate. In addition, he pointed out that the equipment arrangement inside containment prevents all but a very limited view of the supports of these large components. A more detailed screening would be done by calculations based on plant drawings. Although this would provide greater assurance, he felt that this was not an area of potential vulnerability and that additional analysis is probably not warranted. Discussion with Surry's seismic personnel revealed more information. The Surry vulnerabilities discussed in NUREG-4551 and cited in the peer review comment appear inconsistent with other analyses performed on Surry's RCP and Steam Generator supports. A Westinghouse analysis was performed using the WESTDYNE computer code (reviewed and found acceptable by the NRC for use on Surry Units). Elastic analysis using 90% of yield allowable showed that the RCP and steam generator supports had a factor of safety of at least 2.5 for the DBE. Based on this analysis, Surry is also screening NSSS components in their seismic PRA. In light of the above, BGE's position is that there is not enough information available to indicate that the screening of the CCNPP NSSS components is inappropriate. We feel that further analysis is not warranted.

Table 6.3 IPEEE Peer Review Comments and Resolution (Continued)

Subject	Comment	Resolution
Seismic	The documents reviewed in the course of this review provided no information on the treatment of passive structures in the screening process. The disposition of these items should be documented.	This comment was incorporated and is explained in the second paragraph of Section 3.1.2.
Seismic	The hazard curve used in the CCSPRA is based on the mean LLNL curve presented in NUREG-1488, and is CCNPP site specific. Exceedance frequencies within the 50 to 1000 cm/s^2 not explicitly provided are calculated by linear interpolation. In the reviewers judgment based on a plot of the mean hazard curve, logarithmic interpolation, rather than linear, would have been a more appropriate method for calculating the exceedance frequencies.	In hindsight, a logarithmic interpolation would have been more appropriate. The linear interpolation used tends to overestimate the hazard frequencies of the interpolated points. A sensitivity analysis showed that the linear interpolation may cause us to overestimate (more conservative) the seismic CDF by around 5% compared to results obtained using a logarithmic interpolation.
Seismic	The CCSPRA defined thirty seismic initiating events. These initiating events span seismic events from approximately 0.01 g to 6.12 g. The partitioning into thirty initiating events introduces a level of complexity into the model that the reviewers believe is not required. The thirty bins were selected with the intent of keeping the contribution of any given initiator to less than five percent of the total seismic core damage frequency. RISKMAN's fragility calculation option uses a piecewise integration scheme which effectively normalizes the calculated failure fraction to generate the best estimate for the failure fraction, regardless of the bin width. This feature combined with the fact that most of the seismic core damage frequency comes from the guaranteed failure bin, and from sequences involving the failure of a single fragility item (e.g., the surrogate Top Event ZA) lead to the observation that many fewer bins could be used to obtain the same degree of accuracy.	Some of the bins could probably have been combined without a significant increase in core damage frequency. This will be considered in future improvements to the Seismic model.
Seismic	Top Event LE (0C EDG Sustains a Seismic Event) models the failure of item 13 from Table 3 of Ref. 3-2. Items 14 and 15 from Table 3 are also related to the diesel generator HVAC and have HCLPF values of less than 0.3g. Why have the items been screened from the model?	These items were missed initially but are now included in the fragility of the 0C-EDG. Top event LE now includes items 13, 14 and 15. Refer to Table 3-3 and Section 3.1.5.2 of this report for a more detailed explanation of Top Event LE.
Seismic	Top Event LG includes the "U1 Turbine Lube Oil Cooler." This item has a HCLPF greater than 0.3g, and therefore is in theory, included in the surrogate Top Event LA and can be excluded from Top Event LG.	This item could have been excluded. It is not wrong to include it because its failure would still lead to SRW failure. It does not have a significant impact on the fragility for top event LG because it is only one of a composite of seven lower capacity component fragilities.

Table 6.3 IPEEE Peer Review Comments and Resolution (Continued)

Subject	Comment	Resolution
Seismic	The EXCEL file indicates that item 10 from Table 3 of the EQE report (Ref. 3-2) is included in Top Event LG. An e-mail received from John Koelbel on 6/12/97 indicated that item 10 is not included. The disposition of item 10 should be verified.	Item 10 is included and should be included in Top Event LG.
Seismic	A review of the split fraction assignment rules for the Seismic Event Tree shows that split fraction values for the seismic top events are set equal to 1.0 when the calculated split fraction value is greater than some cutoff value. This is certainly conservative and simplifies the event tree quantification, but this process should be documented. Most top events (e.g., LA, LE, LG, LJ) are guaranteed failed when the calculated split fraction is greater than 0.95, while others (e.g., LB, LH) are guaranteed failed when the split fraction is greater than 0.5.	This comment has been incorporated. The truncations are explained in Sections 3.1.5.3 and 3.1.5.5.
Seismic	SMCGT1 Event Tree. According to the table from Ref. 6, Top Event SV should be guaranteed failed when Top Event LA is failed. It appears that the rules as written accomplish this, but one has to trace back through the rules for several other top events to verify that SV will indeed be failed when LA is failed. It would be much clearer, and easier for someone less familiar with the model to understand, if the split fraction SVF given LA=F was entered as the first SV split fraction, and split fraction SV1 was left as the default. A similar comment applies to many top events in several event trees. The model currently modifies all rules for a top event, by adding the condition that the related seismic top event must be successful, rather than add a single guaranteed failed split fraction to the top of the list that is dependent on the failure of the related seismic top event. The latter method requires many fewer changes to the base general transient model, and is easier to review.	The intent of the rule changes in question were not intended to guarantee failure of the top event when LA failed, but to speed the model by forcing use of either the guaranteed success or guaranteed failure split fractions for sequences which were guaranteed to go to core damage (sequences where LA is failed). For example, before the change to top event SV rules, split fraction SV1 would be evaluated in both the failed and success states. After the rule change, when LA is failed, either SVS will be used, which has only one path (SV success), or SVF will be used, which also has only one path (SV failure). SV1, which has an intermediate failure probability and would normally be evaluated in both success and failed states, will not be used when LA is failed. The same type of rule change was made for several other top events.

Table 6.3 IPEEE Peer Review Comments and Resolution (Continued)

Subject	Comment	Resolution
Seismic	SMCGT2 Event Tree. It appears that Top Event PT should be guaranteed failed if Top Event LA is failed. But unless the analyst or reviewer is very familiar with the dependencies modeled for a number of previous top events, it is very difficult to determine whether PT will be guaranteed failed when Top Event LA is failed. This may not be strictly a question of style. A subtle change in the modeling assumptions related to the "previous" top events, may have a sneaky impact on the current rules for Top Event PT. It would be simpler and safer to add a rule at the beginning of the split fractions for Top Event PT that fails PT given LA=F.	Top Event PT is not meant to be guaranteed failed if top event LA is failed. A guaranteed failure split fraction, PTF, was added to speed the model. The explanation for Top Event SV, above, applies here as well.
Seismic	A mission time of 24 hours is assumed for the diesel generators following a seismically induced loss of offsite power. As evidenced during the hurricane event at Turkey Point when the plant was dependent on diesel generators for seven days or more, an external event induced loss of offsite power may last much longer than a "typical" loss of offsite power event. Part of the argument for using 24 hours in the internal events analysis (which has been way overused for a long time) is the high probability of recovery of the internal event initiator. While the 24 hour success criterion is typical of previous seismic PRAs, we believe it is difficult to defend especially for the high intensity seismic events. Something like 72 hours might be more appropriate for some of the larger seismic events. The use of 24 hours for the remaining success criteria other than diesel generators is probably acceptable because you are only crediting such system success when there is no station blackout, and it appears that you have been careful not to inadvertently recover seismically damaged equipment.	For internal events, the 24 hour mission time is used in part to account for the repairable nature of redundant equipment. In the seismic analyses, the same logic is applied. So even though a seismic induced loss of off-site power is likely to last more than 24 hours, we use a 24 hour mission time to account for the likely repairs that could be performed on diesels that failed in the 24 hour mission time. At CCNPP, we have five emergency diesel generators; two are self-cooled and three are cooled by Service Water. For long term AFW flow control indication, only one EDG is required. For EDG failures that occur after the 24 hr mission time, it is likely that one of the initial EDG failures would be recovered, and at any time, at least one EDG would be available. For example, one of the likely causes of losing the three SRW dependent EDG's is a loss of SRW cooling. EDG's 2A, 2B and 1B fail at relatively low g levels due to failure of low-seismic-capacity SRW coolers in the Turbine Building. Recovery efforts would be to restore SRW. Turbine Building SRW could be isolated and the more rugged Auxiliary Building portion refilled and restarted to restore SRW to one of the SRW-dependent EDG's. Let us consider extending the mission time for the EDG's. This would require extending the mission times for EDG support systems such as SRW, SW, and 4KV Busses. AFW and other equipment should really be extended as well, for this option. The longer mission time tends to

Table 6.3 IPEEE Peer Review Comments and Resolution (Continued)

Subject	Comment	Resolution
		increase CDF. However, a longer mission time also allows more recovery options, which tends to decrease CDF. We currently do not model EDG recovery in either the internal events or external events models. Recovery scenarios would have to be modeled to avoid getting an overly-conservative increase in CDF. However, these are likely to be very complex.
		We believe that the current modeling yields realistic failure probabilities. In addition, the rest of the industry uses 24 hours also, and this allows for fairer comparisons. Therefore, we do not currently plan on extending the mission time beyond 24 hours for external events.

SECTION 7

PLANT IMPROVEMENTS and UNIQUE SAFETY FEATURES

The results of the IPEEE identified several plant improvements at CCNPP from external events. The plant improvements and corrective actions are listed below:

7.1 Helicopter Flights

Description of Issue

A helipad is located 1000 feet from the northern edge of the nearest vital structure. An average of seventeen corporate flights per year originating from BGE headquarters (about 50 miles north of the plant, near Baltimore, MD) make use of the helipad. There are no specific flight paths or exclusion areas for helicopter flights in the vicinity of the plant.

The expected number of flights over the plant is believed to be a small percentage of the total flights and according to Helicopter Transport Services, Inc., the operator of the BGE corporate helicopters, represents much less than one-half the number of flights. Thus, the vast majority of the helicopter landings and takeoffs likely occur at least 1000 feet from any vital structure.

However, due to the lack of specific flight paths or exclusion requirements, this analysis considered all flights capable of damaging vital structures. With this in mind, approximately 74% of the total probability of air crash impacting vital structures comes from helicopter operations which serve the CCNPP site.

Action

It has always been the policy for helicopter pilots to avoid over flights of the Plant's protected area at all times, unless it was absolutely necessary for reasons of safety or at specific BGE requests. This policy is now enforced by a restriction included in BGE's contract with the Helicopter Transport Services to accordingly limit the number of flights over the Plant's protected area. With this restriction in place, there is now a high degree of confidence that the number of over flights will be significantly less than six per year. As a consequence, the total annual aircraft crash frequency will be less than 1.0E-6. This reduction is considered in the results of this IPEEE submittal. No additional action is required.

7.2 Switchgear Room Ventilation Recovery During A Hurricane

Description of Issue

The air conditioning (A/C) condenser units and air intakes for the switchgear rooms for both units are located on the roof of the Auxiliary Building, adjacent to the Turbine Building. The Turbine Building however, is a potential source of missiles during hurricanes and tornadoes, in that the sheet metal siding is designed to collapse when a differential pressure of 0.45 psi exists (for the Main Steam Line Break event). This differential pressure is assumed to occur with a hurricane wind speed of 100 mph or greater. The collapsed metal siding is assumed to damage the roof-top A/C equipment (i.e., the striking probability is assumed to be 1.0) which in turn fails the switchgear room HVAC systems.

The switchgear room HVAC systems can also be lost for both units due to the loss of ducts and dampers inside the main plant exhaust equipment rooms on the 69-foot level. These rooms are part of the Auxiliary Building and also have metal siding walls. The metal sidings are assumed to collapse at a wind speed of 100 mph or greater, exposing the ducts and dampers to damage from the hurricane winds.

Due to the frequency of hurricanes and the importance of maintaining cooling to the switchgear rooms, the loss of switchgear room ventilation without a high likelihood of recovery is risk significant.

Action

A new realistic switchgear room heat-up model is being developed. An evaluation of the results will determine if procedure changes are necessary to partially stage emergency ventilation. If required, an Emergency Response Plan Implementation Procedure (ERPIP) procedure change will be initiated that considers having personnel partially set up emergency ventilation. Set up of fans and associated portable generators to cool both Units 1 and 2 27' and 45' Switchgear Rooms will be performed within eight hours of the anticipated hurricane arrival. In addition, since fans and their associated portable generators are available for only two of the four switchgear rooms, additional fans and generators will be procured. Actions for effectively cooling the switchgear rooms are credited in this IPEEE submittal. All necessary actions will be implemented by November 30, 1998.

7.3 Smoke Infiltration into the Control Room via Ventilation Intake

Description of Issue

The Control Room and Cable Spreading Room Ventilation system normally takes a suction on a combination of outside air and recirculated air and discharges to a common supply header. This header branches to the Control Room and each of the Cable Spreading Rooms. A fire which produces a significant amount of smoke could migrate to the intakes of the ventilation system, resulting in smoke infiltration into the Control Room and Cable Spreading Rooms. There is no automatic system nor adequate procedural direction to shift the ventilation system into a recirculation mode in order to maintain Control Room habitability. The operators may be forced to abandon the Control Room if significant smoke accumulates. The frequency of such an occurrence from either a Turbine Building or outside transformer fire is risk significant. Note that the impact on the Cable Spreading Room is addressed in Improvement 7.4.

Action

Improvement in procedural direction and training for the operators has been initiated. The revised procedures will direct the Control Room operator to place the Control Room and Cable Spreading Room HVAC into recirculation on the likelihood that smoke could be drawn into the Control Room ventilation intake. Emergency Response Plan Implementation Procedure (ERPIP) procedure changes will be completed by November 27, 1997. Associated OI changes and operator training will be completed by July 31, 1998.

This IPEEE submittal includes the benefit of placing the ventilation system in recirculation. An estimated failure rate of 5E-3 is used for this action.

7.4 Inadvertent Isolation of Switchgear Room and Cable Spreading Room Ventilation

Description of Issue

A total flooding Halon suppression system exists for Units 1 and 2 Switchgear Rooms (one system for each pair of rooms) and for the Unit 1 and 2 Cable Spreading Rooms. Halon will discharge in a room whose detectors sense smoke. Ventilation to the room where smoke is sensed will also be isolated to enable the Halon to be effective. Although these fire protection features are effective when there is a fire in the room, they could result in an inadvertent loss of ventilation as a result of a fire in an area outside of these rooms. A fire which produces a significant amount of smoke near the intakes of the ventilation system could result in the infiltration of smoke into these Halon protected rooms and result in Halon actuation and the loss of ventilation. The smoke could be the result of a Turbine Building or outside transformer fire. Such a fire could challenge all the rooms, both switchgear and cable spreading rooms, at the same time. To prevent increased failure rates of equipment in the cable spreading rooms and the potential failure of the 4KV and 480VAC buses in the switchgear rooms, recovery of the ventilation to these rooms is needed.

Due to the estimated frequency of the smoke infiltration into the rooms and importance of the ventilation systems, this issue is considered risk significant.

Action

Improvement in procedural direction and training for the operators has been initiated. The procedures will direct that the cable spreading rooms and switchgear ventilation systems be restored following an inadvertent actuation. Measures are also being taken to place these ventilation systems in recirculation for fires outside the Cable Spreading Room and Switchgear Rooms if smoke from a fire external to them has potential for getting into the ventilation intakes. Changes to the Fire Fighting Strategies for Yard Areas and Turbine Building will be completed by December 31, 1997. Changes to Operating Instructions and Alarm Manual procedures, and associated operator training will be completed by July 31, 1998.

In addition, the switchgear room ventilation systems will be evaluated for a method to ensure effective recovery from inadvertent actuation of Halon. A realistic switchgear room heat-up model will be developed and is expected to show that there is considerable time available to recover the loss of ventilation. An evaluation based on the new heat-up model will determine if modifications or procedure changes are required to: 1) either prevent the loss of ventilation or 2) restore ventilation. Corrective actions which effectively reduce the risk associated with this issue will be completed by November 30, 1998.

This IPEEE submittal includes the benefit of restoring the switchgear ventilation systems following Halon actuation from an external fire. An estimated failure rate to recover all the switchgear rooms of 5E-3 is used.

7.5 Barrier Inspections

Description of Issue

Some fire barriers and components such as fire dampers, fire penetration seals and fire doors for fire barriers considered in the CCFPRA are not included in a plant surveillance and maintenance program. These barriers were evaluated to ensure they represented effective boundaries between PRA fire areas. To ensure continued effectiveness of these barriers, a configuration control and inspection program is needed.

Action

BGE will incorporate the fire barriers and components such as fire dampers, fire penetration seals and fire doors for fire barriers considered in the CCFPRA - which are not included in a plant surveillance and maintenance program - into an appropriate control and/or inspection program. See Tables 4.3.3a and 4.3.3b. Fire PRA fire barriers will be incorporated into a barrier control process by March 31, 1998. A performance evaluation (PE) procedure to inspect Fire PRA barriers will be completed by July 31, 1998.

7.6 Control of Transient Ignition Sources in the Cable Chases

Description of Issue

A fire in a cable chase could cause significant loss in the ability of the plant to safely shutdown. The Fire Prevention Administrative Procedure, SA-1-100, does not explicitly preclude hot work from the cable chase while at-power. However, SA-1-100 does indicate the following areas where hot work activities are extremely restricted (e.g., review by POSRC, authorization by Superintendent - Nuclear Operations):

- Control Room
- Cable Spreading Rooms
- Switchgear Rooms
- DAS Computer Rooms

Allowance of hot work in the cable chases at power is risk significant.

Action

A procedure change has been initiated to explicitly include Cable Chases in the list of restricted areas in SA-1-100. This procedure change will be completed by July 31, 1998.

SECTION 8

SUMMARY and CONCLUSIONS

This report examines the plant-specific relationships of external events to severe accidents at CCNPP. It provides a comprehensive evaluation of the relative contribution to core damage for each external event. The results indicate a relatively low risk from external events. Several improvements are identified. These either have been implemented or are planned and tracked for resolution.

8.1 Seismic Analysis

The seismic analysis shows a low CDF: Unit 1: 1.29E-5/yr and Unit 2: 1.52E-5/yr. The difference between the units is due to Emergency Diesel Generator 1A being self-cooled while the other three site safety-related emergency diesels are not. A fifth SBO diesel is available equally to both units and therefore does not contribute to this difference. Since SRW has a relatively low fragility, the diesels dedicated to Unit 2 are more susceptible to failure from a seismic event.

Most of the top 100 seismic sequences fall into one of two main categories. The first type is associated with a Spurious Safety System Actuation (SSSA) and a loss of AFW. The second category is a seismic-induced SBO and loss of AFW.

An SSSA is the spurious actuation of the Engineered Safety Features Actuation System, Auxiliary Feed Actuation System (AFAS) and Reactor Protection System (RPS). It occurs when two of four Vital 120VAC buses de-energize.

In most of the seismic sequences involving the SSSA, it is caused by a seismic-induced Loss of Offsite Power and loss of all but one EDG. The three SRW dependent EDGs fail whenever SRW fails. If either of the two remaining self-cooled EDGs fail, an SSSA could occur (if both of the self-cooled EDGs fail at this point, a SBO would result).

An SSSA causes these significant events to occur.

- UV Channels A and B activate and lock in: This opens the feeder breakers to safety-related 4KV buses and sheds all major 4KV loads, including SRW and Safety Injection and the motor-driven AFW pump. Lock-in of the UV signal prevents the 4KV loads from re-starting. The three SRW-dependent EDGs (1B, 2A, 2B) will fail in 10 to 20 minutes because they have no SRW cooling (in most of the seismic sequences SRW is lost anyway due to failure of one of the various SRW coolers/piping in the Turbine Building).

- Steam Generator Isolation Signal (SGIS) Channels A and B activate: This shuts both Main Steam Isolation Valves (MSIVs), which fails MFW.

- Reactor Protection System (RPS) activates: This sends an open signal to both Power Operated Relief Valves (PORVs). However, when the SSSA is caused by a loss of all power except EDG 1A - as in many of the seismic sequences - the PORVs (on both units) lose 125VDC power and remain closed. The HPSI pumps cannot provide make-up to the RCS due to the locked in UV signal.

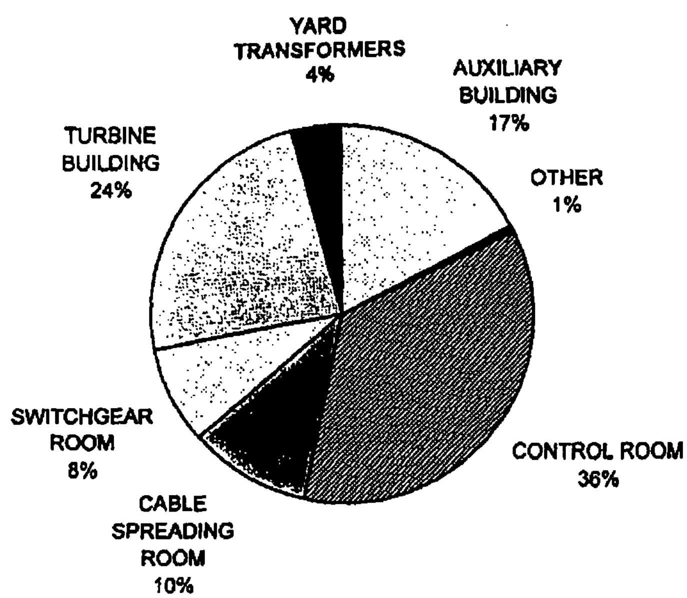
- AFAS Block Channels A & B activate: This isolates AFW flow to both steam generators.

The second most common type of seismic sequence is a seismic-induced SBO and an associated AFW flow control failure.

8.2 Fire Analysis

The fire analysis shows a relatively low CDF: Unit 1: 7.29E-05 and Unit 2: 9.6E-05. Figure 8.2-1 depicts the breakdown of Unit 1 areas according to the initiating events that contribute to CDF.

FIGURE 8.2-1
UNIT 1 FIRE CONTRIBUTORS BY AREA



The difference between the Units is due to Emergency Diesel Generator 1A being self cooled and due to some cable routing differences in key compartments. Note that only those differences which are not bounded by the Unit 1 analysis are evaluated. Therefore, the results for Unit 2 are conservative since advantageous changes are not addressed.

A review of the top 100 sequences for Unit 1 identify the following contributing issues:

- A large Turbine Building fire causes the loss of off-site power and prevents several key human actions from being accomplished. It also challenges the Control Room and switchgear rooms due to smoke. Smoke could cause the evacuation of the Control Room and the loss of ventilation to the cable spreading and switchgear rooms. The human actions impacted include: aligning the OC EDG, providing make-up to the SRW head tanks, allowing AFW Turbine Pump Room cooling by opening the doors to the Turbine Building, the alignment of AFW Pump 23 to Unit 1 and the use of portable ventilation.

- A fire on Control Room Panel 1C04 disables Main Feedwater Water, Once-Through-Core-Cooling, and AFW. If the operator does not man the auxiliary shutdown panel (1C43) and establish AFW flow in a timely fashion, core damage occurs.

- The loss of a battery room is a significant contributor primarily due to the fire degradation of the human actions.

- Every Control Room panel fire can lead to Control Room evacuation. If the fire is not suppressed, then the operators are forced to evacuate. Once the Control Room is evacuated, the operators are required to load shed most of the electrical loads, and manually re-start these loads. If the re-start is not done, then the site is in a self-induced SBO condition and a SSSA could result.

8.3 Other Events Analysis

Each of the other external hazards listed in the PRA Procedures Guide is reviewed. Based on information in the UFSAR and on the information gathered during walkdowns, it is determined that there are no other known plant-unique external events that pose a significant threat of severe accidents within the context of the NUREG–1407 screening approach. The potential for significant impact of the three hazards identified by NUREG–1407 are evaluated, namely, high winds, external floods, and transportation and nearby facility accidents. A detailed analysis of turbine missile is included and determined that this issue is not risk significant.

The CDF due to high winds from hurricanes or tornadoes and associated missiles together with a LOOP is 4.35E-6 per reactor year. The combined hurricane sequences account for approximately 65% of the cumulative importance. This is due to the higher initiating event frequency of the hurricane. The dominant hurricane sequences are generally related to the failure of the Switchgear Room HVAC recovery actions which fails electrical support systems.

The site ponding due to Probable Maximum Precipitation is not a concern at CCNPP. The roof drains have adequate capacity and are inspected monthly to ensure the integrity of the critical roofs. Therefore, it is concluded that the external flooding analyzed presents no risk to plant safety.

Approximately 74% of the total probability of an air crash impacting vital structures comes from helicopter operations which serve the CCNPP site. A restriction is now included in BGE's contract with the Helicopter Transport Services to limit the number of flights over the plants protected area. With this restriction in place, there is now a high degree of confidence that the number of over flights will be significantly less than 6 per year, and as a consequence, the total annual aircraft crash frequency will be less than 1.0×10^{-6}. No other transportation risks are identified.

8.4 Proposed Resolution of Unresolved Safety Issues and Generic Issues

The results of the CCNPP IPEEE show that there are no decay heat removal vulnerabilities from external events. USI A-45, "Shutdown Decay Heat Removal Requirements," is considered to be resolved. Further, the IPEEE considers the following issues resolved: the Eastern US Seismicity (The Charleston Earthquake) Issue, NUREG/CR-5088, "Fire Risk Scoping Study" and GI-57, "Effects of Fire Protection Systems Actuation on Safety Related Equipment."

CALVERT CLIFFS NUCLEAR POWER PLANT

RAN: 98-062
REV: 1

RELIABILITY ENGINEERING
REU QUALITY RECORD

INTERNAL FLOOD INITIATING EVENT FREQUENCIES

ORIGINATOR: Jeffrey L. Stone — J. L. Stone — 6/17/02 DATE

REVIEWER: James B. Landale — J. B. Landale — 6/17/02 DATE

APPROVAL: B. B. Mrowca — 6/17/2002 DATE

LIST OF EFFECTIVE PAGES

LIST OF EFFECTIVE SOFTWARE FILES

Reliability Software Number: 247

Directory: REV 1

File Name	Software	Description
RAN98-062-R1.DOC	MICROSOFT WORD 97	MAIN REPORT
FREQ-R1.XLS	MICROSOFT EXCEL 97	PIPE BREAK FREQUENCIES RAW DATA EPRI DATA FAILURE CAUSE SCREENING TOTAL FLOOD FREQ. MODEL INITIATING EVENT FREQ.
MAINTIND-R1.XLS	MICROSOFT EXCEL 97	MAINTENANCE INDUCED FLOOD DATA COLLECTION AND FREQ. CALCULATION. DATA REFERENCES
WALKDWN.DOC	MICROSOFT WORD 97	DOCUMENTATION OF PLANT WALKDOWN.
PRA_Rooms.XLS	MICROSOFT EXCEL 97	LIST OF FLOOD AREAS

REVISION HISTORY

Revision	Description
0	Initial issue.
1	CRMP 256 notes that this RAN uses calendar years in calculating flood initiating event frequencies, but uses critical years when calculating the maintenance-induced contribution to the flood initiating event frequencies. This revision addresses this CRMP by calculating both using critical years. The following changes are made: • Added Attachment 9, EPRI Data, with revised failure rates on piping using critical hours for some piping systems. Systems that are assumed to be in service close to 100% (Operating and Shutdown) are not adjusted. System that are often available during extended shutdown are adjusted to compensate. Systems that are generally not available during extended outages are adjusted for percentage of critical hours in the data used. Calendar hours were converted to critical hours using the correction factor documented in Key Input 1395. • Attachment 1, Piping Failure Frequencies, has new failure rates, calculated using the revised data from Attachment 9. • Attachment 5, Total Flood Frequencies, has new pipe rupture frequencies and total flood frequencies, using the revised failure rates from Attachment 1. • Attachment 6, Model Frequencies, has new flood initiating event frequencies, using the revised flood frequencies from Attachment 5. • Attachment 6 adds new data showing the 0Q (no maintenance) flood initiating event frequencies. The 0Q initiating event frequencies are calculated using the pipe rupture frequencies from Attachment 5. • In Attachment 6, the Excel spreadsheet calculation for initiating event ST12AM was found to have an error (it referenced the maintenance induced frequency only, not the total frequency). The calculation was corrected resulting in an IE change from 2.501E-6 to 8.413E-6.

TABLE OF CONTENTS

1.0 PURPOSE

The purpose of this report is to document the development of the flood initiating event frequencies from pipe ruptures and maintenance activities. These frequencies are used in the flood module of the Calvert Cliffs Probabilistic Risk Assessment (CCPRA).

2.0 INTRODUCTION

In order to quantify the contribution of internal flooding events to total core damage frequency, the pipe-break and maintenance-induced flooding frequencies must be calculated or estimated. This RAN documents the methodology used to determine those frequencies.

3.0 APPROACH

Total flood frequency may be thought of as composed of two parts. One part is the frequency associated with pipe ruptures, and the second part is the frequency associated with maintenance errors resulting in significant floods.

This total flood frequency is calculated using a four-step process. First, the pipe-break frequency for a particular system in a particular room is calculated using the EPRI methodology found in Ref 1. This method permits the calculation of frequencies for various break sizes which may be used by the modeler in the last step when he/she assigns frequencies to similar impacts (discussed further in section 4.2).

The next step involves correcting the calculated frequency in order to credit maintenance and engineering programs which aim to reduce failures due to particular causes.

In the third step, a maintenance-induced flooding frequency for a particular system in a particular room is calculated. This frequency estimates the occurrences of plant flooding brought on by maintenance activities. This is further discussed in sections 4.3 and 6.

The last step is to calculate a total flood frequency by summing the frequencies calculated in the previous three steps. What the analyst is left with in this case is a matrix of flood frequencies particular to a system and an individual room.

Once total flood frequencies are calculated by the analyst, the initiating event frequencies may be determined based on the needs of the CCPRA modeler. For instance, the modeler may choose to combine different flood scenarios based on similar plant impacts (determined in RAN 96-024 FLOOD). In this manner, floods of different pipe break sizes and different rooms may be combined to arrive at particular initiating event frequency for use in the CCPRA model. Attachment 6 is the result of this interactive process and is the end product of this RAN.

4.0 PIPE BREAK FREQUENCIES

Pipe break frequencies are calculated using Ref. 1, which should be consulted for a fuller understanding of its application to this report. Its updated methodology correlates the number of system piping segments with the pipe-rupture probability. The calculation methodology is described in sections 4.1 & 4.2.

4.1 Pipe Segment Count

Pipe segments are counted, primarily utilizing O&M prints and valve line-ups. This activity is supplemented by checking isokinetics, elevation-piping plans, sprinkler drawings, etc. When necessary the analyst validates information collected from segment counting with plant walkdowns (see Attachment 6).

The first step in calculating system pipe break frequencies requires a careful count of system piping segments in each susceptible room. Various conservative assumptions and estimates are used in determining the pipe segment counts. The paragraphs below discuss, where appropriate, those flood scenarios in which amplifying information is useful. Susceptible rooms and scenarios refer to those documented in Ref. 6.

Referring to flood-scenario designators, the first single character represents the system experiencing the rupture (e.g. C=CCW, S=SW, W=SRW, F=FP, D=DW, I=CW, Q=MFW, N=CST, R=RWT, W=SRW). The second set of characters represents the room (e.g. C228 is a CCW rupture in room 228). The next single character represents the relative pipe size (S=small, M=medium, L=large, A=all three sizes). The last single character represents the flood duration based on whether or not i.) the flood is recovered by a human action -- normally within a few hours (R), ii.) not recovered by a human action – but not going beyond 12 hours (N), or iii.) the flood duration was terminated early (i.e. <12 hours) was probably unrecovered by human action and self-terminated by its own limited flood source (M).

C228AM – CCW pipe break in room 228

Due to difficulty experienced in counting pipe segments from the prints, five additional segments are added to the total for conservatism.

W319AM -- SRW pipe break in room 319.

Same piping installation as for SFP cooling room – separated from the main corridor by a screen door, therefore no piping is counted in room 319 to prevent the double counting of the associated impact.

W320AM -- SRW pipe break in room 320.

Eleven (11) sections for Unit 1. Because this room is susceptible to rupture from both units, the Unit 1 number is multiplied by two to arrive at twenty-two (22) total sections.

C419AM -- CCW pipe break in room 419.

RCW Evaporators are assumed to be secured since they typically may be operated about 12 hrs. per week.

F419AM -- Fire System pipe break in room 419.

In counting small diameter (i.e. <2") piping, the Unit 1 drawing indicates a total of 148 sprinklers for the whole Unit 1 elevation. Subtracting 47 (both Penn. Rms) and 48 sprinklers (piping area) leaves 101 sprinkler heads. To this is added the 48 U-2 sprinkler heads to arrive at the total number of heads in the room.

W419AM -- SRW pipe break in room 419.

There are two SRW head tank lines per unit (4 total) in the vicinity of this room.

R530A -- RWT pipe break in Room 530.

There actually is no RWT piping in room 530. The frequency appears in the attachments for further development of the total flood frequency which includes RWT overfill events (maintenance-induced flooding). As it turns out, this flood scenario has minor impact and most industry floods of this type have not been significant. Engineering judgement was used to eliminate this flood from the CCPRA.

D605AM -- Demineralized Water pipe break in room 605.

There is no Demin. Water piping in room 605 (Ref. 2).

FT27AM -- Fire Protection system pipe break on 27' of the TB.

Ref. 9 documents the methodology used to determine the pipe segment counts for this flood. Fire Protection prints are not generally maintained current at CCNPP. For this reason, construction prints were consulted for counting segments.

IT12AM/ST12AM -- CW/SW pipe break on 12' of the TB

These pipe segment counts include Unit 2 pipe segments. This is due to the open bay architecture of the turbine building.

FT12AM -- Fire Protection system pipe break on 12' of the TB.

Ref. 9 documents the methodology used to determine the pipe segment counts for this flood. Fire Protection prints are not generally maintained current at CCNPP. For this reason, construction prints were consulted for counting segments.

IISPAM/SISPAM -- CW/SW pipe break in the Intake Structure

Unit 1 numbers are multiplied by two (2) to account for a Unit 2 rupture flooding the Unit 1 side. This assumes that the flood height exceeds the 3' height of the free-communication swing door.

4.2 Break Frequency Calculations

Ref. 1 provides industry-wide break frequencies per pipe segment for various pipe sizes and systems. The specific CCNPP frequency is obtained by multiplying the industry segment frequency by the number of CCNPP segments for a particular system. Further refinement of the break frequencies is possible by using conditional probabilities provided to calculate the frequency of specific break sizes. Frequency reduction is also possible as diminishing factors are provided where plant processes (e.g. erosion-corrosion inspection programs) mitigate particular break mechanisms.

Note – Revision 1 revised the failure rates from Reference 1 as these appear to be developed per Calendar Year instead of per Critical Year. These changes are reflected in the failure rates shown on Attachment 9. The percentage of critical time is adjusted per **Key Input 1395**.

Attachment 1 shows the results of the pipe break frequency analysis. It provides specific break frequencies for small, medium, large, and total break sizes. Mostly, however, it is the total break frequencies that are currently used in the current CCPRA, as most impacts are bounded by the largest break. It is anticipated that the flood impact analysis will utilize separate analyses for each break-size category in the future. The small, medium, and large break-size frequencies for each flood are calculated for this purpose. Nevertheless, there are certain flood scenarios (e.g. N205LN) that have separate impacts that differ from other system pipe-size floods and are broken out for separate treatment.

The methodology for calculating size-dependent rupture frequencies is described on the following pages:

Frequency Calculation Methodology

f_{sm} = $(FR_{sm} \times P_{sm/sm} \times N_{sm}) + (FR_{med} \times P_{sm/med} \times N_{med}) + (FR_{lrg} \times P_{sm/lrg} \times N_{lrg})$

f_{med} = $(FR_{med} \times P_{med/med} \times N_{med}) + (FR_{lrg} \times P_{med/lg} \times N_{lrg})$

f_{lrg} = $(FR_{lrg} \times P_{lrg/lrg} \times N_{lrg})$

f_{total} = $f_{sm} + f_{med} + f_{lrg}$

Where:

f_X is the frequency of a break occurring of size "x"

FR_X is the failure rate of a pipe segment of size "x" (i.e. small, medium, or large)

$P_{Y/X}$ is the conditional segment-rupture probability of size "y" given pipe size "x" (obviously x is always greater than, or equal to, y).

N_X is the number of segments of size "x"

FR_X and $P_{Y/X}$ are provided in Ref. 1.

For reference, the associated pipe-segment rupture frequencies given in Ref. 1 Table 4.4 are shown below:

	Failure Rate (per segment/year)		
System Group	**S**	**M**	**L**
SI/Recirc.	1.244E-05	9.899E-07	1.682E-06
Other SR	6.21E-06	6.16E-07	1.218E-06
Feed/Cond.	6.474E-06	1.025E-05	5.606E-06

The analyst calculates the break frequencies in accordance with the methodology outlined. The frequencies shown above (previous page)are applied based on the following rationale:

- SI/Recirc. group frequencies: Used for RWT floods since the RWT is part of the SI system.

- Other SR group frequencies: Used for all other SR system floods as well as Fire Protection, CW and Demin. Ref. 1 suggests the analyst select the most appropriate frequency based on the system's particulars. For all three non-safety related systems, this frequency group was assigned because of the similarities with other systems that do use them. This group of frequencies is a better fit than SI/Recirc and Feed/Condensate groups.

- Cond/Feed group frequencies: Used for CST and Feed system floods.

4.3 Arriving at Corrected Pipe-break Frequencies using Reduction Factors

Following calculation of the break frequencies, adjustment is made for break-mechanisms that are not likely. Ref. 1 provides for this where justifiable. Two such justified adjustments in the Calvert Cliffs internal flooding study are for the erosion corrosion and cyclic fatigue break mechanisms.

Erosion corrosion is discounted as a concern in all pipe systems with the exception of Main Feed. This is because of the predominant requirement for wet steam transport. MN-3-202, Erosion Corrosion Monitoring of Secondary Piping, identifies the systems which are vulnerable to erosion/corrosion failure. With respect to potential internal flooding, only the Main Feed system is considered susceptible to this type of failure. Additionally, the encapsulated feed piping from the MSIV room to the 5' Piping Area cannot be checked by the site's monitoring program and is therefore most susceptible.

Cyclic fatigue is also not considered a significant failure mechanism. The following rationale is provided:

MN-1-209, Rotating Machinery Condition Monitoring, provides engineering attention in the area of low-cycle fatigue failure. All flood susceptible systems are included in this program and therefore are at a much-reduced risk for this type of failure. Additionally, due to the nature of low-cycle fatigue, the probability of its occurrence is extremely low during the course of a reactor year, which is how pipe-break failures are measured, so as to warrant their reduced consideration. As for high-cycle fatigue, the Main Feed system is the only susceptible system due to its thermal transient nature. So much so that it is covered by Life Cycle Management's fatigue monitoring program.

Only the Main Feed System receives no compensatory reduction in pipe-rupture frequency quantification from cyclic fatigue or erosion corrosion program implementation. See Attachment 4 for further information.

For all systems with the exception of Main Feed, we derive corrected frequencies by lowering the calculated frequencies by a reduction factor of 50% of the combined evaluated failure-mechanism percentages, as described below:

Corrected Frequency Methodology

$$f_{corrected} = f_{calculated} \times (1 - 0.5*((\%_{errosion\text{-}corrosion} + \%_{cyclic})/100\%))$$

Where:

$f_{corrected}$ is the corrected pipe-break frequency

$f_{calculated}$ is the pipe-break freq. calculated strictly using the EPRI guidelines.

$\%_{errosion\text{-}corrosion}$ is the %-age of all size-related breaks associated with errosion corrosion mechanisms.

$\%_{cyclic}$ is the %-age of all size-related breaks associated with cyclic fatigue.

5.0 MAINTENANCE-INDUCED INTERNAL FLOOD FREQUENCIES

Maintenance induced flooding frequencies are calculated by conducting a study using INPO's web-based Plant Event and LER databases (see Attachment 8). We limit these databases to the period from 1985 on -- when the modern 10CFR50.73 (i.e. LER) reporting rule was in effect. System flooding frequency for a generic reactor unit is calculated by dividing the total number of incidents by the number of U. S. reactor operating years. This is done for those systems identified as flood sources (Salt Water, Service Water, Circ. Water, RWT, and Fire Protection).

5.1 Flood Calculations and Assumptions

To apply industry data to Calvert Cliffs, a correlation has to be made between the site flood frequency and the frequency for a particular CCNPP room for the system in question. Two frequencies were calculated:

- one for the critical operating year (hence the term, Operating Year) which is calculated using only events that occurred while critical, and

- one for the total year which uses all of the event data (hence the term All-Event).

This is important in that many maintenance-induced floods occurred during intensive maintenance periods in which reactors were shutdown. Since the CCPRA initial condition is a critical reactor, the data must be appropriately discriminating.

It is interesting to note that there are no reported Service Water floods while at power. This could be due to many factors related to the technical specification treatment of this system or its special maintenance treatment as a critical safety system. Based on engineering judgment, a unit-specific frequency of 1×10^{-4}/yr is used for this system. This is only one-twentieth the frequency for SRW floods occurring during outages and is considered a reasonable assumption.

Due to the dissimilarities between system functional names within the industry (e.g. application of the names "component cooling water" and "service water") and the fact no CCW floods were reported, the SRW frequency is also utilized for the component cooling water system which also presented no industry data.

There are no reported CST leaks in the data, operating or otherwise. Considering the fact that there is very little piping associated with these flood scenarios, and very few valves associated with the subject system piping, a CST system maintenance-induced flood frequency will be extremely low. The unit-specific frequency 1×10^{-4}/yr assumed for SRW will also be used for the CST system.

Additionally, there are no reported Demineralized Water system floods reported, most probably due to the non-safety-related nature of the system. A conservative unit-specific frequency of 1×10^{-3}/yr is assumed. This number is in-line with the majority of the most conservative numbers determined by the study. The lone exception to this is the RWT system (2.7×10^{-3}/yr) whose data included significant numbers of Spent Fuel Pool overflow events and is therefore considered not truly comparable to the other maintenance-induced flood data.

Because potential flood-causing maintenance on the MFW system is not conducted at power, no maintenance-induced flood frequency is calculated for this system.

After evaluating the data, it was apparent that the majority of events occurred when system components were removed or disassembled (i.e., there were very few pipe ruptures).

5.2 Apportioning of the Plant Maintenance-Induced Flood Frequencies

For most systems , it can be assumed that the relative flood frequency for a particular room is directly related to the ratio of maintainable components[1] in the room to the maintainable components in the overall system. Generally speaking, piping segments are separated by components that can be considered "maintainable". The number of piping segments can therefore be considered approximate to the number of maintainable components in a room. Additionally, the number of valves in a system can also be considered approximately equal to the total number of maintainable components in the system, as the vast majority of such components are valves and strainers.

A rough study for these systems (CCW, SRW, RWT, SW, CST, Demin., and CW), was conducted to arrive at the relative percentages, and the total frequency was subdivided to achieve an appropriate apportionment to each applicable room. Table 1 (next page) shows the estimates of the number of maintainable components in each system as determined by counting the number of valves cataloged in the system Operating Instruction. This information, along with the number of system pipe segments for the applicable rooms (Attachment 1) are used to determine the final maintenance induced flood frequencies.

In the case of the Fire Protection System (for which complete drawings do not exist), it is assumed that 1% of the associated site frequency may be applied to each CCPRA modeled room, since there are at least 116 plant rooms found to have Fire Protection piping. This was determined after evaluating Table 9-20 of Ref. 8 as well as Attachment 1 of this RAN which shows some rooms are used only for piping runs.

The overall plant Fire Protection maintenance-induced flooding frequency was therefore multiplied by 0.01 to arrive at the frequency in each modeled room -- resulting in a uniform frequency in each room for this one system.

Additionally, since half of reported RWT events were SFP overfills, the plant frequencies are adjusted such that half the total frequency is apportioned to the SFP room (530).

See Attachments 2 and 3 for reference to the actual study data and references, Attachment 5 contains the study results.

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[1] Anything that is routinely opened up for purposes of performing routine maintenance is considered a maintainable component, including valves, strainers, heat exchangers, etc..

TABLE 1 Maintainable Components for Flood Modeled Systems		
System	**Number of Components (e.g. valves)**	**Reference**
CCW	420	OI-16
FP	891	OI-20
MFW	NA	NA
SW	893	OI-29
DW	777	OI-23B
CW	288 (both units counted)	OI-14A
CST	625	OI-11A
RWT	658	OI-3A/3B
SRW	700	OI-15

6.0 INITIATING EVENT FREQUENCIES

Flood impacts on the plant are determined in RAN 96-024 FLOOD. From these impacts, the modeler combines like impacts into bins for calculating the Initiating Event frequencies. Attachment 6 is based on this binning and the appropriate flood constituents are combined to arrive at the CCPRA model initiating event frequencies.

7.0 REFERENCES

1. EPRI TR 102266
2. Design Calc M-90-191 sect. 2.15
3. Drawing 12759-0002
4. NFPA STD 13, Installation of Sprinkler Systems, Ref. 6.4
5. Unused
6. RAN 94-001 Rev. 0
7. RAN 96-024 FLOOD Rev. 0
8. CCNPP UFSAR Rev. 26 CH. 9
9. Key Input 36
10. Key Input 1395

ATTACHMENT 1

Piping Failure Frequencies

Format Example

ROOM NUMBER ROOM FLOOD ID	SYSTEM	PIPE GROUP	NO. OF SECTIONS	BASIS	FREQ.	CORR. FREQ.	
101 (U-2 Ea. ECCS Pp Rm)					2.669E-04	8.656E-05	M118
S101	SW	S	12	60708/M-295 & 291	9.076E-05	6.217E-05	
		M	5	60708/M-295 & 291	6.375E-06	4.686E-06	
		L	8	60708/M-295 & 291	5.631E-06	3.885E-06	
					1.028E-04	7.074E-05	
F101					1.550E-04	7.694E-05	
					7.076E-06	7.073E-06	
					2.082E-06	2.545E-06	
					1.641E-04	8.656E-05	
C101							
M101 (F101 & C101)					2.669E-04	1.573E-04	

Rm Number & Name

Pipe Size Catagories
S -- <2"
M -- 2" to <6"
L -- 6" or >

drawing ref's

rupture freq. for equivalent break size

total freq. -- all break sizes considered

Flood ID

Rupture freqs. for all systems considered in this room.

Some rooms have rupture freqs. @ bottom of page for various system combinations.

ATTACHMENT 1

Piping Failure Frequencies

ROOM NUMBER ROOM FLOOD ID	SYSTEM	PIPE GROUP	NO. OF SECTIONS	BASIS	FREQ.	CORR. FREQ.	
118					6.024E-04	4.139E-04	M118
(W. ECCS Pp. Rm.)							
S118	SW	S	11	60708/M-295 & 291	8.322E-05	5.701E-05	
		M	4	60708/M-295 & 291	5.663E-06	4.162E-06	
		L	8	60708/M-295 & 291	5.631E-06	3.885E-06	
					9.451E-05	6.505E-05	
F118	FIRE	S	0	W/D	0.000E+00	0.000E+00	
		M	0	W/D	0.000E+00	0.000E+00	
		L	0	W/D	0.000E+00	0.000E+00	
					0.000E+00	0.000E+00	
C118	CCW	S	66	60710/60410	4.842E-04	3.317E-04	
		M	20	60710/60410	1.741E-05	1.279E-05	
		L	9	60710/60410	6.335E-06	4.371E-06	
					5.079E-04	3.488E-04	

ATTACHMENT 1

Piping Failure Frequencies

ROOM NUMBER ROOM FLOOD ID	SYSTEM	PIPE GROUP	NO. OF SECTIONS	BASIS	FREQ.	CORR. FREQ.	
119					6.655E-04	4.574E-04	← M119
(Ea. ECCS Pp. Rm.)							
S119	SW	S	10	60708/M-295 & 297	7.639E-05	5.233E-05	
		M	4	60708/M-295 & 297	6.015E-06	4.421E-06	
		L	9	60708/M-295 & 297	6.335E-06	4.371E-06	
					8.874E-05	6.112E-05	
F119	FIRE	S	42	12261-58SH11 & W/D	3.048E-04	2.088E-04	
		M	9	12261-58SH11 & W/D	6.407E-06	4.709E-06	
		L	0	12261-58SH11 & W/D	0.000E+00	0.000E+00	
					3.112E-04	2.135E-04	
C119	CCW	S	32	60710/60410	2.404E-04	1.647E-04	
		M	19	60710/60410/60744	1.740E-05	1.279E-05	
		L	11	60710/60410	7.742E-06	5.342E-06	
					2.655E-04	1.828E-04	

ATTACHMENT 1
Piping Failure Frequencies

ROOM NUMBER ROOM FLOOD ID	SYSTEM	PIPE GROUP	NO. OF SECTIONS	BASIS	FREQ.	CORR. FREQ.	
221					2.717E-04	1.874E-04	← M221
(5' W. Penn. Rm.)							
D221	DEMIN	S	0	60-439 & 440	3.560E-07	2.705E-07	
		M	1	60-439 & 440	7.119E-07	6.372E-07	
		L	0	60-439 & 440	0.000E+00	0.000E+00	
					1.068E-06	9.077E-07	
F221	FIRE	S	5	60-714/12261-27	3.661E-05	2.508E-05	
		M	1	60-714/12261-27	1.064E-06	7.819E-07	
		L	1	60-714/M-182	7.038E-07	4.856E-07	
					3.838E-05	2.634E-05	
C221	CCW	S	20	60710	1.454E-04	9.959E-05	
		M	5	60710	3.560E-06	2.616E-06	
		L	0	60710	0.000E+00	0.000E+00	
					1.489E-04	1.022E-04	
W221	SRW	S	4	60706	4.247E-05	2.909E-05	
		M	6	60706	1.588E-05	1.168E-05	
		L	33	60706	2.323E-05	1.603E-05	
					8.158E-05	5.679E-05	
R221	RWT	S	0	60716/60440	4.205E-07	2.880E-07	
		M	0	60716/60440	4.205E-07	3.091E-07	
		L	1	60716/60440	8.410E-07	5.803E-07	
					1.682E-06	1.177E-06	

ATTACHMENT 1

Piping Failure Frequencies

ROOM NUM./FLD ID	SYSTEM	PIPE GP	SECTIONS	BASIS	FREQ.	CORR. FREQ.	
224					6.585E-04	4.772E-04	← M224
(U-1 5' Piping Area)							
C224	CCW	S	2	60-710/432/417	1.542E-05	1.056E-05	
		M	0	60-710/432/417	1.056E-06	7.760E-07	
		L	3	60-710/432/417	2.112E-06	1.457E-06	
					1.858E-05	1.279E-05	
N224	CST	S	1	60583	1.973E-05	1.351E-05	
		M	1	60432	1.788E-05	1.314E-05	
		L	2	60432	7.680E-06	5.299E-06	
					4.529E-05	3.195E-05	
D224	DEMIN	S	9	60717/60432	6.569E-05	4.992E-05	
		M	3	60717/60432	2.136E-06	1.912E-06	
		L	0	60717/60432	0.000E+00	0.000E+00	
					6.783E-05	5.184E-05	

Continued on Next Page

ATTACHMENT 1

Piping Failure Frequencies

ROOM NUM./FLD ID	SYSTEM	PIPE GP	SECTIONS	BASIS	FREQ.	CORR. FREQ.
F224	FIRE	S	42	60714,12261-41,W/D	3.122E-04	2.139E-04
		M	30	60714,12261-41,W/D	2.136E-05	1.570E-05
		L	0	60714,12261-41,W/D	0.000E+00	0.000E+00
					3.336E-04	2.296E-04
R224	RWT	S	0	60432& 91489	4.205E-07	2.880E-07
		M	0	60432& 91489	4.205E-07	3.091E-07
		L	1	60432& 91489	8.410E-07	5.803E-07
					1.682E-06	1.177E-06
W224	SRW	S	13	60706	1.035E-04	7.093E-05
		M	0	60706	1.021E-05	7.501E-06
		L	29	60706	2.041E-05	1.408E-05
					1.342E-04	9.252E-05
Q224	FEED	S	3	60-702 & 432	3.428E-05	3.428E-05
		M	0	60-702 & 432	7.680E-06	7.680E-06
		L	4	60-702 & 432	1.536E-05	1.536E-05
					5.732E-05	5.732E-05

ATTACHMENT 1

Piping Failure Frequencies

ROOM NUMBER ROOM FLOOD ID	SYSTEM	PIPE GROUP	NO. OF SECTIONS	BASIS	FREQ.	CORR. FREQ.	
225					2.739E-04	1.894E-04	M225
(U-1 5' FAN RM)							
N225	CST	S	1	60583	2.291E-05	1.569E-05	
		M	2	60432	2.808E-05	2.064E-05	
		L	0	60432	0.000E+00	0.000E+00	
					5.099E-05	3.633E-05	
D225	DEMIN	S	0	60432	3.560E-07	2.705E-07	
		M	1	60432	7.119E-07	6.372E-07	
		L	0	60432	0.000E+00	0.000E+00	
					1.068E-06	9.077E-07	
F225	FIRE	S	30	60-714/12261-27	2.175E-04	1.490E-04	
		M	6	60-714/12261-27	4.272E-06	3.140E-06	
		L	0	60-714/12261-27	0.000E+00	0.000E+00	
					2.218E-04	1.522E-04	

ATTACHMENT 1

Piping Failure Frequencies

ROOM NUMBER ROOM FLOOD ID	SYSTEM	PIPE GROUP	NO. OF SECTIONS	BASIS	FREQ.	CORR. FREQ.	
226					1.731E-03	1.358E-03	← M226
(U-1 SRW Pp. Rm.)							
S226	SW	S	10	60708	1.014E-04	6.948E-05	
		M	16	60708	3.532E-05	2.596E-05	
		L	68	60708	4.786E-05	3.302E-05	
					1.846E-04	1.285E-04	
F226	FIRE	S	80	60-714/12261-27	5.851E-04	4.008E-04	
		M	30	60-714/12261-27	2.136E-05	1.570E-05	
		L	0	60-714/12261-27	0.000E+00	0.000E+00	
					6.065E-04	4.165E-04	
W226	SRW	S	71	60706	5.802E-04	4.845E-04	
		M	98	60706	1.053E-04	1.111E-04	
		L	101	60706	7.109E-05	8.673E-05	
					7.566E-04	6.823E-04	
N226	CST	S	3	60-202, 499, & 583	7.640E-05	5.234E-05	
		M	6	60-202, 499, & 583	9.192E-05	6.756E-05	
		L	4	60-202, 499, & 583	1.536E-05	1.060E-05	
					1.837E-04	1.305E-04	

ATTACHMENT 1

Piping Failure Frequencies

ROOM NUMBER ROOM FLOOD ID	SYSTEM	PIPE GROUP	NO. OF SECTIONS	BASIS	FREQ.	CORR. FREQ.	
227					8.747E-04	6.051E-04	M227
(5' Ea. Penn. Rm.)							
C227	CCW	S	4	60-710/497/499	3.083E-05	2.112E-05	
		M	0	60-710/497/499	2.112E-06	1.552E-06	
		L	6	60-710/497/499	4.223E-06	2.914E-06	
					3.717E-05	2.559E-05	
D227	DEMIN	S	3	60-497, 499, &729	2.296E-05	1.745E-05	
		M	4	60-497, 499, &729	2.848E-06	2.549E-06	
		L	0	60-497, 499, &729	0.000E+00	0.000E+00	
					2.581E-05	2.000E-05	
F227	FIRE	S	68	60714,12261-47,W/D	5.039E-04	3.452E-04	
		M	40	60714,12261-47,W/D	2.989E-05	2.197E-05	
		L	4	60714,12261-47,W/D	2.815E-06	1.943E-06	
					5.366E-04	3.691E-04	
W227	SRW	S	7	60706	6.542E-05	4.481E-05	
		M	7	60706	1.765E-05	1.297E-05	
		L	36	60706	2.534E-05	1.748E-05	
					1.084E-04	7.527E-05	
N227	COND	S	11	60-497&761	1.148E-04	7.866E-05	
		M	0	60-497&761	1.728E-05	1.270E-05	
		L	9	60-497&761	3.456E-05	2.385E-05	
					1.667E-04	1.152E-04	

ATTACHMENT 1

Piping Failure Frequencies

ROOM NUMBER ROOM FLOOD ID	SYSTEM	PIPE GROUP	NO. OF SECTIONS	BASIS	FREQ.	CORR. FREQ.	
228					1.349E-03	9.300E-04	M228
(CCW Pp Rm.)							
C228	CCW	S	45	60710, 738	3.576E-04	2.450E-04	
		M	9	60710	3.773E-05	2.773E-05	
		L	89	60710	6.264E-05	4.322E-05	
					4.580E-04	3.159E-04	
D228	DEMIN	S	2	60-717, 411, &413	1.507E-05	1.145E-05	
		M	2	60-717, 411, &413	1.424E-06	1.274E-06	
		L	0	60-717, 411, &413	0.000E+00	0.000E+00	
					1.650E-05	1.273E-05	
F228	FIRE	S	78	60-714/12261-27	5.704E-04	3.907E-04	
		M	28	60-714/12261-27	2.029E-05	1.491E-05	
		L	1	60-714/12261-27	7.038E-07	4.856E-07	
					5.914E-04	4.061E-04	
R228	RWT	S	1	60432	1.286E-05	8.809E-06	
		M	0	60432	4.205E-07	3.091E-07	
		L	1	60432	8.410E-07	5.803E-07	
					1.412E-05	9.698E-06	
S228	SW	S	28	60708	2.183E-04	1.495E-04	
		M	2	60708	1.796E-05	1.320E-05	
		L	47	60708	3.308E-05	2.283E-05	
					2.693E-04	1.856E-04	

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Piping Failure Frequencies

ROOM NUMBER ROOM FLOOD ID	SYSTEM	PIPE GROUP	NO. OF SECTIONS	BASIS	FREQ.	CORR. FREQ.	
315					3.613E-04	2.481E-04	M315
(MSIV Rm.)							
F315	FIRE	S	47	60-714/12261-15	3.442E-04	2.358E-04	
		M	17	60-714/12261-15	1.281E-05	9.413E-06	
		L	2	60-714/12261-15	1.408E-06	9.713E-07	
					3.584E-04	2.462E-04	
W315	SRW	S	0	12535A-34& 35	7.038E-07	4.821E-07	
		M	0	12535A-34& 35	7.038E-07	5.173E-07	
		L	2	12535A-34& 35	1.408E-06	9.713E-07	
					2.815E-06	1.971E-06	
317					3.204E-06	2.301E-06	M317
(27' Swgr. Rm.)							
F317	FIRE	S	0	60-714/12261-15	1.068E-06	7.315E-07	
		M	3	60-714/12261-15	2.136E-06	1.570E-06	
		L	0	60-714/12261-15	0.000E+00	0.000E+00	
					3.204E-06	2.301E-06	

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Piping Failure Frequencies

ROOM NUM./FLD ID	SYSTEM	PIPE GP	SECTIONS	BASIS	FREQ.	CORR. FREQ.	
318					3.749E-05	2.679E-05	M318
(Purge Air Supply Rm.)							
N318	CST	S	0	60-482&499	1.086E-05	7.439E-06	
		M	1	60-482&499	1.788E-05	1.314E-05	
		L	2	60-482&499	7.680E-06	5.299E-06	
					3.642E-05	2.588E-05	
D318	DEMIN	S	0	60499	3.560E-07	2.705E-07	
		M	1	60499	7.119E-07	6.372E-07	
		L	0	60499	0.000E+00	0.000E+00	
					1.068E-06	9.077E-07	
319					7.264E-05	5.659E-05	M319
(27' Mn. Hallway)							
D319	DEMIN	S	7	60426 & W/D	5.631E-05	4.280E-05	
		M	17	60426 & W/D	1.210E-05	1.083E-05	
		L	0	60426 & W/D	0.000E+00	0.000E+00	
					6.842E-05	5.363E-05	
F319	FIRE	S	0	60-714/	1.056E-06	7.232E-07	
		M	0	60-714/	1.056E-06	7.760E-07	
		L	3	60-714/M-182	2.112E-06	1.457E-06	
					4.223E-06	2.956E-06	
W319	SRW	S	0	12531A-06 & 08	0.000E+00	0.000E+00	
		M	0	12531A-06 & 08	0.000E+00	0.000E+00	
		L	0	12531A-06 & 08	0.000E+00	0.000E+00	
					0.000E+00	0.000E+00	

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Piping Failure Frequencies

ROOM NUMBER ROOM FLOOD ID	SYSTEM	PIPE GROUP	NO. OF SECTIONS	BASIS	FREQ.	CORR. FREQ.	
320					3.310E-05	2.349E-05	← M320
(SFP Cooler Rm.)							
D320	DEMIN	S	0	W/D	7.119E-07	5.411E-07	
		M	2	W/D	1.424E-06	1.274E-06	
		L	0	W/D	0.000E+00	0.000E+00	
					2.136E-06	1.815E-06	
W320	SRW	S	0	12531A-06 & 08	7.742E-06	5.303E-06	
		M	0	12531A-06 & 08	7.742E-06	5.691E-06	
		L	22	12531A-06 & 08	1.548E-05	1.068E-05	
					3.097E-05	2.168E-05	
324					1.699E-05	1.205E-05	← M324
(LDHX Rm.)							
C324	CCW	S	0	60710	4.959E-06	3.397E-06	
		M	8	60710	7.807E-06	5.738E-06	
		L	6	60710	4.223E-06	2.914E-06	
					1.699E-05	1.205E-05	

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Piping Failure Frequencies

ROOM NUMBER ROOM FLOOD ID	SYSTEM	PIPE GROUP	NO. OF SECTIONS	BASIS	FREQ.	CORR. FREQ.	
419					1.509E-03	1.041E-03	← M419
(45' Truck Bay)							
C419	CCW	S	8	60-710/734/427	6.307E-05	4.320E-05	
		M	0	60-710/734/427	5.631E-06	4.139E-06	
		L	16	60-710/734/427	1.126E-05	7.770E-06	
					7.996E-05	5.511E-05	
D419	DEMIN	S	9	60-427&734,W/D	6.498E-05	4.938E-05	
		M	1	60-427&734,W/D	7.119E-07	6.372E-07	
		L	0	60-427&734,W/D	0.000E+00	0.000E+00	
					6.569E-05	5.002E-05	
F419	FIRE	S	180	60-714/12261-01/07	1.312E-03	8.987E-04	
		M	36	60-714/12261-01/07	3.232E-05	2.375E-05	
		L	19	60-714/12261-01/07	1.337E-05	9.227E-06	
					1.358E-03	9.317E-04	
W419	SRW	S	0	60419	1.408E-06	9.643E-07	
		M	0	60419	1.408E-06	1.035E-06	
		L	4	60419	2.815E-06	1.943E-06	
					5.631E-06	3.941E-06	
M419 {C419, D419, AND F419} →					1.503E-03	1.037E-03	

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Piping Failure Frequencies

ROOM NUMBER ROOM FLOOD ID	SYSTEM	PIPE GROUP	NO. OF SECTIONS	BASIS	FREQ.	CORR. FREQ.	
421					1.848E-04	1.267E-04	← M421
(1B DG Rm.)							
F421	FIRE	S	25	60-714/12759-02	1.799E-04	1.232E-04	
		M	1	60-714/12759-02	7.119E-07	5.233E-07	
		L	0	60-714/12759-02	0.000E+00	0.000E+00	
					1.806E-04	1.237E-04	
W421	SRW	S	0	60490	1.056E-06	7.232E-07	
		M	0	60490	1.056E-06	7.760E-07	
		L	3	60490	2.112E-06	1.457E-06	
					4.223E-06	2.956E-06	
422					5.033E-05	3.536E-05	← M422
(2A DG Rm.)							
F422	FIRE	S	3	60-714/12759-02	2.973E-05	2.036E-05	
		M	23	60-714/12759-02	1.637E-05	1.204E-05	
		L	0	60-714/12759-02	0.000E+00	0.000E+00	
					4.610E-05	3.240E-05	
W422	SRW	S	0	62490	1.056E-06	7.232E-07	
		M	0	62490	1.056E-06	7.760E-07	
		L	3	62490	2.112E-06	1.457E-06	
					4.223E-06	2.956E-06	

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Piping Failure Frequencies

ROOM NUMBER ROOM FLOOD ID	SYSTEM	PIPE GROUP	NO. OF SECTIONS	BASIS	FREQ.	CORR. FREQ.	
428					5.372E-04	3.688E-04	← M428
(U-1 45' PIPING AREA)							
F428	FIRE	S	71	60-714/12261-01	5.180E-04	3.548E-04	
		M	23	60-714/12261-01	1.637E-05	1.204E-05	
		L	0	60-714/12261-01	0.000E+00	0.000E+00	
					5.344E-04	3.669E-04	
W428	SRW	S	0	60434	7.038E-07	4.821E-07	
		M	0	60434	7.038E-07	5.173E-07	
		L	2	60434	1.408E-06	9.713E-07	
					2.815E-06	1.971E-06	
429					2.878E-04	1.976E-04	← M429
(45' Ea. Penn. Rm.)							
F429	FIRE	S	38	60-714/12261-01	2.778E-04	1.903E-04	
		M	14	60-714/12261-01	9.967E-06	7.326E-06	
		L	0	60-714/12261-01	0.000E+00	0.000E+00	
					2.878E-04	1.976E-04	

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Piping Failure Frequencies

ROOM NUMBER ROOM FLOOD ID	SYSTEM	PIPE GROUP	NO. OF SECTIONS	BASIS	FREQ.	CORR. FREQ.	
520					2.649E-05	2.016E-05	← M520
(SFP Vent. Rm.)							
C520	CCW	S	0	60-710/435	7.038E-07	4.821E-07	
		M	0	60-710/435	7.038E-07	5.173E-07	
		L	2	60-710/435	1.408E-06	9.713E-07	
					2.815E-06	1.971E-06	
D520	DEMIN	S	3	60435	2.225E-05	1.691E-05	
		M	2	60435	1.424E-06	1.274E-06	
		L	0	60435	0.000E+00	0.000E+00	
					2.368E-05	1.819E-05	

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Piping Failure Frequencies

ROOM NUMBER ROOM FLOOD ID	SYSTEM	PIPE GROUP	NO. OF SECTIONS	BASIS	FREQ.	CORR. FREQ.	
524					1.738E-05	1.365E-05	← M524
(Mn. Plant Exh. Rm.)							
F524	FIRE	S	0	60-714 & W/D	0.000E+00	0.000E+00	
		M	0	60-714 & W/D	0.000E+00	0.000E+00	
		L	0	60-714 & W/D	0.000E+00	0.000E+00	
					0.000E+00	0.000E+00	
W524	SRW	S	0	60435	2.120E-06	1.452E-06	
		M	2	60435	2.832E-06	2.081E-06	
		L	4	60435	2.815E-06	1.943E-06	
					7.767E-06	5.476E-06	
D524	DEMIN	S	0	60706/60435	3.204E-06	2.435E-06	
		M	9	60706/60435	6.407E-06	5.735E-06	
		L	0	60706/60435	0.000E+00	0.000E+00	
					9.611E-06	8.170E-06	

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Piping Failure Frequencies

ROOM NUMBER ROOM FLOOD ID	SYSTEM	PIPE GROUP	NO. OF SECTIONS	BASIS	FREQ.	CORR. FREQ.	
525					3.776E-05	2.897E-05	M525
(69' PAL Area)							
F525	FIRE	S	0	60-714 & W/D	1.068E-06	7.315E-07	
		M	3	60-714 & W/D	2.136E-06	1.570E-06	
		L	0	60-714 & W/D	0.000E+00	0.000E+00	
					3.204E-06	2.301E-06	
W525	SRW	S	0	60435 & 60205	7.038E-07	4.821E-07	
		M	0	60435 & 60205	7.038E-07	5.173E-07	
		L	2	60435 & 60205	1.408E-06	9.713E-07	
					2.815E-06	1.971E-06	
C525	CCW	S	0	60-710/435	1.764E-06	1.208E-06	
		M	1	60-710/435	2.120E-06	1.558E-06	
		L	4	60-710/435	2.815E-06	1.943E-06	
					6.699E-06	4.709E-06	
D525	DEMIN	S	2	60435	1.792E-05	1.362E-05	
		M	10	60435	7.119E-06	6.372E-06	
		L	0	60435	0.000E+00	0.000E+00	
					2.504E-05	1.999E-05	

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Piping Failure Frequencies

ROOM NUMBER ROOM FLOOD ID	SYSTEM	PIPE GROUP	NO. OF SECTIONS	BASIS	FREQ.	CORR. FREQ.	
530					3.174E-05	2.329E-05	← M530
(SFP Cask Handling Area)							
F530	FIRE	S	0	60-714	3.916E-06	2.682E-06	
		M	11	60-714	7.831E-06	5.756E-06	
		L	0	60-714	0.000E+00	0.000E+00	
					1.175E-05	8.438E-06	
C530	CCW	S	0	60-710/422 & W/D	1.408E-06	9.643E-07	
		M	0	60-710/422 & W/D	1.408E-06	1.035E-06	
		L	4	60-710/422 & W/D	2.815E-06	1.943E-06	
					5.631E-06	3.941E-06	
D530	DEMIN	S	2	60717 & W/D	1.436E-05	1.091E-05	
		M	0	60717 & W/D	0.000E+00	0.000E+00	
		L	0	60717 & W/D	0.000E+00	0.000E+00	
					1.436E-05	1.091E-05	

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Piping Failure Frequencies

ROOM NUMBER ROOM FLOOD ID	SYSTEM	PIPE GROUP	NO. OF SECTIONS	BASIS	FREQ.	CORR. FREQ.	
536					4.871E-05	3.668E-05	← M536
Misc. Waste Evap. CP Rm.)							
C536	CCW	S	0	W/D	1.408E-06	9.643E-07	
		M	0	W/D	1.408E-06	1.035E-06	
		L	4	W/D	2.815E-06	1.943E-06	
					5.631E-06	3.941E-06	
D536	DEMIN	S	6	W/D	4.308E-05	3.274E-05	
		M	0	W/D	0.000E+00	0.000E+00	
		L	0	W/D	0.000E+00	0.000E+00	
					4.308E-05	3.274E-05	

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Piping Failure Frequencies

ROOM NUMBER ROOM FLOOD ID	SYSTEM	PIPE GROUP	NO. OF SECTIONS	BASIS	FREQ.	CORR. FREQ.	
537					9.897E-05	7.488E-05	← M537
(Misc. Waste Evap. Rm.)							
C537	CCW	S	0	W/D	1.408E-06	9.643E-07	
		M	0	W/D	1.408E-06	1.035E-06	
		L	4	W/D	2.815E-06	1.943E-06	
					5.631E-06	3.941E-06	
D537	DEMIN	S	13	60422 & W/D	9.334E-05	7.094E-05	
		M	0	60422 & W/D	0.000E+00	0.000E+00	
		L	0	60422 & W/D	0.000E+00	0.000E+00	
					9.334E-05	7.094E-05	

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Piping Failure Frequencies

ROOM NUMBER ROOM FLOOD ID	SYSTEM	PIPE GROUP	NO. OF SECTIONS	BASIS	FREQ.	CORR. FREQ.	
603					4.916E-04	3.395E-04	← M603
(U-1 AFW Pp. Rm.)							
N603	CST	S	18	60583	1.909E-04	1.308E-04	
		M	2	60583	4.536E-05	3.334E-05	
		L	9	60583	3.456E-05	2.385E-05	
					2.709E-04	1.880E-04	
F603	FIRE	S	30	60-714	2.168E-04	1.485E-04	
		M	4	60-714	2.848E-06	2.093E-06	
		L	0	60-714	0.000E+00	0.000E+00	
					2.197E-04	1.506E-04	
D603	DEMIN	S	0	60387	3.560E-07	2.705E-07	
		M	1	60387	7.119E-07	6.372E-07	
		L	0	60387	0.000E+00	0.000E+00	
					1.068E-06	9.077E-07	

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Piping Failure Frequencies

ROOM NUMBER ROOM FLOOD ID	SYSTEM	PIPE GROUP	NO. OF SECTIONS	BASIS	FREQ.	CORR. FREQ.	
605					6.060E-04	4.184E-04	← M605
(U-2 AFW Pp. Rm.)							
N605	CST	S	18	62583	1.980E-04	1.356E-04	
		M	3	62583	5.940E-05	4.366E-05	
		L	9	62583	3.456E-05	2.385E-05	
					2.919E-04	2.031E-04	
F605	FIRE	S	43	12261-14	3.105E-04	2.127E-04	
		M	5	12261-14	3.560E-06	2.616E-06	
		L	0	12261-14	0.000E+00	0.000E+00	
					3.141E-04	2.153E-04	
D605	DEMIN	S	0	62387	0.000E+00	0.000E+00	
		M	0	62387	0.000E+00	0.000E+00	
		L	0	62387	0.000E+00	0.000E+00	
					0.000E+00	0.000E+00	

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Piping Failure Frequencies

ROOM NUMBER ROOM FLOOD ID	SYSTEM	PIPE GROUP	NO. OF SECTIONS	BASIS	FREQ.	CORR. FREQ.	
205					1.863E-03	1.288E-03	← M205
(U-2 SRW Pp. Rm.)							
S205	SW	S	8	60708	9.974E-05	6.832E-05	
		M	16	60708	4.799E-05	3.527E-05	
		L	104	60708	7.320E-05	5.051E-05	
					2.209E-04	1.541E-04	
F205	FIRE	S	95	12261-28	6.953E-04	4.763E-04	
		M	37	12261-28	2.634E-05	1.936E-05	
		L	0	12261-28	0.000E+00	0.000E+00	
					7.216E-04	4.956E-04	
W205	SRW	S	71	60706	5.802E-04	3.975E-04	
		M	98	60706	1.053E-04	7.741E-05	
		L	101	60706	7.109E-05	4.905E-05	
					7.566E-04	5.239E-04	
N205	CST	S	7	62583	9.274E-05	6.352E-05	
		M	3	62583	5.172E-05	3.801E-05	
		L	5	62583	1.920E-05	1.325E-05	
					1.637E-04	1.148E-04	

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Piping Failure Frequencies

ROOM NUMBER ROOM FLOOD ID	SYSTEM	PIPE GROUP	NO. OF SECTIONS	BASIS	FREQ.	CORR. FREQ.	
12' TB					2.600E-03	1.793E-03	← MT12
IT12	CW	S	24	60708	2.699E-04	1.849E-04	
		M	120	60708/60406	1.403E-04	1.031E-04	
		L	156	60708/60406	1.098E-04	7.576E-05	
					5.201E-04	3.638E-04	
FT12	FIRE	S	271	60-714/12759-0002	1.987E-03	1.361E-03	
		M	114	60-714/12759-0002	8.222E-05	6.043E-05	
		L	3	60-714/12759-0002	2.112E-06	1.457E-06	
					2.072E-03	1.423E-03	
ST12	SW	S	0	60708	2.112E-06	1.446E-06	
		M	0	60708	2.112E-06	1.552E-06	
		L	6	60708	4.223E-06	2.914E-06	
					8.446E-06	5.912E-06	
27' TB					2.983E-03	2.047E-03	← MT27
FT27	FIRE	S	400	60-714/12759-0002	2.908E-03	1.992E-03	
		M	88	60-714/12759-0002	6.687E-05	4.915E-05	
		L	12	60-714/12759-0002	8.446E-06	5.828E-06	
					2.983E-03	2.047E-03	

ATTACHMENT 1

Piping Failure Frequencies

ROOM NUMBER ROOM FLOOD ID	SYSTEM	PIPE GROUP	NO. OF SECTIONS	BASIS	FREQ.	CORR. FREQ.	
INTAKE					1.271E-03	8.730E-04	MISP
IISP	CW	S	0	60708	4.223E-06	2.893E-06	
		M	0	60708	4.223E-06	3.104E-06	
		L	12	60708	8.446E-06	5.828E-06	
					1.689E-05	1.182E-05	
FISP	FIRE	S	4	60-714	3.086E-05	2.114E-05	
		M	6	60-714	4.272E-06	3.140E-06	
		L	0	60-714	0.000E+00	0.000E+00	
					3.513E-05	2.428E-05	
SISP	SW	S	154	60708	1.136E-03	7.782E-04	
		M	24	60708	3.891E-05	2.860E-05	
		L	62	60708	4.364E-05	3.011E-05	
					1.219E-03	8.369E-04	

ATTACHMENT 2
Operating Data

UNIT	S/U DATE	OP. DAYS > 1985	
ANO I	74 12 19	5,788	
ANO II	80 3 26	5,788	
BEAVER VALLEY I	76 10 1	5,788	
BEAVER VALLEY II	87 11 17	4,738	
BELLEFONTE I	NA	0	
BELLEFONTE II	NA	0	
BIG ROCK PT	63 3 29	5,788	
BRAIDWOOD I	88 7 29	4,483	
BRAIDWOOD II	88 10 17	4,403	
BROWNS FERRY I	74 8 1	5,788	
BROWNS FERRY II	75 3 1	5,788	
BROWNS FERRY III	77 3 1	5,788	
BRUNSWICK I	77 3 18	5,788	
BRUNSWICK II	75 11 3	5,788	
BYRON I	85 9 16	5,530	
BYRON II	87 8 21	4,826	
CALLAWAY I	84 12 19	5,788	
CALVERT CLIFFS I	75 5 8	5,788	
CALVERT CLIFFS II	77 4 1	5,788	
CARROL COUNTY I	NA	0	
CARROL COUNTY II	NA	0	
CATAWBA I	85 6 29	5,609	
CATAWBA II	86 8 19	5,193	
CLINTON	87 11 24	4,731	
COMMANCHE PK I	90 8 13	3,738	
COMMANCHE PK II	88 8 3	4,478	
COOK I	75 8 28	5,788	
COOK II	78 7 1	5,788	
COOPER	74 7 1	5,788	
CRYSTAL RIVER III	77 3 13	5,788	
DAVIS-BESSE	78 7 31	5,788	
DIABLO CANYON I	85 5 7	5,662	
DIABLO CANYON II	86 3 13	5,352	
DRESDEN I	59 10 15	0	S/D < 1985
DRESDEN II	70 6 9	5,788	
DRESDEN III	71 11 16	5,788	
DUANE ARNOLD	75 2 1	5,788	
FARLEY I	77 12 1	5,788	
FARLEY II	81 7 30	5,788	
FERMI I	63 5 10	0	S/D < 1985
FERMI II	88 1 23	4,671	
FITZPATRICK	75 7 28	5,788	
FT CALHOUN	74 6 20	5,788	
FT ST VRAIN	79 7 1	1,680	
GINNA	70 7 1	5,788	

ATTACHMENT 2
Operating Data

UNIT	S/U DATE	OP. DAYS > 1985	
GRAND GULF I	85 7 1	5,607	
GRAND GULF II	NA	0	
HADDAM NECK	68 1 1	4,355	S/D 1996
HARRIS 1	87 5 2	4,937	
HATCH I	75 12 31	5,788	
HATCH II	79 9 5	5,788	
HOPE CREEK	86 12 20	5,070	
HUMBOLDT BAY	63 8 1	0	S/D < 1985
INDIAN PT I	62 3 26	0	S/D < 1985
INDIAN PT II	74 8 1	5,788	
INDIAN PT III	76 8 30	5,788	
KEWAUNEE	74 6 16	5,788	
LA CROSSE I	69 11 1	849	S/D 1987
LASALLE I	84 1 1	5,788	
LASALLE II	84 10 19	5,788	
LIMERICK I	86 2 1	5,392	
LIMERICK II	90 1 8	3,955	
MCGUIRE I	81 12 1	5,788	
MCGUIRE II	84 3 1	5,788	
MIDLAND I	NA	0	
MIDLAND II	NA	0	
MILLSTONE I	71 3 1	5,788	
MILLSTONE II	75 12 26	5,788	
MILLSTONE III	86 4 23	5,311	
MN YANKEE	72 12 28	5,788	
MONTICELLO	71 6 30	5,788	
NINE MILE PT I	69 12 1	5,788	
NINE MILE PT II	88 3 11	4,623	
NORTH ANNA I	78 6 6	5,788	
NORTH ANNA II	80 12 14	5,788	
OCONEE I	73 7 15	5,788	
OCONEE II	74 9 9	5,788	
OCONEE III	74 12 16	5,788	
OYSTER CREEK	69 12 1	5,788	
PALISADES	71 12 31	5,788	
PALO VERDE I	86 1 28	5,396	
PALO VERDE II	86 9 19	5,162	
PALO VERDE III	88 1 8	4,686	
PEACH BOTTOM I	66 1 24	0	S/D < 1985
PEACH BOTTOM II	74 7 5	5,788	
PEACH BOTTOM III	74 12 23	5,788	
PERRY 1	87 11 18	4,737	
PERRY 2	NA	0	
PILGRIM	72 12 1	5,788	
PRAIRIE ISLAND I	73 12 16	5,788	
PRAIRIE ISLAND II	74 12 21	5,788	
PT BEACH I	70 12 21	5,788	
PT BEACH II	72 10 1	5,788	
QUAD CITIES I	73 2 18	5,788	
QUAD CITIES II	73 3 10	5,788	
RANCHO SECO	75 4 17	1,618	S/D < 1985

ATTACHMENT 2
Operating Data

UNIT	S/U DATE	OP. DAYS > 1985	
RIVER BEND	86 6 16	5,257	
ROBINSON II	71 3 7	5,788	
SALEM I	77 6 30	5,788	
SALEM II	81 10 13	5,788	
SAN ONOFRE I	68 1 1	2,890	S/D 1993
SAN ONOFRE II	83 8 8	5,788	
SAN ONOFRE III	84 4 1	5,788	
SEABROOK I	90 8 19	3,732	
SEABROOK II	NA	0	
SEQUOYAH I	81 7 1	5,788	
SEQUOYAH II	82 6 1	5,788	
SHIPPINGPORT	57 12 23	0	S/D < 1985
SHOREHAM	NA	0	
SOUTH TEXAS I	88 8 25	4,456	
SOUTH TEXAS II	89 6 19	4,158	
ST LUCIE I	76 12 21	5,788	
ST LUCIE II	83 8 8	5,788	
SUMMER	84 1 1	5,788	
SURRY I	72 12 22	5,788	
SURRY II	73 5 1	5,788	
SUSQUEHANNA I	83 6 8	5,788	
SUSQUEHANNA II	85 2 12	5,746	
TMI I	74 9 2	0	S/D < 1985
TMI II	78 10 30	5,788	
TROJAN	76 5 20	2,869	S/D 1992
TURKEY PT III	72 12 14	5,788	
TURKEY PT IV	73 9 7	5,788	
VERMONT YANKEE	72 11 30	5,788	
VOGTLE I	87 6 1	4,907	
VOGTLE II	89 5 20	4,188	
WATERFORD III	85 9 24	5,522	
WATTS BAR I	96 2 7	1,734	
WATTS BAR II	NA	0	
WNP I	NA	0	
WNP II	84 12 13	5,788	
WNP III	NA	0	
WOLF CREEK I	85 9 3	5,543	
YANKEE-ROWE	61 7 1	2,464	S/D 1991
ZION I	73 12 31	5,788	
ZION II	74 9 17	5,788	

SINCE DATA COLLECTION BEGAN IN 1985:

TOTAL OP. DAYS	**614358.0**
TOTAL OP. YEARS	**1682.0**
TOTAL CRITICAL DAYS	**490589.3**
TOTAL CRITICAL YEARS	**1343.2**

ATTACHMENT 3
Event Data

UNIT	Event Date	Affected System*	Operating Mode	Flood Source	Reference
ANO I					
ANO II					
BEAVER VALLEY I	UNKNOWN	FP	UNKNOWN	UNKNOWN	15
BEAVER VALLEY II	UNKNOWN	FP	UNKNOWN	UNKNOWN	15
BELLEFONTE I					
BELLEFONTE II					
BIG ROCK PT					
BRAIDWOOD I					
BRAIDWOOD II					
BROWNS FERRY I	5/19/1987	SW	UNKNOWN	remv'd hx head	9
BROWNS FERRY II					
BROWNS FERRY III					
BRUNSWICK I					
BRUNSWICK II					
BYRON I					
BYRON II					
CALLAWAY I					
CALVERT CLIFFS I					
CALVERT CLIFFS II	10/17/1990	RWT	NA	SFP Overfill	10
CARROL COUNTY I					
CARROL COUNTY II					
CATAWBA I					
CATAWBA II					
CLINTON I	1/4/1991	SRW	5	remv'd valve internals	4
COMMANCHE PK I					
COMMANCHE PK II					
COOK I					
COOK II					
COOPER					
CRYSTAL RIVER III	3/13/1993	CW	5	various system openings	6
DAVIS-BESSE					
DIABLO CANYON I					
DIABLO CANYON II					
DRESDEN I					
DRESDEN II	10/14/1990	RWT	NA	SFP Overfill	10
DRESDEN III					
DUANE ARNOLD					
FARLEY I					
FARLEY II					
FERMI II					
FITZPATRICK					

*CCNPP equivalent system

ATTACHMENT 3
Event Data

UNIT	Event Date	Affected System*	Operating Mode	Flood Source	Reference
FT CALHOUN	3/12/1990	RWT	NA	SFP Overfill	10
FT ST VRAIN					
GINNA					
GRAND GULF I					
GRAND GULF II					
HADDAM NECK					
HARRIS 1					
HATCH I	12/21/1985	RWT	6	remv'd valve internals	3
HATCH II					
HOPE CREEK					
HUMBOLDT BAY					
INDIAN PT II	8/13/1984	SW	6	remv'd valve internals	1
INDIAN PT III					
KEWAUNEE					
LASALLE I					
LASALLE II					
LIMERICK I					
LIMERICK II					
MCGUIRE I	9/5/1989	RWT	4	system overpressure (failed component)	7
MCGUIRE II					
MIDLAND I					
MIDLAND II					
MILLSTONE I					
MILLSTONE II					
MILLSTONE III					
MN YANKEE					
MONTICELLO					
NINE MILE PT I					
NINE MILE PT II					
NORTH ANNA I					
NORTH ANNA II					
OCONEE I	10/7/1987	RWT	5	remv'd pipe	2
"	5/17/1990	RWT	NA	SFP Overfill	10
OCONEE II					
OCONEE III					
OYSTER CREEK					
PALISADES					
PALO VERDE I					
PALO VERDE II					
PALO VERDE III					
PEACH BOTTOM III	1/14/1984	CW	1	mispos. valve	1
"	2/25/1997	CW	1	mispos. valve	11

*CCNPP equivalent system

ATTACHMENT 3
Event Data

UNIT	Event Date	Affected System*	Operating Mode	Flood Source	Reference
PERRY 1					
PERRY 2					
PILGRIM					
PRAIRIE ISLAND I					
PRAIRIE ISLAND II					
PT BEACH I					
PT BEACH II					
QUAD CITIES I					
QUAD CITIES II					
RANCHO SECO					
RIVER BEND	4/19/1989	SRW	5	remv'd valve internals	2
ROBINSON II					
SALEM I	12/22/1987	SRW	5	isol. valve failure	14
SALEM II					
SAN ONOFRE I					
SAN ONOFRE II					
SAN ONOFRE III	7/15/1991	RWT	NA	SFP Overfill	8
SEABROOK I					
SEABROOK II					
SEQUOYAH I					
SEQUOYAH II					
SHIPPINGPORT					
SHOREHAM					
SOUTH TEXAS I	7/14/1993	SRW	6	isol. valve failure	12
SOUTH TEXAS II					
ST LUCIE I					
ST LUCIE II					
SUMMER					
SURRY I					
SURRY II					
SUSQUEHANNA I					
SUSQUEHANNA II					
TMI I					
TMI I					
TMI II					
TROJAN					
TURKEY PT III					
TURKEY PT IV					
VERMONT YANKEE					
VOGTLE I					
VOGTLE II					
WATERFORD III					
WATTS BAR I					
WATTS BAR II					
WNP I					

*CCNPP equivalent system

ATTACHMENT 3
Event Data

UNIT	Event Date	Affected System*	Operating Mode	Flood Source	Reference
WNP II					
WNP III					
WOLF CREEK					
YANKEE-ROWE					
ZION I					
ZION II					

Total Number of Events Occurring @ Power	10
Total Number of Events Occurring in All Modes	20
Operating Year Event Frequency	7.445E-03
All Events, All Modes Frequency	1.189E-02
SRW Operating Year Event Frequency	1.000E-04
SRW All Events, All Modes Frequency	2.378E-03
CST Operating Year Event Frequency	1.000E-04
CST All Events, All Modes Frequency	0.000E+00
RWT Operating Year Event Frequency	3.723E-03
RWT All Events, All Modes Frequency	4.756E-03
SW Operating Year Event Frequency	7.445E-04
SW All Events, All Modes Frequency	1.189E-03
CW Operating Year Event Frequency	1.489E-03
CW All Events, All Modes Frequency	1.784E-03
FP Operating Year Event Frequency	1.489E-03
FP All Events, All Modes Frequency	1.189E-03

Note: A frequency of 1E-4 is conservatively used for SRW and CST since there are no reported floods at power for these systems. A conservative frequency of 1E-3 is used for Demineralized Water, which is not likely to be a reported system (i.e. NSR). 1E-3 is acceptable in that it is in the same range as for other frequencies.
There is no frequency associated with MFW flooding since this system cannot be maintained at power in such a manner as to lead to flooding.

Maintenance Frequencies are calculated using data terminated on 11/06/2000.

*CCNPP equivalent system

ATTACHMENT 4
Failure Cause Screening

SYSTEM	MN-1-209 MONITORING	MN-3-202 MONITORING
CW	CW PUMPS	NO
SW	SW PUMPS	NO
FIRE PROT.	FIRE PROT. PUMPS	NO
CCW	CCW PUMPS	NO
DEMIN	NONE*	NO
SRW	SRW PUMPS	NO
RWT (SI)	CS, HPSI, AND LPSI PUMPS	NO
CST (COND)	AFW PUMPS	NO
COND.	COND. & BOOSTER PUMPS	NO
MN FEED	SG FEED PUMPS	YES**

MN-1-209 monitoring indicates that the system receives engineering attention in the area of low-cycle fatigue failure and therefore is at a much reduced risk for this type of failure. Low cycle fatigue is a concern where maintenance affects motor to pump alignments, resulting in vibration.

MN-1-209 describes the site's vibration monitoring program, which periodically verifies good alignment. Additionally, the site provides for vibration testing after maintenance activity.

Because the Main Feed system is included in Life Cycle Mngt's fatigue monitoring program, it is determined to be susceptible to this failure mode and thereby is not screened. Only the Main Feed System is vulnerable to both high- and low-cycle fatigue mechanisms.

MN-3-202 monitoring implies that the system is vulnerable to erosion/corrosion failure. The Main Feed system is also monitored for this failure mechanism (see footnote below).

*Not a fatigue concern per reference 7.

**Encapsulated feed piping from the MSIV room to the 5' Piping Area cannot be inspected for errosion/corrosion, and therefore cannot be screened.

ATTACHMENT 5

Total Flood Frequencies

ROOM NUMBER ROOM FLOOD ID	CORRECTED PIPE-RUPTURE FREQ.	MAINT-INDUCED FREQUENCY	TOTAL FLOOD FREQ.
118			
S118A	6.505E-05	1.918E-05	8.423E-05
F118A	0.000E+00	0.000E+00	0.000E+00
C118A	3.488E-04	2.048E-05	3.693E-04
M118A	4.139E-04	3.965E-05	4.535E-04
119			
S119A	6.112E-05	1.918E-05	8.030E-05
F119A	2.135E-04	1.489E-05	2.284E-04
C119A	1.828E-04	1.476E-05	1.976E-04
M119A	4.574E-04	4.883E-05	5.062E-04
221			
D221A	9.077E-07	1.287E-06	2.195E-06
F221A	2.634E-05	1.489E-05	4.123E-05
C221A	1.022E-04	5.952E-06	1.082E-04
W221A	5.679E-05	6.143E-06	6.294E-05
R221A	1.177E-06	2.829E-06	4.006E-06
M221A	1.874E-04	3.110E-05	2.185E-04
224			
C224A	1.279E-05	1.190E-06	1.398E-05
N224A	3.195E-05	6.400E-07	3.259E-05
D224A	5.184E-05	1.544E-05	6.728E-05
F224A	2.296E-04	1.489E-05	2.445E-04
R224A	1.177E-06	2.829E-06	4.006E-06
W224A	9.252E-05	6.000E-06	9.852E-05
Q224A	5.732E-05	0.000E+00	5.732E-05
M224A	4.772E-04	4.099E-05	5.182E-04
225			
N225A	3.633E-05	4.800E-07	3.681E-05
D225A	9.077E-07	1.287E-06	2.195E-06
F225A	1.522E-04	1.489E-05	1.670E-04
M225A	1.894E-04	1.666E-05	2.061E-04

ATTACHMENT 5

Total Flood Frequencies

ROOM NUMBER ROOM FLOOD ID	CORRECTED PIPE-RUPTURE FREQ.	MAINT-INDUCED FREQUENCY	TOTAL FLOOD FREQ.
226			
S226A	1.285E-04	7.837E-05	2.068E-04
F226A	4.165E-04	1.489E-05	4.314E-04
W226A	6.823E-04	3.857E-05	7.209E-04
N226A	1.305E-04	2.080E-06	1.326E-04
M226A	1.358E-03	1.339E-04	1.492E-03
N226L	1.060E-05	6.400E-07	1.124E-05
N226M	6.756E-05	9.600E-07	6.852E-05
N226S	5.234E-05	4.800E-07	5.282E-05
227			
C227A	2.559E-05	2.381E-06	2.797E-05
D227A	2.000E-05	9.009E-06	2.901E-05
F227A	3.691E-04	1.489E-05	3.840E-04
W227A	7.527E-05	7.143E-06	8.241E-05
N227A	1.152E-04	3.200E-06	1.184E-04
M227A	6.051E-04	3.662E-05	6.418E-04
228			
C228A	3.159E-04	3.405E-05	3.500E-04
D228A	1.273E-05	5.148E-06	1.788E-05
F228A	4.061E-04	1.489E-05	4.210E-04
R228A	9.698E-06	5.657E-06	1.536E-05
S228A	1.856E-04	6.420E-05	2.498E-04
M228A	9.300E-04	1.239E-04	1.054E-03
S228L	2.283E-05	3.918E-05	6.201E-05
S228M	1.320E-05	1.667E-06	1.487E-05
S228S	1.495E-04	2.334E-05	1.729E-04
315			
F315A	2.462E-04	1.489E-05	2.611E-04
W315A	1.971E-06	2.857E-07	2.256E-06
M315A	2.481E-04	1.518E-05	2.633E-04
317			
F317A/M317A	2.301E-06	5.014E-06	7.315E-06

ATTACHMENT 5
Total Flood Frequencies

ROOM NUMBER ROOM FLOOD ID	CORRECTED PIPE-RUPTURE FREQ.	MAINT-INDUCED FREQUENCY	TOTAL FLOOD FREQ.
318			
N318A	2.588E-05	0.000E+00	2.588E-05
D318A	9.077E-07	1.287E-06	2.195E-06
M318A	2.679E-05	1.287E-06	2.807E-05
N318L	5.299E-06	0.000E+00	5.299E-06
N318M	1.314E-05	0.000E+00	1.314E-05
N318S	7.439E-06	0.000E+00	7.439E-06
319			
D319A	5.363E-05	3.089E-05	8.452E-05
F319A	2.956E-06	1.489E-05	1.785E-05
W319A	0.000E+00	0.000E+00	0.000E+00
M319A	5.659E-05	4.578E-05	1.024E-04
320			
D320A	1.815E-06	2.574E-06	4.389E-06
W320A	2.168E-05	3.143E-06	2.482E-05
M320A	2.349E-05	5.717E-06	2.921E-05
324			
C324A	1.205E-05	3.333E-06	1.538E-05
M324A	1.205E-05	3.333E-06	1.538E-05
419			
C419A	5.511E-05	5.714E-06	6.083E-05
D419A	5.002E-05	1.287E-05	6.289E-05
F419A	9.317E-04	1.489E-05	9.465E-04
W419A	3.941E-06	5.714E-07	4.513E-06
M419A	1.037E-03	3.347E-05	1.070E-03
421			
F421A	1.237E-04	1.489E-05	1.386E-04
W421A	2.956E-06	4.286E-07	3.385E-06
M421A	1.267E-04	1.532E-05	1.420E-04
422			
F422A	3.240E-05	1.489E-05	4.729E-05
W422A	2.956E-06	4.286E-07	3.385E-06
M422A	3.536E-05	1.532E-05	5.067E-05

ATTACHMENT 5

Total Flood Frequencies

ROOM NUMBER ROOM FLOOD ID	CORRECTED PIPE-RUPTURE FREQ.	MAINT-INDUCED FREQUENCY	TOTAL FLOOD FREQ.
428			
F428A	3.669E-04	1.489E-05	3.817E-04
W428A	1.971E-06	2.857E-07	2.256E-06
M428A	3.688E-04	1.518E-05	3.840E-04
429			
F429A	1.976E-04	1.489E-05	2.125E-04
M429A	1.976E-04	1.976E-04	3.953E-04
520			
C520A	1.971E-06	4.762E-07	2.447E-06
D520A	1.819E-05	6.435E-06	2.462E-05
M520A	2.016E-05	6.911E-06	2.707E-05
524			
F524A	0.000E+00	0.000E+00	0.000E+00
W524A	5.476E-06	8.571E-07	6.333E-06
D524A	8.170E-06	1.158E-05	1.975E-05
M524A	1.365E-05	1.244E-05	2.609E-05
525			
F525A	2.301E-06	1.489E-05	1.719E-05
W525A	1.971E-06	2.857E-07	2.256E-06
C525A	4.709E-06	1.190E-06	5.899E-06
D525A	1.999E-05	1.544E-05	3.544E-05
M525A	2.897E-05	3.181E-05	6.078E-05
530			
F530A	8.438E-06	1.489E-05	2.333E-05
C530A	3.941E-06	9.524E-07	4.894E-06
D530A	1.091E-05	2.574E-06	1.349E-05
R530A	0.000E+00	1.861E-03	1.861E-03 *
M530A	2.329E-05	1.880E-03	1.903E-03

* R530A Eliminated from CCPRA based on Engineering judgement. Minimal impact on plant model.

ATTACHMENT 5
Total Flood Frequencies

ROOM NUMBER ROOM FLOOD ID	CORRECTED PIPE-RUPTURE FREQ.	MAINT-INDUCED FREQUENCY	TOTAL FLOOD FREQ.
536			
C536A	3.941E-06	9.524E-07	4.894E-06
D536A	3.274E-05	7.722E-06	4.046E-05
M536A	3.668E-05	8.674E-06	4.536E-05
537			
C537A	3.941E-06	9.524E-07	4.894E-06
D537A	7.094E-05	1.673E-05	8.767E-05
M537A	7.488E-05	1.768E-05	9.256E-05
603			
N603A	1.880E-04	4.640E-06	1.926E-04
F603A	1.506E-04	1.489E-05	1.655E-04
D603A	9.077E-07	1.287E-06	2.195E-06
M603A	3.395E-04	2.082E-05	3.603E-04
605			
N605A	2.031E-04	4.800E-06	2.079E-04
F605A	2.153E-04	1.489E-05	2.302E-04
D605A	0.000E+00	0.000E+00	0.000E+00
M605A	4.184E-04	1.969E-05	4.381E-04
205			
S205A	1.541E-04	1.067E-04	2.608E-04
F205A	4.956E-04	1.489E-05	5.105E-04
W205A	5.239E-04	3.857E-05	5.625E-04
N205A	1.148E-04	2.400E-06	1.172E-04
M205A	1.288E-03	1.626E-04	1.451E-03
N205L	1.325E-05	8.000E-07	1.405E-05

ATTACHMENT 5

Total Flood Frequencies

ROOM NUMBER ROOM FLOOD ID	CORRECTED PIPE-RUPTURE FREQ.	MAINT-INDUCED FREQUENCY	TOTAL FLOOD FREQ.
12' TB			
IT12A	3.638E-04	1.551E-03	1.915E-03
FT12A	1.423E-03	1.489E-05	1.438E-03
ST12A	5.912E-06	2.501E-06	8.413E-06
MT12A	1.793E-03	1.568E-03	3.361E-03
27' TB			
FT27A	2.047E-03	1.489E-05	2.062E-03
MT27A	2.047E-03	1.489E-05	2.062E-03
INTAKE			
IISPA	1.182E-05	6.204E-05	7.387E-05
FISPA	2.428E-05	1.489E-05	3.917E-05
SISPA	8.369E-04	1.000E-04	9.370E-04
MISPA	8.730E-04	1.770E-04	1.050E-03

Flood ID	INITIATING EVENT FREQUENCY FORMULA CONTSTITUENTS 1	2	3	4	5	6	7	HA #1	HA #2	INITIATING EVENT FREQUENCY FORMULA	RESULTS	0Q Results
S118XR	S118A	S119A						1-BHF118	NA	freq = (S118A + S119A)*(1-BHF118)	1.644E-04	1.261E-04
	8.423E-05	8.030E-05						9.990E-01				
0Q	6.505E-05	6.112E-05										
S118XN	S118A	S119A						BHF118	NA	freq = (S118A + S119A)*(BHF118)	1.573E-07	1.206E-07
	8.423E-05	8.030E-05						9.560E-04				
0Q	6.505E-05	6.112E-05										
C118XR	C118A	C119A						1-BHF118	NA	freq = (C118A + C119A)*(1-BHF118)	5.663E-04	5.311E-04
	3.693E-04	1.976E-04						9.990E-01				
0Q	3.488E-04	1.828E-04										
C118XN	C118A	C119A						BHF118	NA	freq = (C118A + C119A)*(BHF118)	5.419E-07	5.223E-07
	3.693E-04	1.976E-04						9.560E-04				
0Q	3.488E-04	1.976E-04										
F119AM	F119A							NA	NA	freq = F119A	2.284E-04	2.135E-04
	2.284E-04											
0Q	2.135E-04											
S205AR	S205A							1-BHF118	NA	freq = F205A * (1-BHF118)	2.606E-04	1.540E-04
	2.608E-04							9.990E-01				
0Q	1.541E-04											
S205AN	S205A							BHF118	NA	freq = F205A * (BHF118)	2.493E-07	1.473E-07
	2.608E-04							9.560E-04				
0Q	1.541E-04											
N205LN	N205L							NA	NA	freq = N205L	1.405E-05	1.325E-05
	1.405E-05											
0Q	1.325E-05											
F205AN	F205A							NA	NA	freq = F205AN [covers both the successful/not successful HA cases in CCPRA]	5.105E-04	4.956E-04
	5.105E-04							1.000E+00				
0Q	4.956E-04											
F221AM	F221A							NA	NA	freq = F221A	4.123E-05	2.634E-05
	4.123E-05											
0Q	2.634E-05											
C221AM	C221A							NA	NA	freq = C221A	1.082E-04	1.022E-04
	1.082E-04											
0Q	1.022E-04											
W221AM	W221A							NA	NA	freq = W221A	6.294E-05	5.679E-05
	6.294E-05											
0Q	5.679E-05											
D221AM	D221A							NA	NA	freq = D221A	2.195E-06	9.077E-07
	2.195E-06											
0Q	9.077E-07											
R221AM	R221A							NA	NA	freq = R221A	4.006E-06	1.177E-06
	4.006E-06											
0Q	1.177E-06											

Flood ID	INITIATING EVENT FREQUENCY FORMULA CONTSTITUENTS 1	2	3	4	5	6	7	HA #1	HA #2	INITIATING EVENT FREQUENCY FORMULA	RESULTS	0Q Results
N224AM	N224A							NA	NA	freq = N224A	3.259E-05	3.195E-05
	3.259E-05											
0Q	3.195E-05											
R224AM	R224A							NA	NA	freq = R224A	4.006E-06	1.177E-06
	4.006E-06											
0Q	1.177E-06											
F224AM	F224A							NA	NA	freq = F224A	2.445E-04	2.296E-04
	2.445E-04											
0Q	2.296E-04											
C224AM	C224A							NA	NA	freq = C224A	1.398E-05	1.279E-05
	1.398E-05											
0Q	1.279E-05											
W224AM	W224A							NA	NA	freq = W224A	9.852E-05	9.252E-05
	9.852E-05											
0Q	9.252E-05											
Q224AM	Q224A							NA	NA	freq = Q224A	5.732E-05	5.732E-05
	5.732E-05											
0Q	5.732E-05											
F225AM	F225A							NA	NA	freq = F225A	1.670E-04	1.522E-04
	1.670E-04											
0Q	1.522E-04											
N225AM	N225A							NA	NA	freq = N225A	3.681E-05	3.633E-05
	3.681E-05											
0Q	3.633E-05											
D225AM	D225A							NA	NA	freq = D225A	2.195E-06	9.077E-07
	2.195E-06											
0Q	9.077E-07											
S226AR	S226A							1-BHF118	NA	freq = S226A*(1-BHF118)	2.066E-04	1.283E-04
	2.068E-04							9.990E-01				
0Q	1.285E-04											
S226AN	S226A							BHF118	NA	freq = S226A*(BHF118)	1.977E-07	1.228E-07
	2.068E-04							9.560E-04				
0Q	1.285E-04											
X226AM	F226A	W226A						NA	NA	freq = F226A + W226A	1.152E-03	1.099E-03
	4.314E-04	7.209E-04										
0Q	4.165E-04	6.823E-04										
N226AM	N226							NA	NA	freq = N226A	1.326E-04	1.305E-04
	1.326E-04											
0Q	1.305E-04											
F227AM	F227A							NA	NA	freq = F227A	3.840E-04	3.691E-04
	3.840E-04											
0Q	3.691E-04											

Flood ID	INITIATING EVENT FREQUENCY FORMULA CONTSTITUENTS 1	2	3	4	5	6	7	HA #1	HA #2	INITIATING EVENT FREQUENCY FORMULA	RESULTS	0Q Results
C227AM	C227A							NA	NA	freq =C227A	2.797E-05	2.559E-05
	2.797E-05											
0Q	2.559E-05											
W227AM	W227A							NA	NA	freq =W227A	8.241E-05	7.527E-05
	8.241E-05											
0Q	7.527E-05											
D227AM	D227A							NA	NA	freq =D227A	2.901E-05	2.000E-05
	2.901E-05											
0Q	2.000E-05											
X228AM	F228A	C228A	D228A					NA	NA	freq = F228A + C228A + D228A	7.888E-04	7.348E-04
	4.210E-04	3.500E-04	1.788E-05									
0Q	4.061E-04	3.159E-04	1.273E-05									
S228AR	S228L	S228M	S228S					1-BHF118	NA	freq = S228L*(1-BHF118)	6.195E-05	2.280E-05
	6.201E-05	1.487E-05	1.729E-04					9.990E-01				
0Q	2.283E-05	1.320E-05	1.495E-04									
S228AN	S228L	S228M	S228S					BHF118	NA	freq = (S228L + S228M + S228S)*BHF118	2.388E-07	1.774E-07
	6.201E-05	1.487E-05	1.729E-04					9.560E-04				
0Q	2.283E-05	1.320E-05	1.495E-04									
R228AM	R228A							NA	NA	freq = R228A	1.536E-05	9.698E-06
	1.536E-05											
0Q	9.698E-06											
F315AM	F315A							NA	NA	freq = F315A	2.611E-04	2.462E-04
	2.611E-04											
0Q	2.462E-04											
W315AM	W315A							NA	NA	freq = W315A	2.256E-06	1.971E-06
	2.256E-06											
0Q	1.971E-06											
F317AM	F317A							NA	NA	freq = F317A	7.315E-06	2.301E-06
	7.315E-06											
0Q	2.301E-06											
N318AM	N318A							NA	NA	freq = N318A	2.588E-05	2.588E-05
	2.588E-05											
0Q	2.588E-05											
D318AM	D318A							NA	NA	freq = D318A	2.195E-06	9.077E-07
	2.195E-06											
0Q	9.077E-07											

Flood ID	INITIATING EVENT FREQUENCY FORMULA CONTSTITUENTS									INITIATING EVENT FREQUENCY FORMULA	RESULTS	0Q Results
	1	2	3	4	5	6	7	HA #1	HA #2			
X319AM	D224A	F319A	D319A	D320A				NA	NA	freq = D224A + F319A + D319A + D320A	1.740E-04	1.102E-04
	6.728E-05	1.785E-05	8.452E-05	4.389E-06								
0Q	5.164E-05	2.956E-06	5.363E-05	1.815E-06								
W320AM	W320A							NA	NA	freq = W320A	2.482E-05	2.168E-05
	2.482E-05											
0Q	2.168E-05											
C324AM	C324A							NA	NA	freq = C324A	1.538E-05	1.205E-05
	1.538E-05											
0Q	1.205E-05											
M419AM	F419A	C419A	D419A					NA	NA	freq = F419A + C419A + D419A	1.070E-03	1.037E-03
	9.465E-04	6.083E-05	6.289E-05									
0Q	9.317E-04	5.511E-05	5.002E-05									
W419AM	W419A							NA	NA	freq = W419A	4.513E-06	3.941E-06
	4.513E-06											
0Q	3.941E-06											
M421AM	F421A	W421A						NA	NA	freq = F421A + W421A	1.420E-04	1.267E-04
	1.386E-04	3.385E-06										
0Q	1.237E-04	2.956E-06										
M422AM	F422A	W422A						NA	NA	freq = F422A + W422A	5.067E-05	3.536E-05
	4.729E-05	3.385E-06										
0Q	3.240E-05	2.956E-06										
F428AM	F428A							NA	NA	freq = F428A	3.817E-04	3.669E-04
	3.817E-04											
0Q	3.669E-04											
W428AM	W428A							NA	NA	freq = W428A	2.256E-06	1.971E-06
	2.256E-06											
0Q	1.971E-06											
F429AM	F429A							NA	NA	freq = F429A	2.125E-04	1.976E-04
	2.125E-04											
	1.976E-04											
X524AM	F524A	D524A						NA	NA	freq = F524A + D524A	1.975E-05	8.170E-06
	0.000E+00	1.975E-05										
	0.000E+00	8.170E-06										
W524AM	W524A							NA	NA	freq =W524A	6.333E-06	5.476E-06
	6.333E-06											
	5.476E-06											
W525AM	W525A							NA	NA	freq = W525A	2.256E-06	1.971E-06
	2.256E-06											
	1.971E-06											
C537XM	C520A	C525A	C530A	C536A	C537A			NA	NA	freq = C520A + C525A + C530A + C536A + C537A	2.303E-05	1.860E-05
	2.447E-06	5.899E-06	4.894E-06	4.894E-06	4.894E-06							
	1.971E-06	4.709E-06	3.941E-06	3.941E-06	3.941E-06							

Flood ID	INITIATING EVENT FREQUENCY FORMULA CONTSTITUENTS											
	1	2	3	4	5	6	7	HA #1	HA #2	INITIATING EVENT FREQUENCY FORMULA	RESULTS	0Q Results
X537AM	D520A	F525A	D525A	F530A	D530A	D536A	D537A	NA	NA	freq = D520A + F525A + D525A + F530A + D530A + D536A + D537A	2.422E-04	1.635E-04
	2.462E-05	1.719E-05	3.544E-05	2.333E-05	1.349E-05	4.046E-05	8.767E-05					
	1.819E-05	2.301E-06	1.999E-05	8.438E-06	1.091E-05	3.274E-05	7.094E-05					
F603AM	F603A							NA	NA	freq = F603A	1.655E-04	1.506E-04
	1.655E-04											
	1.506E-04											
D603AM	D603A							NA	NA	freq = D603A	2.195E-06	9.077E-07
	2.195E-06											
	9.077E-07											
N603AM	N603A							NA	NA	freq = N603A	1.926E-04	1.880E-04
	1.926E-04											
	1.880E-04											
N605AM	N605A							NA	NA	freq = N605A	2.079E-04	2.031E-04
	2.079E-04											
	2.031E-04											
F605AM	F605A							NA	NA	freq = F605A	2.302E-04	2.153E-04
	2.302E-04											
	2.153E-04											
SISPAR	SISPA							1-BHF121	NA	freq = SISPA*(1-BHF121)	9.166E-04	8.187E-04
	9.370E-04							9.782E-01				
	8.369E-04											
SISPAN	SISPA							BHF121	NA	freq = SISPA*(BHF121)	2.043E-05	1.825E-05
	9.370E-04							2.180E-02				
	8.369E-04											
FISPAM	FISPA							NA	NA	freq = FISPA	3.917E-05	2.428E-05
	3.917E-05											
	2.428E-05											
IISPAM	IISPA							NA	NA	freq = IISPA	7.387E-05	1.182E-05
	7.387E-05											
	1.182E-05											
ST12AM	ST12A							NA	NA	freq = ST12A	8.413E-06	5.912E-06
	8.413E-06											
	5.912E-06											
XT27AM	FT12A	FT27A						NA	NA	freq = FT12A + FT27A	3.500E-03	3.470E-03
	1.438E-03	2.062E-03										
	1.423E-03	2.047E-03										
IT12AM	IT12A							NA	NA	freq = IT12A	1.915E-03	3.638E-04
	1.915E-03											
	3.638E-04											

BGE

RELIABILITY ENGINEERING

June 29, 1998
RE98-066

TO: **File**

FROM: **R. E. Franke**

SUBJECT: **RAN 98-062 Attachment 7: Record of Plant Walkdown**

On Friday, September 4, 1998, I conducted a plant walkdown to verify the correspondence between my pipe segment counts from P&IDs to the plant. As I had anticipated, there were some discrepancies in the area of the Demineralized Water and Fire Suppression system pipe counts. In general. all other counts appear accurate.

The walkdown record appears as RAN 98-062 Attachment 7.

Enclosure

REF/hcb

Attachment 7

Record of Plant Walkdown

ROOM	FINDINGS
119	Verified the absence of fire suppression. There were no locatable drawings for this room related to fire suppression.
224	Verified the presence of two fire suppression hose stations in the area.
227	Verified four fire-suppression large diameter pipe segments (system utilizes 8" piping in the main distribution network) as well as two small diameter lines not detected.
319	Detected seven segments of small diameter Demin. Water piping and another fifteen segments of intermediate piping. This room also had another two segments of small diameter fire-suppression piping and seven segments of inter. piping.
320	Found two segments of small diameter Demin. Water piping.
419	Found nine additional small diameter Demin. Water pipe segments.
524	Verified the total absence of fire-suppression piping.
525	Found three segments of intermediate diameter fire-suppression piping.
536	Located six segments of small-dia. Demin. Water piping and no fire-suppression piping. There are four segments of large diameter CCW piping in the room.

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Contact Events Analysis **Department Manager:** lynchje@inponn.org.

Tell us what you think about **this page.**

Plant Events Database

Search Plant Events Database

Select search criteria for desired fields by clicking on the underlined heading.

Then click "Start Search" to search on selected criteria.

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Unit:	
NSSS Vendor:	All
NSSS Type:	All
Country:	All
Unit:	All
Event Data:	
INPO Event Number:	All
Event Date:	All
INPO Change Date:	All
Narrative Words:	All
Event Codes:	
INPO Significance:	All
Event Descriptors:	All
Keywords:	All
Human Performance Causal Factors:	All
Component/System Failure:	
Systems:	All
Components:	All
Equipment Causal Factors:	All
Scrams:	
Signals:	All
Initiating Transients:	All

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SEE-IN

Search Licensee Event Reports (LERs)

Click on the underlined row heading to display the selection screen for that field.

The query is currently empty, please make your selections.

Note: If an LER search result appears without a hyperlink, it is not yet available in electronic version.

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Primary Unit:	
NSSS Vendor	All
NSSS Type	All
Unit:	All
Other Affected Units:	All
Specific LER:	All
Event Date:	All
Report Date:	All
Title contains	All
Abstract contains	All
Requirement	All
Component/System Failure:	
Cause Code	All
System	All
Component	All
Component Manufacturer	All

Start Search | Reset Query | Help

Rupture Failure Rates for each Generic PWR System and Pipe-Size Group from Analysis

	Failure Rate (per section/year)		
Group Name	S	M	L
SI/Recirc.	1.244E-05	9.899E-07	1.682E-06
Other SR	7.180E-06	7.119E-07	1.408E-06
Feed/Cond.	8.868E-06	1.404E-05	7.680E-06

Conditional Probabilities of Eq. Break Sizes	
P_{SM}	0.5
P_{MM}	0.5
P_{SL}	0.25
P_{ML}	0.25
P_{LL}	0.5

Percentiles of Failure by Cause Catagory

Catagory	S	M	L	Total
Fatigue	0.48	0.21	0.09	0.31
Erosion/Corr	0.15	0.32	0.53	0.3

		50% S/D impact
Critical to Calendar Hour Ratio (See Key Input 1395)	73%	86.5%

(As EPRI Manual appear to use Calendar Hours for Failure Rates the data is revised per this Criticaility Factor as noted below)

SI/Recirc Class	No adjustment made - HPSI and LPSI are generally available close to 100% of operational hours.
Other SR	This includes SW/SRW/Fire Protection systems. Generally SW and SRW are running during outages - but only one header. Fire Protection is always available. As it is uncertain which other Safety Related Systems are covered in the data, and SW/SRW headers are OOS 50% during outages - this is adjusted assuming out of service 50% of time during outage.
Feed/Cond.	For a very large percentage of planned outages Feedwater and Condensate will be out of service. This will be adjusted assuming out of service when not critical - as there are some periods these will be in service when not critical it is slightly conservative.