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PG&E Letter DCL-09-044

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

Docket No. 50-275, OL-DPR-80
Docket No. 50-323, OL-DPR-82
Diablo Canyon Units 1 and 2
Emergency Plan Update

Dear Commissioners and Staff:

In accordance with 10 CFR 50.4 and 10 CFR 50.54(q), Pacific Gas and Electric Company is submitting changes to the Emergency Plan (E-Plan) for Diablo Canyon Power Plant, and including the requirements of 10 CFR 72.32 for the Independent Spent Fuel Storage Installation. This submittal includes a complete E-Plan binder. Enclosure 1 provides a description of the changes.

As provided under 10 CFR 50.54(q), these changes have been made without prior NRC approval since they do not decrease the effectiveness of the E-Plan, and the Plan, as changed, continues to meet 10 CFR 72.32, 10 CFR 50.47(b), and 10 CFR 50, Appendix E. Revision bars in the right-hand margin note the changes.

These changes do not contain privacy/proprietary information identified in accordance with NRC Generic Letter 81-27. Enclosure 2 provides a listing of the applicable revision/change for each section of the E-Plan.

If there are any further questions regarding these changes, please contact Mr. Michael A. Ginn of my staff at (805) 545-6300.

Sincerely,

James R. Becker

ddm/4509/50197009
Enclosures
cc: Alan B. Wang, NRC Project Manager
Shana R. Helton, NRC Senior Project Manager, ISFSI
cc/enc: Elmo E. Collins, (2) NRC Regional Administrator, Region IV
Michael S. Peck, NRC Senior Resident Inspector

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DIABLO CANYON POWER PLANT, UNITS 1 AND 2, EMERGENCY PLAN CHANGE SUMMARY

GENERAL CHANGE DESCRIPTION

The attached Emergency Plan sections are changed as a result of an upgrade to the Emergency Response Organization (ERO) including several editorial changes regarding position titles. Specific changes are summarized as follows:

Section 1, Revision 4.05, "Definitions and Acronyms," was not revised but is included for completeness.

Section 2, Revision 4.03, "Scope and Applicability," was not revised but is included for completeness.

Section 3, Revision 4.00, "Summary of Emergency Plan," was not revised but is included for completeness.

Section 4, Revision 4.11, "Emergency Conditions," was revised to reflect the change that the Recovery Manager (new title, Emergency Director) is now a minimum staff position.

Section 5, Revision 4.11, "Organizational Control of Emergencies," was revised to clarify functional titles, ERO (emergency response organization) responsibilities, and emergency actions. Reformatted On-shift and ERO staffing tables to be aligned with NUREG-0654, Table B-1, in new Section 5.18, "Table B-1, DCPP On-Shift and ERO Staffing."

Section 6, Revision 4.09, "Emergency Measures," was revised to reflect the new ERO titles and associated responsibilities. Removed references to allow Recovery Manager (new title, Emergency Director (ED)) up to 2.5 hours to respond since the ED is now a minimum staff position.

Section 7, Revision 4.13, "Emergency Facilities and Equipment," was revised to reflect the new ERO titles and associated responsibilities.

Section 8, Revision 4.08, "Maintaining Emergency Preparedness," was revised to reflect the new ERO titles and associated responsibilities.

Section 9, Revision 4.02, "Recovery," Section 9.1, "Recovery Phase," was revised to reflect the new ERO titles and associated responsibilities.

Section 10, Revision 4.01, "References," was not revised, but is included with this submittal for completeness.

Appendix A, Revision 4.04, “Procedures,” was not revised, but is included with this submittal for completeness.

Appendix B, Revision 4.00, “Offsite Agency Support Documents,” was not revised, but is included with this submittal for completeness.

Appendix C, Revision 4.00, “Non-Applicable NUREG-0654 Standards,” was not revised, but is included with this submittal for completeness.

Appendix D, Revision 4.01, “Table of Contents,” was not revised, but is included in this submittal for completeness.

Appendix D, Revision 4.01, “Introduction,”was revised to reflect the new ERO titles.

Appendix D, Category R, Revision 4.01, "Abnormal Rad Levels/Radiological Effluent," "Basis,” was revised to reflect the new ERO titles.

Appendix D, Category C, Revision 4.01, "Cold Shutdown/Refueling, System Malfunction," “Basis,” was revised to reflect the new ERO titles.

Appendix D, Category H, Revision 4.01, "Hazards," “Basis,” was revised to reflect the new ERO titles.

Appendix D, Category S, Revision 4.01, "System Malfunction," was revised to reflect the new ERO titles.

Appendix D, Category F, Revision 4.01, "Fission Product Barriers," was revised to reflect the new ERO titles.

Appendix D, Category E, Revision 4.00, “Independent Spent Fuel Storage Installation (ISFSI),” was not revised but is included for completeness.

Appendix D, Bases, Revision 4.01, “Fission Product Barrier Loss/Potential Loss Matrix and Bases,” was revised to reflect the new ERO titles.

The Diablo Canyon Emergency Plan – Applicable Revision/Change by Section

• Section 1 – Definitions & Acronyms	Rev. 4, Change 05
• Section 2 - Scope and Applicability	Rev. 4, Change 03
• Section 3 - Summary of Emergency Plan	Rev. 4, Change 00
• Section 4 - Emergency Conditions	Rev. 4, Change 11
• Section 5 - Organizational Control of Emergencies	Rev. 4, Change 11
• Section 6 - Emergency Measures	Rev. 4, Change 09
• Section 7 - Emergency Facilities and Equipment	Rev. 4, Change 13
• Section 8 - Maintaining Emergency Preparedness	Rev. 4, Change 08
• Section 9 – Recovery	Rev. 4, Change 02
• Section 10 – References	Rev. 4, Change 01
• Appendix A – Procedures	Rev. 4, Change 04
• Appendix B – Offsite Agency Support Documents	Rev. 4, Change 00
• Appendix C – Non-Applicable NUREG-0654 Standards	Rev. 4, Change 00
• Appendix D – Emergency Action Level Technical Basis Manual	Rev. 4, Change 01

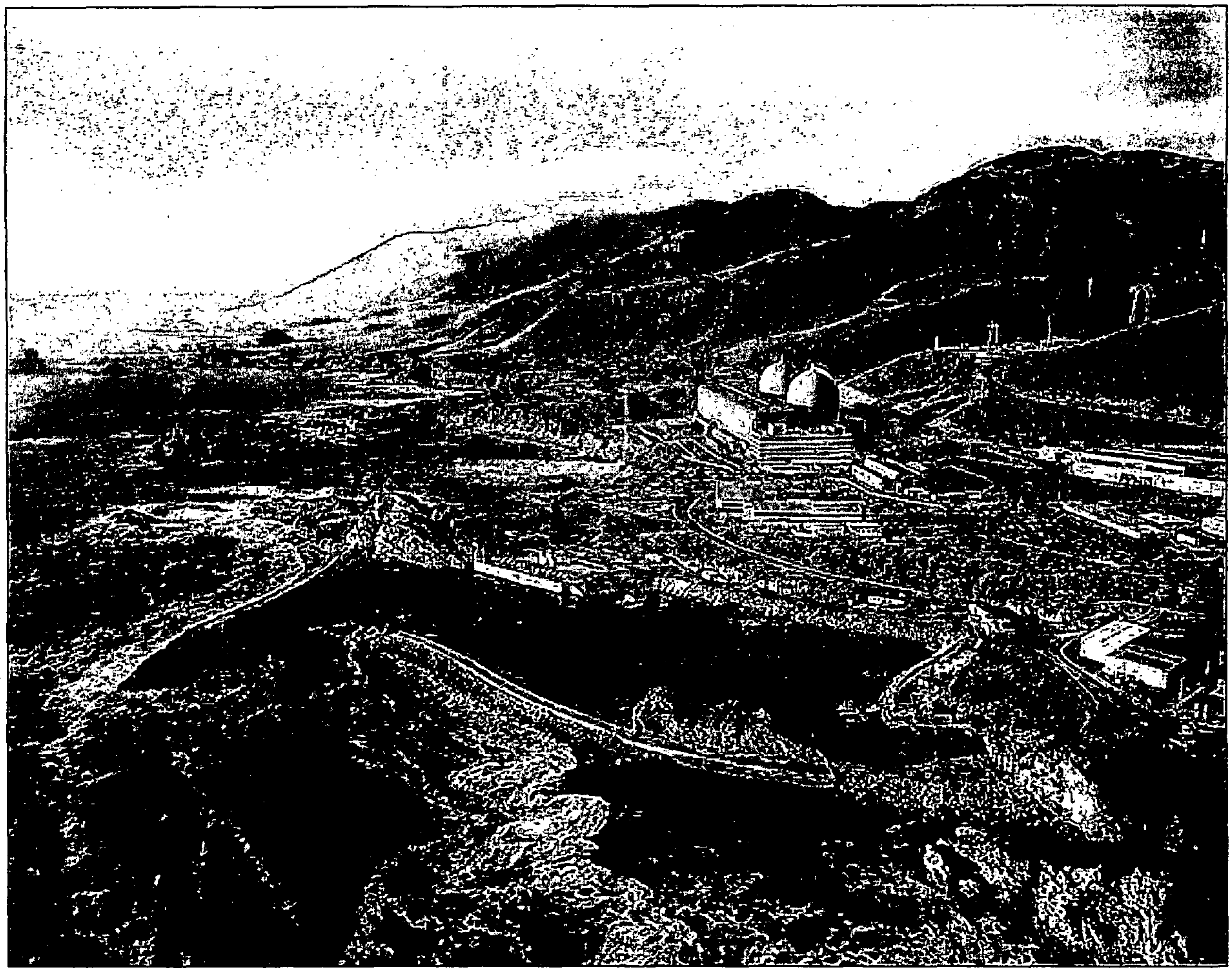
Pacific Gas & Electric Company
Nuclear Power Generation

Diablo Canyon Power Plant

Revision 4

Emergency Plan

10-3-01
Effective Date



O. H. Oatley
Vice President, Diablo Canyon Operations

9/26/01
Date

The Diablo Canyon Emergency Plan is organized as shown below.

1. DEFINITIONS AND ACRONYMS

1.1 DEFINITIONS

This section defines terms used in the Diablo Canyon Power Plant Emergency Plan.

Annual
A calendar year beginning January 1st and ending December 31st.

Assessment Actions
Those actions taken during or after an accident to obtain and process information necessary for decisions implementing specific emergency measures.

Basic Emergency Planning Zone
The Diablo Canyon Basic Emergency Planning Zone is the area surrounding Diablo Canyon Power Plant defined by the State of California that is divided into twelve smaller Protective Action Zones (PAZs).

Collective Dose
The sum of the individual doses received in a given period of time by a specified population from exposure to a specified source of ionizing radiation.

Committed Dose Equivalent ($H_{t,50}$)
The dose equivalent to organs or tissues of reference (T) that will be received from an intake of radioactive material by an individual during the 50-year period following the intake.

Committed Effective Dose Equivalent ($H_{e,50}$)
The sum of the products of the weighting factors applicable to each of the body organs or tissues that are irradiated and the committed dose equivalent to these organs or tissues ($H_{E,50} = \Sigma W_T H_{T,50}$).

Company
Refers to Pacific Gas and Electric Company.

Controlled Area
An area outside of a restricted area but inside the site boundary, access to which can be limited by the licensee for any reason.

Corrective Actions
Those emergency measures taken to ameliorate or terminate an emergency situation at or near the source of the problem in order to prevent an uncontrolled release of radioactive material or to reduce the magnitude of a release (e.g., shutting down equipment, fire fighting, repair, and damage control).

Declared Pregnant Woman
A woman who has voluntarily informed her employer, in writing, of her pregnancy and the estimated date of conception.

Deep Dose Equivalent (H_d)
Which applies to external whole-body exposure, is the dose equivalent at a tissue depth of 1 cm (1000 mg/cm^2).

Derived Air Concentration (DAC)
The concentration of a given radionuclide in air which, if breathed by the referenced man for a working year of 2,000 hours under conditions of light work (inhalation rate 1.2 cubic meters of air per hour), results in an intake of one ALI (Annual Limit of Intake). DAC values are given in Table 1, Column 3 of Appendix B to §§ 20.1001 - 20.2401.

Dose Conversion Factor
Any factor that is used to change an environmental measurement to dose in the units of concern.

Early Phase
The period at the beginning of a nuclear power plant incident when immediate decisions for effective use of protective actions are required, and must be based primarily on predictions of radiological conditions in the environment. This phase may last from hours to days. For the purpose of dose projection, it is assumed to last for four days.

Emergency Action Levels (EAL)
Specific radiation levels associated with airborne, waterborne, or surface-deposited concentrations of radioactive materials; or specific instrument indications (including their rates of change) that may be used as thresholds for initiating a particular emergency classification level which then requires initiating a notification procedure, or initiating a particular protective action.

Emergency Classification Levels (ECL)
The four levels of nuclear power plant emergencies: Unusual Event, Alert, Site Area Emergency, and General Emergency.

Emergency Planning Zone (EPZ)
A nominal ten-mile radius around the plant which potentially could be in the plume exposure pathway. (Established by federal criteria, 10 CFR 50.33.)

Emergency Worker Dose
The dose received by a DCPP employee under emergency conditions. Emergency worker dose does not include public or occupational dose.

Exclusion Area Boundary
An exclusion area of such size that an individual located at any point on its boundary for two hours immediately following onset of the postulated fission product release would not receive a total radiation dose to the whole body in excess of 25 rem or a total radiation dose in excess of 300 rem to the thyroid from iodine exposure.

Evacuation
The urgent removal of people from an area to avoid or reduce high-level, short-term exposure, usually from the plume or from deposited activity. Evacuation may be a preemptive action taken in response to a facility condition rather than an actual release.

Extended Emergency Planning Zone
The San Luis Obispo County Nuclear Power Plant Emergency Response Plan area which coincides with the Public Education Zone.

Independent Spent Fuel Storage Installation (ISFSI)
The facility shown on ISFSI Final Safety Analysis Report (FSAR) Figure 4.1-1 that is used for dry storage of spent fuel.

Intermediate Phase
The period beginning after the incident source and releases have been brought under control and reliable environmental measurements are available for use as a basis for decisions on additional protective actions and extending until these protective actions are terminated. This phase may overlap the early and late phases and may last from weeks to many months. For the purpose of dose projection, it is assumed to last for one year.

Late Phase
The period beginning when recovery action designed to reduce radiation levels in the environment to permanently acceptable levels are commenced, and ending when all recovery actions have been completed. This period may extend from months to years (also referred to as the recovery phase).

Low Population Zone (Lpz)
The area immediately surrounding the exclusion area which contains residents, the total number and density of which are such that there is a reasonable probability appropriate protective measures could be taken in their behalf in the event of a serious accident (10 CFR 100.3). For Diablo Canyon Power Plant this is an area encompassed by a radius of 10,000 meters (6.2 statute miles).

Member of the Public
An individual in a controlled or unrestricted area. However, an individual is not a member of the public during any period in which the individual receives an occupational dose.

Non-Essential Personnel
DCPP personnel that do not have assigned emergency response duties or are not required for maintaining the safe operation of the plant.

Owner Controlled Area (OCA)
The land area(s) adjacent to the site boundary that are owned and controlled by the licensee, whereby access can be limited by the licensee for any reason. Generally described, the DCPP OCA is the area between the Port San Luis gate and security gate A, bounded by the eastern hills directly adjacent to the site access road and the northern evacuation route, and bounded to the west by the Pacific Ocean.

PF_i
Protection factor for isotope is for the least effective respiratory equipment employed by persons in an area.

Population at Risk
Those persons for whom protective actions are being or would be taken. (Reference 5.)

Projected Dose
Future dose calculated for a specified time period on the basis of estimated or measured initial concentrations of radionuclides or exposure rates and in the absence of protective actions.

Protected Area
A security area encompassed by physical barriers and to which access is controlled (ANSI N 18.17-1973). At Diablo Canyon Power Plant, this is the secured areas inside the double fence.

Protective Action Recommendations (PARs)
Those recommended emergency measures taken before or after an uncontrolled release of radioactive material has occurred to prevent or minimize radiological exposures to persons likely to occur if the actions were not taken.

Protective Action Guide (PAG)
The projected dose to an individual, based on reference man, from an accidental release of radioactive material at which a specific protective action to reduce or avoid that dose is warranted.

Public Dose
The dose received by a member of the public from exposure to ionizing radiation and to radioactive material released by licensee, or to another source of radiation either within a licensee's controlled area or in unrestricted areas. It does not include occupational dose or doses received from background radiation, as a patient from medical practices, or from voluntary participation in medical research programs.

Public Education Zone
The State of California Nuclear Power Plant Emergency Response Plan area enclosed by a boundary beyond the Basic EPZ to include the area where public education is required, but planning for public protective actions is not required.

Radiologically Controlled Area (RCA)
An area which is established for the protection of personnel from radiological hazards.

Recovery
The process of reducing radiation exposure rates and concentrations of radioactive material in the environment to levels acceptable for unconditional occupancy or use.

Recovery Actions
Those actions taken after the emergency to restore the plant as nearly as possible to its pre-emergency condition.

Reentry
Temporary entry into a restricted zone under controlled conditions.

Restricted Area
Synonymous with DCPP Protected Area.

Restricted Zone
An area with controlled access from which the population has been relocated.

Return
The reoccupation of areas cleared for unrestricted residence or use.

Sheltering
The use of structure for radiation protection from an airborne plume and/or deposited radioactive materials.

Shallow-Dose Equivalent (H_s)
Which applies to the external exposure of the skin or an extremity, is taken as the dose equivalent at a tissue depth of 0.007 centimeter (7 mg/cm^2) averaged over an area of 1 square centimeter.

Site Boundary
The site boundary and the location of principal structures are shown in Figure 1.1-2 of the DCPP Final Safety Analysis Report Update. A portion of the site is bounded by the Pacific Ocean.

Total Effective Dose Equivalent (TEDE)
The sum of the deep dose equivalent (for external exposures) and the committed effective dose equivalent (for internal exposures).

Turnover
The exchange of the appropriate amount of information to allow another person or team to assume the responsibility for providing a function.

Unrestricted Area
An area, access to which is neither limited nor controlled by the licensee.

Whole Body
For purposes of external exposure, head, trunk (including male gonads), arms above the elbow, or legs above the knee.

Whole Body Dose
Dose resulting from uniform exposure of the entire body to either internal or external sources of radiation.

1.2 ACRONYMS

This section provides acronyms and symbols used in the Diablo Canyon Power Plant Emergency Plan.

ASW	Auxiliary Salt Water
BEPZ	Basic Emergency Planning Zone
CCW	Component Cooling Water
CFR	Code of Federal Regulations
Ci	Curie
CRT	Cathode Ray Tube
CVCS	Chemical and Volume Control System
DCPP	Diablo Canyon Power Plant
DHS	Department of Health Services (California State)
DNB	Departure from Nucleate Boiling
EAL	Emergency Action Level
EARS	Emergency Assessment and Response System
ECL	Emergency Classification Level
ECCS	Emergency Core Cooling System
EOC	Emergency Operations Center (County)
EOF	Emergency Operations Facility (Offsite)
EPZ	Emergency Planning Zone
ERDS	Emergency Response Data System
ERFDS	Emergency Response Facility Data System
ERP	Emergency Response Plan
ESF	Engineered Safety Features
FEMA	Federal Emergency Management Agency
FSAR	Final Safety Analysis Report
FTS	Federal Telecommunications System
GE	General Emergency
GM	Geiger Mueller Radiation Monitor
I	Iodine
IPZ	Ingestion Planning Zone
ISFSI	Independent Spent Fuel Storage Installation
JIC	Joint Information Center (A.K.A., JMC)
JMC	Joint Media Center
KI	Potassium Iodide

LOCA	Loss of Coolant Accident
LPZ	Low Population Zone
m	meter
NRC	U.S. Nuclear Regulatory Commission
UE	Unusual Event
OCA	Owner Controlled Area
OEL	Offsite Emergency Laboratory
OES	Office of Emergency Services, SLO County or CA State
OSC	Operational Support Center (Onsite)
PAG	Protective Action Guideline
PAR	Protective Action Recommendation (Utility)
PAZ	Protective Action Zone
PAD	Protective Action Decision (Offsite Agency)
PEZ	Public Education Zone
PG&E EOC	Corporate Emergency Operations Center, Conference Rm. A & B, 245 Market Street, San Francisco
PIC	Pressurized Ion Chamber
RCA	Radiological Controlled Area
RCS	Reactor Coolant System
SAE	Site Area Emergency
SCADA	Supervisory Control and Data Acquisition (Siren Query)
SPDS	Safety Parameter Display System
SRO	Senior Reactor Operator
SSPS	Solid State Protection System
TLD	Thermo-luminescent Dosimeter
TSC	Technical Support Center
TSC-CC	Technical Support Center Computation Center
UDAC	Unified Dose Assessment Center
μCi/cc	microcuries per cubic centimeter

2. SCOPE AND APPLICABILITY

There are several documents which, when taken together, describe the manner in which government agencies and Pacific Gas and Electric Company will respond to an emergency at Diablo Canyon Power Plant. These documents include the State of California and San Luis Obispo County / Cities Nuclear Power Plant Emergency Response Plans and the Pacific Gas and Electric Diablo Canyon Emergency Plan.

The Emergency Plan for Diablo Canyon Power Plant is an integral part of the licensing documentation developed to meet the requirements for an operating license. The Emergency Plan is included in Volume 11 of the Plant Manual. The Diablo Canyon Power Plant Emergency Plan, per se, is a summary document which describes the general manner in which Pacific Gas and Electric Company corporate and plant staff personnel will respond to an emergency situation at the site. Their responses are detailed in the Emergency Plan implementing procedures.

Revision 1 was issued in September 1977. Revision 2 was promulgated in February 1980 as a result of major changes in content and format to meet the requirements of NRC Regulatory Guide 1.101, "Emergency Planning for Nuclear Power Plants."

New emergency planning regulations went into effect in late 1980 which were codified in Appendix E of Title 10, Part 50 of the Code of the Federal Regulation (10 CFR 50), effective November 3, 1980. These changes were implemented by the issuance of NUREG-0654, Revision 1 (Reference 2). Consequently, it was necessary to again revise the Emergency Plan to incorporate the requirements of NUREG-0654.

All nuclear power plant emergency response plans are written to carry out the provisions and intent of 10 CFR 50, Appendix E. This statute requires nuclear power plant licensees to prepare and submit their emergency plans together with state and local (county) emergency response plans, to the Nuclear Regulatory Commission (NRC). The Federal Emergency Management Agency (FEMA) is charged to review the state and county plans for adequacy, measured according to the standards listed in joint NRC-FEMA documentation (Reference 2). In this capacity, FEMA required state and county nuclear power plant emergency response to be revised to meet the new federal standards. The NRC is charged to assess the Diablo Canyon Power Plant Emergency Plan and it's implementation and to review the FEMA findings to determine the adequacy of the overall state of emergency preparedness as a condition for an operating license.
Diablo Canyon has an Independent Spent Fuel Storage Installation (ISFSI). The Diablo Canyon ISFSI site is located east of the plant near the raw water reservoir and the 230-kV switchyard. Thus, in addition to the requirements discussed above, the DCPP Emergency Plan meets the requirements of 10 CFR 72.32(c).

The provisions of this document and its implementing procedures carry out the policy for an emergency organization as stipulated by the chief nuclear officer, the Program Directive, "Emergency Preparedness," OM10. The Emergency Plan is intended to serve several purposes, including:

1) Establish the emergency duties and responsibilities of the various members of the plant and corporate support staff.

2) Inform all affected agencies and organizations of the interfaces which have been established between the plant staff and participating company and noncompany support groups.

3) Provide a convenient means for gathering together, by way of appendices to the Plan, the plans of the various participating offsite agencies such that plant staff personnel are made aware of the basic responsibilities and capabilities of these agencies.

4) Provide an overview of the facilities, equipment, and procedures utilized in an emergency to inform and assist those offsite agencies who must coordinate their planning and response activities with those of the plant and corporate staff.

5) Fulfill licensing requirements of the Nuclear Regulatory Commission.

This plan, in conjunction with the San Luis Obispo County / Cities Nuclear Power Plant Emergency Response Plan, provides for planning in the geographical area bounded by possible plume exposure pathways. According to federal guidelines, this is, as a minimum, an area covered by a radius of approximately 10 miles from the power plant. With this federally defined radius as minimum, the federal government has delegated to the states the responsibility to define the Emergency Planning Zone unique to each nuclear power plant site within their individual state boundaries. As a result, in the State of California for the San Luis Obispo area, the State Basic Emergency Planning Zone is used for emergency response planning.

3. SUMMARY OF EMERGENCY PLAN

There are several documents which, when taken together, describe the manner in which The Diablo Canyon Power Plant Emergency Plan provides guidance to company personnel in emergency classification, appropriate response actions, and outside agency relationship and response. Pacific Gas and Electric Company will be referred to in this text as "The Company".

The Plan and its Appendices are included as one volume of the Diablo Canyon Plant Manual. Other volumes of the Plant Manual contain the implementing procedures which provide detailed instructions to plant staff personnel for responding to various postulated emergency situations and for performing the various analytical assessments which are required to support the response.

The identification and classification of an emergency situation is described in Section 4, EMERGENCY CONDITIONS. Standard NRC emergency classifications are used to facilitate identification and non-technical assessment of emergency conditions. The use of emergency classifications assist various affected parties in understanding the potential severity and in initiating a pre-planned response during the early stages of an emergency situation.

The normal plant operating and emergency organizations are described in Section 5, ORGANIZATIONAL CONTROL OF EMERGENCIES. This Section also summarizes the relationship between the plant, corporate staff, and participating offsite emergency authorities, and defines responsibilities of individuals and organizations involved in emergency response organizations.

In Section 6, EMERGENCY MEASURES, specific emergency response measures are outlined. Also addressed are the activation and actions of the emergency organization and procedures for handling onsite evacuation as well as personnel injury.

Emergency response facilities, communication systems, radiological dose assessment, protective measures and medical facilities are described in Section 7, EMERGENCY FACILITIES AND EQUIPMENT.

The means to ensure emergency plans continue to be effective are identified in Section 8, MAINTAINING EMERGENCY PREPAREDNESS.

Section 9, RECOVERY, describes general, long term arrangements for restoring the plant to a safe status.

Section 10, REFERENCES, provides a listing of material referred to in other sections.

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4. EMERGENCY CONDITIONS

4.1 EMERGENCY CLASSIFICATIONS

Events having actual or potential emergency implications are placed into one of four emergency classifications in accordance with established criteria. The classifications are:

1. UNUSUAL EVENT (UE)
2. ALERT
3. SITE AREA EMERGENCY (SAE)
4. GENERAL EMERGENCY (GE)

The principal purposes of the standardized classification system are threefold:

1. To assure timely notification of particular events which could lead to significant consequences given subsequent operator error or equipment failure, or which might be indicative of more serious conditions not at the time fully appreciated.
2. To provide a non-technical description of the actual or likely implications of the event which can be easily communicated to and understood by various affected parties during the early stages of the event.
3. To provide a vehicle for setting in motion prearranged emergency response activities by all affected parties.

All the initiating conditions of NEI 99-01, "Methodology for Development of Emergency Action Levels", and the postulated accidents in the Diablo Canyon Final Safety Analysis Report (FSAR) have been classified in accordance with the criteria discussed in Sections 4.1.1, 4.1.2, 4.1.3, and 4.1.4 and are listed in Appendix D and Section 4.3, Table 4.1-1. The Emergency Plan Implementing Procedures (EPIPs) are listed in Appendix A.

Appendix D provides specific instruments, parameters, or equipment status used at Diablo Canyon Power Plant to classify events specified in NEI 99-01. Many of the "UNUSUAL EVENTS" included in NEI 99-01 also are reportable under the reporting requirements of 10 CFR 50.72. Diablo Canyon Administrative Procedures contain the detailed criteria and instructions necessary to evaluate and report operating events not classified as emergencies to the NRC.

Table 4.1-1 lists the postulated transients in the FSAR (Reference 3) and classifies them in accordance with the emergency classification criteria discussed in Sections 4.1.1, 4.1.2, 4.1.3, and 4.1.4. Many of the transients analyzed in Chapter 15 of the FSAR do not result in degraded plant safety and therefore need not be included in the emergency classification. Those transients, which do indicate a potential degradation in the level of plant safety, are either included in the emergency classification or the effect of the transient (e.g., fuel failure, abnormal primary system temperature and/or pressure, abnormal primary system leakage, unusual radiation levels) which causes it to degrade plant safety is included in the Table 4.1-1 listing.

Table 4.1-2 lists the credible hypothetical ISFSI accidents in the Diablo Canyon ISFSI FSAR and classifies them in accordance with the emergency classification criteria discussed in Section 4.1.1. Many of the accidents analyzed in Chapter 8 of the Diablo Canyon ISFSI FSAR do not result in degraded ISFSI safety and therefore need not be included in the emergency classification. Those accidents, which do indicate a potential hypothetical degradation in the level of ISFSI safety, are included in the Table 4.1-2 listing.

The bases for notifying off-site response organizations are established by the identification of Emergency Action Levels. These in turn determine which one of the classifications is an appropriate assessment of the severity of the accident. The classification is then the non-technical assessment transmitted to off-site government agencies which provides the vehicle mentioned above for setting emergency response activities in motion.

4.1.1 UNUSUAL EVENT (UE)

1) Description

UE, as used in the context of the Emergency Plan, generally characterizes off-normal plant conditions that may not in themselves be particularly significant from an emergency preparedness standpoint, but could reasonably indicate a potential degradation of the level of safety of the plant or indicate a security threat to facility protection if proper action is not taken or if circumstances beyond the control of the operating staff render the situation more serious from a safety standpoint.

2) Release Potential and Significance

No releases of radioactive material requiring off-site response or monitoring are expected unless further degradation of safety systems occur.

3) Purpose

The purpose of an off-site UE is to:

a) Assure the first step in any response later found necessary has been carried out.

b) Bring the operating staff to a state of readiness.

c) Provide systematic handling of UE information and decision making.

4) General Actions of Plant Staff

a) Promptly inform local, state, and off-site company support agencies of the nature of the UNUSUAL EVENT.

b) Augment on-shift resources as required.

c) Assess and respond.

d) Close out with verbal summary to off-site support agencies; followed by a written summary within 24 hours (or the next working day).

OR

e) Escalate to a more severe class.

5) General Actions of Local and State Off-Site Authorities

a) Provide medical, fire or security assistance if requested.

b) Standby until verbal close-out.

OR

c) Take appropriate actions for event.

4.1.2 ALERT

1) Description

Events are in progress, or have occurred, which involve an actual or potentially 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 intentional malicious dedicated efforts of a hostile action. At the ALERT action level, small releases of radioactivity may occur (greater than Technical Specification limits for normal operation, but only a small fraction of the EPA Protective Action Guideline (PAG) exposure levels at the site boundary). It is the lowest level where emergency off-site response may be anticipated. However, for most of the ALERT events, the plant would be quickly brought to a safe condition and releases, if any, would be minimal.

2) Release Potential and Significance

Such a release would ordinarily not require near-term protective measures, (such as evacuation) beyond the site boundary, although some action within the LPZ might be taken as a precautionary measure if a release near the Technical Specification maximum was actually expected and the potential existed for a release of extended duration or for escalation to a more severe class. The need for near-term protective action beyond the boundary of the LPZ would be unnecessary.

3) Purpose

The purpose of the ALERT classification is to:

a) Assure emergency personnel are readily available to respond if the situation becomes more serious, or to perform confirmatory radiation monitoring if required.

b) Provide off-site authorities with current status information.

4) General Actions of Plant Staff

   a) Promptly inform local, state, and off-site company support agencies of the nature and status of the ALERT condition.

   b) Augment on shift resources by activating the Technical Support Center (TSC), Operational Support Center (OSC), the Emergency Operations Facility (EOF) and Joint Information Center (JIC).

   c) Provide a dedicated individual (Advisor to County) for plant status up-dates to off-site authorities.

   d) PG&E News Department will provide periodic media briefings (joint with off-site authorities).

   e) Dispatch on-site and off-site monitoring teams and associated communications as required.

   f) Provide periodic plant status updates to off-site authorities (approximately every 30 minutes).

   g) Provide periodic meteorological assessments to off-site authorities and, if any releases are occurring, dose estimates for actual releases.

   h) Close out or recommend reduction in emergency classification by verbal summary to off-site authorities, followed by written summary within 8 hours.

   OR

   i) Escalate to a more severe class.

5) General Actions of Local and State Off-Site Authorities

   a) Provide fire or security assistance if requested.

   b) Augment resources by activating the County Emergency Operations Center (EOC).

   c) Alert to standby status key emergency personnel including monitoring teams and associated communications.

   d) Provide confirmatory off-site radiation monitoring and ingestion pathway dose projections if actual releases substantially exceed Technical Specification limits.

   e) Maintain ALERT status until verbal close-out or reduction in emergency classification.

   OR

   f) Take appropriate actions for event.

4.1.3 SITE AREA EMERGENCY (SAE)

1) Description

The SITE AREA EMERGENCY classification reflects events which are in progress or have occurred involving actual or likely major failures of plant functions needed for protection of the public or security events that result in intentional damage or malicious acts; (1) toward site personnel or equipment that could lead to the likely failure of or; (2) prevents effective access to equipment needed for the protection of the public, but a core meltdown situation is not indicated based on current information. Any releases are not expected to exceed EPA Protective Action Guides beyond the site boundary. However, because the possible release associated with a SITE AREA EMERGENCY is significant, care must be taken in alerting off-site authorities to distinguish whether the release is merely potential, likely, or actually occurring. Response of off-site authorities will be guided initially by this determination.

2) Release Potential and Significance

Events in which projected dose assessment results of ≥100 mRem TEDE or ≥500 mRem Thyroid CDE at or beyond the site boundary were indicated, for actual or expected releases, would lead to a SITE AREA EMERGENCY classification.

Such a release would almost certainly require that protective measures be taken in the vicinity of the site and may require some precautionary measures to be taken in the downwind LPZ sectors. The appropriate near-term response for such an occurrence is to make an assessment of conditions as they actually exist and take action based on this assessment, as discussed below.

3) Purpose

The purpose of the SITE AREA EMERGENCY classification is to:

(1) Assure that all response centers are activated.

(2) Assure that monitoring teams are dispatched.

(3) Assure that personnel required for evacuation of near-site areas are at duty stations if the situation becomes more serious.

(4) Provide current information for off-site authorities and the public.

4) General Actions of Plant Staff

a) Promptly inform local, state, and off-site company support agencies of the nature of the SITE AREA EMERGENCY condition and its status.

b) Augment resources by activating the TSC, OSC, EOF, and JIC if not activated earlier. Resources may be provided to corporate emergency response facilities, if needed.

c) Provide a dedicated individual (Advisor to the County) for plant status updates to off-site authorities.

d) PG&E News Department will provide periodic media briefings (joint with off-site authorities).

e) Dispatch on-site and off-site monitoring teams and associated communications.

f) Evacuate non-essential people from the Site.

g) Make senior technical and management staff on-site available for consultation with NRC and state authorities on a periodic basis.

h) Provide meteorological and dose estimates to off-site authorities for actual releases.

i) Provide release and dose projections based on available plant condition information and foreseeable contingencies.

j) Close-out or recommend reduction in emergency class by briefing off-site authorities at County Emergency Operations Center (EOC) by phone, followed by written summary within 8 hours.

OR

k) Escalate to GENERAL EMERGENCY class.

5) General Actions of Local and State Off-Site Authorities

a) Provide assistance if possible.

b) Activate immediate public notification of emergency status and provide public periodic updates.

c) Augment resources by activating the County EOC if not activated earlier.

d) Dispatch key emergency personnel including monitoring teams and associated communications.

e) Alert to standby status other emergency personnel (e.g., those needed for evacuation) and dispatch personnel to near-site duty stations.

f) Provide off-site monitoring results to the Company and others. Jointly assess monitoring results with the Company.

g) Continuously assess information from the Company and off-site monitoring regarding changes to protective actions already initiated for public and mobilizing evacuation resources.

h) Assess need for action to prevent or mitigate ingestion pathway exposure.

i) Provide joint media briefings with the Company.

j) Maintain site emergency status until close-out or reduction of emergency class.

OR

k) Take appropriate actions for event.

4.1.4 GENERAL EMERGENCY

1) Description

The GENERAL EMERGENCY classification reflects accident situations involving actual or imminent substantial core degradation or melting with potential loss of containment integrity or security events that result 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.

2) Release Potential and Significance

The GENERAL EMERGENCY classification includes any releases which exceed 1 Rem TEDE and/or 5 Rem thyroid CEDE at or beyond the site boundary.

A condition where an actual release exceeds the levels for a GENERAL EMERGENCY would almost certainly require some protective action on-site and in the downwind areas of the LPZ. Lack of available shelter for many of the persons in this area (agricultural workers, visitors to Montana de Oro State Park) makes precautionary evacuation of the LPZ the appropriate response following a declaration of GENERAL EMERGENCY. For areas beyond the LPZ, directing transients to return to their homes outside the BEPZ, sheltering of permanent residents, and deployment of law enforcement agencies in preparation for possible evacuation are the appropriate responses while actual conditions are assessed.

3) Purpose

The purpose of the GENERAL EMERGENCY classification is to:

a) Initiate predetermined protective actions for the public.
b) Provide continuous assessment of information from the Company and off-site measurements.
c) Initiate additional measures as indicated by event releases or potential releases.
d) Provide current information for off-site authorities and the public.

4) General Actions of Plant Staff

a) Promptly inform local, state, and off-site company support agencies of the nature of the GENERAL EMERGENCY condition and status.
b) Augment resources by activating the TSC, OSC, EOF, and JIC if not activated earlier. Resources may be provided to corporate emergency response facilities, if needed.
c) Assess and respond.
d) Evacuate nonessential people from the site.
e) Dispatch on-site and off-site monitoring teams and associated communications.
f) Provide a dedicated individual (Advisor to County) for plant status up-dates to off-site authorities.
g) The JIC is activated to provide periodic media briefings (joint with off-site authorities).
h) Make senior technical and management staff on-site available for consultation with NRC and the State on a periodic basis.
i) Provide meteorological and dose estimates to off-site authorities for actual releases.

j) Provide release and dose projections based on available plant condition information and foreseeable contingencies.

k) Close out or recommend reduction of emergency class by briefing of off-site authorities at County EOC by phone; followed by written summary within 8 hours.

5) General Actions of Local and State Off-Site Authorities

a) Provide assistance if possible.

b) Activate immediate public notification of emergency status and provide public periodic updates.

c) Evacuate the LPZ. Place other areas of the Basic Emergency Planning Zone on ALERT status and assess need to extend evacuation distance beyond the LPZ.

d) Augment resources by activating the County EOC if not activated earlier.

e) Dispatch key emergency personnel including monitoring teams and associated communications.

f) Dispatch other emergency personnel to duty stations within Basic Emergency Planning Zone and alert all others to standby status.

g) Provide off-site monitoring results to the Company and others. Jointly assess monitoring results.

h) Continuously assess information from the Company and off-site monitoring regarding changes to protective actions already initiated for the public and mobilizing evacuation resources.

i) Assess need for action to prevent or mitigate ingestion pathway exposure.

j) Provide media briefings jointly with the Company.

k) Maintain GENERAL EMERGENCY status until close-out or reduction of emergency class.

4.2 SPECTRUM OF POSTULATED ACCIDENTS

The parameter values, equipment status and initiating conditions for each of the four classifications are identified in Appendix D. A more comprehensive discussion of various accident assessment equipment and capabilities is given in Chapters 6 and 7.

4.3 DCPP AND ISFSI FSAR ANALYZED ACCIDENTS CORRELATED TO EMERGENCY CLASSIFICATIONS

Table 4.1-1

DCPP FSAR Accidents	Classification
Condition II Accidents	
1. Uncontrolled rod cluster control assembly bank withdrawal from a sub-critical condition.	UNUSUAL EVENT
2. Uncontrolled rod cluster control assembly bank withdrawal at power.	None
3. Rod cluster control assembly mis-operation.	None
4. Uncontrolled boron dilution.	UNUSUAL EVENT
5. Partial loss of forced reactor coolant flow.	None
6. Startup of an inactive reactor coolant loop.	None
7. Loss of external electrical and/or turbine trip.	None
8. Loss of normal feedwater.	None
9. Loss of off-site power to the station auxiliaries (station blackout).	UNUSUAL EVENT
10. Excessive heat removal due to feedwater system malfunctions.	None
11. Sudden feedwater temperature reduction	None
12. Excessive load increase incident.	None
13. Accidental depressurization of the Reactor Coolant system.	UNUSUAL EVENT
14. Accidental depressurization of the main steam system.	None
15. Spurious operation of the Safety Injection system.	None
Condition III Accidents	
1. Loss of reactor coolant from small ruptured pipes or for cracks in large pipes which actuates emergency core cooling system.	ALERT/SITE AREA EMERGENCY
2. Minor secondary system pipe breaks.	None
3. Inadvertent loading of a fuel assembly into an improper position.	None
4. Complete loss of forced reactor coolant flow.	None
5. Single rod cluster control assembly withdrawal at full power.	UNUSUAL EVENT/ALERT (if Fuel Failure Evident)
Condition IV Accidents	
1. Major reactor coolant system pipe ruptures (Loss of coolant accident).	SITE AREA EMERGENCY
2. Major secondary system pipe rupture.	UNUSUAL EVENT
3. Steam generator tube rupture.	ALERT/SITE AREA EMERGENCY
4. Single reactor coolant pump locked rotor.	UNUSUAL EVENT /ALERT (If Fuel Failure Evident)
5. Fuel handling accident.	UNUSUAL EVENT /ALERT
6. Rupture of control rod drive mechanism housing (Rod cluster control assembly ejection).	ALERT/SITE AREA EMERGENCY
7. Rupture of a Waste Gas Decay Tank.	UNUSUAL EVENT (if a radiological release occurs)
8. Rupture of a Liquid Holdup Tank.	UNUSUAL EVENT (if a radiological release occurs)
9. Rupture of Volume Control Tank.	UNUSUAL EVENT (if a radiological release occurs)

Table 4.1-2

DCPP ISFSI FSAR Accidents	Classification
Accidents	
1. Earthquake	UNUSUAL EVENT
2. Tornado	UNUSUAL EVENT
3. Cask drop and tip-over	UNUSUAL EVENT
4. Fire accident	UNUSUAL EVENT
5. Explosion	UNUSUAL EVENT
6. Lightning	UNUSUAL EVENT
a. Transmission line strikes	UNUSUAL EVENT
7. Transfer cask loss of shielding	None
8. Partial blockage of canister basket vent holes	UNUSUAL EVENT
9. Transmission tower collapse	UNUSUAL EVENT
10. 100 percent blockage of air inlet vents	UNUSUAL EVENT

4.4 CROSS REFERENCE TO NUREG-0654*

NUREG 0654	DCPP Emergency Plan
D.1 & D2	4.2, 4.3
E.1	4.1.1, 4.1.2, 4.1.3, 4.1.4
H.5	4.3
I.1	4.2, 4.3
J.7	4.1
J.10.m	4.1, 4.2

* EAL cross reference to NEI 99-01 contained in Appendix D.

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5. ORGANIZATIONAL CONTROL OF EMERGENCIES

In the event of an emergency, the DCPP normal plant operational organization is supplemented with an organization specifically designed to control emergency conditions. The emergency response organization may, depending upon the severity of the incident, consist of shift operating staff or a comprehensive emergency force composed of plant, corporate, and contract personnel. This section describes the normal plant organization, the onsite emergency organization and governmental agencies responsible for directing offsite emergency response activities.

5.1 NORMAL PLANT ORGANIZATION

The chief nuclear officer, has executive responsibility for Diablo Canyon Power Plant (DCPP) and the Diablo Canyon ISFSI. The site vice president reports to the chief nuclear officer and has overall responsibility for in-plant functions, ISFSI functions, and plant-corporate interfaces. The organizational structure for DCPP is shown in the DCPP FSAR Update. The organizational structure for the ISFSI is shown in the DCPP ISFSI FSAR.

5.1.1 Emergency Planning

The site vice president has responsibility for overall emergency preparedness at DCPP and Diablo Canyon ISFSI. The organizational structure is described in DCPP and ISFSI FSARs.

The emergency planning manager is assigned the responsibility for preparation and maintenance of the DCPP Emergency Plan and its implementing procedures. The emergency planning manager also provides the corporate interface with offsite emergency response organizations including San Luis Obispo County, the State of California, the Nuclear Regulatory Commission, the Federal Emergency Management Agency, the U.S. Coast Guard, the Environmental Protection Agency and other Federal, State and local agencies involved in DCPP emergency response.

The emergency planning supervisor is assigned the responsibility for onsite coordination of drills and exercises, maintenance of emergency response facilities and the Early Warning System (EWS), review and revision of the DCPP emergency plan and implementing procedures, and maintenance of the emergency response organization. This position coordinates onsite emergency preparedness issues with the Nuclear Regulatory Commission as required.

The emergency planning group provides training to the site emergency response organization, the company emergency response organization and assists, as required, with training of local emergency response organizations.

5.1.2 Balance of Normal Station Organization

See DCPP FSAR Chapter 13 for further organization details.

5.1.3 Minimum On-Shift Staffing Requirements

At DCPP, a minimum staff of nineteen personnel are required to be on shift during operation of both units and the ISFSI. This implements the criteria of NUREG 0654, Table B-1 and NUREG 0737 Supplement 1, Table 2. The normal shift complement provides staffing for the initial on-shift emergency response organization. Positions in this organization are filled as required by the emergency. Shift functional assignments are shown in Section 5.18 Table B-1.

5.2 NORMAL CORPORATE ORGANIZATION

5.2.1 Nuclear Power Generation Business Unit

The chief nuclear officer has executive responsibility for the safe operation of DCPP. An important part of this responsibility focuses on the coordination and development of both company and DCPP emergency plans and preparedness as an essential element in assuring the public health and safety.

5.2.2 Other Supporting Departments

Other company departments provide support services along normal departmental areas of responsibility. Some examples are governmental relations, news department, law and insurance.

5.3 DIABLO CANYON INTERIM EMERGENCY RESPONSE ORGANIZATION

5.3.1 Control Room

The fundamental Control Room responsibilities during an emergency are to operate the plant systems, monitor unit conditions and take corrective actions to regain control, minimize accident consequences, and terminate the incident.

5.3.2 Shift Manager

The Shift Manager is responsible for activating the DCPP emergency plan and assumes emergency response command and control until relieved by the Site Emergency Coordinator or Emergency Director. The Shift Technical Advisor functional qualification may also reside with the Shift Manager as a dual role. Once the TSC and/or EOF is activated, the Shift Manager can fill the role of Emergency Operations Coordinator.

1) Prior to being relieved, the Shift Manaer is responsible for implementing the following non-delegable activities and authorizations:
   a) Perform the initial evaluation and classification of the event.
   b) Assign plant personnel to positions in the Site Emergency Organization.
   c) Provide Protective Action Recommendations regarding evacuation, sheltering, or other emergency measures to local government agencies.
   d) Authorize the sounding of the site emergency signal.
   e) Authorize the evacuation of the plant site and specify the appropriate evacuation route.
   f) Authorize overtime and other expenses associated with establishing and maintaining an appropriate site emergency organization.
   g) Provide direction for all emergency response operations performed by Company personnel in the San Luis Obispo County area.
   h) Authorize any extraordinary emergency measures, such as the use of company emergency personnel exposure limits.
   i) Authorize release of Public Information notices.
2) The Shift Manager is responsible for ensuring the performance of the following delegable activities:
   a) Notification of:
      (1) Plant personnel
      (2) Company offsite emergency organizations
      (3) Local non-company emergency support groups
      (4) San Luis Obispo County, California Office of Emergency Services and the Nuclear Regulatory Commission
   b) Maintain liaison with offsite emergency support groups.

5.3.3 Shift Foreman

Two Shift Foremen are on-shift in the normal operating organization. One Shift Foreman is assigned to each unit. A third Shift Foreman is normally staffed on-shift for work control.

During an emergency, the Shift Manager assigns each Shift Foreman to an emergency position. One shift foreman is responsible for emergency communications.

The other shift foreman assumes the role of the interim Emergency Operations Coordinator. This position provides plant management representation in the Control Room:

1) Manages operational activities.

2) Supervises the control room management in the operational control of the plant.

3) Advise the Site Emergency Coordinator on operational matters.

4) This position may be assigned other operational duties such as radwaste management as required by the situation.

The third Shift Foreman position is normally staffed and assigned the role of an Emergency Evaluation Coordinator (EEC). The Shift Manager may assign the role of the EEC to another on-shift individual with appropriate functional qualifications that is not already filling a minimum on shift staff position.

5.3.4 Operators

These are positions in the normal plant organization that continue to perform plant operational manipulations in the emergency organization when not otherwise assigned.

5.3.5 Shift Technical Advisor

The Shift Technical Advisor (STA) is a functional requirement (not a position) for the normal operating organization. The function of the STA is to provide engineering expertise on-shift, which may be fulfilled by a licensed Senior Reactor Operator (SRO) acting in dual functional role (preferably the Shift Manager with functional qualifications, but may be an on shift individual with functional qualifications). The STA functional qualification is to ensure technical support on plant system engineering, repair, and corrective actions.

5.3.6 Security Watch Commander

This is a normal plant organization position located in the Security Building and which is responsible for onsite emergency assembly and accountability, the results of which are reported to the Site Emergency Coordinator.

5.3.7 Evacuation Coordinator

This is a temporary position that coordinates evacuation of nonessential personnel from the site if warranted by the situation. It would normally be assigned to a member of the security staff, such as the Security Advisor, as per EP G-5.

5.3.8 Evacuation Team

These are temporary positions consisting of a C&RP Technician and an assigned individual who will accompany the evacuees in the event a site evacuation is necessary. The basic functions of this team are:

1) Assure evacuees stay together and take the correct route.

2) Assist in personnel accountability at the evacuation offsite assembly area.

3) Secure radiation survey equipment and survey personnel and vehicles at the collection area and arrange for decontamination as required.

5.3.9 First-Aid and Medical

The importance of providing prompt first-aid is well recognized. First aid training is provided for plant personnel and includes voluntary certification for Cardio-pulmonary resuscitation. An onsite medical facility is available to provide basic life support capabilities at all times. During off hours, staffing is limited to an EMT qualified Fire Fighter.

5.3.10 Fire Department

This department is responsible for onsite fire fighting, rescue operations, and first-aid.

5.4 DIABLO CANYON EMERGENCY RESPONSE ORGANIZATION

5.4.1 Emergency Response Staffing

1) The Emergency Response Organization (ERO) is grouped into assigned teams for rotating on call duties and to ensure that continuous 24-hour operation can be sustained. The ERO is fully activated and staffed for emergencies classified as Alert, Site Area Emergency, or General Emergency.

2) The Emergency Response Organization minimum staffing requirements and functional responsibilities are provided in Section 5.18 Table B-1

5.5 EMERGENCY RESPONSE FACILITIES AND ORGANIZATIONS

The Diablo Canyon Power Plant emergency organization operates from onsite emergency centers -- the Control Room (CR), the Technical Support Center (TSC), and the Operational Support Center (OSC). The organization onsite is supported by offsite emergency response facilities -- the Emergency Operations Facility (EOF) and the Joint Information Center (JIC).

Section 5.19 Figure 5-1, Overall Command and Control Structure illustrates these organizations

5.6 TECHNICAL SUPPORT CENTER (TSC)

The primary function of the TSC is to provide an onsite location, independent of the Control Room, where the overall coordination of the onsite emergency response effort takes place. The TSC staff augments the Control Room in the diagnosis of plant conditions, recommends corrective actions, and coordinates plant and unit emergency activities.

More specifically, functional assignments at the TSC include (primary Emergency Response Organization assignments are indicated in parenthesis):

1) Engineering and Technical Analysis (Engineering Advisor)

   Perform systems analysis, resolve core/thermal hydraulics, electrical, instrumentation, and mechanical engineering problems, and diagnose plant conditions. Maintain liaison with offsite technical support. Provide coordination and supervision of all company support teams operating at or in the vicinity of the site.

2) Radiological Protection (Radiological Advisor)

   Assess onsite radiological conditions, recommend radiation protection measures, direct radiological surveys and decontamination actions, assist in assessment of offsite consequences as requested by the response organization at the EOF.

   Coordinate and supervise all onsite radiological surveys and investigations and provide management of the onsite radiation protection program.

3) Communicator

   Notify affected individuals and agencies of the emergency, establish initial contact and maintain communications with offsite groups, transmit instructions and information to and from the Site Emergency Coordinator.

Section 5.20 Figure 5-2, TSC Organization, illustrates the TSC organization chart. Specific TSC position functions are:

5.6.1 Site Emergency Coordinator

The Site Emergency Coordinator is responsible, under the general direction of the Emergency Director, for directing onsite emergency actions from his position at the Technical Support Center.

At DCPP the line of succession for the Site Emergency Coordinator position is the Shift Manager, department directors, and plant manager. Only higher-level utility officials may relieve the plant manager or the alternates listed above in the Site Emergency Coordinator position when the chief nuclear officer specifically orders it. In making personnel assignments in the emergency response organization, the Site Emergency Coordinator has the authority to make assignments according to availability and qualification as appropriate to carry out response activities.

1) The Site Emergency Coordinator will establish the emergency response organization in the TSC.

2) After the TSC is activated, the Site Emergency Coordinator will relieve the Shift Manager

3) Prior to the Emergency Director assuming his position at the Emergency Operation Facility, the Site Emergency Coordinator is responsible for implementing the following **non-delegable** activities and authorizations:

   a) Provide direction for all emergency response operations performed by Company personnel in the San Luis Obispo County area.

   b) Request assistance as necessary for onsite or offsite radiation monitoring from federal agencies, either through the county / state emergency response organization once established, or directly.

   c) Authorize any Protective Action Recommendations (PARs).

   d) Authorize changes in the Emergency Classification and notification of offsite authorities.

   e) Authorize any extraordinary emergency measures, such as the use of company emergency personnel exposure limits.

   f) Authorize the administration of KI to onsite personnel.

   g) Authorize the administration of KI to offsite monitoring personnel.

   h) Authorize the release of Public Information notices.

   i) Authorize overtime and other expenses associated with maintaining an appropriate Onsite Emergency Organization throughout the emergency response period.

   j) Authorize the evacuation of the plant site and specify the appropriate evacuation route.

4) The Site Emergency Coordinator is responsible for ensuring the performance of the following delegable activities:

a) Coordinate and direct all onsite activities.

b) Maintain liaison with offsite emergency support groups providing onsite assistance and support the Emergency Director in the development of a coordinated recovery action plan for onsite.

c) Recommend changes in Emergency Classification to the Emergency Director.

d) Manage TSC Operations. This includes:

- Collecting and analyzing technical information to assess plant operations.
- Providing technical counsel in support of the Control Room (CR).
- Assessing radiological release potential.
- Determining actual or potential release rates.
- On-site exposure monitoring and contamination control.
- Repairing plant components or systems as required by the emergency and / or consequences.

e) Provide management direction to the Control Room (CR) through the Operations Advisor (TSC) or Emergency Operations Coordinator (CR).

f) Provide management direction to the Operational Support Center (OSC) through the OSC Director.

g) Assign plant staff personnel to positions in the Onsite Emergency Organization as appropriate.

h) Establish and maintain onsite personnel accountability.

5.6.2 TSC Director

1) Staffing and activation of the TSC and providing assistance to the SEC in performance of their response activities.

2) Ensuring adequate staffing is available in the TSC.

3) Coordinating ongoing TSC operations

5.6.3 Radiological Advisor

1) Coordinating emergency radiological response efforts.

2) Advise the Site Emergency Coordinator on matters relating to radiological safety

3) Providing Radiological Engineering support to TSC

4) Coordinate and supervise operation and control of radiological emergency data transmission systems, review and evaluation of data from these systems, and development of data and status updates for transmission offsite.

5) Prior to UDAC activation, assist with PAR development based on Control Room dose assessment.

5.6.4 Radiological Data Processor

1) Enabling TSC radiation monitors and verifies habitability.
2) Trending radiation levels in the plant.
3) Serving as the primary contact for Site RP Coordinator in the OSC.
4) Perform Offsite Dose Calculations if EOF is unavailable.

5.6.5 Maintenance Advisor

1) Coordinates maintenance activities between TSC and OSC
2) Advises the TSC staff on plant equipment repairs assessment, site materials capabilities and resource status as required.

5.6.6 Engineering Advisor

1) Ensuring adequate Engineering Support is available to support emergency response based on plant conditions and events
2) Evaluate the safety consequences of the occurrence and advise the Site Emergency Coordinator of appropriate response actions and onsite and offsite recommended protective measures.
3) Advise the Site Emergency Coordinator on technical matters relating to nuclear and radiological safety.
4) Coordination plant technical support.
5) Advise the Site Emergency Coordinator of actions and findings of support groups.
6) Assist the Site Emergency Coordinator in determining personnel deployment for emergency support assignments.
7) Provide operation and control of emergency data transmission systems, and review and evaluate plant data.

5.6.7 Plant Process Computer (PPC) Operator

1) Assists the Reactor Engineer in reading the PPC data.and preparing release pathway information for dose assessment purposes.

5.6.8 Mechanical Engineer

1) Performing mechanical engineering assessments, trends and recommendations.

5.6.9 Electrical Engineer

1) Performing mechanical engineering assessments, trends and recommendations.

5.6.10 Reactor Engineer

1) Performing mechanical engineering assessments, trends and recommendations.

5.6.11 Operations Advisor

1) Providing general operational advice and assistance to the SEC and TSC Staff.

5.6.12 Communications Advisor

1) Oversees TSC communications staff.

2) Performs or oversees all off-site notifications when the SEC has responsibilities for classifications, notifications and PARs.

3) Maintaining continuous communications with the Control Room, transmitting and receiving technical data and procedural activities between the facilities.

4) As directed by the Site Emergency Coordinator, notify plant staff and other affected individuals and organizations of the emergency and their assignments.

5.6.13 Ops Communicator – CR

1) From the Control Room maintaining continuous communications with the TSC, transmitting and receiving technical data and procedural activities between the facilities

5.6.14 Agency / ENS Communicator

1) Maintaining communications with the NRC over the Emergency Notification System (ENS) telephone line.

2) Performing State and Local notifications if EOF is not available.

5.6.15 Security Advisor

1) Coordinating site security activities and advises the SEC on security matters.

2) Performing accountability in the TSC.

3) Coordinating evacuation or early dismissal of nonessential site personnel.

4) Overseeing the fitness for duty program in the TSC.

5.6.16 Administrative Advisor

1) Providing administrative support.

2) Directing clerical staff to requested emergency response facilities.

3) Establishing 24 hour shift schedules for all emergency response facilities.

4) Executing administrative/logistical functions as directed.

5.6.17 TSC Clerical Support

1) Providing clerical support to the TSC staff.

5.6.18 SEC Admin Assistant

1) Providing administrative support to the SEC.

2) Maintaining SEC log.

3) Assisting SEC with communications.

5.7 OPERATIONAL SUPPORT CENTER (OSC)

The functions of the OSC include:

1) Provides a location for staging trained personnel for assignment to relieve personnel and staff special emergency positions on an as-needed basis for:
   - emergency maintenance, repair and damage control
   - fire fighting, search and rescue and first aid
   - emergency sampling of plant fluids
   - a location for storage of selected emergency response equipment
   - personnel decontamination

2) Provides an office for the OSC Director who determines and recommends repair/damage control and corrective actions for plant mechanical, instrumentation, and electrical systems.

3) Section 5.21 Figure 5-3, OSC Organization, illustrates the OSC organization chart. Specific OSC position functions are:

5.7.2 OSC Director

1) Directing activities of OSC personnel.

2) Coordinating a repair plan to recover from the emergency, in cooperation with the SEC and Maintenance Advisor.

3) Coordinating the fabrication and sets up of any special equipment necessary at the direction of the SEC and Maintenance Advisor.

4) Coordinating the movement and accountability of maintenance teams.

5) Providing OSC status updates to the TSC.

5.7.3 Maintenance Coordinators (Mechanical, Instrument and Control, and Electrical)

1) Planning and coordinating resources to conduct assessment, maintenance, repair or installation of special equipment.
2) Providing team status updates to the Team Coordinator.
3) Providing technical advice to evaluation personnel in the TSC.

5.7.4 Team Coordinator

1) Coordinates plant access and ensures personnel entering a potentially hazardous plant area are informed of:
    a) Plant status.
    b) Potential hazards.
    c) Safety and radiation protection provisions.
    d) Appropriate protective equipment required.
2) Maintains accountability of personnel dispatched from the OSC.
3) Maintains the capability of communicating with personnel engaged in operations, maintenance or chemistry and radiation protection emergency response activities.
4) Ensures response teams have been briefed on plant conditions prior to dispatch.
5) Assists the Control Room and TSC/OSC in communicating with response teams.

5.7.5 Site Radiation Protection Coordinator

1) Provides personnel exposure monitoring and record keeping.
2) Directs In-plant surveys and establishment of radiation and/or contamination control area boundaries.
3) Determines radiation protection access requirements for entry to radiologically controlled areas.
4) Determines when an emergency exposure authorization is required and provides justification to the SEC or ED.
5) Keeps the Radiological Advisor, OSC Director and Team Coordinator informed of actions and findings.
6) Coordinate briefing and dispatch of personnel into affected plant areas with the OSC Access Supervisor.

5.7.6 Chemistry Coordinator

1) Directs radiological and chemical analysis of in plant samples.

2) Maintain proper records and logs.

3) Keeps the Radiological Advisor informed of actions and findings.

4) Coordinates briefing and dispatch of personnel into the plant for sampling or analysis with the Site Radiation Protection Coordinator and Team Coordinator.

5.7.7 Operations Coordinator

1) Coordinates operation's response outside the Control Room.

2) Ensure Control Room is updated on status of OSC Team assignments.

5.7.8 Onsite Field Monitoring Teams

1) Perform radiation surveys around the plant site and obtain appropriate samples for analysis.

2) Maintain communications with the FMT Coordinator for reporting monitoring results and maintaining cognizance of the emergency situation.

5.7.9 OSC Clerical Support

1) Provide support as needed to maintain OSC records and status boards.

5.7.10 Maintenance, RP, Chemistry and Operations Personnel

The following personnel provide support in planning and performing tasks in their disciplines.

1) Mechanical Maintenance personnel

2) Electrical Maintenance personnel

3) Technical Maintenance personnel

4) Radiation Protection personnel

5) Chemistry personnel

6) Operations personnel

5.8 EMERGENCY OPERATIONS FACILITY (EOF)

1) The principal functions of the EOF are to provide for strategic planning and control. The principal functions of the utility staff are:

   a) Control and Coordination

      The EOF utility staff, under the direction of the Emergency Director, manages and coordinates all PG&E technical direction and control of the integrated emergency response effort.

   b) Liaison/Interface

      The EOF provides management level interfaces with San Luis Obispo County through the Advisor to the County and subordinate managers who report to the Emergency Director.

      When activated, the EOF determines status notifications, provides Protective Action Recommendations (PARs) based on plant conditions, and Public Information releases under the ED's authority. The TSC may communicate the Notifications and PARs to the State and NRC.

   c) Assessment

      The EOF provides a centralized location, immediately available to county decision makers in the County EOC for unified dose assessments.

   d) Decision-Maker Staff Support

      The EOF provides a center for response organizations to make staff specialists available to provide data and recommendations to County decision makers for their use in taking protective actions relating to either dose assessment evaluations or predictions of degraded conditions.

   e) Company Interface

      Company emergency response activities are communicated by the Government Relations Coordinator located at the EOF.

      The Government Relations Coordinator is provided with communications via the Company telephone system and data links through the local area network (LAN)/wide area network (WAN).

Section 5.22 Figure 5-4, EOF Organization, illustrates the EOF organization chart. Specific EOF position functions are as follows:

5.8.1 Emergency Director

After arrival at the Emergency Operations Facility, the Emergency Director maintains overall command of PG&E emergency response operations during the Alert, Site Area Emergency, and General Emergency classifications. The Emergency Director provides direction and support for onsite emergency response actions to the Site Emergency Coordinator, and ensures coordination of DCPP emergency actions with those of government, and General Office support through the PG&E Emergency Operations Center (PG&E EOC). The Emergency Director also approves plant status updates and information for media release.

Emergency Director responsibilities and authorities are separated into two categories. The first category involves decision making for overall emergency response direction and protective action recommendations that may impact the general public. The second category relates to the administration and management of the utility response organization. Duties in the latter category include:

- Interface between DCPP/PG&E and federal/state/local emergency response agencies.
- Communicate plant status updates and radiological release data to NRC/FEMA, county/state EOC, PG&E EOC, and Joint Information Center (JIC) personnel.
- Provide administrative, technical, and logistical support to station emergency operations.
- Ensure continuity of emergency organization resources.
- Determine when to deactivate the Corporate Emergency Response Organization.

Emergency Director responsibility and authority for overall response direction and for recommending protective actions to offsite authorities is initially assumed by the Site Emergency Coordinator until the Emergency Director position is assumed at the EOF by the assigned corporate staff member.

EOF administrative and facility management functions are assumed by the EOF Director. The Emergency Director's primary responsibility is to place the EOF in an operational status capable of early field monitoring team direction and control, Emergency Assessment and Response System (EARS) operation, and plant data analysis.

1) The Emergency Director is responsible for implementing the following **non-delegable** activities and authorizations:
    a) Provide direction for all emergency response operations performed by Company personnel.
    b) Authorize any recommendations of the Company regarding evacuation, or other emergency measures including PARs, to non company emergency support groups.
    c) Authorize changes in the Emergency Classification and notification of offsite authorities.
    d) Authorize any extraordinary emergency measures, such as the use of company emergency personnel exposure limits.
    e) Authorize the administration of KI to offsite monitoring personnel.

2) The Emergency Director is responsible for ensuring the performance of the following delegable activities:

   a) Authorize the release of Public Information notices (or designated alternate).

   b) Request assistance as necessary for onsite or offsite radiation monitoring from federal agencies, either through the county/state emergency response organization once established, or directly.

   c) Maintain liaison with offsite emergency support groups providing onsite assistance and of a coordinated recovery action plan for onsite.

   d) Modify or re-organize the Emergency Response Organization based upon the type of accident and specific accident emergency response needs.

   e) De-escalate the emergency classification and/or establish a recovery organization.

5.8.2 EOF Director

1) Directs the activation of the utility portion of the building and appropriately establishing communications.

2) Provides administrative and management direction of the EOF staff in support of the event response.

3) Keeps the Site Emergency Coordinator informed and serve as his contact at the EOF.

4) Coordinate access control of the EOF and UDAC portion of the building as well as the JIC with the facility security and Sheriff's Watch Commander.

5.8.3 Radiological Manager

1) Develops radiological data and status information for evaluation by UDAC personnel and distribution to EOF and EOC personnel.

2) Direct the activities of offsite monitoring teams and Offsite Emergency Laboratory (OEL) in coordination with the UDAC Coordinator, maintain records, and provide findings in status reports.

3) Perform dose projections and provide radiological assessment information for the determination of Protective Action Recommendations. Advise the TSC of PARs (RM not in EOF command), or formulate PARs for RM approval.

4) Ensures proper exposure controls are taken for DCPP offsite emergency response personnel.

5.8.4 Dose Assessor

1) Performs dose projections using the dose assessment computer models or manual dose calculations.

2) Updates the Radiological Manager, Dose Assessment Coordinator and UDAC staff on the status of any plant release and of dose projection results.

5.8.5 Dose Assessment Coordinator

1) Monitors the plant telephone bridge line.

2) Directs and coordinates the DCNPP offsite dose assessment activities.

3) Coordinates the deployment of Field Monitoring Teams in support of dose assessment activities.

4) Coordinates assessment and communications with offsite agency personnel in the UDAC.

5.8.6 Field Monitoring Team Coordinator

1) Dispatches and coordinates the activities of the Field Monitoring Teams and offsite emergency lab.

2) Coordinates activities with the offsite agency environmental field teams.

3) Provide recommendations regarding establishing controlled access areas and determining the boundaries of such areas in cooperation with county personnel assigned to the monitoring team.

4) Coordinates the monitoring activities for the Congregate Care Center.

5.8.7 Field Monitoring Team Communicator

1) Establishes and maintains communications with the Field Monitoring Teams.

2) Records and reports field monitoring survey, sample and exposure information.

5.8.8 Off-Site Field Monitoring Teams

1) Performs radiation surveys and obtains environmental samples to monitor and track any releases of radioactivity outside the protected area.

2) Maintains communications with the FMT Communicator to report environmental conditions and obtains event information and instructions.

3) Coordinate monitoring activities and reporting of results with the county personnel assigned to the monitoring team.

4) Provide recommendations regarding establishing controlled access areas and determining the boundaries of such areas in cooperation with county personnel assigned to the monitoring team.

5) Assist in monitoring personnel and evaluating their exposure as required.

5.8.9 HPN Communicator

1) Monitors plant and environment radiological data.

2) Maintains radiological communications with the NRC on the HPN line.

3) Keeps the Radiological Manager informed on NRC request and reported activities.

5.8.10 Off-site Emergency Lab Analyst

1) Performs environmental sample analysis to monitor and track any releases of radioactivity outside the protected area.

2) Maintains communications with the FMT Communicator to report sample results and obtain event information and instructions.

3) Performs habitability surveys and maintains contaminations controls at the Lab.

5.8.11 Engineering Liaison

1) Activation and operation of plant data computer systems for obtaining plant data.

2) Provides technical assistance to response personnel.

3) Notify and update INPO for any event of Alert or higher.

4) Disseminate plant status updates if automated systems fail.

5.8.12 Communications Coordinator

1) Assists the Emergency Director in preparing and approving notifications.

2) Oversees the transmittal and distribution of notifications.

5.8.13 Offsite Communicator

1) Transmits, receives and documents notification information.

2) Notifies corporate offices and individuals of the event.

5.8.14 Advisor to the County

1) Assists the EOF Director with setup and activation of the EOF.

2) Keep the senior county response staff members advised of plant conditions and recommended protective actions.

3) Maintains awareness of plant conditions and emergency response activities.

4) Assists in the interface between DCPP and county EOC personnel.

5.8.15 Government Relations Coordinator

1) Monitors actions and events from the EOF to provide information to other GR Reps and JIC personnel.

2) Coordinates communication between PG&E offices in Washington, D.C., Sacramento and San Francisco.

3) Evaluates and responds to external company governmental issues and informs the ED of corporate status and decisions.

4) Communicates PG&E public information from the JIC to County EOC and EOF personnel.

5) Supports the development and review of information for use in press statements.

6) Coordinates with JIC staff for tracking and controlling rumors at the EOF.

5.8.16 EOF Security Support

1) Provides building access controls for the EOF/County EOC.

5.8.17 EOF Clerical Support

1) Provides communications, copying, distribution and other administrative support to the EOF staff.

5.8.18 ED Admin Assistant

1) Providing administrative support to the ED.

2) Maintaining ED log.

3) Assisting ED with communications.

5.8.19 UDAC Meteorologist

1) Determines current and forecast meteorological information.

5.9 JOINT INFORMATION CENTER (JIC)

The principal function of the JIC is to provide information to the general public through the media.

Section 5.23 Figure 5-5, JIC Organization, illustrates the JIC organization chart. Specific JIC position functions are:

5.9.1 Company Spokesperson

1) The Company Spokesperson consults with the Public Information Officer (PIO) and other company officials to provide the following in an emergency:

   a) Characterizes company responsibility for the event.

   b) Characterizes concern for any impacts to public.

   c) Describes company emergency response actions

2) When available and requested by the Emergency Director or PIO, the Company Spokesperson may participate in news briefings to provide additional technical expertise and representation of the company at an officer level.

3) When available participate in the Emergency Director video conference briefings and is available to assist with the development of news briefing talking points and news briefing summaries.

4) Works closely with company government and public relations staff to assist with community issues related to the emergency.

5.9.2 JIC Director

1) Acts as a point of contact for other facility managers and maintains current emergency status information throughout the event.

2) Oversees the functional organization of the JIC, including facility setup and activation, fitness-for-duty of ERO personnel, coordinating additional resources and personnel, JIC security, clerical support, parking, etc.

3) Participates in intra facility videoconference briefings to obtain approved public information.

4) Coordinates the collection of information and the development of draft news releases, news briefing talking points, and summaries by the news writer and Technical Assistants for approval by the PIO.

5) Coordinates with the PIO on any changes, corrections or additions to news releases and news briefing summaries and ensures that they are available for use in time for media briefings.

6) Coordinates with the Government Relations Coordinator for investigating unanswered questions posed by the media and resolving reported rumors.

7) Ensures the JIC is adequately staffed at all times during an emergency.

8) Communicates with the Company Spokesperson while in transit for providing status of facility readiness, where to go first, key issues for the NRC, etc.

5.9.3 Public Information Officer (PIO)

1) Coordinates news briefing presentations and the development of communication strategy.

2) Reviews public information prior to its release to the media and coordinates with the JIC Director for clarity.

3) Participates in intra facility videoconference briefings to obtain approved public information.

4) Coordinates news conferences with the Company Spokesperson, company officials, the SLO County Public Information Officer (PIO), and other participating government agencies.

5) Develops presentation strategies for the media and coordinates key messages with Company Spokespersons.

6) Identifies Company Spokesperson(s) at news briefing.

7) Communicates with the company news department regarding the status of media briefings, approved and released information, and additional staffing needs.

5.9.4 Assistant PIO

1) Serves as the JIC news spokesperson and lead moderator for the Company Spokespersons.

2) Coordinates with the SLO County PIO and other agency representatives at the JIC to assure timely and accurate news media briefings are conducted.

3) Supports pre-briefings conducted by the SLO County PIO to define speaker conduct (what to say, were to stand, order of presentation) in front of media.

4) Anchors company presentations in front of the media.

5) In conjunction with the PIO, provides or arranges additional media interviews.

6) Advises the PIO of rumors as reported by the media.

7) Ensures formality and consistency of company presentations during media briefings.

5.9.5 Technical Advisor – Health Physics

1) Assists the News Writer in verifying and writing of approved news briefing talking points, news briefing summaries and other news releases.

2) Coordinates with the EOF and JIC staff for maintaining current status of projected dose, radiological plant conditions, and field monitoring.

3) Analyzes radiological data for the JIC news personnel.

5.9.6 Technical Advisor – Operations

1) Assists the News Writer in verifying and writing of approved news briefing talking points, news briefing summaries and other news releases.

2) Provides technical assistance to assure that approved information is accurately described in general and understandable terms for the news briefing talking points and news briefing summaries.

3) Observes the Emergency Director videoconference briefings and interprets technical information for the PIO, Assistant PIO, and the News Writer.

4) Maintains current information on plant conditions and provides updates to the JIC Director, JPIM and the Company Spokesperson.

5) Reviews NRC Event Reports for information on event and reports discrepancies to the PIO.

6) Assists JIC staff to ensure technically correct information is released.

7) Presents plant information to the media as directed by the PIO.

5.9.7 News Media Liaison – Site

1) Operates the computer and projector system used to display graphics and other information during news briefings.

2) Assumes "News Media Liaison" responsibilities on an interim basis until relieved by company news personnel.

3) Records any unanswered questions by the company spokespersons during media briefings.

4) Distributes approved news documents.

5.9.8 News Media Liaison – Corporate

1) Coordinates with media personnel arriving at the JIC and provides approved background information and facility orientation.

2) Acts as an interface for the company news department for the media in between briefings.

3) Records any unanswered questions by the company spokespersons during media briefings.

4) Advises the PIO of relevant information obtained from the media.

5.9.9 News Writer

1) Collects facts for insertion into news briefing talking points and summaries as well as other news releases, as directed by JIC Director and the TA-PIO using information from video conferences and approved tailboard forms.

5.9.10 JIC Clerical Support

1) Provides administrative and distribution support as needed for the JIC Staff.

5.9.11 JIC Security

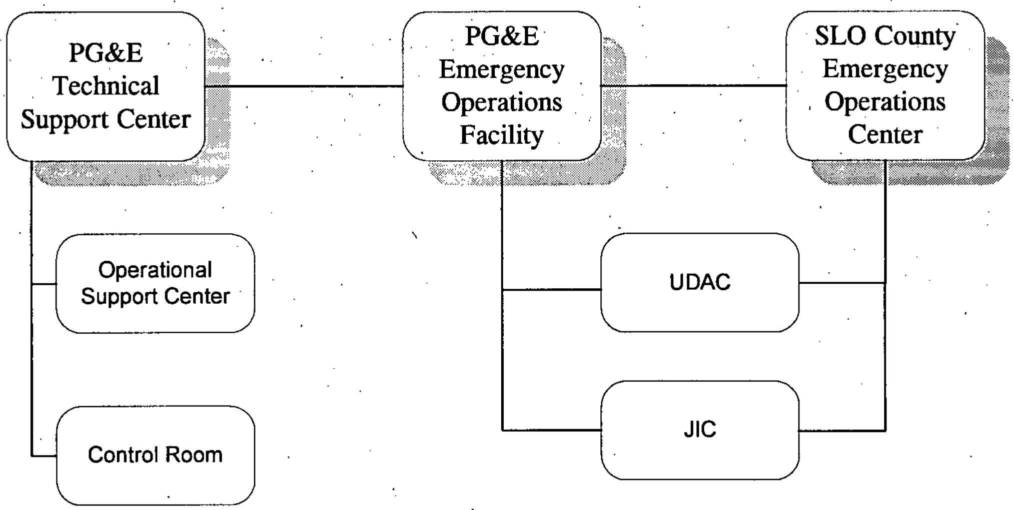
1) Provides access control for JIC areas.

5.10 FACILITY FUNCTIONAL RELATIONSHIPS

The representation of the integrated relationships between government response organizations and the licensee is shown in Figure 1 below.

The individual in charge at each center is responsible for center communications, recording important events, decisions, and actions, and assuring resource continuity (technical, administrative, and material).

FIGURE 1



5.11 COMPANY HEADQUARTERS SUPPORT

Company emergency response capabilities are identified in several areas both within and outside the nuclear generation department. Capabilities as diverse as public information, procurement of materials, contract services, construction services and meteorology have been identified. These capabilities, for the most part, are concentrated at the company headquarters. Due to the physical distance between the company headquarters and Diablo Canyon Power Plant, an extensive communications system is used to enable rapid initial corporate response as well as adequate coordination once all emergency response centers are operational and appropriately staffed. To permit the plant staff to devote the majority of its personnel and resources to mitigating and recovering from the emergency condition, the primary company role will be to assist the overall recovery effort, particularly those activities which are conducted offsite.

The company emergency response organization is structured along existing organizational lines so that the emergency organization can function in a manner similar to the normal routine company operation. To the maximum extent possible, the functional responsibilities and management functions have not changed.

To maintain the emergency preparedness of the company emergency response organization, the company emergency plan makes provisions for conducting periodic drills and exercises of simulated emergency conditions.

In an emergency action level of Alert or higher the Emergency Director will establish headquarters at the EOF and will direct all company emergency response activities. The Emergency Director will work with the Site Emergency Coordinator, as well as local, state, and federal officials, to develop a comprehensive response for the specific emergency situation. The Emergency Director will interface with the other key members of the company emergency response organization to arrange for the necessary company resources to support the short-and/or long-term emergency response effort.

5.11.1 Company Emergency Response Organization

The company emergency response organization is comprised of all company personnel outside of the plant operations staff who could provide management, technical, logistical, or liaison support services during a nuclear plant emergency. Due to demands that a particular emergency may place on the company emergency response organization, it has been developed on a functional basis along the lines of normal company duties.

5.11.2 Notification of Company Departments

It is company policy that selected departments be routinely informed of plant activities and/or notified of unusual events. Nuclear generation is routinely informed of overall plant operations and promptly notified of all emergencies.

The chief nuclear officer is the primary Emergency Director designee. If the Emergency Director cannot be contacted directly by the plant staff via company or commercial telephone systems, notification will be through the use of a paging system. The company emergency planning organization is also immediately notified of any occurrence that influences the plant's load carrying capability.

Company emergency response activities are directed by the Government Relations located at the EOF.

The Government Relations Coordinator is provided with communications via the Company telephone system and data links through the local area network (LAN)/wide area network (WAN).

5.12 NON-COMPANY ORGANIZATIONS SUPPORT

1) Other organizations involved in emergency responses include government agencies, organizations at the Federal, State and Local levels and commercial enterprises. The various non-company groups have written agreements outlining their emergency response commitments. When an exchange of information, response action or concept of operation is governed by a regulation, law or executive order, the commitments are documented by either signature papers within the body of the planning documents or by written agreements. This commitment will outline specific work to be performed or be an agreement to respond to an emergency. These agreements are maintained and verified yearly.

5.12.2 Local Services Support

Additional support from regional and local services has been contracted to further enhance PG&E's capability to handle all types of onsite emergencies. These services include medical, hospital, ambulance and fire fighting support. These services provide immediate 24-hour-a-day, on-call support.

5.12.3 Letters of Agreement

Letters of Agreement for services are maintained and reviewed annually with the organizations listed below.

1) Cambria Community Healthcare District
2) California Department of Forestry / San Luis Obispo County Fire Dept.
3) Department of California Highway Patrol
4) DOE Oak Ridge Operations Office
5) French Hospital Medical Center
6) INPO Nuclear Power Plant Agreement
7) Marian Medical Center
8) Rogers Helicopters, Inc.
9) San Luis Ambulance Service, Inc.
10) United States Coast Guard

5.12.4 Medical Consultants

A number of physicians are retained on a PG&E medical panel. Panel physicians in the San Luis Obispo area and their specialties are discussed in Section 6.

5.12.5 Hospitals

French Hospital in San Luis Obispo will handle cases of both radiological and non-radiological injuries occurring at Diablo Canyon Power Plant.

The company also has an agreement with Marian Medical Center in Santa Maria to handle injured persons who may be contaminated with radioactive materials.

5.12.6 Ambulance Service

Arrangements have been made with ambulance services in San Luis Obispo County to handle cases of personnel injury at Diablo Canyon, including those involving radioactive contamination.

5.12.7 Fire Fighting Support

The San Luis Obispo County Fire Department will be called to assist in case of fire in the plant, or the California Department of Forestry and Fire Protection (CDF) will be called upon to assist in case of fire in the surrounding grasslands, which cannot be controlled by personnel onsite. Designated members of the County Fire Department are trained in radiation protection practices. DCPP provides trained personnel to monitor potential radiation exposures to offsite fire agency personnel. The SLO County Fire Department is a participating agency in the San Luis Obispo County / Cities "Nuclear Power Plant Emergency Response Plan."

5.12.8 Air Transportation

Various air transportation services are available to the Company under emergency response conditions. Arrangements with commercial enterprises are discussed in Section 7.

5.13 NON-COMPANY INDUSTRIAL SUPPORT ORGANIZATION

5.13.1 Notification

Plant implementing procedures make provisions for early notification and information transfer to these various support organizations. The list of designated persons and emergency phone numbers are used by the plant emergency response staff to make early contact to place these organizations on alert status. This listing would include primary manufacturing suppliers, other regional nuclear utilities, and specialty nuclear utility response organizations.

The principal response organizations in the non-company-industrial organization category include:

1) Westinghouse Electric Corporation, Pittsburgh, Pa.

   The DCPP Nuclear Steam Supply Systems were supplied by Westinghouse. In its capacity as a supplier, Westinghouse can lend emergency assistance to the company on an around-the-clock basis.

"Westinghouse Electric Company Emergency Response Plan"

a) Defines the Westinghouse emergency response organization, role, scope, functions and responsibilities, and how it is activated.

b) Identifies key Westinghouse personnel available in the early phases of an emergency response.

c) Defines the primary Westinghouse interfaces with involved parties.

d) Defines the Westinghouse role in emergency news communications and their interrelationship with the company and news media.

2) Institute of Nuclear Plant Operations (INPO), Atlanta, GA.

INPO operates and maintains an Emergency Response Center with 24-hour day coverage. Through this center, INPO offers:

Assistance in locating sources of emergency personnel and equipment, analysis of the operational aspects of the incident, dissemination of information concerning the incident to other utilities and organization of industry experts who can advise on technical matters. INPO's role is detailed in, "Institute of Nuclear Power Operation INPO's Role in an Emergency."

5.14 COORDINATION WITH PARTICIPATING SAN LUIS OBISPO COUNTY AGENCIES

San Luis Obispo County authorities are assigned the lead role in coordinating offsite emergency activities. The county has prepared a plan specifically applicable to DCPP: "San Luis Obispo County/Cities Nuclear Power Plant Emergency Response Plan."

The San Luis Obispo County/Cities Plan will be activated upon notification by PG&E of an event at the Alert or higher Emergency Classification level, or by an independent remote alarm indication located in the State Warning Center in Sacramento staffed 24 hours a day by the California Office of Emergency Services. Initial PG&E notification will be received by the County Sheriff's Office Watch Commander from the plant Control Room via dedicated phone line, commercial phone, or radio.

Upon notice of a potential accident, PG&E will immediately transmit to the County all of the information required for initial assessment including protective measures recommended at that time. The response taken will be in accordance with the joint County/PG&E Emergency Classification system ensuring there are no misunderstandings of event severity.

The County Emergency Organization has overall responsibility for authorization and coordination of offsite emergency response activities, including:

1) Coordination with State Office of Emergency Services (OES) and other participating State and Local agencies.

2) Determining the appropriate response and implementing appropriate protective actions for the general public.

3) Monitoring offsite locations and controlling access to problem areas. The Company will assist with offsite environmental monitoring as well as provide monitoring for evacuated populations and offsite emergency workers.

4) Arranging for medical treatment, health, and sanitation services for the general public. Procuring and distributing stable iodine thyroid blocking tablets for institutional populations as well as emergency workers. Initial stocks of KI for Emergency Workers are provided by PG&E.

5) Public information releases regarding offsite response and protective measures to be taken. (The Company is responsible for news releases concerning onsite events and protective measures. News releases pertaining to onsite conditions will be coordinated with County public information personnel.)

Since the Sheriff's Office Watch Commander is staffed 24-hours-a-day the Control Room or TSC will make initial notification to the Sheriff's Office. The Sheriff's Office, in turn, will notify selected members of the County Emergency Organization. Since both the County EOC and the PG&E EOF are co-located, communication channels will remain open from plant site to this offsite location during County participation.

Initially, the Site Emergency Coordinator, or, once he arrives at the EOF, the Emergency Director will:

- Keep the County informed of meteorological conditions, results from pertinent surveys by company personnel, and conditions at the plant. An Advisor to the County, a Company representative, will be assigned to the County Emergency Organization to assist in interpretation of technical information received from DCPP.
- Recommend appropriate protective actions to the County.

The Site Emergency Coordinator and the Emergency Director will be in communication with the technical personnel from the various Federal and State agencies to formulate a coordinated plan of action.

County facilities are used for centralized public information dissemination. A local Company public information specialist will serve as a spokesperson for the onsite and offsite activities of the Company.

Initial news releases and statements will be coordinated by the County Public Information Manager (PIM) in the EOC and the PG&E Public Information Officer. The PG&E Public Information Officer (PIO) will assume responsibility once activated. Contact and briefings with media personnel will be held at the Emergency Response Joint Information Center off Highway 1 (near the EOF). Media briefings may be held at the JIC at the ALERT or higher level classification. Media briefings will also take place at the company headquarters.

The Emergency Director, through the PIO and the Assistant PIO will establish a schedule and format for updates to the PG&E spokesmen, San Luis Obispo County and other agencies at the County EOC. The County will promulgate jointly agreed upon position statements.

Rumor control is under the jurisdiction of the County organization. Centralized and authentic information provides the County information center with consistent, reliable source information.

5.15 COORDINATION WITH PARTICIPATING STATE OF CALIFORNIA AGENCIES

The Governor's Office of Emergency Services (OES) will be notified under the same circumstances as the County. The OES provides assistance as outlined in the State of California "Nuclear Power Plant Emergency Response Plan." The OES and the State Department of Health Services(CA DHS) are the primary state response agencies for nuclear power plant accidents. Summaries of their responsibilities under terms of the State Plan are given below.

5.15.1 Office of Emergency Services (OES)

OES has executive authority for the coordination of state resources and the responsibility of general planning. OES will:

- Alert State agencies, Federal organizations (other than NRC), volunteer organizations and adjacent Counties and States.
- Coordinate State radiological monitoring of areas, personnel, and equipment to support local County authority.
- Assess radiological situations.
- Operate the California State Warning Center (SWC), the State 24-hour communication network.
- Establish liaison with utility representatives.
- Coordinate all requests for Federal assistance.
- Prepare and coordinate public information and media releases with Local, State, and Federal agencies and the Company.
- Coordinate State mutual aid.
- Update and maintain the State of California Nuclear Power Plant Emergency Response Plan.
- Coordinate, review, and approve emergency plans by other State agencies and some local jurisdictions.
- Assist the utility, local and State response agencies in development, conduct and coordination of offsite training programs. Evaluate offsite training programs developed by other agencies.
- Assist in the development, conduct and coordination of offsite drills and periodic exercises. Evaluate and critique the drills and exercises. Recommend improvements in plans, procedures and training programs.

5.15.2 California Department of Health Services (CA DHS)

CA DHS assists OES in maintaining an adequate state of emergency preparedness and provides technical and advisory support. As specified in the State Plan, CA DHS will:

- Assist in monitoring the environment, personnel, and equipment in support of Local authority.
- Establish and direct measures to mitigate radiological impact on public health. Coordinate with Department of Agriculture and local Agricultural Commission to prevent consumption of unacceptably contaminated food and fodder.
- Establish human exposure criteria; assess public impact of radiation levels. Maintain data on population dosimetry.
- Establish maximum contamination levels for controlled areas. Direct and assist Local jurisdictions in defining contaminated areas. Establish measures to limit the spread of contamination.
- Identify laboratories capable of providing radiological support, including assay of ingestion pathway samples.
- Establish levels and guidance for protective action for the ingestion pathway.
- Direct and assist in collection of, and provide for laboratory analysis of, food and fodder samples taken by state and local authority.
- Establish policies and procedures for potential use of prophylactic substances to prevent or reduce the biological effects of radiation.
- Establish radiological safety criteria for recovery and re-occupancy of controlled areas.

5.16 COORDINATION WITH PARTICIPATING FEDERAL GOVERNMENT AGENCIES

The National Response Plan (NRP) integrates Federal domestic prevention, preparedness, response, and recovery plans into a single all-discipline, all-hazards plan. The NRP consists of a "base plan" plus multiple subject-specific annexes that expand on, and further delineate, interagency roles, responsibilities and activities in response to particular incidents.

5.16.1 Nuclear Regulatory Commission (NRC)

The NRC responds to incidents under its own statutory authorities and responsibilities in accordance with the NRP and, if applicable, as an integral part of the overall response by the Federal Government. Under the NRP's Nuclear/Radiological Incident (NRI) Annex, the NRC is the "Coordinating Agency" Federal agency that owns, has custody of, authorizes, regulates, or is otherwise deemed responsible for incidents involving facilities and/or materials licensed by the NRC. Accordingly, the NRC performs the specified Federal-level response functions, as appropriate and consistent with the agency's authorities and responsibilities. For incidents below the threshold of an Incident of National Significance, the NRC, as the Coordinating Agency, performs the Federal-level functions and coordinates the overall Federal response as provided in the NRI Annex. The designated "cooperating agencies" (e.g., Department of Energy, Environmental Protection Agency, Department of Agriculture) provide assistance and support to the NRC. For Incidents of National Significance, DHS is responsible for the overall coordination of Federal response activities and the NRC, as Coordinating Agency, performs the annex-specified Federal-level functions in concert with DHS.

5.16.2 Federal Emergency Management Agency (FEMA)

DHS/EPR/FEMA serves as the NRI "annex coordinator" of Federal resources and coordinates the assistance to affected State, local and tribal governments under the Stafford Act or Federal-to-Federal support provision of the NRP.

5.16.3 Department of Energy (DOE)

Under the NRP NRI Annex ,the DOE will dispatch radiological assistance teams to aid in monitoring and provide technical guidance. These teams are advisory and will not assume control from local authorities. Teams can be dispatched from Oakland, Livermore, Los Angeles, San Diego, and Las Vegas. Normally the State OES will be the primary contact for requesting DOE assistance.

Response resources available from the DOE regional offices include:

- Federal Radiological Monitoring & Assessment Center (FRMAC)
- Radiological Assistance Program Teams (RAP)
- Atmospheric Release Advisory Capability (ARAC)
- Joint Nuclear Accident Coordination Center
- Nuclear Emergency Search Team (NEST)

Although these resources could originate from Lawrence Livermore Laboratory (LLL) near Livermore, California, the DOE aircraft and helicopters providing transportation support would originate from the DOE Las Vegas, Nevada facilities. Transit time for team response, once alerted, would be approximately six hours.

5.16.4 Coast Guard

The U.S. Coast Guard may be requested to assist in emergency actions involving vessels and persons, off-shore and adjacent to the plant. The Coast Guard has one or more vessels based at Morro Bay, approximately eight miles north of the plant. Upon notification, the Coast Guard will notify and attempt to remove persons and vessels from affected coastal waters. They will prevent vessels from entering affected areas and vessels suspected of being radioactively contaminated until they can be surveyed. The Coast Guard maintains " U.S. Coast Guard Procedures to be followed upon Notification of an Emergency at Diablo Canyon Power Plant."

The San Luis Obispo County Emergency Plan provides that the Sheriff will make notifications to the United States Coast Guard (USCG). If the situation at the plant requires immediate protective actions on the part of the general public and the County Emergency Organization has not had time to be activated, the plant will notify the USCG to place them on standby or make recommendations for protective actions as appropriate.

5.16.5 Environmental Protection Agency (EPA)

EPA, through its Nuclear Radiation Assessment Division and its Office of Radiation Programs in Las Vegas, can provide trained health physics personnel, field sampling equipment and laboratory facilities for assessment and monitoring during an emergency. EPA provides radiological monitoring support for DOE in initial stages of an emergency and assumes the lead responsibility in the intermediate to long-term recovery period.

5.17 CROSS REFERENCE TO NUREG-0654

NUREG 0654	DCPP Emergency Plan	NUREG 0654	DCPP Emergency Plan
A.1.a	5.11.1, 5.11.2, 5.11, 5.11.1, 5.13, 5.14, 5.15, 5.16	E.4.a to n	5.14
A.1.b	5.11.1, 5.11.2, 5.11, 5.11.1, 5.13, 5.14, 5.15, 5.16	F.1.a	5.14
A.1.c	5.9	F.1.b	5.14, 5.15
A.1.d	5.4.1, 5.3.2, 5.3.3, 5.6.1, 0	F.1.c	5.16
A.1.e	5.1, 5.4.1, 5.5	F.1.e	5.13
A.3	5.11.1, 5.15, 5.16, 5.12.3	G.3.a	5.14
A.4	5.4.1, 5.5, 5.11, 5.6.1	G.3.b	5.14
B.1	5.1, 5.6.1, 5.6, 5.7	G.4.a	5.11, 5.14, 5.15.1
B.2	5.4.1, 5.3.2, 5.3.3, 5.6.1, 0	G.4.b	5.14, 5.15.1
B.3	5.1, 5.4.1	G.4.c	5.14
B.4	5.4.1, 5.3.2, 5.3.3, 5.6.1, 5.1, 0	H.1	5.6
B.5	5.1, 5.4.1, 5.1.3, 5.6, 5.7, 5.8	H.2	5.8
B.6	5.9, 5.11, 5.12.2, 5.13, 5.14, 5.5, 5.6, 5.7, 5.8, 5.9	I.8	5.5, 5.6, 5.7, 5.8
B.7.a-d	5.11, 5.4.1, 5.14, 5.15, 5.16	K.2	5.6, 5.7, 5.8
B.8	5.11.1, 5.13	L.1	5.12.3, 5.12.5, 5.12.6
B.9	5.11.1, 5.12.3, 5.13, 5.14, 5.15, 5.16	L.4	5.12.3, 5.12.6
C.1.a - c	5.4.1, 5.16	P.3	5.1.1
C.2.b	5.14, 5.8	P.6	5.11, 5.11.1, 5.13, 5.14, 5.15, 5.16
C.4	5.11.1, 5.13, 5.14, 5.15, 5.16	P.7	5.17

5.18 TABLE B-1, DCPP ON-SHIFT AND ERO STAFFING

Functional Area	Major Tasks	Emergency Positions	Minimum Staffing On Shift	Minimum Staffing *60 Minute	Full Augmentation
1. Plant Operations and Assessment of Operational Aspects	Control Room Staff	Shift Manager Shift Foreman Licensed Operator Nuclear Operator	1 2 4 5		
2. Emergency Direction and Control	Command and Control	Shift Manager (CR) Site Emergency Coordinator (TSC) Emergency Director (EOF)	b	 1 1	
	Facility Control	TSC Director (TSC) EOF Director (EOF)		 1	1
3. Notification & Communication	Emergency Communications	Shift Communicator (Licensed Operator) Communications Coordinator(EOF) Agency/ENS Communicator (TSC) HPN Communicator (EOF)	1	 1 1	 1
	Plant Status & Technical Activities	Communications Advisor (TSC) OPs Communicator (CR) Engineering Liaisons (EOF)			1 1 2
	In-Plant Team Control	Team Coordinator (OSC)			1
	Governmental	Offsite Communicator (EOF) Advisor to the County (County EOC) Government Relations Coordinator (EOF)		1	 1 1
4. Radiological Assessment	Offsite Dose Assessment	Radiological Manager (EOF) Dose Assessment Coordinator (EOF) Dose Assessor (EOF) UDAC Meterologist		1 1	 1 1
	Offsite Surveys	FMT Coordinator (EOF) FMT Communicator (EOF) Field Monitoring Team (EOF) Offsite Emergency Lab Analyst (EOF)		 4	1 1 1
	Onsite Surveys	Field Monitoring Team (OSC)		2	
	In-plant Surveys	C&RP Technician (OSC)	1	2	(c)
	Chemistry	C&RP Technician Chemistry Coordinator (OSC)	1		(c) 1
	RP Supervisory	Radiological Advisor (TSC) Site RP Coordinator (OSC)		 1	1

5.18 TABLE B-1, DCPP ON-SHIFT AND ERO STAFFING

Functional Area	Major Tasks	Emergency Positions	Minimum Staffing On Shift	Minimum Staffing *60 Minute	Full Augmentation
5. Plant System Engineering, Repair, and Corrective Actions	Technical Support / Accident Analysis	Shift Technical Advisor (CR)	(b)		
		Shift Forman (CR)	1[(a)]		
		Engineering Advisor (TSC)			1
		Reactor Engineer (TSC)		1	
		Mechanical Engineer (TSC)		1	
		Electrical Engineer (TSC)		1	
		Operations Advisor (TSC)		1	
		Radiological Data Processor (TSC)			1
		PPC Operator (TSC)			1
	Repair and Corrective Actions	Shift Control Technician	1		
		Mechanical Maintenance (OSC)			(c)
		Electrical/I&C Maintenance (SCT)	1		(c)
		Maintenance Advisor (TSC)		1	
		OSC Director (OSC)		1	
		Mechanical Coordinator (OSC)		1	
		Electrical Coordinator (OSC)		1	
		I&C Coordinator (OSC)		1	
		Operations Coordinator (OSC)			1
6. In-Plant Protective Actions	Radiation Protection	C&RP Technician	(b)	4	(c)
7. Fire Fighting	--	Fire Department	(d)		
8. 1st Aid and Rescue Ops	--	Industrial Fire Officers	(b)		(c)
9. Site Access Control and Personnel Accountability	Security & Accountability	Watch Commander	1		
		Security Personnel	(e)	(e)	(e)
		Security Advisor (TSC)			1
10. Resource Allocation and Administration	Logistics	Administrative Advisor (TSC)			1
		Security (EOF)			1
	Administration	ED Admin Assistant (EOF)			1
		SEC Admin Assistant (TSC)			1
		Clerical Staff (TSC/EOF/OSC)			3
		UDAC Clerical Staff (EOF)			1

5.18 TABLE B-1, DCPP ON-SHIFT AND ERO STAFFING

Functional Area	Major Tasks	Emergency Positions	Minimum Staffing On Shift	Minimum Staffing *60 Minute	Full Augmentation
11. Public Information	Media Interface	Company Spokesperson (JIC)			1
		News Media Liaison – Corp (JIC)			1
		News Media Liaison – Site (JIC)			1
	Information Development	Public Info Officer (JIC)		1	
		Technical Advisor – Ops (JIC)			1
		Technical Advisor – HP (JIC)			1
		News Writer (JIC)			1
		Assistant PIO (JIC)			1
	Facility Operation and Control	JIC Director (JIC)		1	
		JIC Security (JIC)			1
		JIC Clerical Staff (JIC)			1
		TOTAL:	19	32	38

Notes:

* Response time is based on optimum travel conditions.

(a) Typical DCPP Position is a 3rd Shift Foreman but can be filled by another on shift individual with functional qualifications that is not already filling a minimum staff position

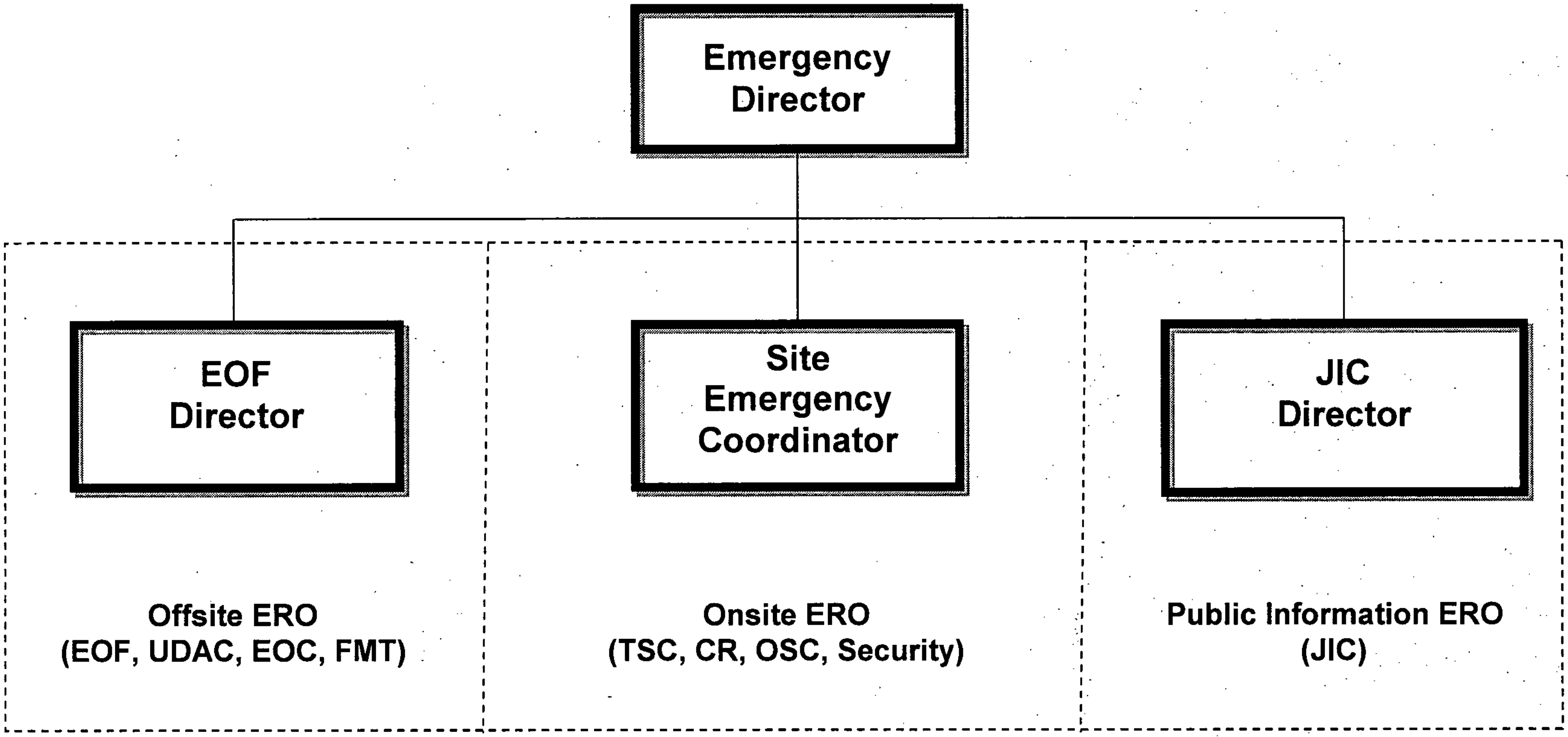
(b) May be filled by someone filling another position having functional qualifications.

(c) Personnel numbers depend on the type and extent of the emergency.

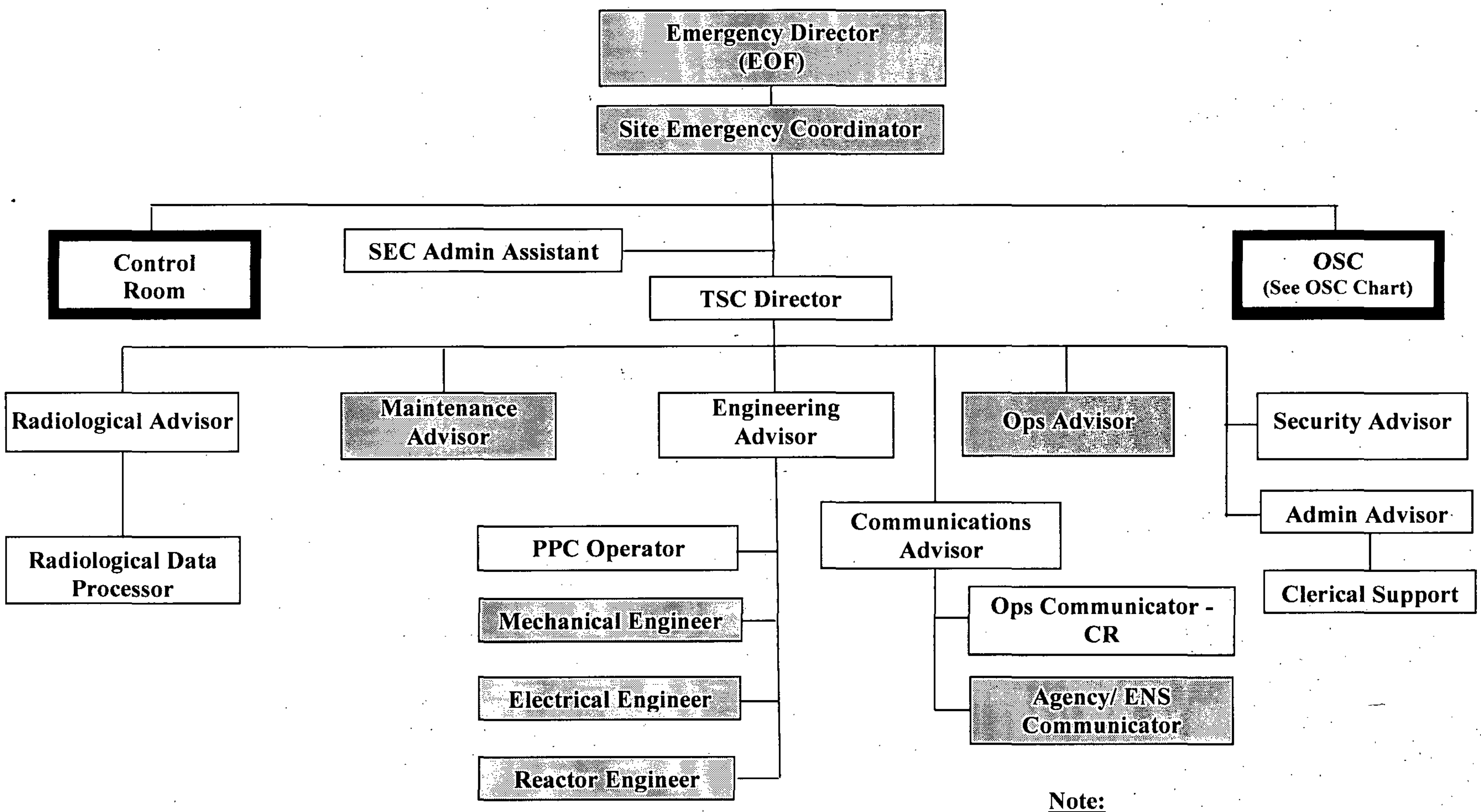
(d) Fire Brigade per FSAR/Technical Specifications, as applicable.

(e) Per DCPP Security Plan.

5.19 FIGURE 5-1, OVERALL COMMAND AND CONTROL STRUCTURE

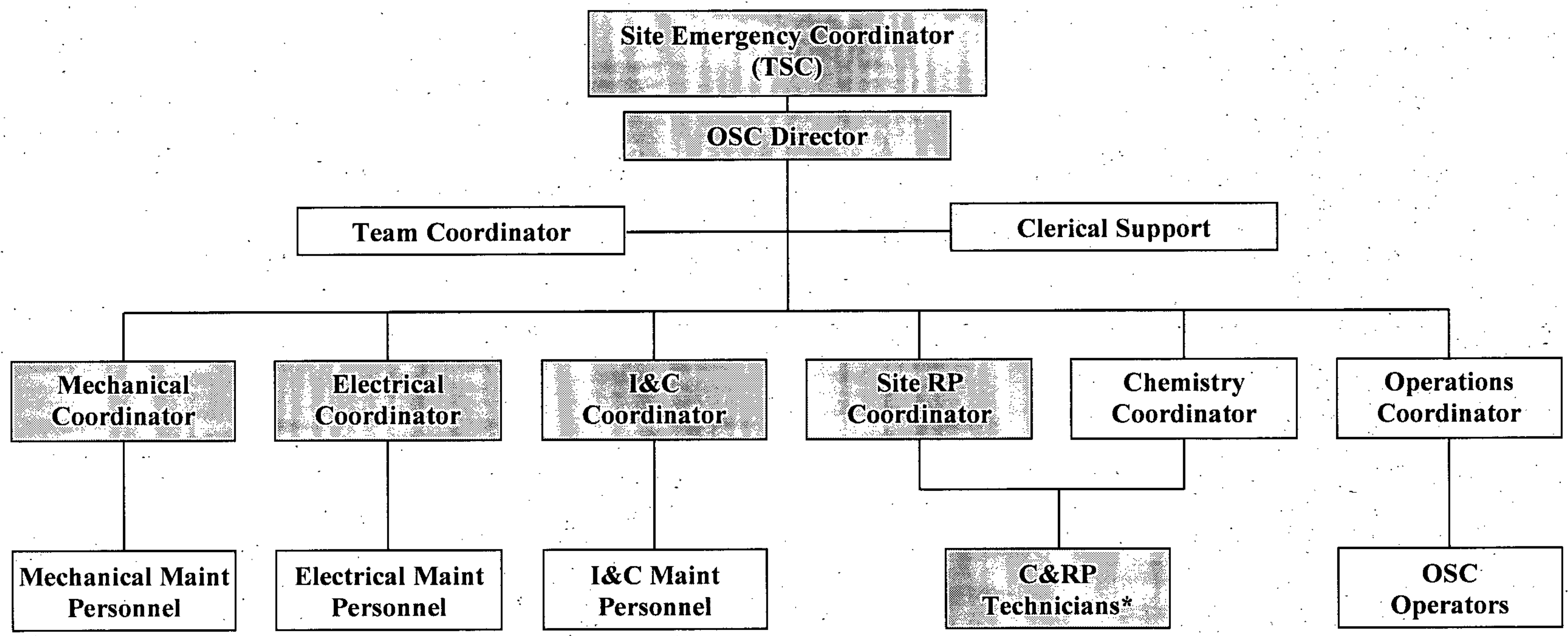


5.20 FIGURE 5-2, TSC ORGANIZATION



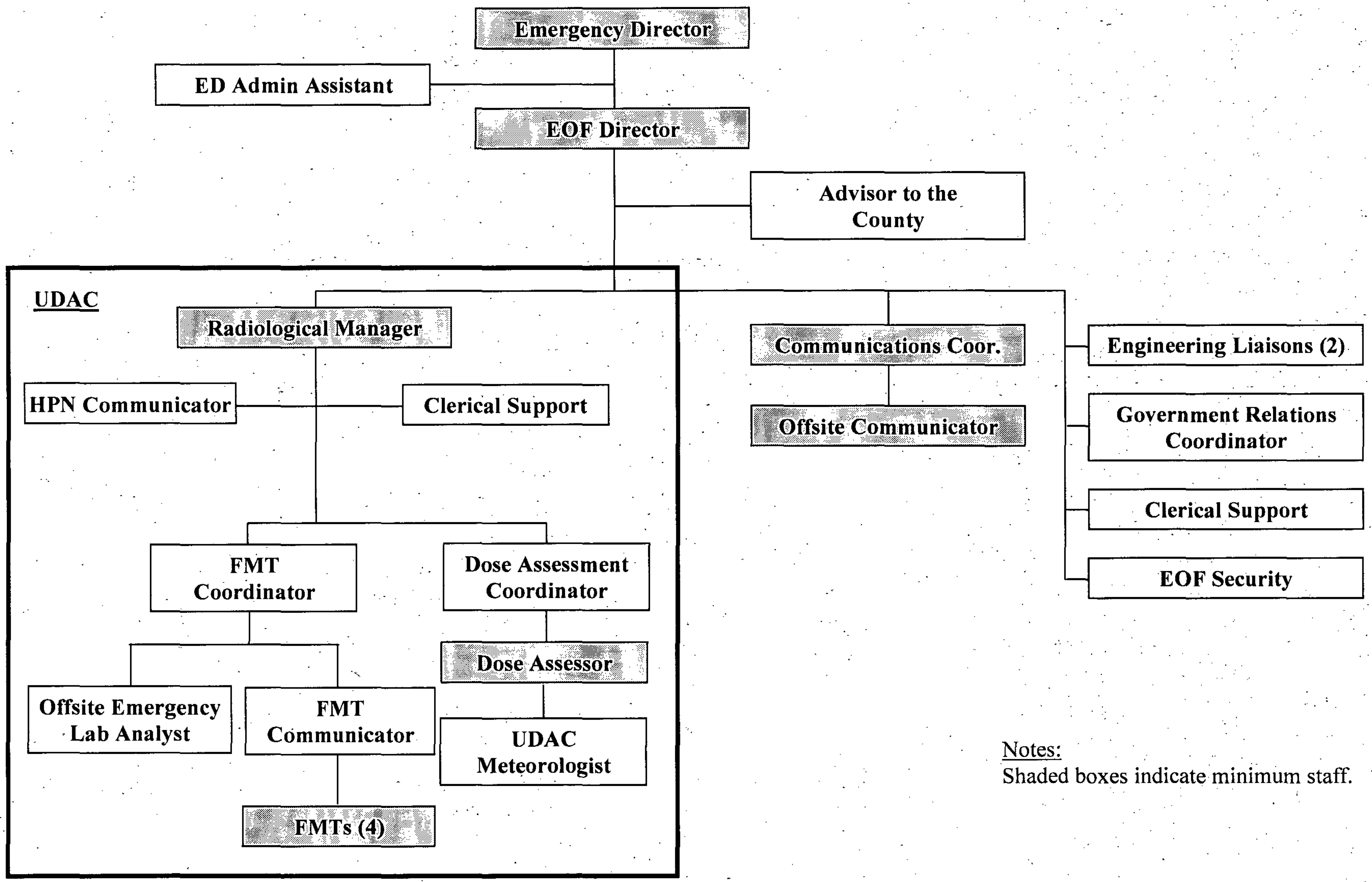
Note:
Shaded boxes indicate minimum staffing.

5.21 FIGURE 5-3, OSC ORGANIZATION



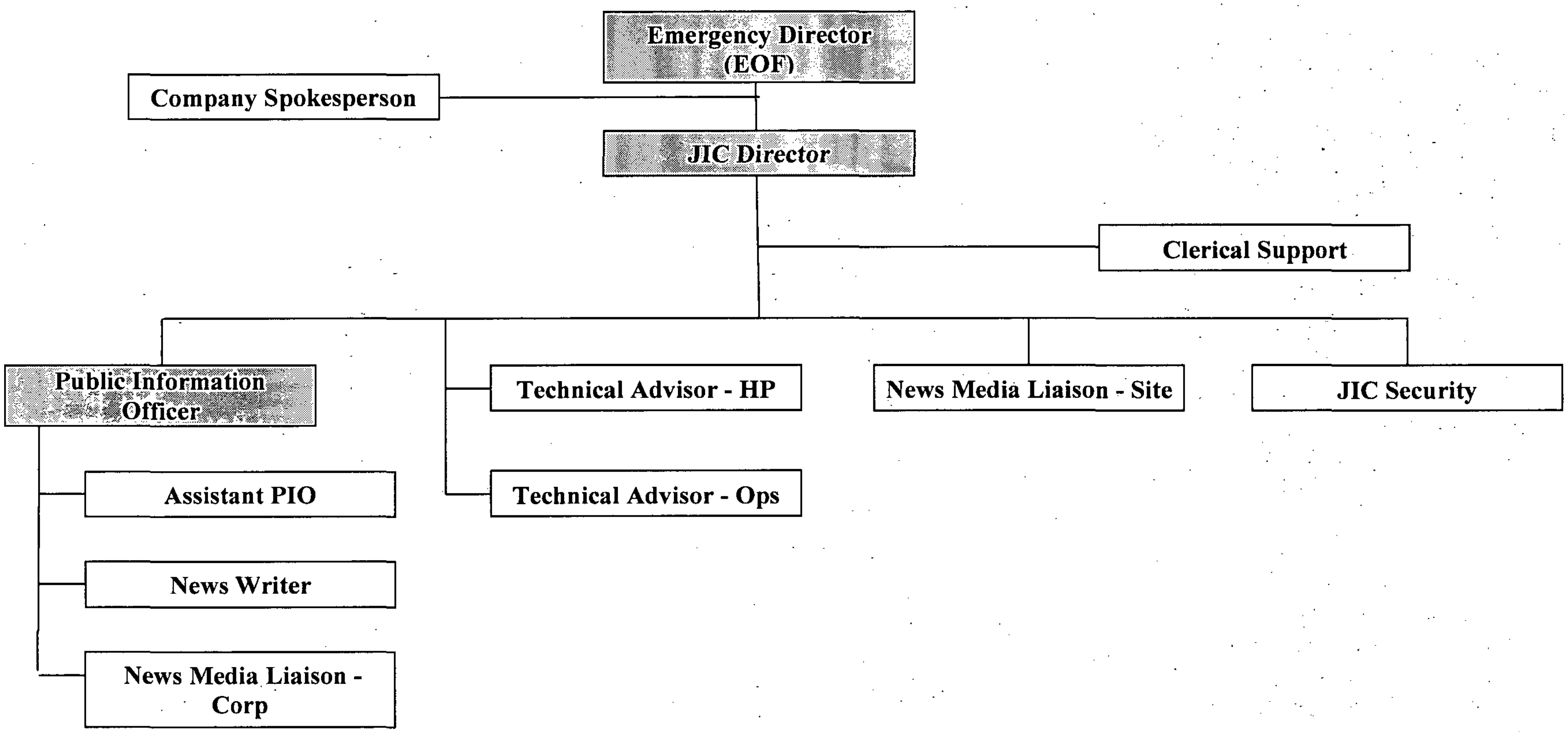
Note:
Shaded boxes indicate minimum staffing.
* 8 C&RP Technicians:
4 In-plant protective actions
2 In-plant surveys
2 FMT members

5.22 FIGURE 5-4, EOF ORGANIZATION



Notes:
Shaded boxes indicate minimum staff.

5.23 FIGURE 5-5, JIC ORGANIZATION



Notes:
Shaded boxes identify minimum staff.

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6. EMERGENCY MEASURES

6.1 ACTIVATION OF EMERGENCY RESPONSE ORGANIZATION

The four emergency classification levels require a varying degree and scope of emergency response. The Shift Manager will immediately initiate actions to limit the consequences of the event and to return the plant to a safe and stable condition. The emergency organization for an Unusual Event consists of the normal shift personnel. Normally, no further site emergency staff augmentation is required, although several members of the plant management are notified and may choose to come to the plant, depending on the circumstances. The Shift Manager may activate or partially activate an Emergency Response Facility to limit the consequences of an event prior to meeting the requirements of a declared emergency and to return the plant to a safe and stable condition.

For Alert, Site Area Emergency, or General Emergency, the Technical Support Center (TSC), Operational Support Center (OSC), and Emergency Operations Facility (EOF) will be staffed and activated by the emergency response organization (ERO).

6.1.1 ERO Notification

When a plant emergency has been declared at the Alert, Site Area Emergency, or General Emergency level, the emergency response organization (ERO) will be notified to activate.

The ERO notification process will be initiated within approximately 10 minutes of the emergency declaration.

6.1.2 Emergency Response Facility Activation

The on shift staff will be augmented by the minimum staff ERO within approximately 60 minutes of the initiation of ERO notification.

The emergency response facilities will be activated when the augmentation by the ERO minimum staff is complete.

Following emergency response facility activation, the transition of emergency responsibilities from the normal operating organization to the emergency response organization will progress as described later in this section, with the ERO assuming responsibilities as described in Section 5.

Minimum staff ERO is defined as DCPP management, administrative, and technical support personnel who will augment the on-shift minimum plant staff in emergency situations as specified in Section 5 and NUREG-0654, Table B-1.

Minimum staff position vacancies may be filled by other qualified individuals not already filling a minimum staff position.

The phrase "approximately 60 minutes" reflects a goal and an expectation, rather than a nominal 60-minute limit. ERO augmentation within 70 minutes, will be considered acceptable towards meeting the goal of approximately one hour.

Other members of the plant staff may be requested to respond by a secondary call out once the initial responders identify the personnel resources and expertise required to mitigate the event in progress. The Atomic Safety and Licensing Board ruled in 1982, that DCPP is not required to augment the on-site staff within 30 minutes. The requirement for 30-minute responders was obviated by increased on-shift staffing as well as increasing the number of 60-minute responders..

6.1.3 Transition from Normal to Emergency On-Shift Staff

The normal and emergency on-site operating organizations are discussed in Section 5. The transition from the normal operating organization to the On-Site Emergency Organization involves three basic steps: 1) filling appropriate positions on an interim basis with personnel who are immediately available at the time of the emergency; 2) notifying plant personnel who are off-site, or are on-site but who may not be aware of the emergency, that their assistance is required; and 3) filling positions in the emergency organization with appropriate plant personnel as they arrive at the TSC or OSC.

1) Notification of Shift Manager/Shift Foreman

   The first step in the event of an emergency is to notify the Shift Manager/Shift Foreman. To accomplish this the individual discovering the emergency shall immediately report it to the Control Room and the Shift Foreman/Shift Manager would be informed.

2) Notification of On-Site Personnel by Site Emergency Signal or Fire Signal

   The Shift Manager (SM) shall make an initial evaluation of the situation and, if warranted, shall authorize the sounding of the emergency signal (described in Section 7).

   If the emergency involves a fire, the person who discovers the fire will dial 779, which rings in the Control Room. The SM will sound the fire alarm and make a PA announcement stating the nature and location of the emergency and the response being requested. The fire alarm is a separate and distinct signal.

6.1.4 Initial Deployment of On-Shift Personnel

The initial deployment of on-shift personnel is strongly dependent upon the extent of the emergency and the time it occurs. To illustrate this, a possible sequence of events is considered below for a major radiological release incident that occurs when a minimum crew is available on-site.

1) The person discovering the incident would communicate it to the Control Room and the Shift Manager would be informed.

2) The Shift Manager would classify the emergency and assume the position of Shift Manager (SM). The Shift Manager would have the Control Operator sound the Emergency Signal. The operators would then report to the Control Room unless they were engaged in a critical operation.

3) The Shift Manager would instruct the Shift Foreman to supervise the operators in making appropriate plant control manipulations to respond to the event.

4) Licensed Operators would form the operations group and operate plant equipment and controls from the Control Room.

5) The third Shift Foreman would act as the Emergency Evaluation Coordinator (EEC) and provide assessment of the incident including initial classification and development of a Protective Action Recommendation. If the emergency involves loss of heat sink (core cooling source), or some other occurrence for which reactor core damage is a possibility, the EEC is primarily responsible for evaluation of this aspect of the emergency.

6) The Shift Manager would assign personnel to perform emergency notifications to San Luis Obispo County, California State OES and the Nuclear Regulatory Commission, until the TSC has been activated.

7) The Shift Manager would assign personnel to perform emergency response organization personnel call-out, PG&E management notifications and notification of the NRC Resident Inspectors until activation of the TSC.

8) The Nuclear Operators and Health Physics Technicians would be available as required for operating equipment, radiological monitoring, notification, or other tasks as they are identified. The Diablo Canyon Security Watch Commander would ordinarily continue normal duties.

9) As other individuals begin to arrive at the site, they will respond to their assigned emergency facility.

6.1.5 Notification of Off-Site Plant Personnel

The Emergency Response Organization (ERO) is grouped in to teams for rotating ERO "on call" duty assignments. On call ERO personnel maintain their availability for callout to ensure that staffing emergency response facilities are available. All ERO personnel (on call and off duty) will be called out for an event at an Alert, or higher emergency classification level. If a minimum staffing position cannot be filled by the person on call, qualified personnel available will be assigned. The "on-call" positions ensure minimum required staffing for emergency response facilities is available. These key positions have been selected to be compatible with the staff augmentation goals recommended by NUREG-0654, Table B-1. The minimum staff requirements are provided in Section 5 per the criteria of NUREG-0654, Table B-1.

Call-out of personnel is accomplished by pagers or telephone. Typical driving time for personnel living in the nearby communities to arrive on-site between 30 to 60 minutes, depending on where they live. On-shift personnel are fully capable of controlling and taking appropriate mitigating actions should those off-site persons called out be delayed.

6.1.6 Response of On-Site Personnel to Emergency Warning Signals

Several warning systems are available to warn on-site personnel of an actual or potential emergency. Section 7 of the Plan describes the physical nature of these warning systems, and this section describes on-site personnel response.

6.1.6.1 Site Emergency Signal

1) The emergency signal consists of electronic warblers and beacon lights manually initiated from the control consoles or the hot shutdown panels. In an emergency, the signal will be sounded continuously for at least one minute. The signal is tested weekly on a seven day routine schedule. Except in cases of a severe emergency when the Shift Manager is not readily available, sounding of the site emergency signal requires Shift Manager approval.

2) All personnel and visitors upon receiving initial site access training are issued a wallet type card which provides site emergency signal response information. Upon receipt of the emergency signal, on-site personnel are trained to immediately report to predestinated assembly areas unless otherwise directed by the site PA system.

3) Personnel are instructed to remain at the assembly area unless directed to leave by the Site Emergency Coordinator. If an assembly area is untenable, the person in charge in the area may direct personnel to leave, but will inform the Site Emergency Coordinator as soon as practicable.

6.1.6.2 Containment Evacuation Signal

The containment evacuation signal utilizes the emergency signal warbler and warning lights within the containment. When this signal is initiated, personnel in the containment are instructed to immediately leave the containment and report to access control.

6.1.6.3 Fire Signal

In the event the fire alarm is sounded, the Fire Fighters are dispatched as required. Other personnel are instructed to remain at their work locations and await further instructions. The fire signal is tested weekly on a seven day routine schedule.

6.1.6.4 Criticality Monitor

The criticality alarms in the fuel handling areas and nearby hot machine shop are horns automatically initiated on high radiation level, as measured by the area monitors in the fuel handling building. Upon receipt of this signal, personnel in the area are instructed to immediately leave and report to Access Control.

6.1.7 Activation of Corporate Emergency Organization

The Corporate Emergency Organization is discussed in Section 5.

The Corporate Emergency Organization can be activated by the Emergency Director by notifying Corporate Security or Corporate Emergency Planning Department in San Francisco. These positions are available on a 24-hour basis.

The Nuclear Power Generation Business Unit is promptly notified of any occurrence that would be reported under the provisions of the Emergency Plan. The extent to which corporate resources are activated is based on staged mobilization depending on the nature of the occurrence.

6.1.8 Activation of County Emergency Organization

Activation of appropriate portions of the County emergency organization is accomplished by telephone or radio communication from the plant Control Room to the Watch Commander at the County Sheriff's Office Dispatch Center. Provisions are included for message authentication.

The San Luis Obispo County emergency organization will activate at the Alert, Site Area Emergency, and General Emergency classifications. The county emergency plan provides for activating the Emergency Operations Center (EOC) and non-utility portion of the EOF under these classifications. The county emergency plan details activation procedures for county emergency response operations. When the EOC is activated for an emergency classification, plant staff personnel will be available at the County EOC to advise the county on plant equipment and plant radiological status.

Initial and follow-up emergency messages for each emergency classification are delivered by the power plant to the Sheriff's Office watch commander until relieved by the Advisor to the County who relays the messages to County Command in the EOC. To ensure that all necessary information is clearly transmitted, a standard form is used. This form provides for such entries as the classification of the emergency, if a radioactive release is taking place, potentially affected population and areas, and what protective measures may be necessary. Follow-up messages provided to off-site authorities provide a comprehensive description of the incident with a characterization of the radioactivity release and appropriate recommended protective measures.

6.2 ASSESSMENT ACTIONS

In Section 4, a brief description of the basic assessment process is discussed for each of the postulated emergencies that were described. This section contains a more detailed discussion of the four most important assessment functions; namely, the proper functioning of emergency cooling systems for emergencies involving possible degradation of the core heat sink, the assessment of core condition in such a circumstance, the estimation of the magnitude of a release, and the determination of the environmental consequences of a release.

6.2.1 Verifying Proper Operation of the Emergency Core Cooling System (ECCS)

The design basis of the ECCS is to prevent a radioactive release by protecting the three major fission product barriers: the fuel cladding, the reactor coolant system piping, and the containment structure. A breach of all three barriers is necessary before radioactive contamination is released to the atmosphere that might pose a hazard to the health and safety of the public. If a Loss of Coolant Accident (LOCA) or Steam Generator Tube Rupture (SGTR) occurs, the reactor protection system will automatically trip the reactor and initiate the ECCS.

In the event of a reactor accident, the Shift Foreman, with the assistance of the operating crew, will ensure the reactor is tripped and will enter the Emergency Operating Procedures (EOPs). The EOPs:

- Verify the reactor is shutdown.
- Verify the operation of the ECCS equipment.
- Diagnose the accident.
- Provide corrective actions to mitigate or alleviate the problem.

If the expected action is not obtained or the equipment does not function properly, the EOPs direct alternate, remedial responses. Criteria for upgrading to a general emergency during a LOCA event are found in the Emergency Action Level Classifications.

6.2.2 Assessing Challenges to Fission Product Barriers

During the implementation of the EOPs, the Emergency Evaluation Coordinator (EEC) is responsible for monitoring the reactor's critical safety functions to ensure that the remaining fission product barriers are not breached. These functions are:

- Subcriticality - Verifying the reactor is shut down - prevent clad failure
- Core Cooling - Ensuring the reactor core is being cooled - prevent clad or RCS failure
- Heat Sink - Ensuring the heat in the reactor is being dissipated - prevent clad or RCS failure or Containment failure
- RCS Integrity - Monitoring the reactor temperature to prevent thermal shock - prevent RCS failure
- Containment - Monitoring containment parameters to ensure containment integrity - prevent Containment failure
- RCS Water Inventory - Monitoring RCS water levels - prevent clad or RCS failure

Upon completion of the initial accident diagnosis, the EEC will notify the Shift Foreman of the critical safety function status. If a safety function is threatened, an alternate EOP functional restoration procedure will be utilized to mitigate the problem.

6.2.3 Assessing Core Damage

Preliminary core damage assessment uses parameters such as reactor vessel water level and core temperatures to confirm that conditions do not exist which can lead to core damage. This is quantified through the use of containment hydrogen and area radiation monitor readings.

Long-term core damage assessment methodology uses reactor coolant and containment air sample analysis to determine the extent of core damage more accurately.

6.2.4 Assessing Release Magnitude

During the initial stage of an off-site release, the Emergency Evaluation Coordinator will make a preliminary dose assessment of any off-site release to determine the accident classification per Emergency Plan Implementing Procedures. After UDAC is activated, they will take over these responsibilities and perform more detailed calculations. These calculations use various radiation monitors, ventilation flow rates, wind speed, direction and stability classification, and the plant parameters to project an estimate of the magnitude, direction and size of the radioactive plume. The results of these calculations will be included in Protective Action Recommendations (PARs) to county personnel.

6.3 OVERVIEW OF THE ASSESSMENT AND MONITORING PROGRAM

The following is a general discussion of the monitoring program. If sufficient personnel are immediately available, or as they become available, several monitoring teams can be formed and several of the steps should be performed simultaneously.

6.3.1 Assessment of Environmental Consequences of Airborne Releases

In the first few hours following a release of airborne radioactive materials to the environment, a monitoring program will be established to assess the extent of the release and to provide guidance for appropriate protective measures. The general program and measurement techniques for environmental monitoring following a suspected airborne release are discussed in this section.

The principal early concerns are thyroid exposure due to inhalation of radioactive iodines and/or whole body exposure from immersion in a cloud of radioactive noble gases. Criteria for taking protective actions such as sheltering and evacuation are expressed in terms of these two variables, and early off-site government agency efforts will be directed toward their assessment. Following this, efforts by off-site authorities will normally be directed toward the evaluation of possible long-term exposures from ground deposition and various food-chain pathways.

6.3.2 Off-Site Dose Calculation

The scope of off-site dose calculation is to establish methodologies for performing early phase dose assessments used for evaluating the need to evacuate, shelter, or implement other appropriate protective actions for individuals located within the DCPP Basic Emergency Planning Zones (EPZ).

In the event of an accident at the Diablo Canyon Power Plant (DCPP) involving an actual or a potential release of radioactive materials, projected off-site doses to members of the public will be determined primarily using the EP R-2 program on a Plant Data Network (PDN) computer in the Control Room, or the Emergency Assessment and Response System (EARS) once UDAC is staffed. As a back-up method, in the event EARS is not operational, a PC-based dose calculation program is available which employs a manual dose calculation methodology. The manual calculations can also be performed without a PC using dose calculation implementing procedures. The EARS and manual dose calculations methodologies are both based on the current stochastic and deterministic risk models, developed originally by the International Commission on Radiation Protection (ICRP) in publication numbers 26, 30, and 60.

The ICRP risks models have been adopted by the Environmental Protection Agency (EPA) Regulatory Guide 400-R-92-001, "EPA Manual of Protective Action Guides and Protective Actions for Nuclear Incidents," which serves as the source document for implementing Protective Action Guides (PAG) to protect the members of the public and emergency workers in the event of a radiological accident at DCPP.

The implementing procedures used for performing manual dose calculations are EP R-2, written for use by Control Room Operators, or EP RB-9 and 11, written for use by the UDAC dose assessment staff.

All manual dose calculation implementing procedures share the same basic methodologies. The differences are:

1) EP R-2 uses a more simplified set of assumptions to allow plant operators to perform manual dose calculations quickly with less chance of error, thus allowing operators to focus their efforts on returning the plant to a safe and stable condition. The R-2 calculation can also be done by invoking a computer based program, which automatically retrieves the necessary radiation monitor, meteorology, and flow rate information to perform the calculation.

2) EP RB-9 and 11 are intended to provide users greater accuracy for projecting dose, and this methodology is employed by the PC-based program installed as a backup to EARS.

6.3.3 Estimate the Magnitude of the Release and/or Release Rate

In most cases, a release to the environment will be monitored by permanently installed, real time monitoring instruments at the effluent release points and at various locations in the environment. These instruments will be promptly checked to estimate the release rate and/or magnitude of the release. In addition, these monitor readings will be correlated with analysis data of the source terms of these releases. In those cases where the unavailability of monitoring instruments does not make the above possible, (due to monitors being off-scale or inoperable) on-site monitoring team data will be used to make the most accurate initial estimate of the magnitude of the release.

6.3.4 Determine If On-Site Personnel Assembly Areas are Affected

In the event of a major release, protection of on-site personnel is a high priority consideration. On-site monitoring is conducted to assure personnel safety in the assembly areas by performing dose rate and air sample surveys. Results of these surveys are transmitted to the Radiological Advisor who may recommend evacuation or other protective measures as warranted.

1) Determine whether the external dose rate has reached an action level for evacuation.

2) Determine whether iodine sampling is necessary.

   The determination of airborne radioiodine concentration takes several minutes, due to the time required to obtain an air sample. In the early stages of assessment, it may not be desirable to collect air samples if it has been determined radioiodine is not a problem. Since noble gases will always accompany iodine in a release, and in general will be released in substantially greater quantities, it is possible to set an upper limit on the possible airborne iodine based upon a measurement of the external dose rate.

   In general, air samples are immediately collected if the general γ dose rate exceeds 3 mR/hr, and can be deferred for dose rates below this. This criterion is based upon predicted noble gas/iodine release rates for major accidents. Air samplers used to detect and measure radioiodine concentrations in the DCPP vicinity have a measuring range between 1E-10 and 1E-6 μCi/cc.

6.3.5 Perform General Monitoring On-Site

Once it has been determined personnel in on-site assembly areas are not endangered, the monitoring program should be directed toward a general assessment of radiological conditions on-site by making appropriate downwind surveys.

Because the location of a plume is often difficult to determine, it is necessary for monitoring teams to "fan out" circumferentially from the measured downwind direction. However, on-site terrain makes it difficult to reach all areas. The preferred monitoring locations would be along the circumference of a circle of approximately 0.5 mile radius from the reactor.

Where it is not possible to reach downwind monitoring locations at the 0.5 mile distance, it will be necessary to move in either closer to the reactor, or farther away. If a ground level release is the most likely, then moving in closer is preferred. If the release is elevated, then moving farther away is preferred.

To reduce confusion during the early stages of on-site data collection, predetermined monitoring locations near the 0.5 mile site boundary have been marked by white with red top poles. These are suggested locations for obtaining initial measurements.

The data gathered from on-site monitoring will be forwarded by satellite telephone to the Control Room or the TSC where that data can be evaluated by responsible assessment personnel. These personnel will then recommend protective actions, as appropriate, to the Site Emergency Coordinator. The Site Emergency Coordinator will make the decision to notify appropriate company or county personnel regarding release status and make appropriate action recommendations.

6.3.6 Establish Off-Site Monitoring

If any of the assessment actions verify that a significant release has occurred, Field Monitoring Teams will be immediately dispatched to perform off-site monitoring. The teams will perform external dose measurements, obtain air samples, and can perform ground and vegetation surveys if required. This monitoring will continue throughout the duration of the accident so that the need for protection measures can be quickly assessed.

6.3.6.1 Identification of Monitoring Locations

Because it is extremely important to clearly and unambiguously identify locations where environmental measurements are made, a systematic approach has been developed to identify monitoring locations that are used throughout the Emergency Plan, showing the site and surrounding areas, respectively.

1) Emergency Monitoring Locations

   Suggested emergency monitoring locations have been established for each sector. These are easily identified locations in the environment to which teams should travel to obtain their initial samples and are chosen to produce data from representative environmental locations. By identifying these locations ahead of time, there is assurance that teams will obtain the maximum information during the early stages with minimum logistical difficulty. Although the Radiological Manager and/or the monitoring team may choose to sample at alternative locations, it must be emphasized that any such "nonstandard" locations must be carefully identified.

   Where fixed instruments are located in the environment, the "emergency monitoring location" usually coincides with the location of the fixed instrument.

2) Real time monitoring and environmental sampling locations

   There is an extensive network of real time monitors, TLDs, and air samplers surrounding the plant. The real time monitors will be automatically interrogated throughout the course of the accident and the environmental assessment. In some circumstances, it may be desirable to have the TLDs from the TLD stations collected and have the data analyzed. Monitoring teams are directed, however, to not collect these TLDs if the EOF Radiological Manager determines that it is necessary to measure the integrated dose over the duration of the emergency where continued releases are a possibility.

3) Dairies

   Because the milk pathway is often the limiting pathway, when actual release has occurred, if a milk dairy is in the area, milk samples will be analyzed as part of a long-term program.

6.3.6.2 Environmental Measurements

The field monitoring teams have a standard series of samples and measurements that they are prepared to make. These samples and the major uses to which the various data can be put are discussed below.

1) External dose rate and/or count rate three (3) feet above ground.

   Measured external dose rates may be used in comparison of Effective Dose Equivalent (EDE) for verification of dose projection models.

2) Ground Surveys

   Count rate measurements made close to the ground using a shielded probe can provide an indication of the magnitude of ground contamination.

3) Air Samples

   Provide airborne radioiodine and particulate data if the plume is present. May be used for estimating internal exposure using dose conversion factors.

4) Vegetation Samples

   Provide information on vegetation contamination for evaluating food chain doses for ingestion pathway monitoring.

5) Soil Samples

   Provide information on ground contamination which is used to estimate food chain doses for ingestion pathway monitoring.

6) Liquid Samples

   Can be used to estimate ingestion doses for long-term program.

7) Smear Surveys

   Indicate need for decontamination measures. Smear samples may also be analyzed by a lab to determine isotopic content of ground deposition.

6.3.6.3 Assessment of Off-Site Field Monitoring Data

Environmental monitoring data serves as part of the basis for determining what protective actions are required to protect the public. The group in the Unified Dose Assessment Center (UDAC) has the function to assess this data and recommend protective actions to the county emergency response director. This group is composed of utility, county, state, and federal personnel. Details on the mechanisms by which this group functions are as follows:

1) Field monitoring teams will meet at a predetermined location to obtain emergency sampling kits, radios, etc., and will be dispatched to standard monitoring sites in affected areas. Samples will be collected and analysis made as discussed earlier in this section.

2) The locally obtained survey data will be communicated by satellite telephone to the Control Room (CR) or TSC initially or, once established, to the Unified Dose Assessment Center (UDAC) at the EOF using the County Brown radio network as the primary communications mode with satellite telephones available as a secondary backup. Results from samples requiring more detailed laboratory analysis will be forwarded to UDAC from the various laboratories using telecommunication links.

3) UDAC personnel will interpret this data and, based on this assessment, make recommendations to the decision-making body. The primary method for handling this data will be through operator use of the Emergency Assessment and Response System (EARS) with backup methods as necessary to provide reliability.

6.3.7 Backup Off-Site Dose Assessment Calculation Methods

Backup methods for assessing radiological doses due to an airborne release provide redundancy to the automated real-time system normally employed. The backup methods (manual calculation procedure, PC-based dose calculation program), utilize simpler models that have been standardized and accepted in the industry for assessing such off-site releases and form the basis for the more time-responsive automated system.

6.4 PROTECTIVE ACTIONS

6.4.1 Alerting of On-Site Personnel

On-site personnel are alerted that an emergency condition exists by the sounding of the site emergency signal. Visitors either have been briefed on the meaning of the plant emergency signals or are escorted by individuals knowledgeable on actions to be taken upon activation of the signals. This applies to visitors and contractors outside and within the plant Protected Area. This signal provides an immediate alert for all on-site personnel.

Supplementary alerting mechanisms for limited areas include the fire alarm, criticality monitor, and containment evacuation signals. A physical description of these alarm signals is included in Section 7 of this Plan.

On-site tests have been performed to verify the response time for on-site personnel when the site emergency signal is sounded. These tests, conducted with considerable numbers of construction workers on-site, have shown assembly and accountability of employees can be expected in approximately 30 minutes. Even with peak construction forces on-site (> 5000 total people), total site accountability required no more than 60 minutes. Agricultural workers on the bench land will be alerted by Security personnel per procedure.

6.4.2 On-Site Personnel Accountability

Several methods for personnel accountability are employed at the plant. Each of these is discussed below.

1) Plant Personnel

The accountability procedures for plant personnel are intended to provide rapid assessment of who is on-site at any given time and where they are located. Several means are employed for personnel accountability, including control of identification badges, supervisory control, and written accountability logs.

As a prelude to the following discussion, it should be noted that plant personnel are provided with automobile passes enabling them to pass the Avila Gate entrance to the site on a 24-hour basis.

a) Control of Identification Badges

Each person requiring long term site access for their employment is issued a Protected Area identification badge for personal identification. The Protected Area identification badge is required for unescorted access into the Protected Area. Personnel visiting the site are provided visitor identification badges that are activated for the visiting period. Visitor badges provide an easily identifiable visual indication when expired. All personnel with authorized site access can be uniquely identified as a visitor or individually using their Protected Area identification badge. Personnel issued Protected Area identification badges are required to maintain control over their badges.

b) Supervisory accountability

In general, it is the responsibility of supervisors to know which of their personnel are on-site and their work location. Personnel report to their designated work headquarters and inform their supervisors of their presence. The supervisors are then responsible for knowing the general whereabouts of their personnel during the remainder of the work period.

c) Computerized Security System

A computerized record is maintained of personnel who enter or leave the Radiological Controls Area, the Power Block, and the Protected Area. The computerized security system can be used to determine personnel accountability inside these areas.

Personnel entering the Radiological Control Area must obtain written authorization in the form of a Routine Work Permit (RWP) or a Special Work Permit (SWP).

A log of all visitors to the Protected Area is maintained by the security staff.

d) Emergency assembly and accountability process

(1) Assembly - At the sounding of the site emergency signal, all non-essential site personnel are to report to their normal work locations and await further instructions. All personnel essential for emergency response and safe operation of the plant (ERO member, Plant Operations, Security, or Medical) will assemble at designated assembly areas or Emergency Response Facilities.

(2) Accountability - At the sounding of the site emergency signal all individuals remaining within the Power Block are accounted for and the names of missing individuals are determined within 30 minutes. Once established, accountability is maintained throughout the course of the event. Should unaccounted for personnel be determined missing, search and rescue operations are initiated. Accountability is coordinated by the Diablo Canyon Watch Commander and results forwarded to the Site Emergency Coordinator.

2) Visitors

The following assembly procedures are employed for visitors at DCPP. A visitor, as used in this section, refers to anyone who is not a member of the plant staff or is not employed with their normal work location at the site. This includes persons from outside the company as well as company personnel who are not assigned to the plant.

a) Assembly Process

(1) Prior to being allowed escorted entry into the Protected Area, visitors are required to sign the Visitor Log at the security building when they arrive.

(2) In general, visitors will have business with one or more plant personnel and their accountability while in the Protected Area will be the responsibility of their escort personnel. During plant start-up, refueling outages, or other special conditions, certain visitors may work for extended periods in the plant. In some cases, visitors may require unescorted access privileges while in the plant. Where these personnel are assigned to established work areas, instructed and badged as plant personnel, they may be treated as members of the plant staff for purposes of personnel accountability.

(3) Each visitor is escorted when in a Radiological Controls Area unless instructed in the basic fundamentals of the company's radiation protection program, including: identification of emergency and evacuation signals and action to be taken if they are sounded, identification of possible radiation hazards locations, and use of applicable protection and monitoring devices.

Visitors are assigned to an assembly area. Accountability is accomplished by checking the names of each person against the visitors log.

6.4.3 Evacuation of On-Site Nonessential Personnel

Evacuation of on-site nonessential personnel is one important protective action considered in emergency situations. Nonessential personnel include visitors and contractor personnel, and any other on-site individuals not having emergency response assignments.

1) Evacuation Criteria

The decision to evacuate nonessential personnel shall be made by the Site Emergency Coordinator.

Since nonessential personnel are not emergency workers, they typically would be evacuated in the event of a SITE AREA or GENERAL EMERGENCY; however, in certain situations it may be desirable to evacuate these individuals at the ALERT level

The Site Emergency Coordinator's decision to evacuate nonessential personnel will be based on the desire to protect the health and safety of these individuals within the constraints of the situation. It is desired to keep nonessential personnel exposures as low as reasonably achievable, and to the extent possible, lower than annual federally established limits for members of the public.

Anytime evacuation is considered it must be weighed against the consequences of not evacuating. In certain cases evacuation may result in higher evacuee exposures than if individuals remain in a shielded or otherwise protected area. In these cases evacuation would be inappropriate.

2) Evacuation Route Considerations

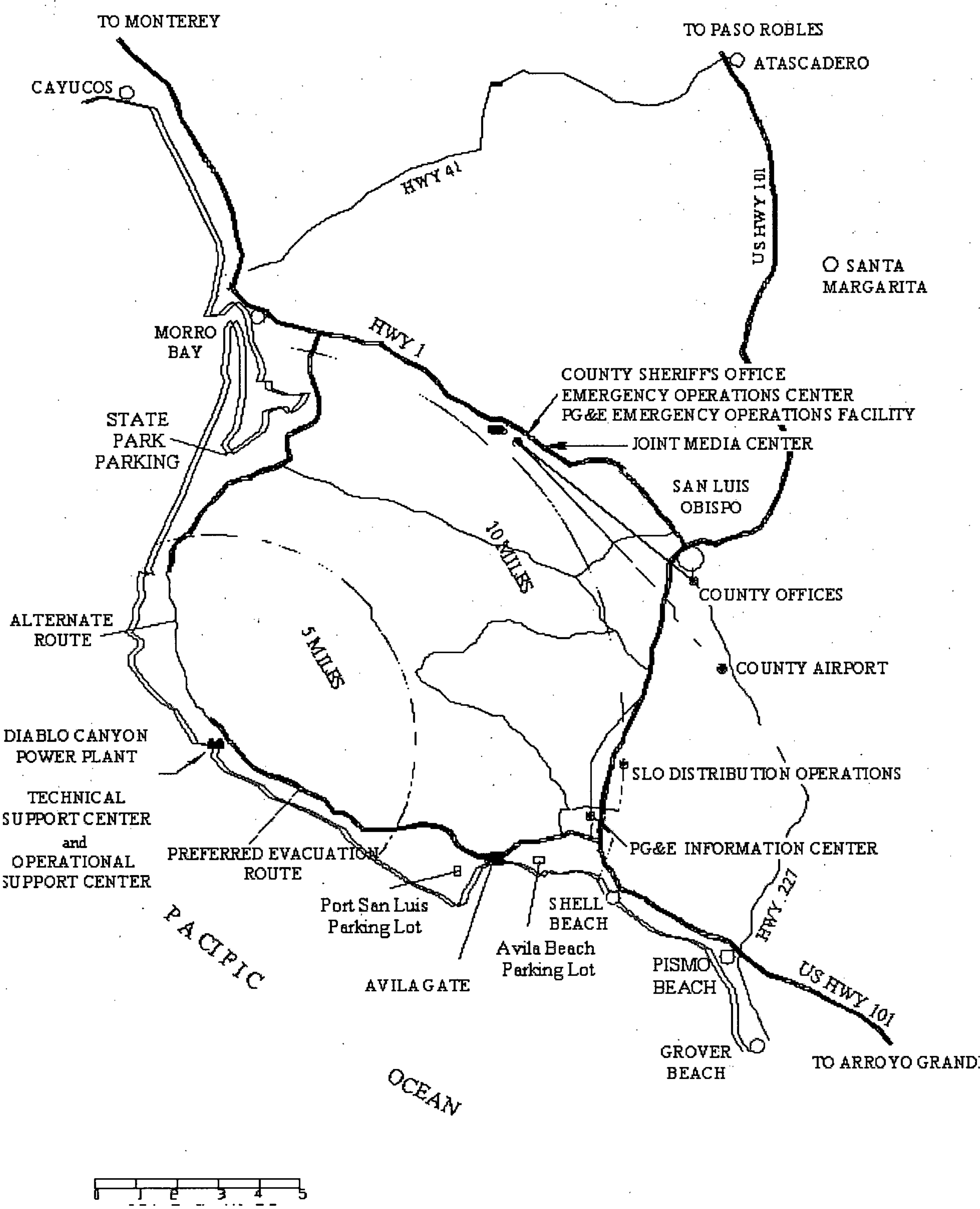
Two routes are available for evacuation from the site. The preferred route is south along the access road to Avila Gate. If conditions prevent the use of the southern route, the evacuation can take a northern route through Montana de Oro State Park. The southern route is preferred because the road is better. The northerly route would only be used in the event of northern wind with a large-scale radiological release or if the southern route is impassable or unsafe.

Alternate off-site assembly areas are available for each route. For the southern route, evacuees can meet at the PG&E Information Center, the frontage road along U.S. 101 north of the Information Center, the Port San Luis Parking Lot, or the parking lot at Avila Beach. For the northern route, evacuees can meet at the Ranger Station in the State Park. Preferred locations would be determined by selecting a direction away from the wind direction of movement. If it is a precautionary evacuation, locations nearer the site would be suitable, otherwise the more distant locations would be preferred.

The decision on which route and rendezvous point to be used shall be made by the Site Emergency Coordinator.

Evacuation routes are illustrated below.

Evacuation Routes for On-Site Personnel



3) Evacuation Procedure

Following the alerting and accounting of on-site personnel, if evacuation is required, the following general steps would occur:

a) The Site Emergency Coordinator will authorize the evacuation and determine which route and assembly areas shall be used. This information will be transmitted to the Evacuation Coordinator, appointed by the Site Emergency Coordinator.

b) The Communications Advisor will notify the Sheriff's Department or the Advisor to the County (coordinates with county Command) of the evacuation, specifying the evacuation route, the assembly areas, the approximate number of cars and individuals being evacuated, pertinent radiological information, and any other information useful in the evacuation.

c) The Evacuation Coordinator is responsible for conducting the evacuation in a safe and orderly fashion. This includes clearing the evacuation route (owner controlled area), personnel accountability of evacuees off-site, assuring transportation traffic control measures, and appointing at least one Radiological Monitor and an Evacuation Leader for each major assembly area.

d) The Evacuation Coordinator is responsible for delivery of the evacuation kits from their storage location to the off-site assembly area. Additional supplies are available in the field monitoring team (FMT) kit storage areas.

e) A C&RP Technician will leave with each group of evacuees, to monitor doses as the evacuation proceeds.

f) At the off-site assembly area the C&RP Technician will be responsible for evacuee dosimetry and contamination control. Typical duties would include the establishment of contamination control areas; surveys of personnel, autos, and other items; decontamination as required; collection and reading of pocket dosimeters; collection of personnel dosimetry devices; and necessary record keeping. Surveys, decontamination techniques, release levels, etc., shall be in accordance with applicable radiation control procedures contained in the plant manual.

g) If evacuated non-essential personnel arriving at the assembly areas are contaminated, actions will be taken to decontaminate the evacuees and to prevent the spread of contamination. Equipment and supplies, along with generalized instructions, necessary to perform these actions are contained in the two decontamination showers located at the PG&E Information Center. The showers are stocked with soap, shampoo, towels, clothing, and other decontamination supplies.

h) The evacuation team leader is responsible for personnel accountability, communication with the Evacuation Coordinator and all other activities at the off-site assembly area.

6.4.4 Use of On-Site Protective Equipment and Supplies

Certain protective measures may be utilized to reduce the exposure to emergency workers.

6.4.4.1 Respiratory Protective Equipment

The quantities and types of respiratory protective equipment available for an emergency are discussed in Section 7. Respirators for routine plant use are also available for emergency use.

Before an emergency worker may use a respirator, prerequisite requirements for respirator training, fitting and medical surveillance must be satisfied. If all respirator program requirements are not satisfied, no credit should be taken for the respirator when estimating exposure reduction prior to exposure (i.e., respirator Protection Factor = 1).

It is the responsibility of the Radiological Advisor or Radiological Manager to determine when respiratory protective equipment use is appropriate, and to select the correct equipment for the expected radiological conditions. The use of respirators should consider maintaining TEDE ALARA for the individual worker; respirators should not be used exclusively for reducing radioiodine or lens of the eye exposure.

6.4.4.2 Protective Clothing

Protective clothing is maintained on-site for routine use and is available in sufficient quantities for use during emergencies.

Protective clothing provides minor protection against penetrating external radiation sources, but is intended to keep contamination off the clothes and skin of individuals and to control the spread of contamination. Protective clothing should be worn when entering known or potentially contaminated areas and should be removed upon exiting.

6.4.4.3 Thyroid Blocking Agent

Stable potassium iodine (KI) tablets are stockpiled and maintained at various on and off-site locations for distribution to emergency workers for emergencies involving significant releases of radioiodine. KI protects an individual's thyroid from airborne radioiodines by blocking the thyroid with stable iodine prior to or during exposure. Since it is an FDA approved drug, the Site Emergency Coordinator or the Emergency Director, with advice from the Radiological Advisor or the Radiological Manager, shall determine when the issue for use of KI would be appropriate.

6.4.4.4 Emergency Dosimetry

An ample supply of dosimetry, both self-reading and various types of thermoluminescent dosimeters (TLDs) are available at the Radiologically Controlled Area (RCA) access control point, ready for immediate issue to emergency workers.

Supplemental emergency dosimetry is stored in kits at various on and off-site locations. The purpose of the kits is for issuing dosimetry quickly to emergency workers that:

- are not RCA qualified, or
- need high range self-reading dosimetry, or
- cannot easily gain access to normal dosimetry storage areas.

Equipment for reading TLDs is available on-site. During an emergency, individuals that routinely operate the TLD reader will either be on-site or are available to be called in to provide the capability of reading TLDs within a few hours.

6.4.5 On-Site Contamination Control Measures

Diablo Canyon's contamination control program consists of radiation control standards that specify measures to minimize the potential for personnel contamination and the spread of contamination. These standards specify criteria for surveys, the establishment of contamination control areas and acceptable surface contamination levels. In the event of an emergency, these same criteria would be used to determine which additional areas of the site would require access control measures. Likewise, these criteria would be used to determine when area and equipment could be returned to normal use.

To assure on-site personnel do not receive excessive exposure from the ingestion pathways, drinking water and food supplies that have been within the boundary of a RCA, should not be consumed.

Equipment and supplies necessary to establish contamination control areas, and for the decontamination of equipment, areas, or personnel are routinely stored in the Auxiliary Building. Additional supplies are stored at the Learning Services Building and various off-site locations. Decontamination supplies are at the PG&E Information Center.

6.4.6 Alerting Off-Site Personnel

Off-site emergency support personnel are alerted by telephone or radio of emergency events and situations as discussed in Section 6.1.

1) Alerting the general public is the responsibility of local governmental authorities. Specifically, it is the responsibility of the SLO County Emergency Organization which is headed by the County Administrator in the role of County Emergency Services Director under the advisory direction of the County Board of Supervisors.

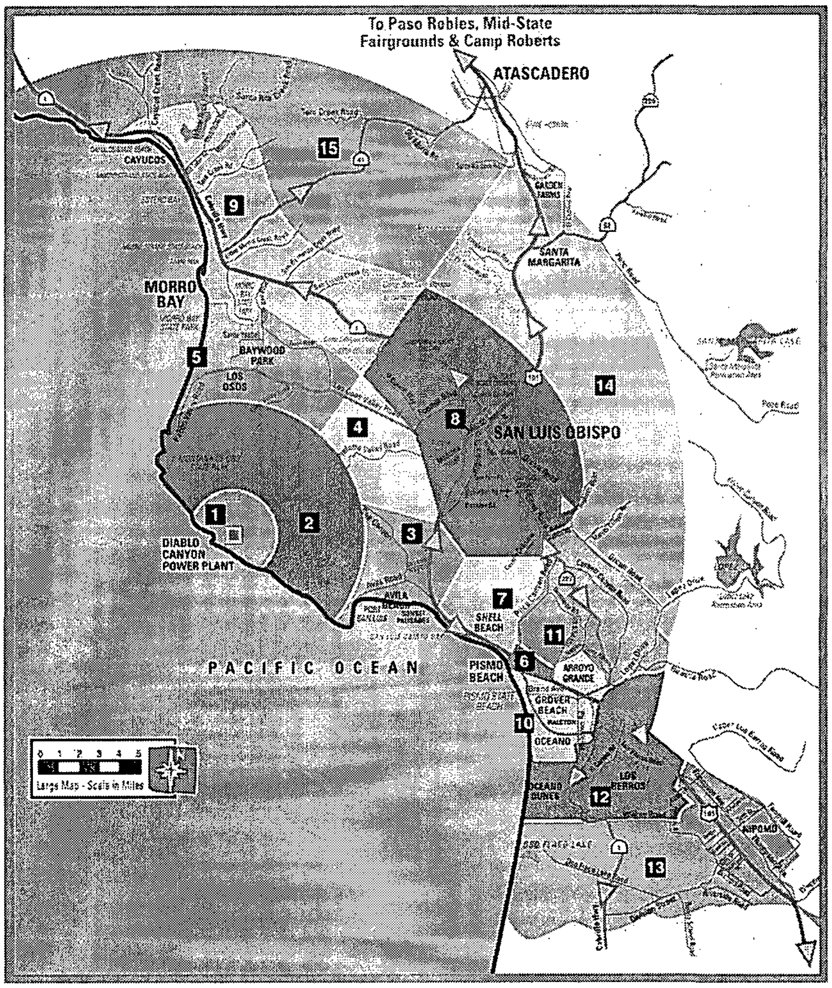
2) The lead agency for implementing the public alerting process is the SLO County Sheriff's office.

3) The Early Warning System is supplemented by special provisions for certain segments of the public as described in the SLO County Emergency Plan. Provisions are included in the plan for the Sheriff's office to promptly warn all persons in the Basic Emergency Planning Zone upon the determination general protective actions are necessary. For events of lesser significance, the timing, extent, and method of an emergency public warning (prior to the issuance of a normal media release) would be at the discretion of the County Emergency Organization.

6.4.7 Protective Actions for the General Public

The responsibility and authority for ordering protective actions for members of the public rests with the state and local emergency organizations. Maps used for describing the population distribution around the nuclear facility by evacuation zone is illustrated in the figures below.

Protective Action Zone and Public Education Zone



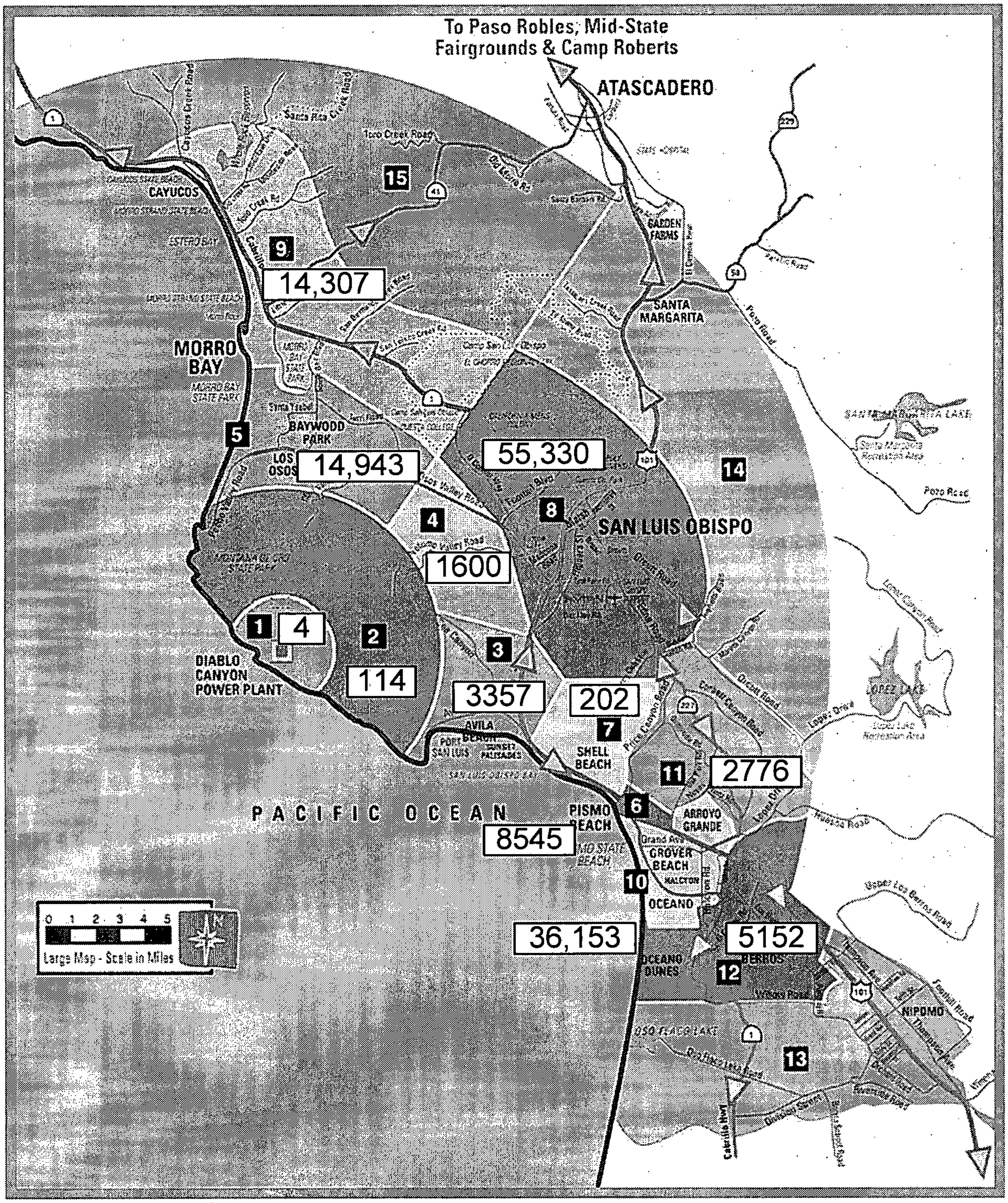
Protective Action Zones (PAZs)

1. 2 - mile
2. 6 - mile
3. Avila/San Luis Bay/See Canyon/Squire Canyon
4. Perfumo Canyon/Los Osos Valley
5. Baywood/Los Osos
6. City of Pismo Beach
7. Indian Knob/Price Canyon
8. San Luis Obispo Area
9. Morro Bay/Cayucos
10. Five Cities, Southern Portion
11. Orcutt Road, Lopez Drive, Route 227
12. Nipomo North of Willow Road

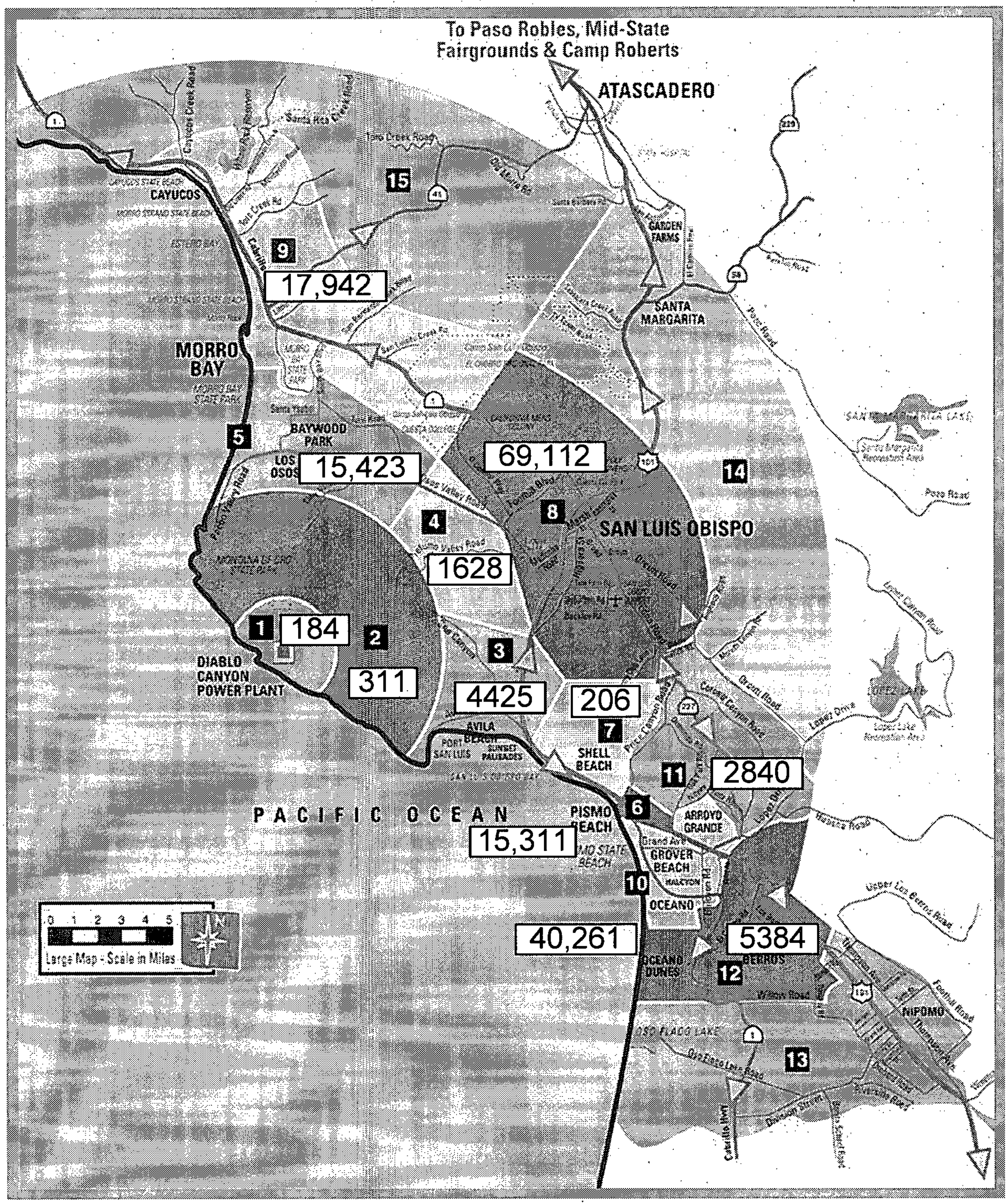
Public Education Zones (PEZs)

13. Nipomo
14. Cuesta Pass/Santa Margarita
15. Route 41/Cypress Mountain Dr.

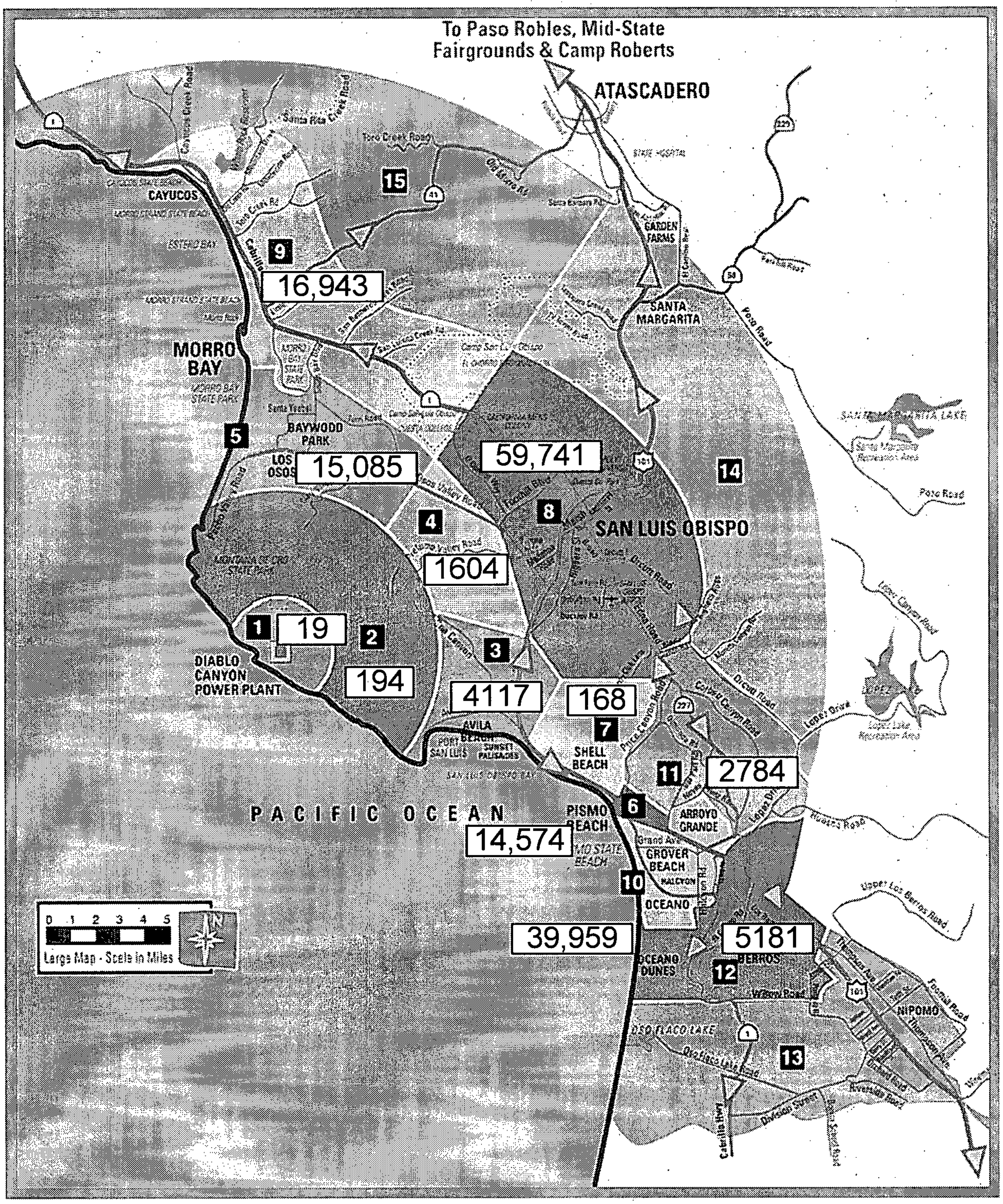
2002 Residential Population by Evacuation Zone



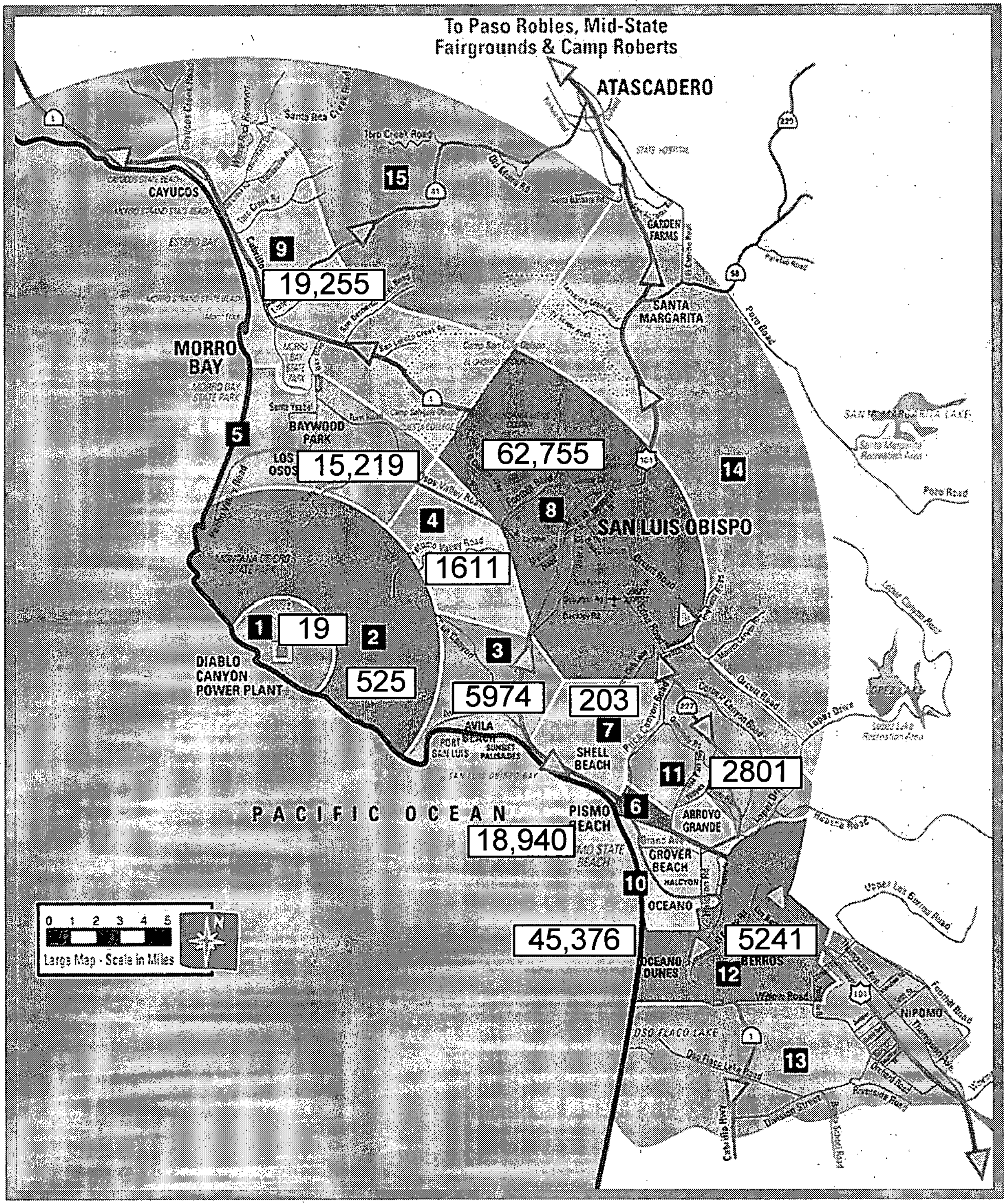
2002 Total Population Including Estimated Transients Normal Weekday



2002 Total Population Including Estimated Transients Nighttime



2002 total population including estimated, transients - Peak summer weekend



2002 Estimated Peak Populations and Evacuation Times by PAZ

Estimated Peak Populations and Evacuation Times by PAZ				
Protective Action Zone		Population	Estimated Cumulative Evacuation Time	
			Normal Weather	Adverse Weather
1	2-mile boundary	184	2.50	2.75
2	6-mile boundary	311	2.50	2.75
3	Avila Beach, Squire Canyon, See Canyon, San Luis Bay	4425	2.75	3.25
4	Los Osos Valley, Perfumo Canyon	1628	2.75	3.25
5	Baywood, Los Osos	15423	11.5	13.75
6	City of Pismo Beach	15311	11.5	13.75
7	Indian Knob, Price Canyon	206	11.50	13.75
8	San Luis Obispo	69112	11.50	13.75
9	Morro Bay, Cayucos	17942	13.00	15.50
10	Five Cities (Southern Portion)	40261	13.00	15.50
11	Orcutt Rd, Lopez Drive, Rt. 227	2840	13.00	15.50
12	Nipomo, North of Willow Rd.	5384	13.00	15.50

SOURCE: Reference 4

Public Protective Actions

The Shift Manager, Site Emergency Coordinator, or Emergency Director should recommend protective actions based on the following criteria.

4) Criteria Based Upon Nature of Emergency

Evacuation of some or all of the persons within protection action zones (PAZs) 1 and 2 may be recommended for any GENERAL EMERGENCY situation regardless of whether or not any radioactive materials have been released from the plant. In situations of a fast breaking event or security related event, sheltering all persons within appropriate PAZs may be recommended to increase their heightened awareness and readiness should immediate protective actions become necessary.

When plant conditions warrant (i.e., a release is imminent or occurring) a recommendation to take immediate actions, such as selective or general sheltering or evacuation may be made by the plant staff. Precautionary actions or general evacuation recommendations will be made by a joint assessment group as described in the County Emergency Plan and would be recommended by the plant staff.

5) Criteria Based Upon Public Exposure

Insofar as possible, evacuation of members of the general population should be carried out to prevent persons from receiving doses in excess of those listed below.

PAGS for the Early Phase of a Nuclear Incident
(Reference EPA 400-R-92-001)

PROTECTIVE ACTION	PAG (Projected Dose)	COMMENTS
Evacuation (or sheltering[a])	1-5 rem[b]	Evacuation (or, for some situations, sheltering[a]) should normally be initiated at 1 rem. Further guidance is provided in Section 2.3.1 of EPA 400
Administration of Stable Iodine	25 rem[c]	Requires approval of State medical officials.

[a] Sheltering may be the preferred protective action when it will provide protection equal to or greater than evacuation, based on consideration of factors such as source term characteristics, and temporal or other site-specific conditions (see Section 2.3.1 of EPA 400).

[b] The sum of the effective dose equivalent resulting from exposure to external sources and the committed effective dose equivalent incurred from all significant inhalation pathways during the early phase. Committed dose equivalents to the thyroid and to the skin may be 5 and 50 times larger, respectively.

[c] Committed dose equivalent to the thyroid from radioiodine.

a) Evacuation Routes

Evacuation routes are dependent on the meteorological conditions at the time of the accident. The meteorological conditions at the Diablo Canyon Power Plant site are very strongly influenced by the local topography. The Irish Hills which run approximately northwest-southeast redirect most onshore windflows in these directions. The diurnal nature of the California coastal meteorology also has a strong influence by causing very frequent weak northerly and easterly offshore drainage winds during the night and early morning hours.

Because of the unusual meteorological characteristics of the Diablo Canyon site, four predominant wind conditions will be used in discussing evacuation routes and procedures taken by outside agencies. These conditions are as follows:

Northwest winds -- These are predominantly daytime winds which occur during fair weather and frequently are very strong.

Southeast winds -- These winds generally are associated with storm conditions and early morning drainage flow. They increase in frequency during the wet winter months.

Offshore northeast winds -- These are predominantly night or early morning winds that are usually weak. Occasional strong offshore winds occur between winter storms where inland high pressure systems dominate the weather.

Onshore southwest winds -- These are the least prevalent winds at Diablo Canyon, occurring less than 5 percent of the time. Onshore winds are highly localized and seldom persist for more than an hour or two.

b) Evacuation Procedures

Evacuation of members of the general public is the responsibility of the County Emergency Organization, working in conjunction with the State Office of Emergency Services, and will be carried out in accordance with their prearranged plans. See the SLO County/Cities Nuclear Power Plant Emergency Response Plan for descriptive text and maps describing evacuation routes, evacuation areas, relocation centers and shelter areas. The general steps to be followed in the event an evacuation is required are as follows:

(1) Based upon plant conditions, on-site and off-site measurements, and meteorological data, the Sheriff's Department will be instructed by the County Emergency Services Director to take protective actions which may include selected or full evacuation. The area to be evacuated, the evacuation routes, and shelter locations will also be agreed upon by the County Emergency Organization Command Group. The Site Emergency Coordinator or Advisor to the County Emergency Organization after EOF activation will keep the County informed of pertinent information regarding the company's evaluation of existing conditions.

(2) The Sheriff's Department assisted by other response agencies, is to carry out the evacuation in accordance with established procedures.

(3) Reentry into the evacuated areas is to be prevented until it is determined that radiological conditions will permit unrestricted access.

c) Evacuation Time Estimates

Studies conducted by demography specialists (see Reference 4) provide information on various evacuation scenarios that could take place as a result of evacuation of the Basic Emergency Planning Zone (BEPZ). A general conclusion for the time required to totally evacuate the BEPZ was 4-1/2 to 10 hours. This applies to normal road conditions. Factors which could increase evacuation times would include time of day (daytime), degraded weather/visibility, and road destruction.

The scenario development and conditions leading to the time estimates, identified by the specialty studies referred to earlier, have been made available to state and county officials for use in their preparation of the emergency response planning documents.

Protective actions to be ordered by county authorities are summarized in the SLO County/Cities Nuclear Power Plant Emergency Response Plan. PG&E cooperates with county and state officials to ensure the county plan reflects appropriate guidelines on how the time estimates will ultimately be used to determine the protective actions to be taken off-site.

Responsibility for ordering protective actions by the public is legally the ultimate responsibility of local government. PG&E will act in an advisory capacity, giving technical assessments of the conditions at the plant and the probabilities for a potential off-site release as well as other pertinent information. This information, along with PG&E's recommended protective actions, will be assessed by responsible county officials in determining appropriate actions to be taken.

6.4.8 Off-Site Contamination Control Measures

The responsibility for ordering and conducting off-site contamination control actions rests with the SLO County Emergency Organization. However, PG&E is prepared to work with the SLO County and other participating governmental agencies to formulate and implement an appropriate program, if required.

6.4.9 Emergency Personnel Exposure

During an emergency, circumstances may dictate personnel receive exposures in excess of the 10 CFR 20 limits. Some examples of the circumstances would be lifesaving actions or other assessment or corrective actions that would serve to mitigate the consequences of the emergency.

During an emergency and prior to arrival of the Emergency Director, the Site Emergency Coordinator can authorize emergency exposure in excess of 10 CFR 20 limits. These emergency exposure limits are described in EPA 400. The Emergency Director assumes this responsibility after he takes his position at the EOF. Emergency workers may receive doses as indicated below.

Guidance on Dose Limits for Workers Performing Emergency Services
(Reference EPA 400-R-92-001)

Dose Limit[a] (rem)	Activity	Condition
5	all	
10	Protecting valuable property	Lower dose not practicable
25	Life saving or protection of large populations	Lower dose not practicable
>25	Life saving or protection of large populations	Only on a voluntary basis to persons fully aware of the risks involved (See Tables 2-3 and 2-4 of EPA 400)

[a] Sum of external effective dose equivalent and committed effective dose equivalent to nonpregnant adults from exposure and intake during an emergency situation. Workers performing services during emergencies should limit dose to the lens of the eye to three times the listed value and doses to any other organ (including skin and body extremities) to ten times the listed value. These limits apply to all doses from an incident, except those received in unrestricted areas as members of the public during the immediate phase of the incident (see Chapters 3 and 4 of EPA 400).

6.4.10 Decontamination

A decontamination shower is located near the access control area of the auxiliary building. Two off-site decontamination showers are located at the Information Center. Supplies include solid waste disposal supplies for contaminated clothing, personnel decontamination supplies, replacement clothing, and other related miscellaneous items.

All radiation protection personnel and licensed operators are trained in decontamination techniques as part of their radiation protection training.

6.4.11 Medical Transportation

Arrangements for medical transportation are discussed in Sections 5 and 7.

6.4.12 Medical Treatment

The company retains a number of physicians, hospitals and ambulances throughout its service area on a medical panel. The panel in the vicinity of Diablo Canyon is given below. French Hospital in San Luis Obispo, San Luis Ambulance, and Marion Medical Center in Santa Maria have agreements with the company for handling accidents involving radioactive contamination.

Physicians, Ambulances, and Hospitals Serving the Immediate Area around Diablo Canyon

Ambulances

NAME	REMARKS
San Luis Ambulance Service	Radiation Exposure Patients

Hospitals

NAME	REMARKS
French Hospital	Radiation Exposure Patients - External Defibrillation Equipped
Marian Hospital	

Physicians

NAME	REMARKS
Doctor's Med Stop	Industrial Injury Treatment
Family Medical Center	Industrial Injury Treatment
Paul Georghiou, M.D.	Medical/Radiation Consultant

6.5 CROSS REFERENCE TO NUREG-0654

NUREG 0654	DCPP Emergency Plan	NUREG 0654	DCPP Emergency Plan
A.1.e	6.1.7	J4	6.3, 6.4.3
B.2	6.1.5	J5	6.4.2
B.6	6.1	J.6.a	6.4.4.1
E.2	6.1, 6.1.7, 6.1.8	J.6.b	6.4.4.2
E3	6.1.8	J.6.c	6.4.4.3
E7	6.1.8	J7	6.1, 6.2.4, 6.4.6, 6.4.7
F.1.e	6.1	J8	6.4.7
H.4	6.1, 6.1.7, 6.1.8	J.10.a	6.4.7
H.11	6.4.5, 6.4.10	J.10.b	6.4.7, 6.4.3
I.2	6.2.3, 6.3	J.10.c	6.4.6
I.3.a	6.3	J.10.m	6.4.6
I.3.b	6.2.4, 6.3	K.1.a to g	6.4.9
I.4	6.3	K.2	6.4.9
I.6	6.3	K.3.a and b	6.4.4.4
I.7	6.3	K.5.a and b	6.4.5, 6.4.10, 6.4.12
I.8	6.3	K.6.a to c	6.4.5
I.9	6.3	K.7	6.4.5, 6.4.10, 6.4.12
I.10	6.2.4, 6.3	L.1	6.4.12
J.1.a to d	6.1.3, 6.1.6, 6.4.1	P.7	6.5
J2	6.4.1, 6.4.3	P.8	6.5
J3	6.4.1, 6.4.4, 6.4.5		

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7. EMERGENCY FACILITIES AND EQUIPMENT

If corrective measures are to be promptly initiated in an emergency situation, it is important required emergency equipment and facilities be readily available. To the maximum extent possible, normal plant equipment and controls will be used to mitigate the consequences of an accident. In some instances, special emergency equipment and facilities have been provided. This section describes this special equipment in conjunction with normal plant equipment which has particular application in an emergency.

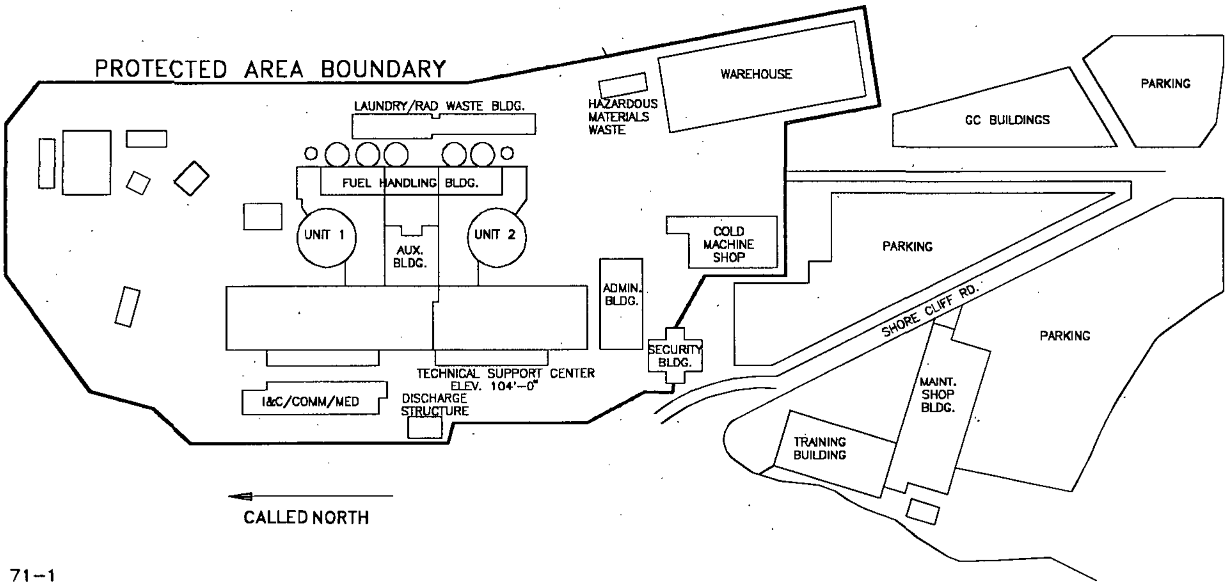
The Emergency Plan provides for a number of on-site and off-site facilities intended for use as accident management centers, and personnel staging and planning areas. These facilities are discussed briefly in the following paragraphs.

7.1 EMERGENCY RESPONSE FACILITIES

7.1.1 Control Room

7.1.1.1 *Location and Description*

The Control Room is common to Units 1 and 2 and is located at the 140' elevation of the Auxiliary Building as shown below. The Shift Manager's office is located adjacent to the Control Room. The Control Room has lavatory and kitchen facilities.



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7.1.1.2 Emergency Function

Prior to the time the Technical Support Center (TSC) is activated (and throughout the course of an emergency in which the TSC is not activated) the Control Room will serve as the headquarters for the Site Emergency Coordinator. All on-site activities are directed from this location, and all communication with off-site agencies will originate from the Control Room. The Control Room has the necessary equipment and instruments to perform accident assessment work involving possible or actual radiological releases and fuel barrier damage.

Following activation of the TSC, overall control of on-site activities will be transferred to the TSC. If the TSC is activated before the EOF, then the TSC will assume responsibilities for communications with off-site agencies until relieved by the EOF. The Control Room will then be headquarters of the on-site Operations Coordinator, and the major Control Room activity will be operation of plant equipment to mitigate the consequences of the emergency.

The Control Room also serves as the backup to the TSC should the latter be unavailable.

7.1.1.3 Habitability Objectives

The Control Room is designed to be habitable throughout the course of a design-basis accident. The Control Room shielding is designed to limit the integrated doses under post-accident conditions to 2.5 rem to the whole body.

The Control Room is provided with a Design Class I Criteria Ventilation System. The design of the system includes provisions for:

1) Protection from smoke generated inside or outside the Control Room area.

2) Protection from airborne radioactivity outside the Control Room and provisions for cleanup of activity trapped in the room.

3) Protection from airborne toxic gas outside the Control Room.

4) Provisions for limiting carbon dioxide buildup inside the Control Room during periods when airborne contaminants prevent use of outside makeup air.

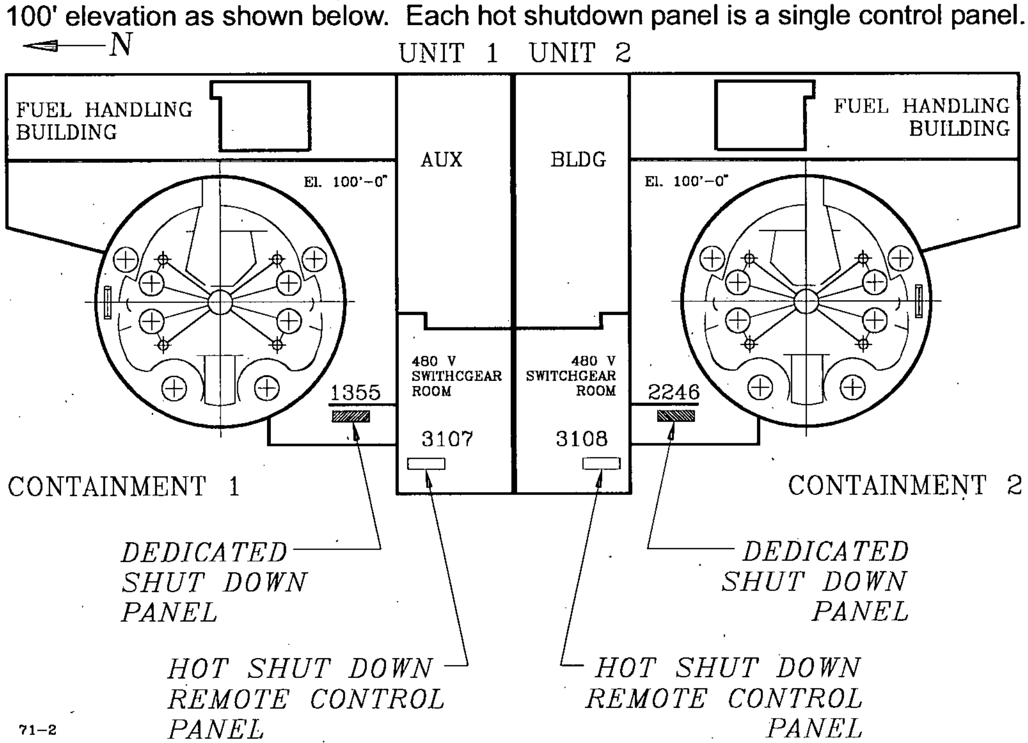
7.1.1.4 Special Equipment

The Control Room is the most completely equipped location in the plant in terms of provisions to monitor the status of plant systems and equipment. The Control Room has complete communication capability; as well as access to meteorological, seismic, and radiological monitoring data.

7.1.2 Hot Shutdown Panel

7.1.2.1 Location and Description

Each unit is provided with a hot shutdown panel located in the Auxiliary Building at the 100' elevation as shown below. Each hot shutdown panel is a single control panel.



7.1.2.2 Emergency Function

The hot shutdown panel contains the essential indicator and controls to maintain a unit in hot standby condition for an extended time period. The hot shutdown panel is primarily intended to be used for a situation in which smoke or toxic gas makes the Control Room temporarily uninhabitable. In such a circumstance, the operators are instructed to trip the reactor as they leave the Control Room and proceed to the hot shutdown panel. From this location, the unit can safely be maintained in the hot standby condition until the Control Room can be reentered.

Any occurrence requiring the use of the hot shutdown panel would also involve activation of the TSC. Overall emergency response actions, including off-site communications, would be handled from the TSC, where the Site Emergency Coordinator would establish his headquarters. A licensed operator would be stationed at the hot shutdown panel.

7.1.2.3 Habitability Objectives

The hot shutdown panels are not intended for use in radiological release type accidents. They are open to the room atmosphere in the "clean" (radiologically) portion of the Auxiliary Building. No special provisions have been made to assure habitability during radiological release emergencies.

7.1.2.4 Special Equipment

Hot Shutdown Panel Instrumentation and Controls are listed below. Each panel has a telephone and an emergency UHF radio for operations frequency.

Auxiliary Feedwater

1. AFW Pump discharge pressure
2. AFW flow indication
3. Turbine Driven AFW Pump Control Valve transfer switch
4. Turbine Driven AFW Pump Control Valve control switch
5. Turbine Driven AFW Pump Control Valve position indicator
6. Motor Driven AFW Pump Control Valve controller
7. Motor Driven AFW Pump Control Valve position indicator
8. Turbine Driven AFW Pump steam supply valve control switch
9. Turbine Driven AFW Pump steam supply valve transfer switch
10. Turbine Driven AFW Pump steam supply valve position indication
11. Motor Driven AFW Pump control switch
12. Motor Driven AFW Pump mode selector switch
13. Motor Driven AFW Pump START/STOP/LOCAL indicator

Auxiliary Saltwater

1. Auxiliary Saltwater Pump control switch
2. Auxiliary Saltwater Pump mode selector switch
3. Auxiliary Saltwater Pump START/STOP/LOCAL status

Chemical and Volume Control System

1. Centrifugal Charging Pump control switch
2. Centrifugal Charging Pump mode selector switch
3. Centrifugal Charging Pump START/STOP/LOCAL indicator
4. Boric Acid Transfer Pump transfer
5. Boric Acid Transfer Pump control switch
6. Boric Acid Transfer Pump ON/OFF indicator
7. Reactor Coolant Pump seal injection back pressure control valve controller
8. Centrifugal Charging Pump flow control valve controller
9. Reactor Coolant Pump seal injection back pressure control valve position indicator
10. Centrifugal Charging Pump flow control valve position indicator
11. Letdown Valve transfer switch
12. Letdown Valve control switch
13. Letdown Valve position indicator
14. Emergency borate valve transfer switch
15. Emergency borate valve control switch
16. Emergency borate valve position indicator
17. Emergency boric acid flow indicator
18. Volume Control Tank level indicator
19. Letdown flow indicator
20. Charging header flow indicator
21. Charging header pressure indicator
22. Reactor Coolant Pump Seal No. 1 pressure indicator

Component Cooling Water

1. Component Cooling Water Pump control switch
2. Component Cooling Water Pump mode selector switch
3. Component Cooling Water Pump START/STOP/LOCAL status

Containment Fan Coolers

1. Containment Fan Cooler transfer switch
2. Containment Fan Cooler control switch
3. Containment Fan Cooler ON/OFF status

Makeup Water

1. Condensate Storage Tank level indicator
2. Raw Water Reservoir level indicator

Reactor Coolant System

1. Pressurizer liquid temperature indicator
2. Pressurizer pressure indicator
3. Pressurizer level indicator (2)
4. Pressurizer heater transfer switch (2)
5. Pressurizer heater control switch (2)
6. Pressurizer heater breaker position indicator (2)
7. Pressurizer Power Operated Relief Valve emergency close switch

Steam Generators

1. Pressure indicator
2. Level indicator
3. Steam dump valve controller
4. Steam dump valve position indication

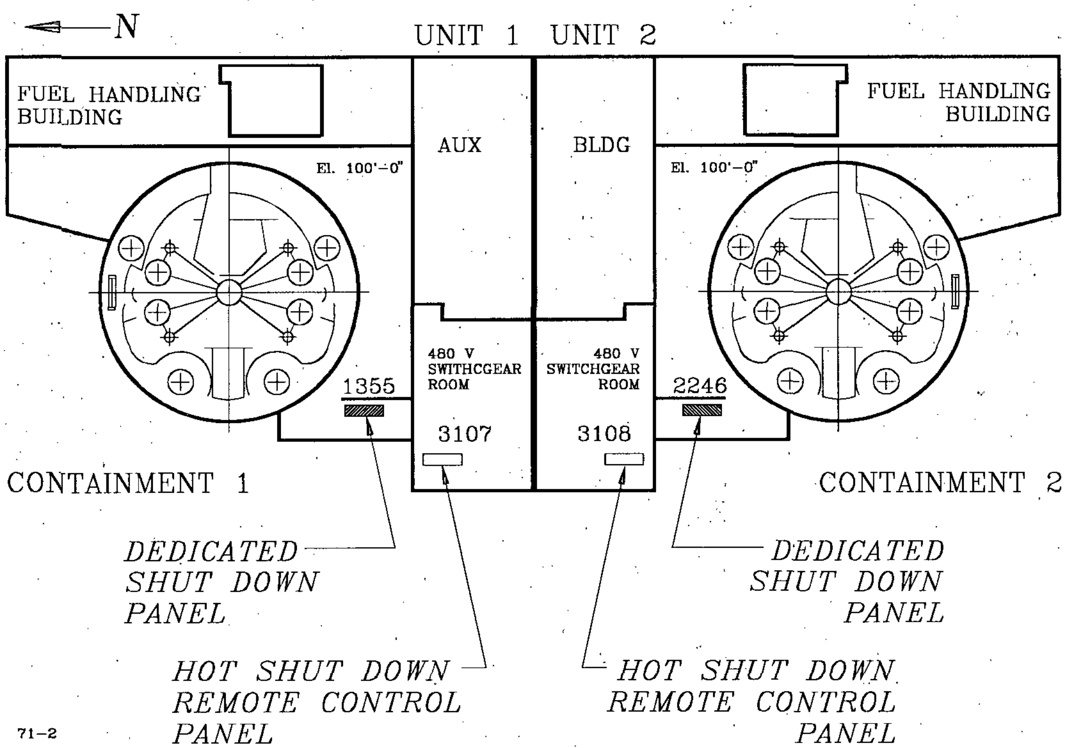
Other

1. Source Range neutron flux indicator (2)
2. Site emergency alarm switch
3. 4kV vital bus voltage indicator

7.1.3 Dedicated Shutdown Panel

7.1.3.1 Location

Each unit is provided with a dedicated shutdown panel located in the 100' elevation of the Auxiliary Building, as shown below.



7.1.3.2 Emergency Function

The dedicated shutdown panel in conjunction with the Hot Shutdown Panel is used if the unit must be taken from the hot shutdown condition to the cold shutdown condition from outside the Control Room. The dedicated shutdown panel contains sufficient instrumentation to follow and direct the cooldown operation and has controls for the pressurizer auxiliary spray control valve operation. The actual manipulation of other controls and valves would be done by operators at appropriate local stations.

Any occurrence requiring the use of the dedicated shutdown panel would also involve activating the TSC. Overall recovery actions, including off-site communications, would be handled from the TSC, where the Site Emergency Coordinator would establish his headquarters. An operator would be stationed at the dedicated shutdown panel.

7.1.3.3 Habitability Objectives

The dedicated shutdown panels are not intended for use in radiological release type accidents. They are open to room atmosphere. No special provisions have been made to assure habitability during radiological emergencies.

7.1.3.4 Special Equipment

Instruments found on each dedicated shutdown panel are listed below. PG&E phone jacks are located near each panel. Portable, hand held radio units may also be used for communications if required.

Steam Generators	
1. Level indicator (each steam generator)	
Reactor Coolant System	
1. Temperature indicator - RCS Loop 1	3. RCS Loop 4 pressure indicator
2. RCS Loop 1 Temperature selector switch	4. Pressurizer level indicator
Chemical and Volume Control System	
1. Auxiliary spray valve transfer switch	
2. Auxiliary spray valve control switch	

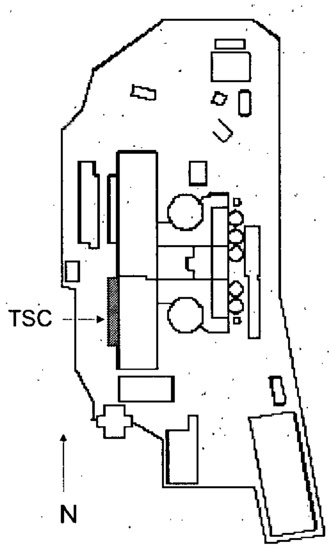
7.1.4 Technical Support Center (TSC)

7.1.4.1 Location and Description

The purpose of the TSC is to provide a facility separate from but in close proximity to the Control Room. The TSC has the capability to display and transmit plant status to personnel responsible for engineering and management support of reactor operations in the event of an accident. This separate facility is needed to house data gathering equipment and the personnel required to assist in an emergency, primarily to reduce Control Room congestion.

The TSC serves both Units 1 and 2 and consists of six rooms. It is located at elevation 104' on the west side of the Unit 2 Turbine Building. It occupies space created as a result of the exterior concrete buttress seismic modification of the turbine building. The thickness of the concrete walls required to enclose the TSC were largely dictated by radiation shielding considerations and the structure is designed to the Hosgri seismic criteria.

The figure below shows the location of the TSC within the plant.



The room layout and description from North to South is:

1) Command Center

   Desks, files and conference table provided for plant operations management, maintenance and technical staff personnel.

2) Operations Center

   Plant parameter data gathering and display equipment is provided for the use of technical staff in assessing the plant condition. The Emergency Response Facility Data System (ERFDS) is located in this room.

3) Computation Center

   Radiological and meteorological data gathering and display equipment and communications equipment is provided for the use of technical staff in assessing radiological conditions on and off-site.

4) Records Management and Reproduction

   Plant manuals, emergency plans and procedures, access to microfilm drawings and other records, and certain hard copy drawings are provided.

5) HVAC Room

   Heating, ventilating, and air-conditioning equipment for the TSC is located in this room.

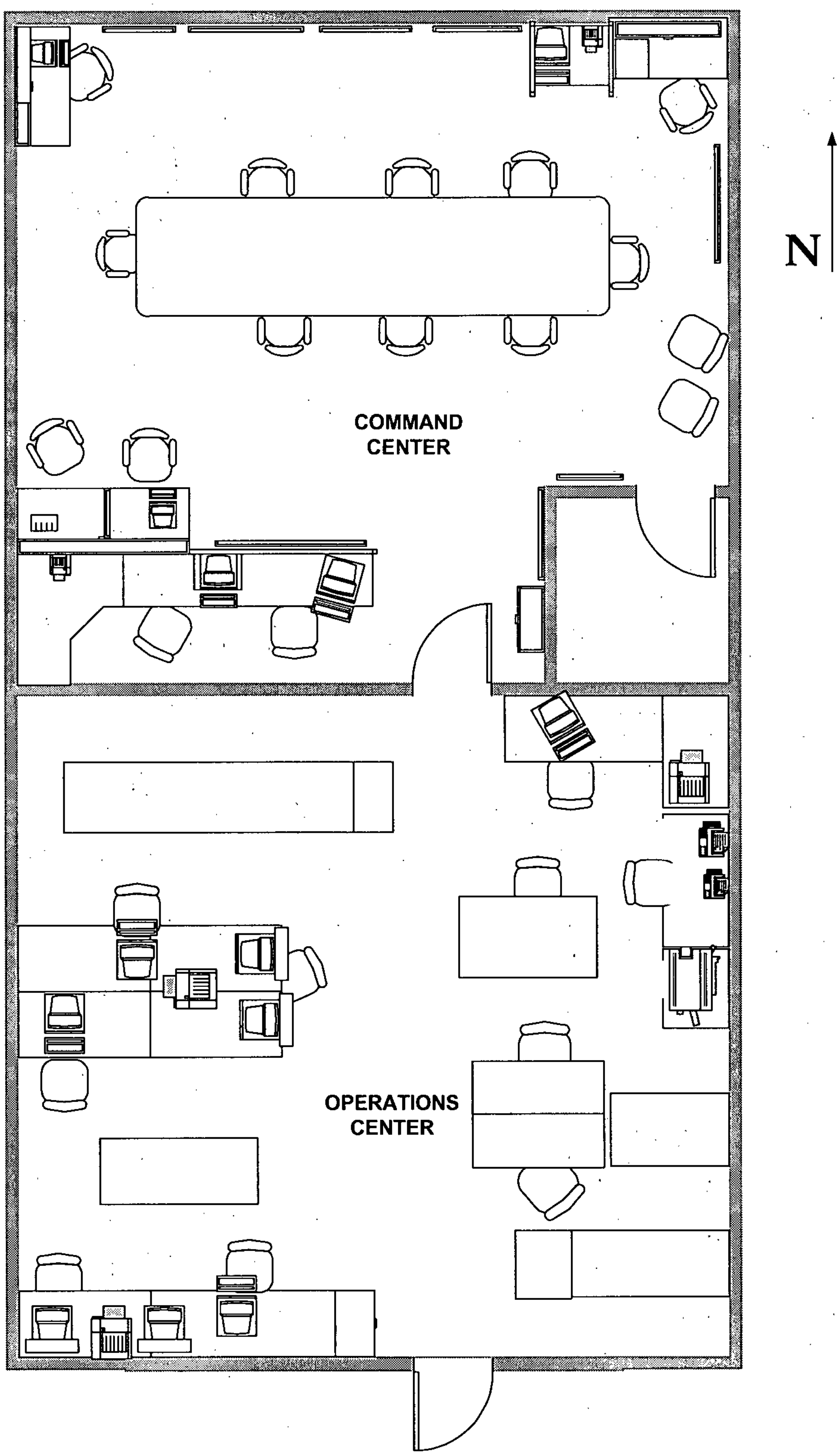
6) Laboratory (Radiological Counting Room)

   In a room adjacent to the TSC, radiological laboratory equipment for analysis of samples is provided.

   The radiological counting room is intended to be a backup location for this type of work in the event the normal counting room is unavailable.

   It is equipped with a multi-channel gamma ray spectroscopy system using a high-resolution intrinsic germanium detector.

The following figures show the general floor plan of the TSC.



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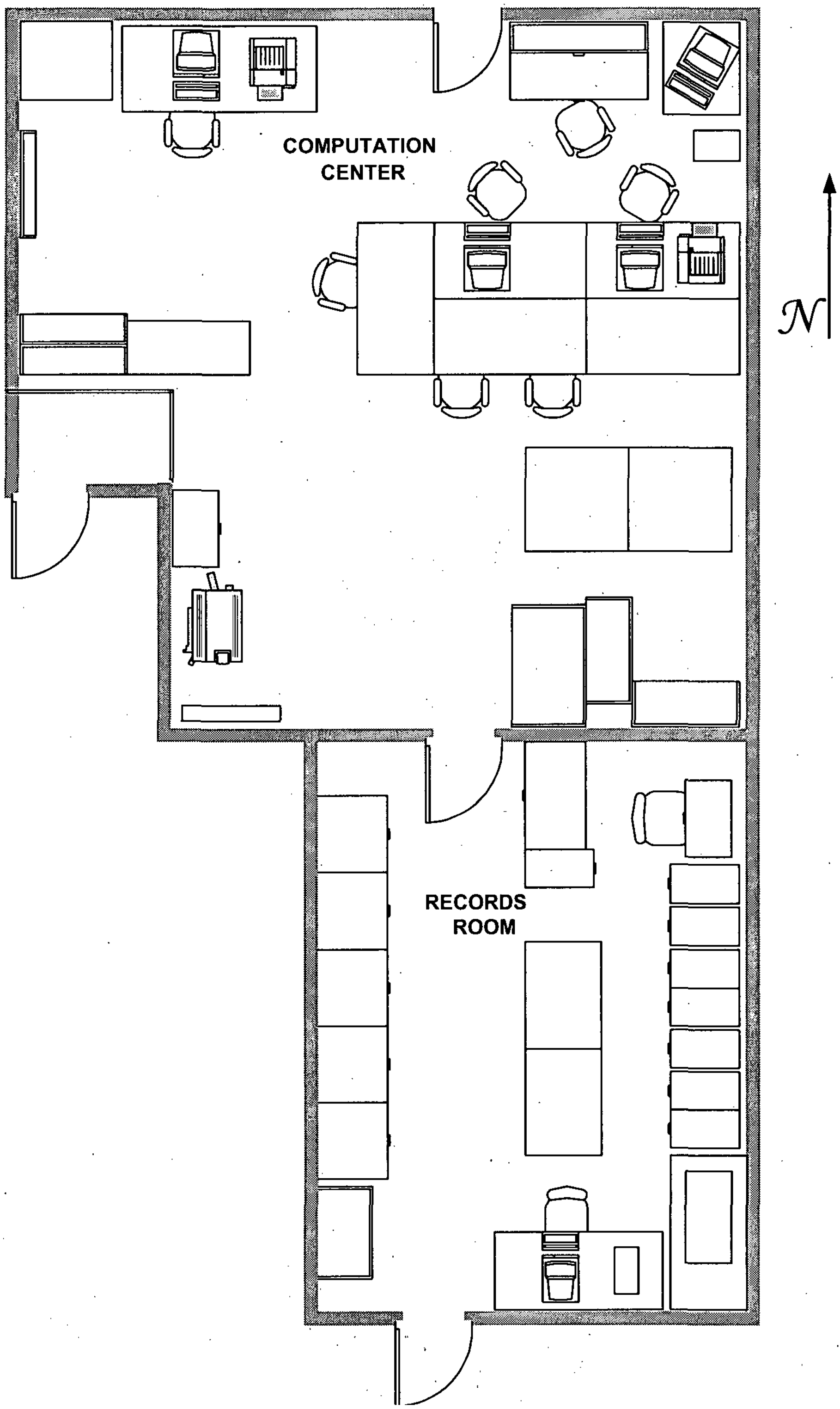
TSC Continued



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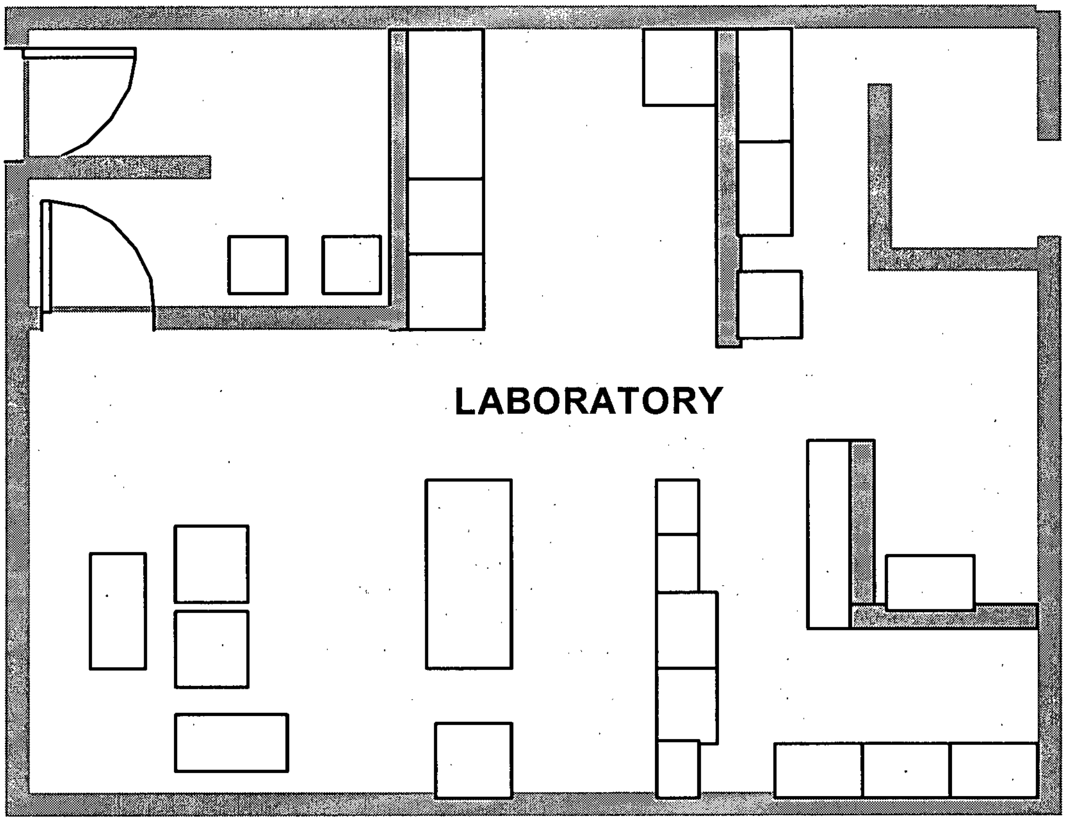
TSC Continued



7.1.4.2 Emergency Function

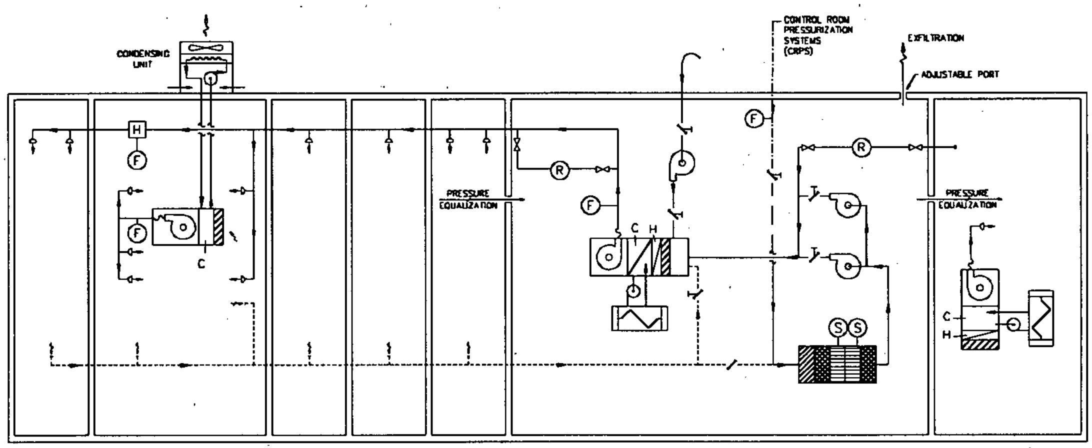
The TSC when activated serves as the headquarters for the Site Emergency Coordinator, Operations Advisor, Radiological Advisor, Communicator, and Engineering Advisor and their staffs throughout an emergency. Provisions have also been made for the establishment of an on-site NRC emergency team co-located in the TSC.

Following activation of the TSC, the overall on-site assessment and recovery programs will be directed from this location. In addition, communications with off-site emergency response locations will be handled through the TSC.

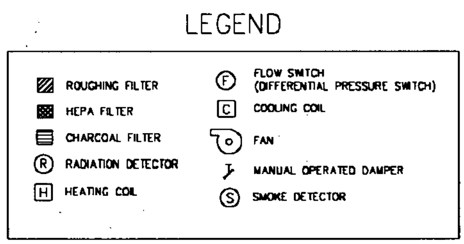
7.1.4.3 Habitability Objectives

The TSC is designed to be habitable throughout the course of a design basis accident. The outside walls, with steel bulkhead doors, form an airtight perimeter. The TSC shielding is designed to limit the integrated doses under post-accident conditions to 2.5 rem to the whole body consistent with the criteria for the Control Room. The TSC structure is designed to Seismic Class I criteria.

The TSC is provided with its own ventilation system. The ventilation system is shown below.



TECHNICAL SUPPORT CENTER HVACS



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Under accident conditions, the supply to the TSC is transferred to the Control Room pressurization system that maintains the TSC at a positive pressure. Intake air is conditioned and internally re-circulated through high efficiency particulate air (HEPA) and charcoal filters within the TSC. The pressurization air filtrates from the TSC to the outside atmosphere. The pressurization portion of the ventilation system, including the duct work, redundant ventilation fans and filter units for the TSC, are designed to Seismic Class I criteria. The fans are powered from 480-volt non-vital buses but can be transferred to a 480-volt vital bus on either unit. The air conditioning units are not designed as seismic structures and are powered from normal AC sources.

The TSC intake air is monitored by GM Detectors with alarm and control capabilities as part of the Control Room ventilation system. It also has area, particulate, iodine, and noble gas monitors with alarm capabilities, which may be backed up by portable equipment. The TSC is also provided with self-contained breathing apparatus and protective clothing for use in an emergency.

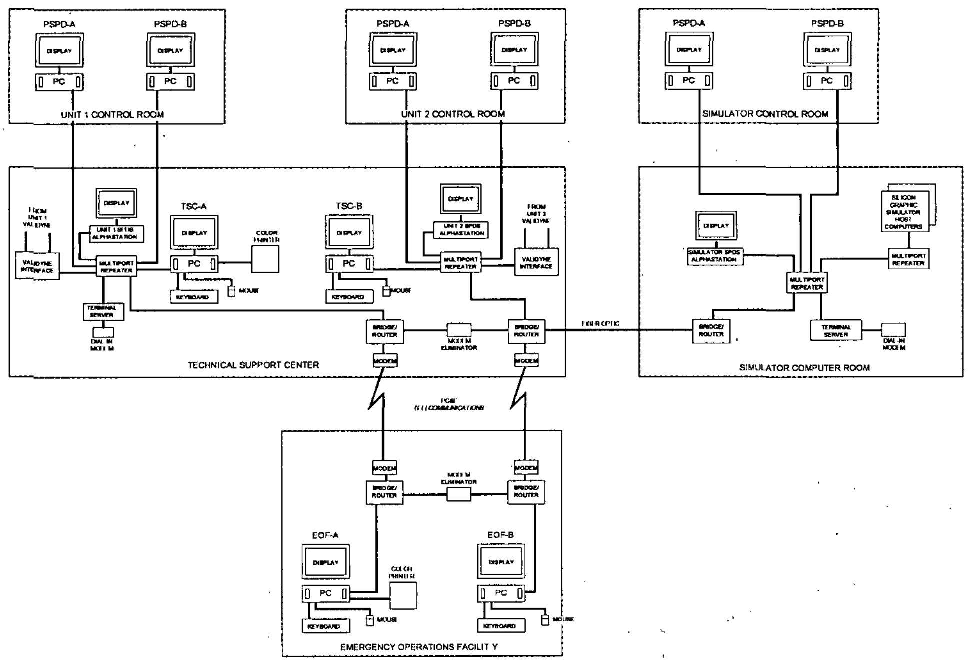
7.1.4.4 Special Equipment

1) Safety Parameter Display System (SPDS)

The SPDS was designed to the guidelines specified in NUREG-0696 and NUREG-0737 Supp. 1. It is part of the Emergency Response Facility Data System (ERFDS).

The SPDS for each unit is a computer-based system consisting of a data acquisition system, server computers, and display computers.

There are two high-resolution color SPDS monitors in the TSC. The displays available for the monitors allow TSC personnel to view plant parameters in real time ("SPDS display"), primary and secondary system mimics, and decision trees.



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Emergency Response Data System (ERDS)

The Emergency Response Data System is a direct near real-time electronic data link between a DCPP installed plant computer system and the Nuclear Regulatory Commission's Operations Center and Regional Office. This system provides for the automated transmission of a limited set of selected parameters and supplements the existing voice transmission over the NRC FTS telephone system. Activation of this system occurs at an Alert or higher emergency classification.

2) Communication

The TSC is provided with full radio and telephone communications capability.

3) Radiological Analysis

The TSC contains radiological laboratory equipment. The TSC is also tied into the emergency radiological monitoring network.

4) Plant Process Computer

The TSC is provided with a display terminal and a printer for each unit's Plant Process Computer (PPC). This provides the ability to monitor and print plant parameters acquired by the PPC.

5) Transient Recording System

The ERFDS recall functionality is provided by the Transient Recording System (TRS). The TRS information is available on any Plant Data Network computer in the TSC.

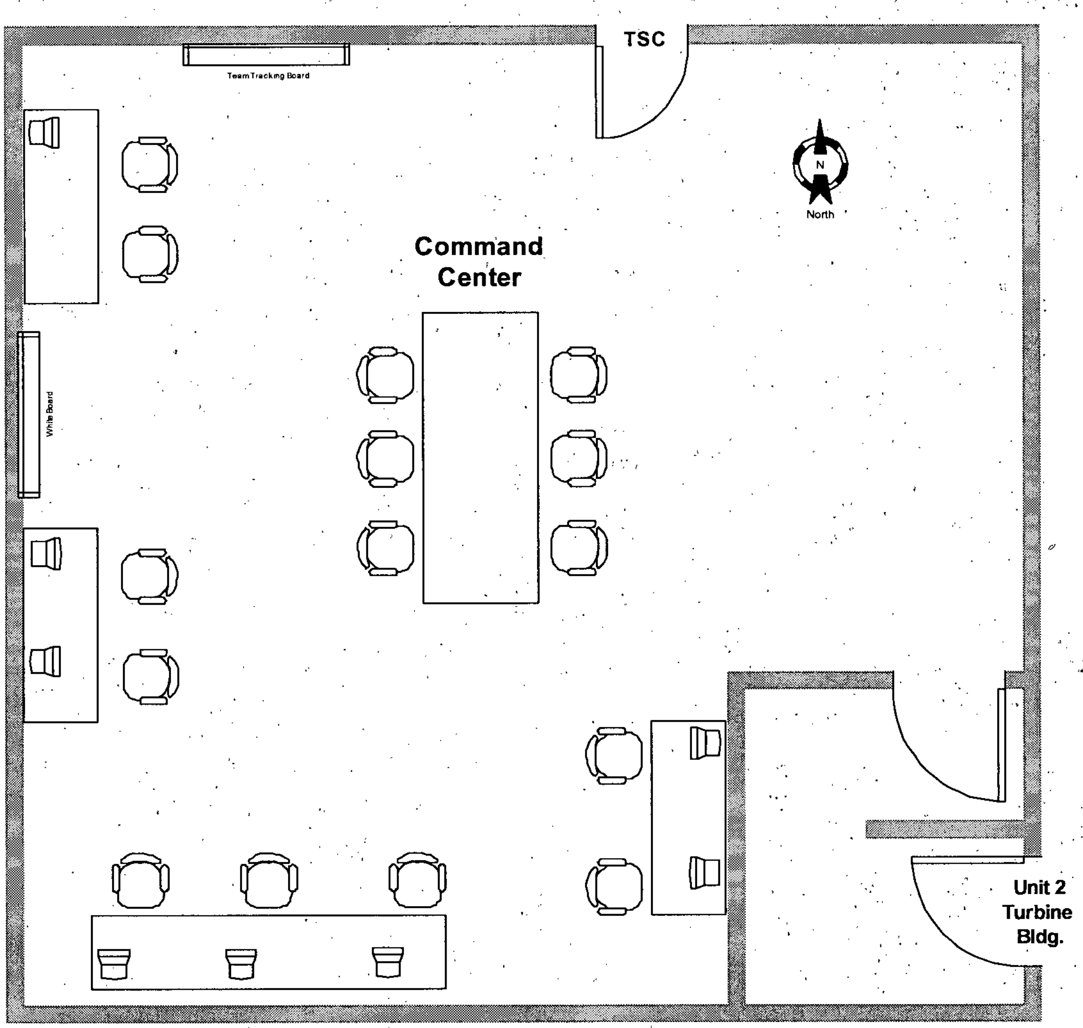
7.1.5 Operational Support Center

7.1.5.1 Location and Description

The Operational Support Center (OSC) provides locations functionally separate from the Control Room and Technical Support Center where designated support personnel assemble and await specific assignment during an emergency. The OSC command center is located in the buttress area on 104-foot elevation, adjacent to the west side of the Unit 2 Turbine Building and the south end of the TSC. Depending on the emergency events and plant conditions, personnel assigned to the OSC may be directed to assemble at the OSC command center, the 85' RCA Access Control, the site medical facility, the firefighters equipment storage area on the 140' elevation. OSC assembly areas serve as team dispatch locations and contain a variety of emergency support equipment immediately available for emergency use. The OSC command center is equipped with a dedicated tie line telephone extension to other facilities. Alternate backup locations for the OSC command center include:

- Elevation 140' Turbine Building northeast corner
- Administration Building (Room 215)

The general floor plan of the OSC command center is shown below.



7.1.5.2 Emergency Functions

Provides a location for staging and dispatching trained emergency workers for assignment to:

- Emergency maintenance, repair, and damage control.
- Fire fighting, search and rescue, and first aid.
- Emergency sampling of plant fluids.

The OSC also functions as locations of selected emergency response equipment and provide facilities for personnel decontamination.

The OSC is intended to eliminate congestion in the Control Room and TSC. This area has a supervisor assigned in the emergency organization.

7.1.5.3 Habitability Objectives

The OSC command and assembly areas are not required to have special provisions for minimizing radiation exposure. Consequently, personnel in these areas may be evacuated under certain emergency situations. If evacuation were necessary, personnel required for emergency response would be relocated to another area. Personnel not immediately essential to the on-site response may be assembled in the Learning Services Building or released.

7.1.5.4 Emergency Equipment and Supplies

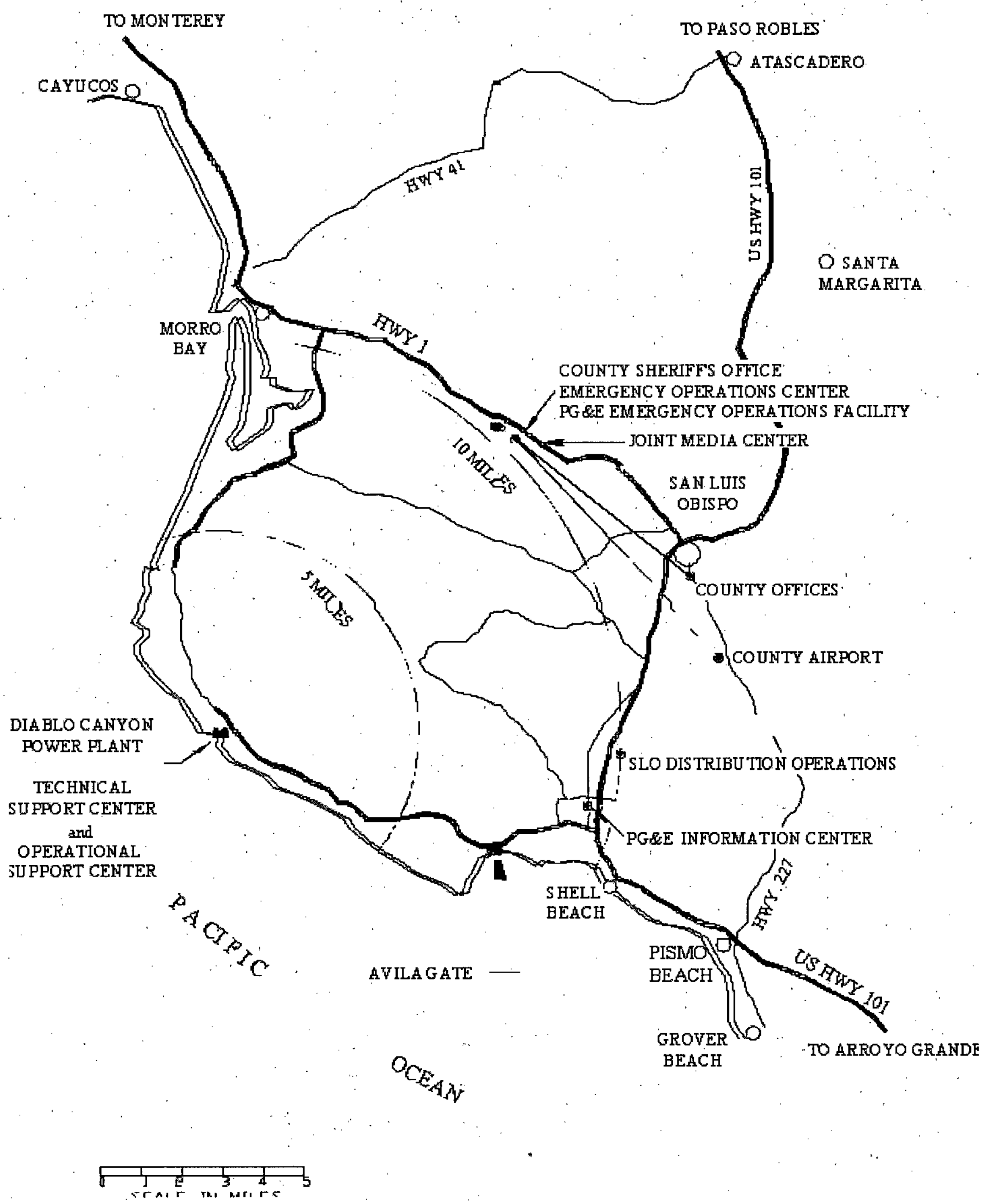
The OSC command center has a dedicated a tie line to the TSC and Control Room and has a CBX telephone with emergency facility priority access to commercial telephone circuits. In addition, there is access to portable radio equipment.

The OSC has ample supplies of respiratory protection equipment, protective clothing, monitoring instruments and other emergency response equipment.

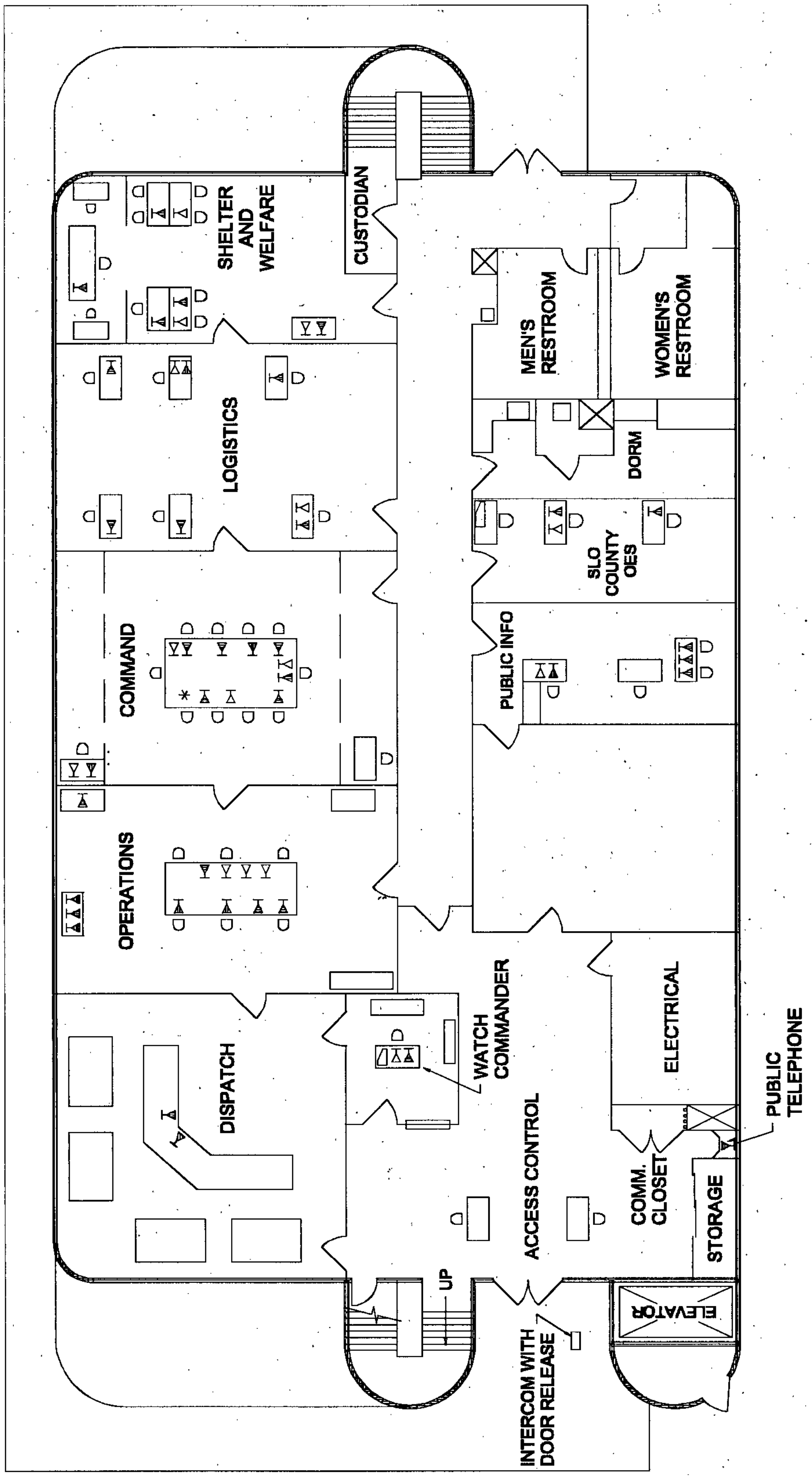
7.1.6 Emergency Operations Facility

7.1.6.1 Location and Description

The Emergency Operations Facility (EOF) and County Emergency Operations Center (EOC) are located approximately 11 miles northeast of the Diablo Canyon Power Plant. See figure below.



The San Luis Obispo County Sheriff's watch commander and dispatch center occupy this building on a 24-hour basis. The first floor of the building provides space for the County EOC, and the Sheriff's watch commander and dispatch center. See figure below.



FIRST FLOOR OF EOC / EOF BUILDING

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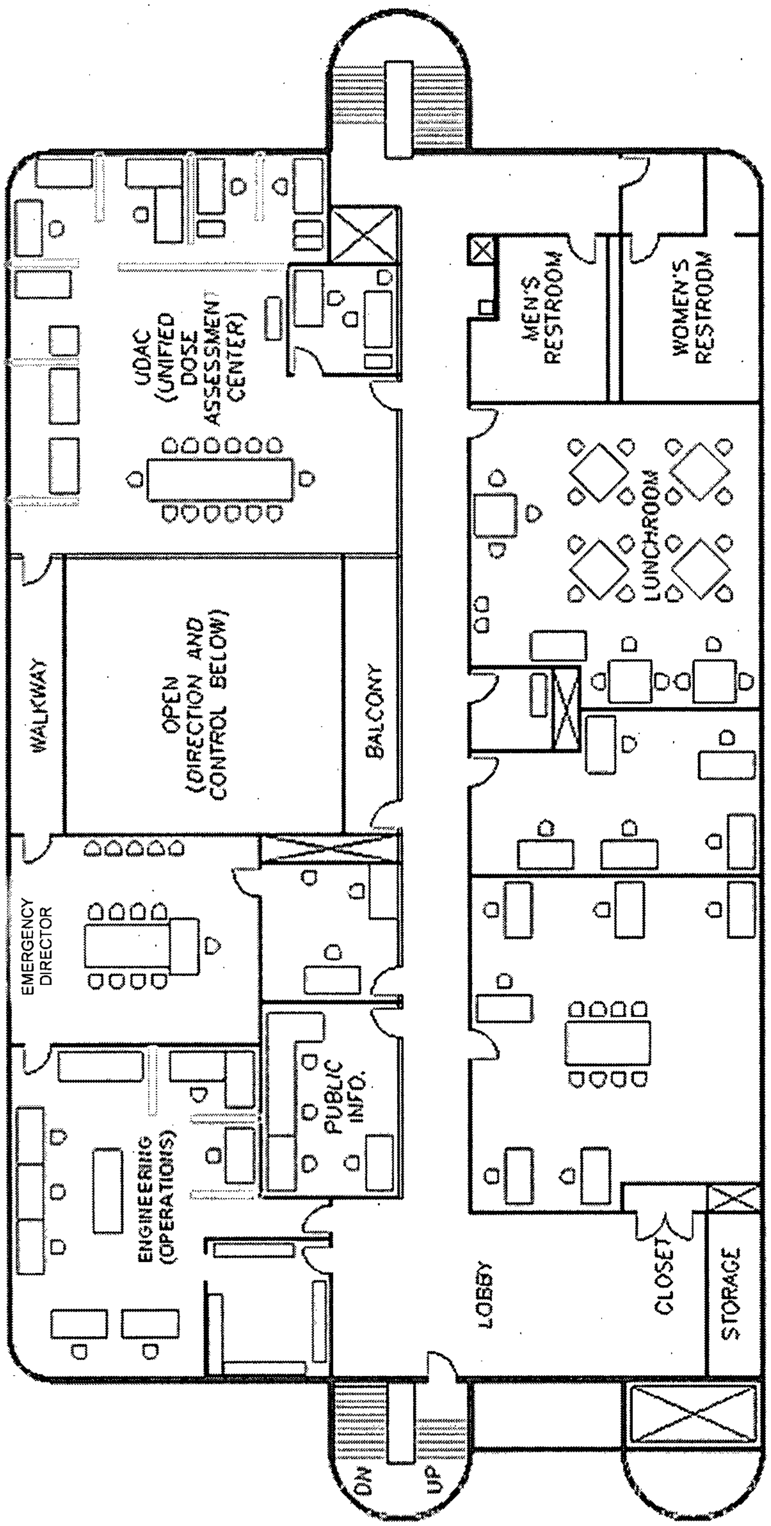
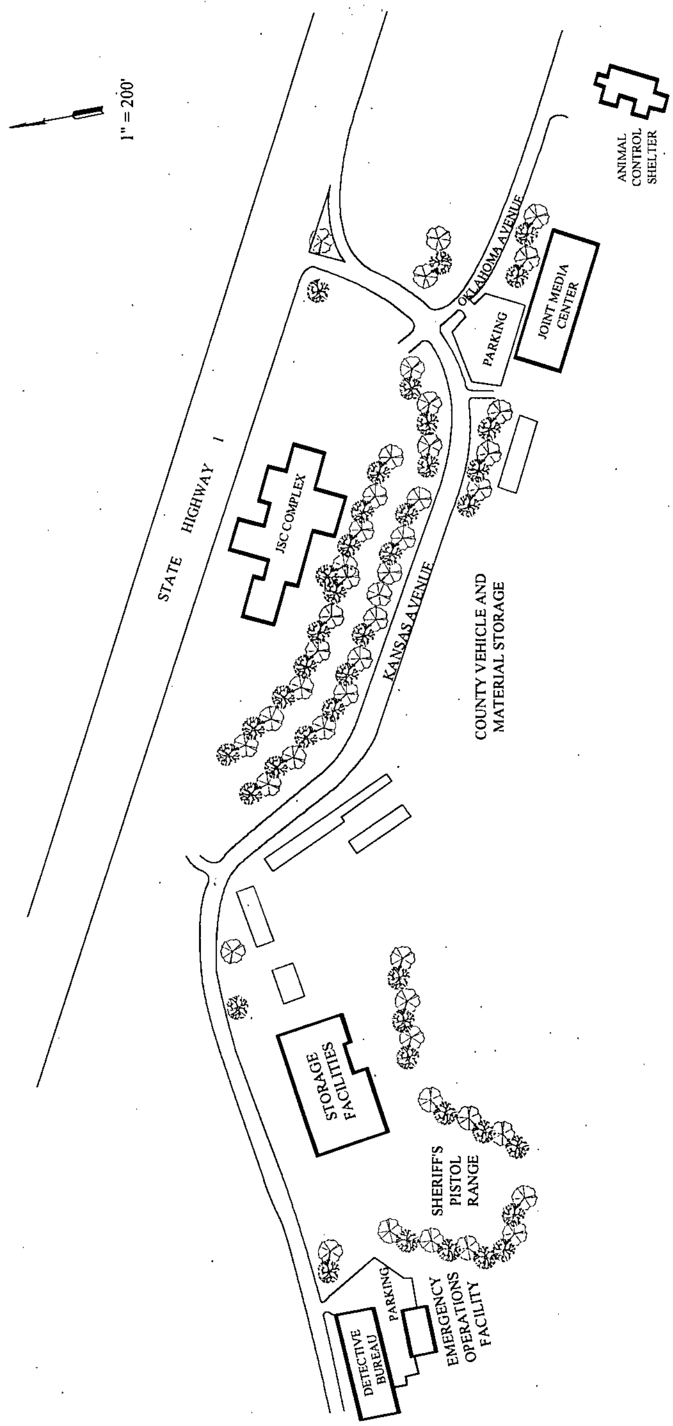
The second floor houses the EOF, the Unified Dose Assessment Center (UDAC) and office space for the State Office of Emergency Services (OES), the Nuclear Regulatory Commission (NRC), the Federal Emergency Management Agency (FEMA) and other responding Federal agencies. See figure below.



SECOND FLOOR OF EOC / EOF BUILDING

The JIC off Highway 1 on Kansas Avenue is approximately 1/4 mile from the EOF. Utility and county public information personnel perform various functions from this facility. The JIC has a briefing room that includes office space for public information officers and a phone assistance center. See figure below.



7.1.6.2 Emergency Function

The EOF when activated serves as the headquarters for the Emergency Director, Advisor to the County and UDAC. The EOF acts as the interface between the Company/County, and the public. The Emergency Director and his staff utilize the EOF as their headquarters to provide overall direction of the recovery effort for Company response personnel for a declared emergency.

7.1.6.3 Habitability Objectives

The distance from the plant to the EOF, the very low frequency of winds in the direction from the site to this location, and the shielding and turbulence produced by the mountainous terrain between the two locations makes special habitability provisions unnecessary.

7.1.6.4 Special Equipment

The EOF is provided with extensive telephone and radio communications capability. Communication systems in the building are redundant and include microwave links to DCPP and the General Office.

The EOF is provided with computer monitors that can display plant parameters received by the Emergency Response Facility Data System (ERFDS). The ERFDS recall functionality is provided by the Transient Recorder System (TRS). The TRS is available on any Plant Data Network Computer in the EOF. The EOF is also a central controlling station for the Emergency Assessment and Response System (EARS) used for off-site dose assessment. The EOF/EOC building is protected by a Wet Pipe Fire Sprinkler System.

Management coordination between the TSC, the EOF and EOC is assured by reliable telephone communications. Telephones at key locations in the EOF/EOC are equipped for building or area paging. Backup electrical power is supplied from a 100-kW diesel-fueled emergency generator.

Resource materials available in the EOF include:

- Plant procedures
- Emergency Plan and implementing procedures
- Maps with sectors indicated (10 and 50 miles)
- Air Sampler
- Portable Count Rate and Dose Rate Survey Instruments
- Plant electrical, piping and instrumentation drawings

The EOF is provided with a display terminal and printer for each unit's Plant Process Computer (PPC). This provides the ability to monitor and print plant parameters acquired by the PPC.

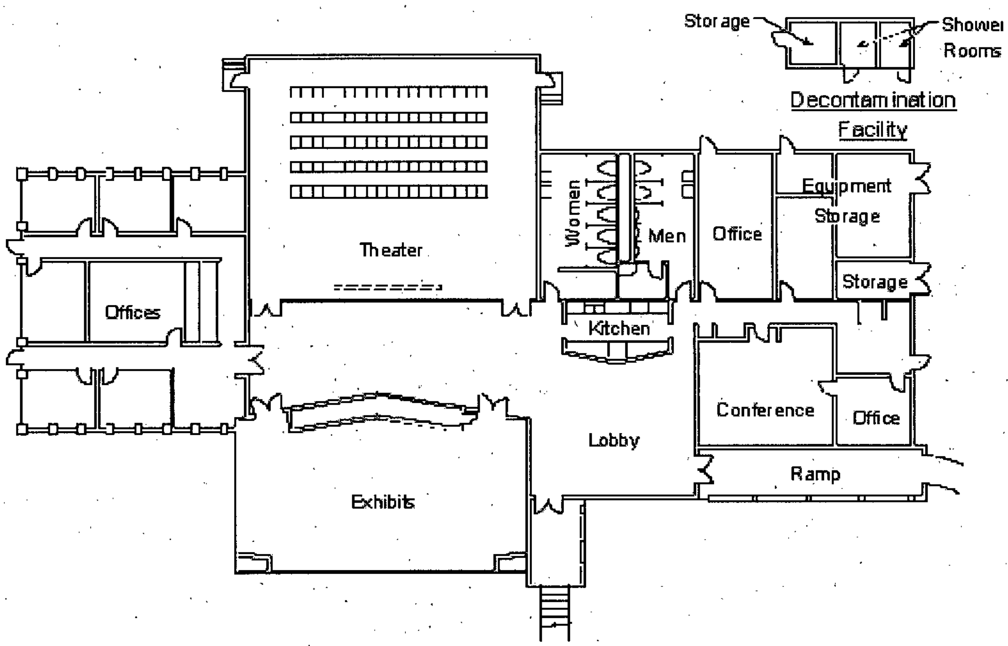
7.1.6.5 PG&E Corporate Interface

The Governerment Relations Coordinator has direct contact with the Corporate Policy Group Chair in the PG&E EOC and the Emergency Director in the EOF. Governerment Relations Coordinator is provided with communications from the Company telephone system and WAN links to Diablo Canyon.

7.1.7 PG&E Information Center

7.1.7.1 Location and Description

The PG&E Information Center is located 12 road miles from DCPP in an ESE direction. The Information Center is approximately 7,000 square feet. The floor plan of the Information Center and adjacent decontamination shower facility is shown below.



7.1.7.2 Emergency Function

The Information Center is the preferred location to gather on-site personnel in the event of a site evacuation, providing meteorological conditions permit. The Information Center can also be used as a headquarters, staging area, and personnel decontamination center. Finally, the Information Center is the first choice for a backup Emergency Operations Facility.

7.1.7.3 Habitability Objectives

No special provisions have been provided to enhance habitability for radiological emergencies. The Information Center has lavatory and kitchen facilities.

7.1.7.4 Special Equipment

The Information Center is provided with a base station emergency radio as well as PG&E and Public Switched Telephone Network (PSTN) telephones. It also has two decontamination showers and related supplies, including replacement clothing.

7.1.8 San Luis Obispo Electric Control Center Operations

7.1.8.1 Location and Description

The San Luis Obispo Electric Control Center Operations (ECCO) is located at the PG&E Service Center in San Luis Obispo approximately 10 miles east of Diablo Canyon. The ECCO is staffed around the clock and monitors distribution outages. There is space for about 15 additional persons in the adjoining Division Emergency Center.

7.1.8.2 Emergency Function

The San Luis Obispo ECCO is primarily intended as a backup headquarters and staging area for Corporate support personnel, in the event the Information Center is unavailable. It is also a back-up off-site assembly area in the event of a site evacuation. It could also be utilized as a backup location for the EOF. It also serves as the backup location for activating the Early Warning System EWS.

7.1.8.3 Habitability Objectives

No special provisions have been provided to enhance habitability for radiological emergencies. The San Luis Obispo ECCO has lavatory and limited kitchen facilities.

7.1.8.4 Special Equipment

The San Luis Obispo ECCO is provided with a base station emergency radio as well as PG&E and PSTN telephones.

7.2 COMMUNICATIONS EQUIPMENT

7.2.1 Plant Telephone Network

The plant telephone network (PTN) is part of the Company owned and operated telecommunications system, which is used for routine inter company communications. Any plant telephone on the company telephone network is capable of communicating with other company telephones, including other plant telephones located at on-site and off-site emergency response facilities. During an emergency, the PTN is the primary communication method for voice communications between emergency response facilities. Plant telephones are located at various locations throughout the plant, including the Control Room, Hot Shutdown panels, Security Building, laboratories, shops, Technical Support Center, Operational Support Center, Emergency Operations Facility, and other work and equipment operating areas. In addition, phone jacks are located in the Control Room and at other strategic operating locations throughout the plant. Each phone jack is associated with a telephone number, and when used in conjunction with a portable telephone unit, can communicate with any other telephone in the power plant as well as the company telephone network.

The PTN is configured to be a highly reliable independent system using a computerized private branch exchange (PBX) switching system to control servers (i.e., nodes) that are linked over two fiber optic network rings. The PTN is comprised of two primary nodes and two redundant secondary nodes that are each capable of controlling the entire PBX network if a control server in the network fails. The fiber optic rings are independent from the plant computer local area network (LAN) and provide two separate redundant paths in the event that one section of network is inoperable.

Telephone calls to and from the PTN are processed via tie trunks over two independent transmission systems. The first link is between the PTN and the company telephone network via a public switched telephone network (PSTN). The other link is via the company-owned and operated microwave system. In addition to these trunks, additional one-way tie trunks are provided to the PBX in the San Francisco Corporate Headquarters. These trunks provide direct dial access to the corporate telephone exchange, bypassing the normal dial traffic, and can be accessed only by high priority telephones. The trunks ensure calls by high-priority telephones can be made to selected Corporate Offices as well as providing an alternate access to the PSTN through San Francisco should the local PSTN Exchange in San Luis Obispo be congested. Also, off-premise extensions (i.e., plant telephones physically located off-site) located in the Corporate Offices provide unrestricted access to the plant PBX excluding the necessity of operating through the PBX in San Francisco.

The PTN has several built in conference call features to allow multiple callers to be connected to the same phone call. One such line is set aside for emergency use and is normally initiated following the sounding of the emergency signal or the fire alarm. Numerous personnel who pick up a company phone and dial a special conference call number will be included in the call.

There will be an available telephone line at the ISFSI to report any emergencies. During cask handling and transportation both security and operations personnel are present with plant radio systems. During normal storage conditions security in the vicinity of the ISFSI is equipped with plant radio systems.

7.2.2 Communication Interface with Public Switched Telephone Network (PSTN)

The DCPP Emergency Response Facilities are served by two trunk groups, the Administrative and Emergency.

The PTN has direct inward dial lines which allow direct access to plant extensions from the PSTN. These lines come in via two separate paths for redundancy. All service from the PSTN to the PTN is distributed among the two PBX nodes. Separate PSTN lines come into the plant for operational use. One line is mounted on the Senior Control Operator's desk in the Control Room to call out from the Control Room in an emergency. This number is unlisted to assure it will not be tied up. A second line runs to the Plant Manager's office, a third line to the Security Supervisor's office, a fourth line is dedicated to the Central Alarm Station (CAS) and the fifth goes to the Secondary Alarm Station (SAS).

7.2.3 Power Supplies

Power is provided by a battery charger where the charger supplies the load and float current for the batteries. The AC for the chargers for the primary and backup telephone system servers located in the Unit 1 communications room are supplied from redundant sources. The AC for the chargers for the primary and backup telephone system servers located in the Building 102 communications room are supplied from a single source. The battery chargers are load-sharing units. The mountain top repeater for the microwave systems are DC powered, employing AC powered battery chargers with batteries which float on the line. An automatic emergency generator at each repeater site supplies the charger if normal AC power is lost.

7.2.4 Control Room Telephone Communication

Numerous phone lines link the Control Room to the plant PBX nodes for redundancy.

In addition, there are several special provisions including:

1) Dedicated dispatch lines that link the Control Room to the Company San Francisco General Office.

2) Dedicated tie lines go from the Control Room and (TSC) Technical Support Center to NRC Headquarters in Bethesda, Maryland (NRC FTS telephone), California Office of Emergency Services (OES) located in Sacramento, the Emergency Operations Facility (EOF) and the Sheriffs Office in San Luis Obispo. Also there are dedicated tie lines from the Operational Support Center (OSC) and the TSC to the Control Room.

7.2.5 *Telephone Communications for the Technical Support Center (TSC)*

The TSC has numerous phone lines to the plant PBX network. Additional telephone communications for the TSC are provided by a PBX Attendant Console. From this console, all incoming calls to the plant emergency phone number can be answered and/or transferred to personnel designated for response. The console can also be used to obtain an outgoing trunk line for use by the TSC. The Attendant's Console will normally be shut-off except during an emergency.

The TSC has dedicated lines to the Control Room, Operational Support Center, Emergency Operations Facility, the San Luis Obispo Count Sheriff's office and the State Office of Emergency Services. The TSC also has a standard unlisted telephone from the NRC FTS telephone system. This telephone provides direct access to an off-site location in the event the Company exchange system is not available.

7.2.6 Operational Support Center Telephone Communications

The Operational Support Center (OSC) has a PBX line and a dedicated auto-tie line to the Control Room and the TSC. In addition, the OSC has access to several telephone extensions installed for routine use.

7.2.7 Telephone Communication at the Emergency Operations Facility (EOF)

The Emergency Operations Facility (EOF) and County Emergency Operations Center (EOC) are co-located near the San Luis Obispo County Sheriff's Office. Included in the building is the County Sheriff's Dispatch Center, which has a dedicated tie line to the Control Room and TSC. This circuit is a common circuit to all these locations and can be accessed from each end.

There is an Off Premise Extension (OPX), in the EOC Command Center. This extension provides unrestricted access to the power plant, via company owned microwave paths, should the local PSTN be congested.

Communication circuits for the EOF include telephone lines from the power plant exchange and additional unlisted telephone lines from the PSTN Network. The NRC FTS telephone lines also tie into the EOF.

Communications between the plant and French Hospital, San Luis Obispo, can be channeled through the EOF. Normal communications will be via commercial telephone. Portable radios may also be used for emergency communications between French Hospital the Plant and EOF.

7.2.8 Data Communication System

The data communication system contains many dedicated circuits that are used in collecting and distributing information. The data network is divided into four areas: 1) records management, 2) time-share operation, 3) operational computer, and 4) health physics.

The majority of the data links radiate from the TSC, which has connectivity to the PG&E LAN/WAN for distribution to many locations in the PG&E network including the PG&E General Office in San Francisco and the EOF.

7.2.9 Nuclear Regulatory Commission Communication Lines

The Nuclear Regulatory Commission (NRC) installed a dedicated telephone system for their use at Diablo Canyon Power Plant.

This system, the Federal Telecommunications System (FTS), provides a separate government network for all the essential communications functions anticipated during an emergency. These essential functions are summarized as follows:

1) Emergency Notification System (ENS): Initial notification by the licensee, as well as ongoing information on plant systems, status, and parameters. The ENS telephones are located in the TSC, Control Room, and EOF.

2) Health Physics Network (HPN): Communication with the licensee on radiological conditions (in-plant and off-site) and meteorological conditions, as well as their assessments of trends and need for protective measures on-site and off-site. The HPN telephones are located in the TSC and EOF.

3) Reactor Safety Counterpart Link (RSCL): This is the channel by which the NRC Operations Center supports NRC reactor safety personnel at the site. In addition, this link may be also used for discussion between the Reactor Safety Team Director and licensee plant management at the site. The RSCL telephones are located in the TSC and EOF.

4) Protective Measures Counterpart Link (PMCL): This is the channel by which the NRC Operations Center supports NRC protective measures personnel at the site. In addition, this link may also be used for discussion between the Protective Measures Team Director and licensee plant management at the site. The PMCL telephones are located in the TSC and EOF.

5) Emergency Response Data System (ERDS) Channel: This is the channel over which the raw reactor parametric data is transmitted from the site. The ERDS lines are located in the TSC computer cabinets.

6) Management Counterpart Link (MCL): Established for any internal discussions between the Executive Team Director or Executive Team members and the NRC Director of Site Operations or top-level licensee management at the site. The MCL telephone is located in the EOF.

7) Local Area Network (LAN) Access: Established for access to any of the products or services provided on the Operations Center's local area network. This includes technical projections, press releases, status reports, E-Mail, and various computerized analytical tools. The LAN line is located in the EOF.

7.2.10 Satellite Telephones

Satellite telephones are available for use for intercommunications between emergency response facilities, along with communicating to Field Monitoring Teams.

7.2.11 UHF and VHF Radio System

7.2.11.1 General

The plant has several voice radio systems available for emergency response use in the UHF (Ultrahigh Frequency) and the VHF (Very High Frequency) radio-frequency bands. In the shorter range UHF band, the plant has voice channels available for Plant Operations and Security use, so that personnel from each department can simultaneously utilize radio communications to perform their duties without interfering with or being interfered with by each other.

The Plant can also communicate by voice radio on VHF channels assigned for Marine use and VHF Aeronautical channel (122.9 Mhz). A channel is also available for use by the California Division of Forestry/County Fire for fire fighting.

7.2.11.2 Plant UHF Radio Systems

The DCPP UHF radio systems can maintain point-to-point communications between the Control Room, the San Luis Obispo ECCO, the PG&E Information Center, the Port San Luis Gate, the San Luis Obispo County Sheriff's Dispatch Center, Technical Support Center, the Emergency Operations Facility, the Security Department, in addition to satellite telephones. Field Monitoring Teams have mobile radios and hand-held radios available for communication to the Emergency Operations Facility using the County Brown radio network.

The plant UHF radio system provides direct radio unit to radio unit usage around the Diablo Canyon site and between emergency response facilities which is independent of the telephone systems. The base station receivers located at the Sheriff's Operations Center, the PG&E Information Center and San Luis Obispo ECCO are equipped for selective calling. The normally silenced receivers may be selectively keyed by the plant Control Room, TSC, or county EOC. This feature may be cut out on the base station receivers to permit continuous, unrestricted monitoring of all units during emergency situations.

The Communications Room base stations are powered from battery backed DC power supplies which are fed from plant vital AC power. The radio system power supply is independent of all other non-vital 48 VDC fed equipment in the room.

Control consoles for this system are located in the Control Room, Technical Support Center, and Emergency Operations Facility.

Additionally, in-plant antenna arrays on radio systems make portable radio communications possible throughout the plant.

7.2.11.3 VHF Systems

The VHF radio system is the Company's Los Padres District Commercial operating network that covers from King City in the north to Solvang in the south. District Company radio-equipped vehicles within the district are normally dispatched by one of the several base stations in this network. The Control Room, TSC, and EOF are equipped with VHF radio consoles that can also be used for dispatch.

7.2.12 Joint Information Center

Telephone service for the Joint Information Center is provided primarily by the PSTN Company with some service from the PG&E network.

7.2.13 News Services Office San Francisco

Telephone service in the News Services Office in San Francisco is provided primarily by the PSTN Company.

7.3 ON-SITE SIGNALS AND ALARMS

Several signals and alarms are provided to alert plant personnel to a possible emergency situation. These include:

7.3.1 Site Emergency Signal

The site emergency signal is a very loud sound produced by electronic warblers placed at numerous locations throughout the plant. The site emergency signal and containment evacuation signal are the only plant signals that are produced by an electronic warbler. The site emergency signal sound is a rapid rise in pitch followed by a slower drop. The cycle repeats itself as long as the signal is energized.

The site emergency signal is initiated manually by the Control Room Operator. Control switches are provided at the Control Room Consoles and the Hot Shutdown Panels for Units 1 and 2. Once initiated, the signal will continue until it is manually turned off.

Due to high background noise levels, flashing red lights as well as a warbler are included in the containment and in some locations of the turbine and auxiliary buildings.

The response of on-site personnel to the site emergency signal, other signals and alarms is discussed in Section 6 of the Emergency Plan.

7.3.2 Fire Signal

The fire signal is produced by sirens placed at numerous locations through out the plant. The fire signal is a steady tone for thirty seconds.

The fire signal may be initiated from any PG&E dial telephone in the plant. A three-digit number is dialed to actuate the signal. The caller is routed to the Control Room where the Unit 1 operator can filter the call and verify the need to sound the Fire Signal. If the operator is delayed from answering, the caller will route to the Emergency Bridge after four rings. *The Fire Signal is actuated as soon as the call enters the Bridge and sounds for* 30 seconds.

The bridge has multiple ports on the initiation side that will allow multiple callers the ability to report a fire or other emergency. The answer side of the bridge has multiple ports so that Operations, Fire and Medical personnel can listen in to see if they need to respond to the fire/emergency.

7.3.3 Criticality Monitor Signal

The criticality monitors are area monitors over the spent fuel pool and the new fuel storage area. These instruments generate an evacuation signal if radiation levels exceed the setpoint value.

The criticality monitor signal consists of horns mounted so that they are audible throughout the fuel handling building and in the hot shop area as well as flashing red lights which are mounted near the doors to the fuel handling building on the +85', +91', and +115' elevations. Flashing red lights over the doors in the hot shop indicate which fuel handling building has been alarmed. Units 1 and 2 have similar systems.

7.3.4 Containment Evacuation Signal

The containment evacuation signal utilizes the same signaling equipment within the containment as is used for the site emergency signal. The electronic warbler on the +140' elevation and flashing red lights on the +91' and +115' elevations are utilized. In addition, a flashing red light is located at the entrance to each personnel airlock.

The containment evacuation alarm is manually actuated from the control console, or from switches mounted in the containment by each airlock. This signal is independent for each unit. The containment evacuation alarm will continue until manually reset in the Control Room.

7.4 OFF-SITE EARLY WARNING SYSTEM

An Early Warning System (EWS) is installed to provide prompt alerting of the public in the event of a major emergency at Diablo Canyon Power Plant. The EWS, used in conjunction with radio and TV broadcasts, allows instructions, information, and necessary actions to be immediately communicated to the general public.

The EWS is designed to meet the requirements of NUREG-0654/FEMA-REP-1 and employs guidelines set forth in FEMA REP-10, "Guide for the Evaluation of Alert and Notification Systems for Nuclear Power Plants."

The EWS design objective specified in Appendix 3 of NUREG-0654 is to provide an alert signal within 15 minutes. This signal is initiated in conjunction with broadcasts providing notification and informational or instructional messages to the population on an area-wide basis throughout the NRC minimum ten-mile radius EPZ. The initial alerting system is designed to assure direct coverage of essentially 100 percent of the population within five miles of the plant site. The SLO County/Cities Nuclear Power Plant Emergency Response Plan provides arrangements to assure 100 percent coverage of the population the NRC designated plume exposure EPZ.

7.4.1 Outdoor Warning

The portion of the EWS providing initial outdoor alerting for the public to turn on their radios consists primarily of a siren system. Alerting provisions for groups that require special contact arrangements and for craft-at-sea are included in the SLO County/Cities Nuclear Power Plant Emergency Response Plan.

The approach taken in designing the siren system is based on the criteria set forth in documents noted above and on substantial practical field experience by the manufacturer. Because of the many factors that can greatly alter sound propagation from a warning device toward a potential listener, the design criteria set forth in the documents identified above are conservative in theoretical terms.

The system employs several siren sizes to attain the desired coverage. Choice of smaller sirens is often dictated by terrain or sparsely populated or isolated areas.

The number of sirens employed in the system was determined by choosing the siren providing the desired radius of coverage deemed necessary to achieve an adequate dissonant alerting signal level.

7.4.2 Indoor Warning

Large groups of people may be gathered together inside structures. Such structures may preclude detection of the warning provided by the outdoor warning system or may present particular response problems. The Emergency Alert System (EAS) is the primary means of providing indoor warning to residents of the DCPP Emergency Planning Zone.

San Luis Obispo Country also has over 100 tone alert monitor receivers in schools, hospitals, convalescent care centers, and other institutions to alert them with special instructions. No tone alert monitor receivers are distributed to residential members of the public. Control consoles for activation of these receivers are located in the Sheriff's Dispatch Center in the EOC/EOF Building.

Pagers have been assigned to key emergency response personnel including governmental officials and to members of the Environmental Health field monitoring teams. The pagers are activated from the Sheriff's Dispatch Center.

7.4.3 Early Warning System-Area of Coverage

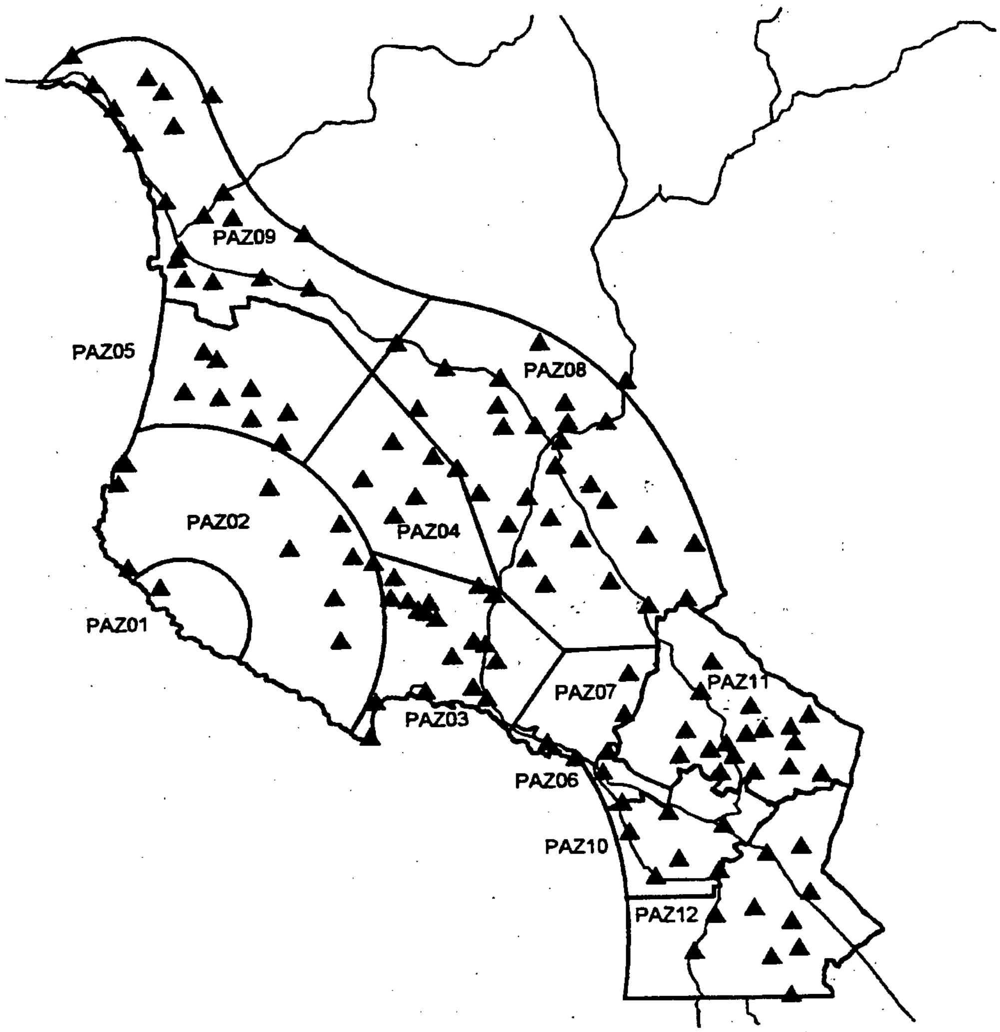
The coverage area of the EWS has been extended beyond the ten-mile radius to include the State of California Basic EPZ. The design coverage objective is based on the following assumptions:

1) All populated areas within the outdoor siren coverage range will be alerted with a 60 dB or greater siren signal.

2) All public institutions and business operations will have and use means for communicating specific instructions to their personnel when they hear the siren signal.

The large area within the five-mile radius of the plant has limited siren sites since it is rugged wilderness and, largely, inaccessible to the public. A portion is identified as a state park with jeep trails. The County Emergency Plan provides special notification arrangements in this area.

There are 131 area sirens of various sizes used in the EWS. The siren locations and approximate areas of coverage are illustrated below.



7.4.4 Siren Location

The location of each siren was determined in accordance with ambient noise measurements and other considerations.

A site survey to measure ambient noise levels was made in the general area of each proposed siren site. These measurements, the topography, the terrain coverage and the population density formed the basis for choosing the siren setting. A substantial portion of the population covered by each siren will be in a sound field that is 10 dB above the average daytime ambient background noise. Each siren site was chosen to minimize the biological, cultural and human impacts consistent with proper acoustic coverage.

7.4.5 Early Warning System Activation

The EWS sirens are activated by a supervisory control and data acquisition (SCADA) control system with primary and backup activation centers. San Luis Obispo County Emergency Operations Center personnel are responsible for system activation. The SCADA control system sends commands to and receives data from the sirens via radio signals from radio base stations such as Tassajera, Black Butte, Davis Peak, and Arroyo Grande Hill. Most of the sirens can be reached from at least two of the base stations.

The SLO County Sheriff Watch Commander, is responsible for EWS activation upon authorization as described in the SLO County/Cities Nuclear Power Plant Emergency Response Plan. The sirens can be activated simultaneously, individually or in designated groups. Security provisions against inadvertent operation of the EWS are built into the SCADA control system.

7.5 ON-SITE ASSESSMENT SYSTEMS AND EQUIPMENT

7.5.1 Seismic Monitoring System (SMS)

In the event of an earthquake, the mitigating actions required are determined by the magnitude of the earthquake. The seismic instrumentation, as described in Section 3 of the FSAR, includes the equipment used as the primary means for timely determination of the magnitude of an earthquake. Emergency classification is based on the earthquake magnitude displayed on the earthquake force monitor (EFM) located in the control room. In the event of an earthquake measuring greater than 0.01 g on the acceleration recorder at 89' elevation of the Unit 1 containment structure, or on the free-field acceleration recorder located north of Unit 1 containment, an annunciator will alert the control room operators and peak ground acceleration (g) values from the seismic monitors will be displayed on the EFM. The Unit 1 acceleration recorders represent on-site ground motion for both Units 1 and 2.

7.5.2 Meteorological Systems

7.5.2.1 Measurement Systems

As shown in FSAR Figure 2.3-4, there are two meteorological (Met) towers located on-site and six supplemental Met towers that are located off-site within the vicinity of the plant. The six supplemental (10-meter) Met towers (Pt. Buchon, Los Osos, Foothill, Service Center, Information Center, and Davis Peak) are run exclusively for DCPP by PG&E. In addition, two other off-site Met locations (Templeton and Santa Maria) are run by PG&E for other projects, but have data available for emergency response. Also available are two 10-meter tower sites (Morro Bay and Grover Beach) that are operated by the San Luis Obispo County Air Pollution Control District and the aviation weather observations at the county airport.

The six supplemental PG&E surface (10-meter) Met towers provide consecutive 15-minute averages of wind speed, wind direction and wind direction standard deviation (sigma theta) based on a 1-2 second sampling interval. The APCD 10-meter sites provide hourly averages of wind speed and direction based on a 1-second sampling interval at Grover Beach and Morro Bay. The Weather Service/Airport site provides standard airways sequence data. All data from the PG&E sites are available in real time at the EOF by PG&E Wide Area Network and dial-up modem.

Site meteorological conditions are monitored continuously by the two on-site meteorological towers that are in close proximity to the plant structures. A primary 76-meter tower system is located about 200 meters SSW of the plant structures. Measurements have been taken and used from this site since July 1967. The following data are provided: wind speed, wind direction, wind direction standard deviation at the 10-meter and 76-meter levels; temperature at 10-meters; temperature difference (delta T) between 10-meter and 46-meter and between 10-meter and 76-meter levels; precipitation and dew point near the tower base; and 10-meter, 46-meter and 76-meter aspirator currents. Meteorological data at the primary site is recorded continuously on strip charts and digitized electronically at two-second intervals. Fifteen-minute mean values are computed from the 2 second values by microprocessor within the primary met facility and are sent to the Unit 1 Transient Recording System (TRS). The Unit 1 TRS is interfaced to the Unit 1 Plant Process Computer (PPC). TRS transmits data to the PPC where they are archived and alarm processed. Primary Met data are sent to the Unit 2 PPC via the backup Met tower computer (FSAR 2.3.3.7) and the Unit 2 TRS. The 15 minute values are also sent to the EARS Computer System and to both unit's Safety Parameters Display System (SPDS), which are located in the TSC. The meteorological data via EARS, PPC, SPDS and TRS are also available to the Emergency Operations Facility. Thirty (30) days of the most recent 15 minute mean values are archived and is maintained on-line for ready access at all times on the EARS computer in the TSC on both the Unit 1 and Unit 2 PPCs. Archives are made approximately every 2 weeks so data from December, 1993 on will always be available.

A backup 60-meter tower system is located about 1.2 KM ESE of the primary tower with two levels of measurement at 10-meters and 60-meters. Measurement, reduction, storage, recording and transmission of backup tower data are continuous with and similar to that of the primary system. The backup Met system measures 60m – 10m temperature difference and sum of aspirator currents and battery voltage (FSAR 2.3.3.6.). It does not measure temperature at the intermediate level and the measurement of precipitation and dew point are available only at the primary site. The backup Met computer is located in the TSC (FSAR 2.3.3.7.). Failure of either the primary or backup systems will be shown on both the Unit 1 and Unit 2 PPC alarm screens and alarm printers. Power for the primary system is derived from the Unit 1 480 volt non vital bus with automatic switching in the event of failure to the Unit 2 480 volt bus. The primary microprocessor and meteorological sensors are backed up by an 8-hour battery source. The backup meteorological system is supplied AC power from the underground 12kV startup bus with backup power in the form of a one-week battery source. To minimize required battery capacity, further backup is provided by a spring-wound strip-chart with mechanically driven pens.

In the event of failure of all electronic measurement systems, a portable weather station is available. This unit is battery powered for independent operation. The instrument provides recording of wind speed, wind direction and ambient temperature, which can be used to estimate off-site effects through manual calculation procedures in the event of failure of the automated assessment process. Should its use be required, (χ/Q would be determined by using wind speed and a stability category based on wind direction range. (See App. 2.3I of Reference 3).

These measurement systems have been upgraded continually and currently satisfy all requirements of NUREG-0737, NUREG-0654 and Regulatory Guide 1.23 of the NRC.

7.5.2.2 Modeling Systems

Dispersion modeling software is operational on the Meteorological tower on-site computer systems to provide required inputs to the Dose Calculation Methodology. The model in use is consistent with characteristics of a Class A model as required by NUREG-0654 and 0737. Normalized ground level plume centerline concentration (dilution) values (χ/Q) are computed by the model as 15-minutes means for ten downwind distances ranging from 0.8 to 100 kilometers. Lateral and vertical plume dimension (sigma y and sigma z) are also provided for each of the same downwind distances. Concentration and plume dimension data generated by the model provide the necessary relationships between effluent monitor readings and off-site exposure and contamination. These data are distributed to recipients in the same manner specified earlier for meteorological measurement data.

Additional details on the meteorological measurement and modeling systems are provided in Section 2.3 of Reference 3.

7.5.3 Area Radiation Monitoring System

The area radiation monitoring system is a system of permanently installed radiation monitoring detectors located throughout the plant. Monitors and locations are listed below. The purpose is to measure the ambient radiation level in each monitored area. Such information is used both for personnel radiation protection purposes and to alert personnel to the release of radioactive materials within a plant structure.

This system includes G-M or Ion chamber detectors that are permanently mounted. Each of these instruments has a readout at the detector and in the Control Room, except as noted below. If an adjustable high radiation setpoint is reached, audible and visual alarms are sounded both at the detector and in the Control Room. The GM detector instruments are calibrated in mR/hr and have a range of 0.1 up to 10,000 mR/hr. The instruments will remain pegged full scale at radiation levels beyond the full-scale value. These instruments are powered from emergency power sources and would be available for post-accident use (except as noted below).

Additional detectors have been added to augment the original system of GM detectors in order to provide area monitoring where new spaces or specialized equipment have been added. The additional areas where these detectors have been added are in the Technical Support Center (G-M Type), and in the Containment (Ion-Chamber Type).

Detector Number	Location	Remarks
U-1 & 2 R-1	Control Room	
U1-R-2 U2-R-2	Containment operating deck (140' elevation) near personnel access hatch.	Not designed to withstand post loss-of-coolant-accident containment temperature.
U1-R-4 U2-R-4	Charging pump room (73' elevation in Auxiliary Building).	
U1-R-6 U2-R-6	NSSS Sampling room (100' elevation in Auxiliary Building).	
U1-R-7 U2-R-7	Incore seal table area (103' elevation in containment).	Not designed to withstand post loss-of-coolant-accident containment temperature.
U-1 & 2 R-10	Auxiliary Building control board (85' elevation in Auxiliary Building).	
U1-R-30 U1-R-31 U2-R-30 U2-R-31	Containment high range radiation (140' elevation in containment).	Designed to withstand post loss-of-coolant-accident (LOCA) conditions in containment.
U1-R-34 U2-R-34	85' Plant Vent RMS Room (ALARA)	
U1-R-48 U2-R-48	HRSS (Sentry) Post Accident Sampling Room	Non-vital power supply Local readout only
U1-R-58 U2-R-58 U1-R-59 U2-R-59	Spent Fuel Pool Storage Area New Fuel Storage Area	
U-1 & 2 R-60 U-1 & 2 R-61 U-1 & 2 R-62 U-1 & 2 R-63 U-1 & 2 R-64 U-1 & 2 R-65	TSC Office Area TSC Operations Area TSC Computer Area TSC NRC Office Area TSC HVAC Equipment Area TSC Lab Area	Local readout only Normal power is from non-vital supply, but can be aligned to vital buses from Switchgear room.

7.5.4 Process Radiological Monitoring System

Several liquid and gaseous plant process streams are continuously monitored for radioactivity to provide an indication of equipment performance and provide a record of radioactivity releases to the environment. The important process monitors from the standpoint of potential usefulness in emergency situations are listed below.

Detector	Process & Location	Type	Range (CPM)	Remarks
1-R-11 2-R-11	Containment air particulate (100' penetration area GE)	Gamma scintillation	10^1 to 10^6 cpm	Continuous sample drawn from containment, passed through moving filter, returned to containment. Detector looks at filter. Sample lines are automatically isolated by R-44A/B in event of major accident in containment.
1-R-12 2-R-12	Containment air gas (100' penetration area GE)	G-M	10^1 to 10^6 cpm	Detector looks at sample flow downstream of particulate monitor. Sample lines are automatically isolated by R-44 A/B in event of major accident in containment.
1-R-13 2-R-13	RHR exhaust duct air particulate (100' Aux. Building)	Gamma scintillation	10^1 to 10^6 cpm	Detector looks at air particulate sample on moving filter.
1-R-14/14R 2-R-14/14R	Plant Vent gas (85' area L)	Beta scintillation	10^1 to 5×10^6 cpm read out in μCi/cc	Detector looks at pressurized sample flow downstream of iodine and particulate monitors.
1-R-15/15R 2-R-15/15R	Condenser air ejector gas (104' Turbine Building)	Beta scintillation	10^1 to 5×10^6 cpm	Detector looks at air ejector off-gas.
1-R-17A,B 2-R-17A,B	Component Cooling Water (73' Aux. Building)	Gamma scintillation	10^1 to 10^6 cpm	Detector looks at sample off CCW discharge header. High alarm isolates CCW surge tank vent.
R-18	Liquid radioactive waste system effluent line to ocean (54' Aux. Building)	Gamma scintillation	10^1 to 10^6 cpm	Looks at waste stream prior to dilution in outfall. High alarm closes waste discharge valve and diverts to EDR.
1-R-19 2-R-19	Steam generator blowdown liquid (100' Aux. Building penetration area GE)	Gamma scintillation	10^1 to 10^6 cpm	Detector looks at combined blowdown from 4 steam generators. Can be valved to look at each steam generator individually. High alarm will isolate Stm. Gen blowdown tank inlet and overboard lines.
R-3	Oily water separator effluent (85' Turbine Building)	Gamma scintillation	10^1 to 10^6 cpm	
1-R-22 2-R-22	Gas decay tank discharge gas (54' Aux. Building)	G-M	10^1 to 10^6 cpm	Detector looks at gas decay tank discharge line and isolates it on high alarm.
1-R-23 2-R-23	Steam generator blowdown to discharge tunnel (100' Aux. Building penetration area GW)	Gamma scintillation	10^0 to 10^6 cpm	Detector looks at liquid blowdown. 10-7 μCi/cc is the detectable level. High alarm will *isolate Stm. Gen blowdown tank* inlet and overboard lines.

(continued next page)

Detector	Process & Location	Type	Range (CPM)	Remarks
1-R-24/24R 2-R-24/24R	Plant Vent iodine (85' area L)	Gamma scintillation	10^1 to 5 x 10^6 cpm (readout in μCi/cc)	Detector looks at plant vent I-131. Sensitivity based on I-131. Continuous sample is drawn through fixed particulate prefilter and silver zeolite cartridge in series. Detector looks at silver zeolite cartridge.
1-R-25 2-R-25 1-R-26 2-R-26	Control room ventilation supply (160' Aux. Building)	Gamma scintillation	10^{-2} to 10^3 mR/hr	Detector looks at Control Room ventilation intake to supply duct. High activity will transfer ventilation system to pressurization mode. (MODE 4)
1-R-28/28R 2-R-28/28R	Plant Vent particulate (85' area L)	Beta scintillation	10^1 to 5 x 10^6 cpm (readout in μCi/cc)	Continuous sample drawn from plant vent is drawn through fixed filter and detector looks at filter
1-R-29 2-R-29	Plant Vent high range gross gamma (outside PV on 155' lvl)	Ion-chamber	10^{-1} to 10^7 mR/hr	Detector monitors plant vent gross gamma by using a shielded detector with a wedge shaped "view window" aimed at the plant vent.
1-R-41 1-R-42 1-R-43 2-R-41 2-R-42 2-R-43	Gas decay tanks (64' Aux Building).	Ion-chamber	10^0 mR/hr to 10^4 mR/hr	Readout is at the Aux. Building Control Panel only. Well mounted with view ports directed at each tank.
R-51 R-52 R-53 R-54	Control room pressurization system inlet (140' Turbine Building)	G-M	10^{-2} to 10^4 cpm	Detector measures gross activity in the intake line. High alarm switches intake to opposite end of Turbine Building.
R-71 R-72 R-73 R-74	Steam lines downstream of containment penetration (130' pipe racks)	G-M	10 to 10^6 cpm	Detector measures gross gamma activity in steam line, including Nitrogen-16.
1-R-44A & 44B 2-R-44A & 44B	Containment Purge Exhaust (100' penetration area L)	Beta-scintillation	10 to 5 x 10^6 cpm	Containment purge exhaust. High alarm causes CVI.
R-66 R-68	TSC HVAC Duct TSC Lab	Beta scintillation	10 to 10^6 cpm	Particulate Monitor
R-67 R-69	TSC HVAC Duct TSC Lab	Beta scintillation	10 to 10^6 cpm	Noble Gas Monitor
R-82 R-83	TSC HVAC Duct TSC Lab	Gamma scintillation	10 to 10^6 cpm	Iodine Monitor
1-R-87 2-R-87	Plant Vent extended range gas (85' area L)	Beta scintillation	10^{-4} to 10^5 μCi/cc	Extended Range Plant Vent Noble Gas Monitor
1-RF-87A & 87B 2-RF-87A & 87B	Plant Vent Sample (85' area L)	Sampler only	N/A	Iodine Grab Sample (High Range)

7.5.5 Radiological Counting Room

The plant has extensive counting room equipment that is used for routine radiochemical determinations. This equipment could also be used during emergencies.

7.5.5.1 Location and Availability

The counting room is located on the +85' elevation of the Auxiliary Building. Although this area would be expected to be available during most emergency conditions, it should be noted that operation of residual heat exchanger 1-1 in the post LOCA recirculation mode (assuming a Regulatory Guide 1.4 source term in the reactor coolant system) produces a gamma background in the facility, which would make it unusable. The TSC also has the capability to accomplish radiochemical analyses.

7.5.5.2 Counting Room Equipment

1) Multi-channel gamma analysis capability

   The counting room has a gamma spectroscopy system that utilizes high purity germanium detectors with high-resolution quality. The data is analyzed (peak search, peak fit, peak identification, peak quantification, etc.) by a host computer.

   Analysis data from the counting room can be transmitted to the on-site Technical Support Center.

2) Liquid Scintillation Spectroscopy

   The counting room employs liquid scintillation spectroscopy. This analyzer is microprocessor controlled. The analyzer is primarily used for tritium (hydrogen - 3) analysis.

3) Proportional Counting Systems

   The counting room has a gas flow proportional counting system that is used for alpha and beta measurements.

7.5.6 Analytical Facilities Associated with On-Site Technical Support Center

One compartment adjacent to the Technical Support Center is set aside for analytical work. The principal purpose of this facility is to provide necessary on-site analytical capability in the event that the normal facilities in and around access control are unavailable. A germanium gamma spectroscopy system similar to those in the counting room is provided for isotopic analysis.

7.5.6.1 Emergency Assessment and Response System (EARS)

The EARS receives input data from a variety of off-site and on-site monitors to a central computer for dispersion dose calculations. The system includes a terrain specific atmospheric dispersion model which accounts for non-linear plume transport due to hill and valley influences. It provides plume projection maps and other assessment related information to centers both on-site and off-site. Overall system design seeks to optimize the diagnostic ability of operators and emergency response groups, maximizing the ability of system operators to understand events as they unfold.

EARS is interfaced to selected channels of the on-site meteorological towers and plant radiation monitoring system. EARS calculates doses for off-site locations based on data received from the various interfacing systems. These computed plume displays, along with input data, source terms, and meteorological parameters, are transmitted via data links to UDAC.

Each EARS workstation has the capability to graphically display plume projection maps and data, and to provide hard copy to the user. Displays at the TSC and EOF can be of an identical form for the purpose of response planning and coordination between the two facilities. Displayed information may be selected from several forms of calculated data and map presentations. Mapping capabilities are provided to superimpose plume boundaries, dose rates, airborne concentrations, population centers, and evacuation routes. Color graphics and printouts are used to increase the information content and readability of information.

7.5.6.2 System Description

The emergency dose assessment system consists of three functional subsystems:

1) Input Data Subsystem

   The data acquisition subsystem provides live-time radiological and meteorological data needed by the central computer as input parameters for the dose projection models. The specific input data systems are described below.

2) Central Computer Subsystem

   The servers, PCs, peripherals, and software needed to manage input and output communications, and to maintain live-time and historical databases. The central computer subsystem is the main data storage and distribution point for EARS. It is located in the Technical Support Center (TSC).

3) EARS Stations

   These are the dedicated PC workstations, with color graphics display monitors, color printers, and the EARS software for execution of the dose projection model, and transmission of results to the central computer and other EARS Stations.

The EARS Stations are distributed at two locations:

- TSC (DCPP)
- EOF (San Luis Obispo)

The TSC and EOF EARS Stations are both Control Enabled stations, capable of performing dose projections. Data transmission and software execution rates provide for updating the graphics display at intervals of not more than fifteen minutes.

4) Input Data Systems

a) Off-Site Radiation Monitors

The off-site monitoring system consists of a "ring" of low-level gamma dose-rate monitors installed at selected locations at distances of 6-17 miles from the Diablo Canyon Power Plant site. These fixed locations have been chosen with a dual purpose of providing measurement stations at population centers, and of including the largest practical number of land based sectors from the site.

b) Meteorological Towers

Meteorological data are provided from the primary on-site meteorological tower. A secondary (backup) meteorological tower located on-site is equipped with similar instrumentation at two levels. The meteorological data from either tower is accessible from any network PC. EARS Stations automatically receive the necessary data from the primary tower, with auto-failover to backup tower data for any missing parameters. Manual entry of current data from the primary or backup meteorological tower is possible in the event of some problem with automatic input or data reduction.

c) On-Site Radiation Monitors and Plant Process Instruments

Parameters from fixed radiation monitoring and process instruments are available to any network PC. EARS Stations automatically receive the radiation monitor and flow element readings necessary for performing dose projections.

No. of Monitors	Monitor Channel	Monitor Function
2	R-02	Containment Area Monitor
2	R-11	Containment Air Particulate Monitor
2	R-12	Containment Noble Gas Monitor
2	R-14	Plant Vent Noble Gas Monitor
2	R-14R	Plant Vent Noble Gas Monitor, redundant
2	R-15	Condenser Air Ejector Gas Monitor
2	R-15R	Condenser Air Ejector Gas Monitor, redundant
2	R-22	Gas Decay Tank Discharge Monitor
2	R-24	Plant Vent Iodine Monitor
2	R-24R	Plant Vent Iodine Monitor, redundant
2	R-28	Plant Vent Air Particulate Monitor
2	R-28R	Plant Vent Air Particulate Monitor, redundant
2	R-29	Plant Vent Gross Gamma Monitor
2	R-30	Containment High Range Area Monitor
2	R-31	Containment High Range Area Monitor
2	R-87	Extended Range Plant Vent Noble Gas Monitor
2	FM-12	Plant Vent Flow Rate
2	FM-700	Containment Purge Flow Rate
2	FIT-81	Condenser Air Ejector Flow Rate
2	R-44A	Containment Purge Exhaust and CVI
2	R-44B	Containment Purge Exhaust and CVI, redundant
2	RE-71	Main Steam Line Monitor
2	RE-72	Main Steam Line Monitor
2	RE-73	Main Steam Line Monitor
2	RE-74	Main Steam Line Monitor

7.5.7 Portable Survey and Dose Rate Instruments

A variety of portable survey instruments (count rate and dose rate) is available at the plant for routine as well as emergency radiological monitoring. The general equipment types are summarized below. It should be noted that this list is intended only to be illustrative of the plant's capabilities. Precise quantities and models of specific equipment may vary from time to time as conditions change, new products appear on the market, etc. The equipment listed in the table is normally located at Access Control when not in use.

Portable Count and Dose Rate Meters

Instrument (Model No. or equivalent)	Detector Type	Radiation Measured	Range	Primary Use
Beta-Gamma Survey Meter (E-140), with the following detectors:			0-600, 0-6000, 0-60,000 CPM	General contamination surveys
Hand probe (HP-260)	GM	Beta, Gamma		
Shielded hand probe, (HP-210)	GM	Beta, Gamma		
Count rate meter (RM-15) for use with GM probes listed above, and:			0-500, 0-5K 050K, 0-500K CPM	Personnel contamination surveys
alpha scintillation probe (AC-3B-7)	ZnS (Ag_2), 59 cm sensitive area	Alpha		
gamma scintillation probe (SPA-3)	NaI (T1) 2" x 2"	Gamma		
Count rate meter (PRM-6) for use with GM probes listed above AC-3B-7 and SPA-3 probes	See above	See above	0-500, 0-5K 050K, 0-500K CPM	General contamination surveys
ASP-1 (HP 270 shielded hand probe)	GM	Beta, Gamma	.1 →10K mR/hr	General environmental radiation surveys
Portable REM Counter (PNR-4)	BF_3	Neutron, thermal to 10 MeV	0-5,0-50,0-500,0-5000 mR/hr	Neutron dose rate
Teletector 6112 OR Equivalent (Johnson Extender, Telescan)	Twin G-M tubes 30 mg/cm beta window	Beta, Gamma	0-2 mR/hr, 0-50 mR/hr 0-2 R/hr 0-50 R/hr 0-1000 R/hr	Beta, gamma dose rate
RO-2	Ion chamber 3.5 mg/cm beta window air fill gas	Beta, Gamma	Dose rate: 0-5, 0-50, 0-500 mR/hr 0-5R/hr	Dose rate
RO-2A	Ion chamber 3.5 mg/cm beta window air fill gas	Beta, Gamma	0-50, 0-500 mR/hr 0-5, 0-50 R/hr	Dose rate

7.5.8 Field Monitoring and Evacuation Kits

1) Field Monitoring Kits

Field Monitoring Kits are stored on-site at the Warehouse B and off-site at the EOF. Miscellaneous equipment is stored at a location near the EOF. The purpose of Field Monitoring Kits is to provide Field Monitoring Teams (FMTs) with the portable supplies in convenient locations for quickly dispatching teams to the field. These kits contain equipment and supplies that allow assessment of ambient exposure rate, airborne, particulate concentration, airborne radioiodine concentration, and ground deposition.

Radiological Emergency Kits Contents

12VDC Air Sampler, sample head	Allen Key for Environmental TLD Holders
Constant Geometry Holder	Decon Center Key
Extra Sample Head	Corporate 909 Key
Air Sample Particulate Filters	Fuses for Air Sampler
TEDA Impregnated Cartridges	Change for Phones
Paper Envelopes for Particulate Filters	Electronic Personal Dosimeters or Pocket Dosimeter (0-200 mR, 0-5R)
Small Ziploc Bags	Dosimetry Charger with battery
Forceps	Count Rate Instrument, E-140N (or equivalent)
Smears	Dose Rate Instrument, ASP-1 (or equivalent)
Timepiece	Pancake GM probe HP-210/260 (or equivalent)
Marking Pens	Extra Batteries for instruments
Tape	Instruction Binder containing the following:
Flashlight	• Procedure EP RB-8
Plastic Bags	• Record of KI Distribution
KI Tablets	• SLO County Map
Protective Clothing	• Field Monitoring Data Sheets
Calculator	

The following miscellaneous equipment is located with the on-site kits.

2-way radios and satellite telephones
Bolt Cutters

The following miscellaneous equipment is located with the off-site kits.

Trowel	Radiation Signs w/Inserts
Grass Shears	Radiation Barrier Tape
Liter Bottles	Air Sample Particulate Filters
Coveralls	AgZ and/or TEDA Impregnated Cartridges
Skin Decon. Soap	Paper Envelopes for Particulate Filters
Scrub Brush	Small Ziploc Bags
Flashlight	Surgical Gloves
Paper Towels	Smears
Plastic Bags	Extra Batteries
Tape	2-way radios and satellite phones

2) Evacuation Kits

Evacuation Kits are available for purpose of providing Evacuation Teams with the necessary portable supplies for performing contamination surveys of evacuated site personnel and their vehicles. The Evacuation Kits are stored at Warehouse B.

If the Site Emergency Coordinator recommends evacuating non-essential personnel on-site, the Evacuation Kits will be obtained by the Evacuation Coordinator and their monitoring staff prior to the evacuation.

Evacuation Kits Contents

Barricade Tape	Decon Center Key
Batteries	Dose rate instrument, ASP-1 (or equivalent)
Binder Containing:	Electronic Personal Dosimeters or Pocket Dosimeters (0-200 mR)
• Procedure G-4	Dosimeter charger
• Procedure G-5	Flashlight
• Form 69-9310	GM Probe, HP-210/260
• Form 69-9311	Pens
Bullhorn	Plastic Bags
Calculator	Protective Clothing
Corporation Key (3A90909)	Smears
Count rate meter, E-140 (or equivalent)	

3) Hospital Kit

Hospital kits containing portable instrumentation, protective clothing and rad protection supplies are stored at French Hospital in San Luis Obispo.

Hospital Kit Contents

Barricade Tape	Personnel Decon. Records (69-9392)
Batteries	Electronic Personal Dosimeters or Pocket Dosimeters (0-200 mR)
Count Rate Instrument, E-140 (or equivalent)	Rad Sign - 6 pocket
Disposal Coveralls	Rad trash tags
Dose rate instrument, ASP-1 (or equivalent)	Radiation and Contamination Survey Sheet (Form 69-11510)
Dosimeter Charger	Shoe Covers
Duct Tape	Smears
GM probe, HP 210/260 (or equivalent)	Yellow/Magenta Tape, 2" width
Latex Gloves	Ziploc bags
Markers	

4) Clothing

Clean overalls and shoe covers are stored with the evacuation kits on-site, at the PG&E Information Center, and at the TSC for use in the event emergency worker evacuee clothing is contaminated.

7.5.9 Fire Detection and Trouble Alarm System

The Fire Detection and Trouble Alarm System uses heat detectors, flame detectors, and smoke detectors to provide the Control Room with an early warning of the existence and location of a fire or potential fire and in certain instances, initiates fire suppression systems.

In addition to the smoke and flame detection system, a vital equipment room temperature monitoring system provides indication of abnormally high compartment temperatures which may result from fire, ventilation system malfunction, or other equipment malfunctions. Individual components are equipped with the usual complement of trouble alarms, some of which (e.g., high temperature bearing alarm) can provide the operators with early warning of fire or potential fires.

7.5.10 Sampling and Analysis Capability

1) Reactor Coolant Sampling

   If emergency conditions require sampling of reactor coolant, sampling may be performed by sampling of reactor coolant system and containment sump sampling systems. In addition to the normal plant laboratory facilities, the TSC has instrumentation for radiological analysis of the samples.

2) Containment Atmosphere Sampling

   If emergency conditions require sampling of the containment atmosphere, the CASP containment atmospheric sampling system may be used. Equipment is provided to determine containment hydrogen. Dilute samples can be collected and prepared for counting. Radiological analyses are performed in either the normal plant laboratory facility or the TSC laboratory.

7.5.11 Miscellaneous Post Accident Assessment Instruments

1) Containment Interior Radiation Monitoring

   These detectors are part of the area and process monitoring systems and include two high range gamma radiation-monitoring instruments that have been provided to monitor post-accident radiation levels inside of the containment. They utilize ion chamber detectors and have a range of 1.0 to 10^7 R/hr. The units read out and alarm in the Control Room. These units are supplied from emergency AC, are seismic design Class I, and are qualified for post-accident containment conditions.

   The radiation level inside the containment can also be estimated from outside the containment using hand held instruments.

2) Containment Water Level

   Two wide range and one narrow range containment water level recorders can be used to indicate water level from the bottom of the reactor cavity (elevation 64 feet) to elevation 98 feet. The volume of water these recorders will represent is greater than 600,000 gallons. The indicators are located in the Control Room.

3) RCS Subcooling Monitor

RCS subcooling instrumentation is provided to determine margins to saturation pressure and temperature using reactor coolant system pressure and temperature instruments and incore thermocouples. A calculating microprocessor module powered from a vital bus is provided. A digital readout is provided on the main control board. The margin, in either degrees Fahrenheit or pounds per square inch, is displayed on a continuous basis. Low and low-low main annunciator inputs are provided.

4) RCS Pressure Indication

Reactor coolant pressure transmitters on loop 4 and loop 3 of the RCS are provided. One transmitter has an indicator in the Control Room with a range of 0 to 3,000 psig. The other transmitter has a recorder with a range of 0 to 3,000 psig and an indicator with a range of 0 to 600 psig.

5) RCS Temperature Indication

RCS temperature is monitored with both narrow and wide range instruments. Each RCS loop has one wide range hot leg RTD and one wide range cold leg RTD. The wide range RTDs are in the main loops and have a temperature range of 0-700F. The narrow range RTDs are in thermowells that protrude into the legs and have a temperature range of 530-630F. The RTDs have Control Room readouts.

Also originally installed is a redundant, two train incore thermocouple system with readout on SPDS monitors in the Control Room, TSC and EOF. The number of operable thermocouples required per core quadrant is governed by the requirements given in the Technical Specifications. The incore thermocouple system is designed to Class IE electrical requirements and uses microprocessor equipment to calculate and readout on the PAMS monitors. Readout is also available on the plant computer and SPDS monitors.

6) Containment Pressure Indication

Seven containment pressure indications are available in the Control Room (each uses its own transmitter):

a) During normal operation a single recorder with a range of -1 to +1.5 psig is utilized.

b) Four indicators are available with a range of -5 to +55 psig.

c) To indicate higher containment pressure two recorders with a range of -5 to 200 psig are available.

d) The instruments described in 2) and 3) are powered from vital electrical power supplies.

7.6 OFF-SITE MONITORING EQUIPMENT

7.6.1 Off-Site Geophysical Monitors

Off-site seismic observation and monitoring facilities in the coastal region are located at the University of California (Berkeley), California Polytechnical University (San Luis Obispo), California Institute of Technology (Pasadena), California Department of Water Resources (Sacramento), the U.S. Geological Survey (Menlo Park), and PG&E Geosciences Department (San Francisco).

7.6.2 Off-Site Meteorological Data

Data from supplemental meteorological sites surrounding Diablo Canyon are available at the Emergency Operations Facility (EOF) and through Meteorological Services at PG&E headquarters in San Francisco and San Ramon. The sites include six surface meteorological sites located at Pt. Buchon, Los Osos, Foothill Blvd., Davis Peak, Information Center, and PG&E Service Center. Additional meteorological information that may be obtained on the PG&E Wide Area Network are: other PG&E power plants, off-site National Weather Service (NWS) data including upper air soundings at Vandenberg AFB, Oakland, San Diego, Las Vegas, Winnemucca and Medford, Oregon, and surface sites located in Monterey, Salinas, Paso Robles, Santa Maria, Santa Barbara, San Luis Obispo and Buoys EB11 and EB23. There are also additional surface meteorological sites available from central and southern California as well as the latest infrared and visual satellite images available at the EOF. Weather forecasts during an Emergency Plan activation are available from the EOF, PG&E forecast office, NWS Los Angeles and Monterey.

7.6.3 Environmental Direct Radiation Monitors and Air Sampling Devices

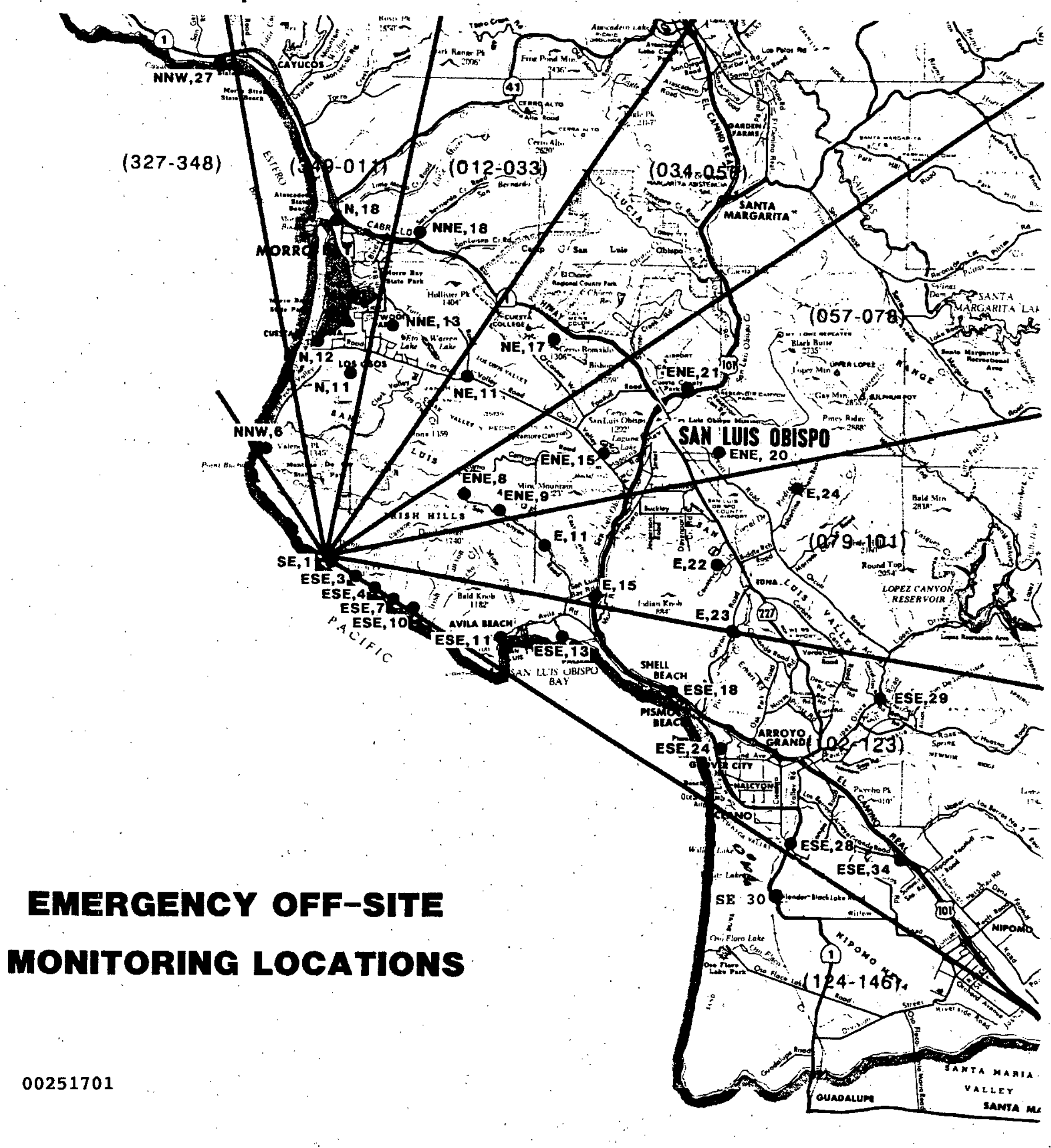
The Company has approximately 31 direct radiation monitoring stations in the vicinity of the plant which are part of its ongoing environmental monitoring program. Each station is equipped with thermoluminescent dosimeters. Fifteen of these stations are located on-site and sixteen are located off-site.

Twenty-one of the stations are located in order to provide an inner ring of stations in the general area of the site boundary and an outer ring in the 4-5 mile range from the site with a station in each land sector of each ring. Since the site is on the coast surrounded by a hilly, generally inaccessible area, some areas of the site boundary and in the range of 4-5 miles are inaccessible. However, a monitoring station has been placed to the closest accessible location to the stated criteria. Some of the above monitoring locations are also equipped with an air sampler fitted with a particulate filter and iodine cartridge.

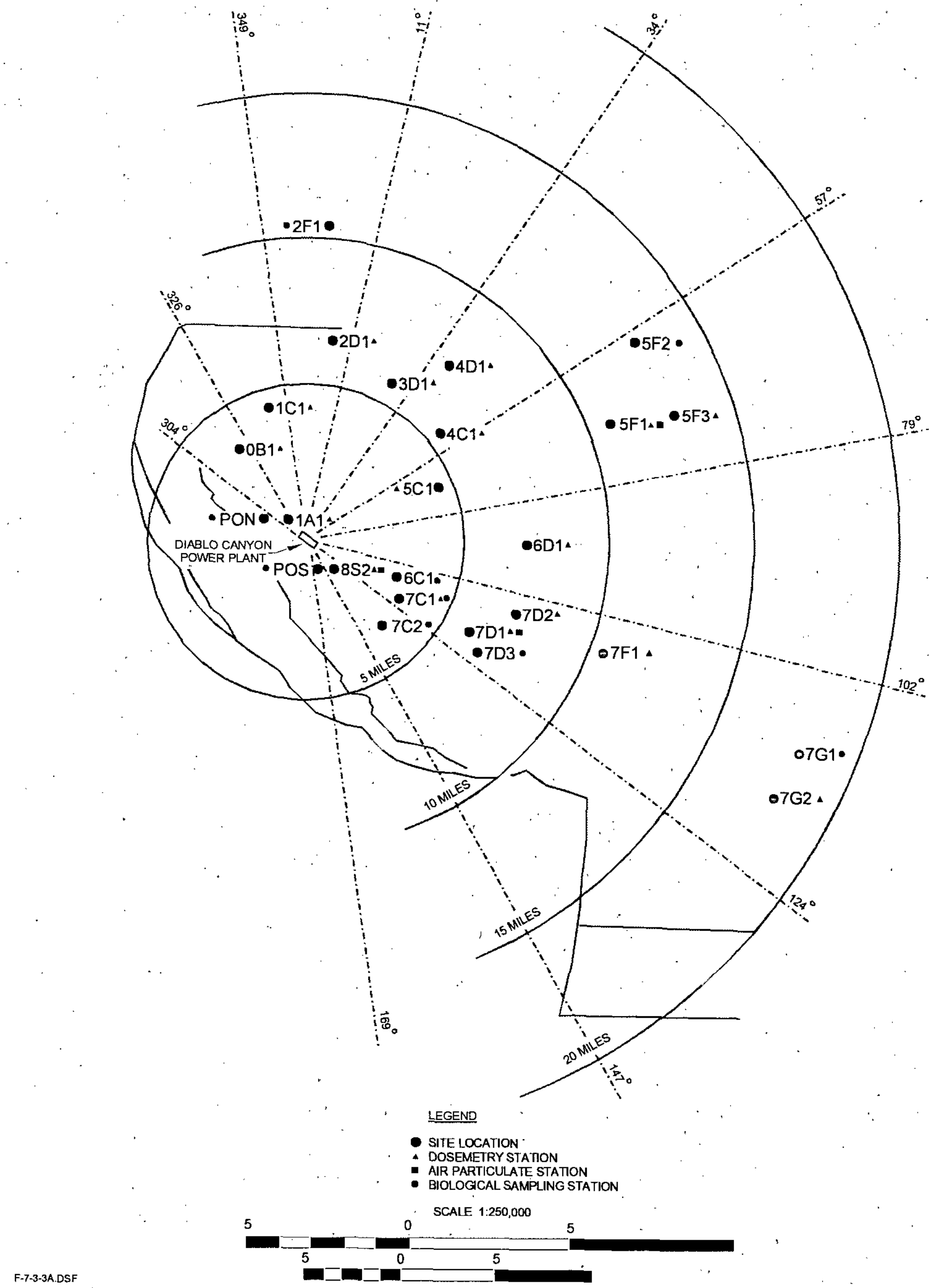
Radiation monitoring stations are located in Santa Barbara County in the cities of Orcutt, Lompoc, and Solvang.

Maps of monitoring locations are shown below.

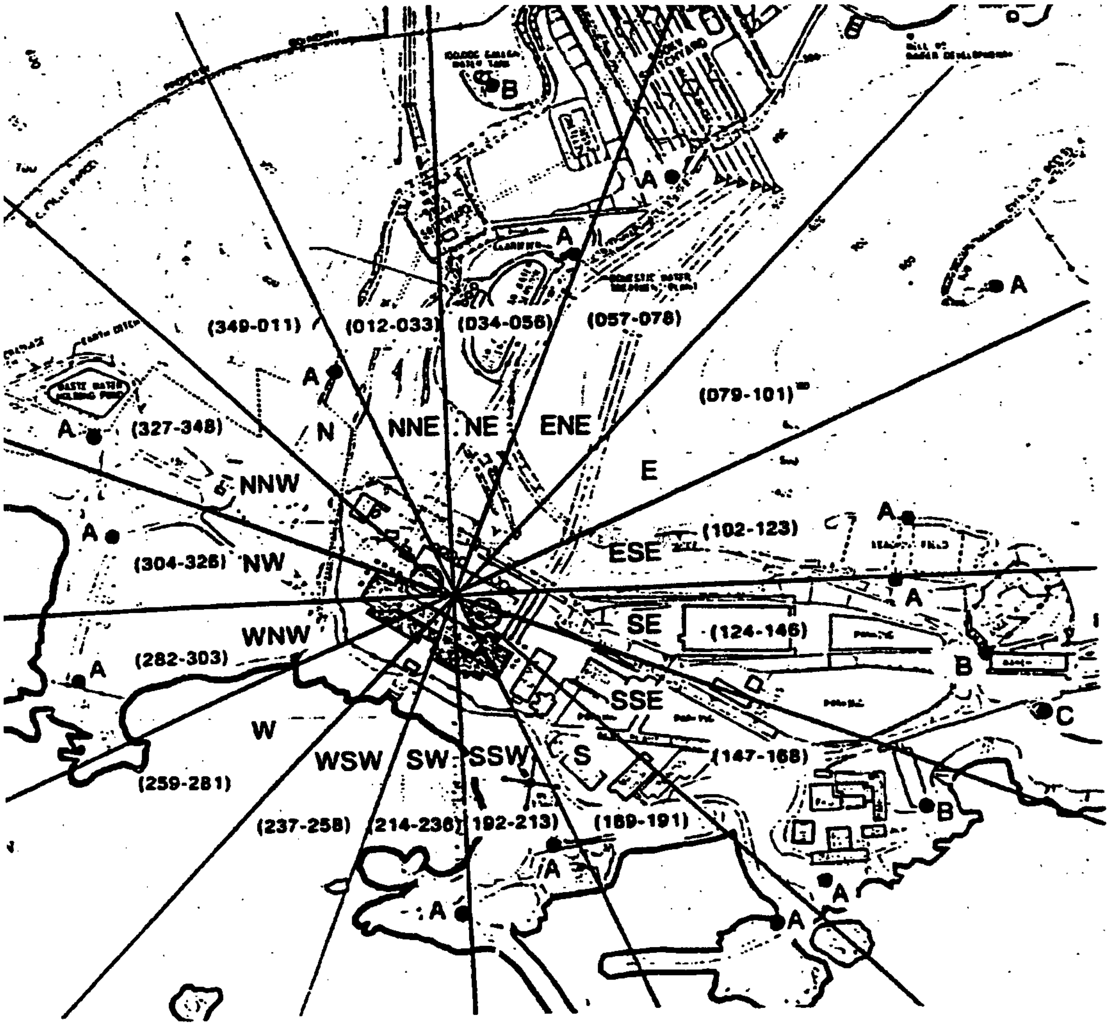
Emergency Off-Site Monitoring Locations



Off-Site Environmental Monitoring Stations



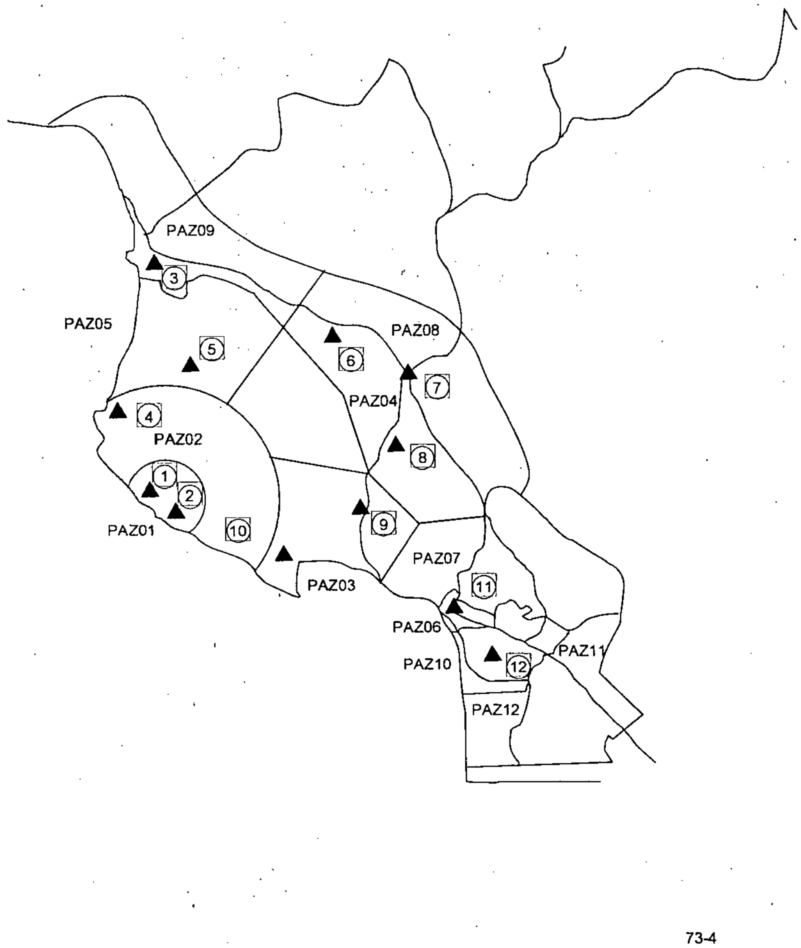
On-Site Monitoring Stations



Monitoring Locations
A, B, and C within Quadrant

7.6.4 Off-Site Radiation Monitoring System

The real-time radiation dose-rate monitoring system consists of twelve low-level gamma radiation dose-rate monitors; ten are installed at the various population centers in the vicinity of Diablo Canyon Power Plant, and two located on the power plant as shown and listed below.



PIC #	Description
1	DCPP North Gate Guard Post.
2	SSW Corner of Target Range (near on-site field monitoring location, SE, B).
3	Morro Bay Power Plant, near front gate.
4	Montana de Oro State Park. At the Park Ranger's residence, adjacent to the siren.
5	South Bay Fire Department.
6	Outside rear entrance to EOF (south side of building) (NE, 17).
7	SLO Police Department. Intersection of Santa Rosa Street and Walnut Street. Behind fence SW of Walnut Street driveway.
8	SLO Service Center, 4325 So. Higuera St., SLO (OEL Garage).
9	PG&E Information Center (E,15).
10	DCPP Front Gate.
11	Pismo Beach at the PG&E Pismo Warehouse on Price Canyon Road, about 0.5 miles NE of the Price Street intersection. PIC is located in the upper parking lot.
12	SLO County Building, Grover Beach. Exit Hwy 101 at 4th street. Take 4th to Longbranch and proceed NE on Longbranch. County Social Services Building is on corner of 16th and Longbranch.

These monitors employ a pressurized ion chamber (PIC) detector for accurate measurement of low-level gamma radiation ranging between background readings (<0.01 mR/hr) up to 10R/hr. Real-time PIC data can be read locally on a graphic display. In addition, PICs are polled by the EARS data acquisition subsystem in the TSC and 1-minute averaged data is made available to all plant computers and emergency response facilities via the Plant Data Network (PDN).

The purpose of the PIC system is to provide continuous measurement and reporting to PG&E, local and state agencies of gamma radiation dose rates in the environment around Diablo Canyon Power Plant. So that informed decisions to protect the public can be made. The technical description of the real-time radiation dose-rate monitoring includes:

1) Detector Type and Specifications

   a) Type: PIC Environmental Radiation Monitoring Station

   b) Sensitivity and Accuracy of the wide range gamma radiation sensor 0-10R/hr.

	Low	High
Range	0 μR/hr to 500 μR/hr	500 μR/hr to 10 R/hr
Accuracy	10 μR/hr: 5% above 10 μR/hr: better than ±5%	500 μR/hr: 8% above 800 μR/hr: better than ±5%
Sensitivity	10mV/R/hr	10 Hz/mR/hr

   NOTE: The sensor will perform within the stated accuracies over a temperature range of -25°C to +55°C degrees.

2) Data Transmitted to EOF

   The data is taken from the real-time monitors by the EARS central computer at the TSC and this data is transmitted by the PDN to the Radiation Monitor Data Display Terminal (RMDDT) stations in the TSC and EOF. The RMDDT stations are able to display map locations with all PIC station readings.

   Monitor data will be stored in a data base on the EARS central computer. Instantaneous readings are available within the previous 2 hours and one hour averages are available on an annual basis.

   Each monitor has a unique ID number which is coded to prepared maps of different scales indicating exact locations of all monitoring stations. Each map can be displayed on the station's CRT's with monitor numbers overlaid at their respective locations and monitor data on one side of the map. The displayed map can also be transformed to hard copy on printer.

3) Detector Locations

The real-time radiation monitors locations were selected on the basis of three criteria. The first is the need to obtain radiation levels in the population centers surrounding Diablo Canyon. The data received from the real-time monitors would not only provide routine background gamma radiation dose rate data but also gamma radiation dose rate in the event of a significant airborne release from Diablo Canyon to augment radiation data obtained by field monitoring teams. The second is the need to distribute the monitors over as many of the land based compass sectors as possible. The final criterion relates to the prevailing wind direction, wind speed and the topography in the area of the Diablo Canyon Power Plant.

7.6.5 Off-Site Laboratories

Either of the following two off-site radiological laboratories may be used in the event of an emergency.

1) PG&E's Off-Site Emergency Laboratory (OEL) at the PG&E Service Center in San Luis Obispo

   a) Location:

      (1) PG&E San Luis Obispo Service Center, 4325 Higuera Street, San Luis Obispo, California.

   b) Capabilities:

      (1) Gamma Spectrum Analysis: 35 samples per 24-hour day; samples may be in the form of filters, cartridges, liquids and solids.

   c) Equipment:

      (1) Intrinsic germanium and/or sodium iodide detectors with associated electronics and desk top computer.

      (2) Radio communications link with the Technical Support Center for voice transmission or analyses results.

      (3) Portable health physics instrumentation.

   d) Response Time:

      (1) During normal hours - two hours following notification.

      (2) During off-normal hours - two hours following notification.

2) Commercial Radiological Laboratory

   a) Location:

      (1) Charleston, SC.

   b) Capabilities:

      (1) Gamma Spectrum Analysis - 50 samples per 24-hour day; samples may be in the form of food, milk, water, air filters and iodine cartridges.

      (2) Gross Alpha/Beta Analysis: 50 samples per 24-hour day; samples may be in the form of air filters and water.

c) Equipment:

(1) Intrinsic germanium detector with associated electronics and computer system.

(2) Liquid scintillation detectors.

(3) Gas proportional detector.

d) Response Time:

(1) During normal hours - staff existing at facility.

(2) During off-normal hours - two hours following notification.

7.6.6 Off-Site Emergency Laboratory (OEL)

Off-site field sample analysis equipment is stored at the PG&E Service Center in San Luis Obispo. The OEL analytical equipment is equipped with a gamma spectroscopy analysis and detection system, portable health physics survey equipment; communications equipment; and independent power supplies.

7.7 MISCELLANEOUS PROTECTIVE FACILITIES AND EQUIPMENT

7.7.1 Installed Smoke, Flame & Heat Detectors

Smoke, flame and heat detectors which are located throughout the plant are designed to give early warning of possible fire conditions. These detectors are annunciated on alarm panels in the Control Room.

When an alarm is received in the Control Room a plant operator will be sent to investigate the cause of the alarm. If a fire condition exists, it should be reported by calling telephone extension 779. Fire suppression system alarms (i.e., sprinkler, deluge, CO2 or halon) also annunciate in the Control Room. These alarms could indicate a potential fire condition and shall be investigated in the same manner as any fire detection alarm.

7.7.2 Fire Detection by Personnel

Reporting of fires takes precedence over fighting a fire. Only personnel trained in fire fighting equipment use should attempt to suppress a fire. The fire alarm signal system is the normal way to report a fire. The fire signal is initiated by dialing from any telephone on the plant site. The call goes to a Control Room Operator who receives information from the caller on a conference line. The fire alarm is sounded by a 30 second monotone sound. The alarm is followed by a public address system message with details about the fire.

The fire signal is a 30-second blast on the fire sirens.

When a fire alarm is sounded, the typical response is to perform the following duties immediately: The Fire Brigade can be made up of IFOs (i.e., Fire Captains and Fire Fighters) and Operations personnel. The Fire Brigade Leader can be a Fire Captain or a licensed operator that is Fire Brigade Leader qualified. A qualified nuclear operator or licensed operator will accompany the Fire Brigade Leader unless the Fire Brigade Leader is the qualified nuclear operator or Licensed Operator. (See FSAR 9.5H.) The Industrial Fire Officers (IFOs) are professional fire fighters and makeup the rest of the Shift Fire Brigade for Fire, Rescue and Hazardous Material Emergencies. The Fire Captain is trained as a Fire Brigade Leader and has the primary role to support the Fire Brigade as an "Operations Officer" in the Incident Command System (ICS). The IFOs will be part of the 5 man Fire Brigade required by the FSAR.

The designated Shift Brigade Members will don protective equipment, pick up radios, proceed to the fire and maintain communication with the Control Room while fighting the fire.

The Security Department will upon notification of a fire, send a security officer to report to the Fire Brigade Leader.

Security will assist with access and the staging of off-site fire response personnel including providing dosimetry and escorts as necessary. Security will also assist the fire brigade with security barriers and ensure that plant security is not compromised during the emergency response.

The SM/SEC will notify California Department of Forestry, San Luis Obispo County Fire (CDF/SLO). If the telephone lines are unavailable, the CDF/SLO radio will be used.

Notification shall be made for any of the following conditions:

- All structure fires or any fire that presents a threat to personnel or plant systems.
- Report of smoke within a structure with no known source for the smoke.
- Any wild land fire.
- Any non-fire emergency that would require a CDF rescue or hazardous materials response.
- Any time the Fire Brigade Leader or SM/SEC recommends additional assistance.

Diablo Canyon control room staff will cancel or modify the original request by contacting the CDF Emergency Command Center utilizing their 24-hour emergency phone line.

7.7.3 Plant Fire Detection and Suppression and Respiratory Protection

Plant fire detection and suppression systems and respiratory protection equipment are described in the FSAR.

7.7.4 Halon 1301 Systems

Halon 1301 Systems are present in but not exclusive to the following:

1) Learning Center Building - There are Halon 1301 cylinders of various sizes set up to protect six different areas within the building. The areas are activated through either a manual pull-station or smoke detectors (photo-electric and ionization) that are cross zoned for automatic release. A full detector activation on the simulator area will open main power breakers for the simulator and the computer, close ventilation dampers and will delay a Halon release for 30 seconds after an alarm. Two different alarms will sound prior to discharge. The first is a ringing bell that is caused from an activated detector. The second alarm is a flashing strobe light and a horn that has been activated from a second detector. The second alarm will bring a discharge of Halon within 30 seconds.

2) The Administration Building - Halon system protects microfiches and sixth floor PC Network rooms and 1st floor Telecommunication room. The discharging of halon is caused by detector activation in the same way as the Learning Services Building.

3) The I&C/Tele Com/Medical Building - The Halon system protects the second floor telecommunication room. Halon discharge is caused by detector activation in the same way as the Training Building.

4) Document Storage Building (Building 604) - Halon discharge is caused by detector activation.

7.7.5 Mobile Fire Fighting Equipment

Mobile fire suppression equipment is provided to support a fire brigade response to areas outside of the main plant structures including construction areas and surrounding wildland. This equipment includes a 1987 class "A" rated fire engine pumper that includes a 1250 gpm two stage fire pump with a 1000 gal water tank. It carries 600 ft of 5", 600 ft of 3" and 600 ft of 1-3/4" fire hose. The engine also includes an installed foam proportioner with two 10 gallon storage tanks (10 gallons of class A foam, 10 gallons of class B foam).

7.7.6 Respiratory Protection Equipment

Respiratory protection equipment as described in the FSAR is available for emergencies involving fires and airborne radioactive materials.

7.7.7 Self-Contained Breathing Apparatus

MSA Model 401 pressure demand self-contained breathing apparatus units (SCBA) or Interspiro 4530 or equivalent are available at the site. Locations and numbers are as follows:

Location	SCBA	Extra 30-Minute Tanks
TSC	10	0
Control Room	10	0
Fire Brigade (various locations)	18	16
Access Control	20	20
Storage (various locations)	50*	64*
Total	108	100

NOTE: Up to one-half of this number may be removed for servicing.

Two on-site facilities are available to recharge SCBA tanks. They are located in: the 85' Aux. Building (RCA and the Unit 2 East Buttress 85'. One portable Poseidon Air Compressor capable of refilling either high pressure (4500 psi) or low pressure (2200 psi) is also available.

7.7.8 Constant Flow Air Line Respirators

The plant employs oil-free air compressors. As a result, the plant service air system can be used to supply breathing air. Approximately 20 MSA "clearvue" full facemasks outfitted as constant flow air line respirators are available. Duo-flow regulators allow the respirators to be used with HEPA filtration or as a constant flow system.

The regulators of these facemasks require a supply pressure of 35 to 40 psig. Since plant air supplies are at a higher pressure than this, pressure regulator boxes are used between the supply and the facemask. These boxes have manifolds where up to six separate air hoses can be plugged into the pressure regulated air. Approximately 10 of these regulator boxes are available at the site. Low pressure alarms are available to let the monitor know when pressure drops below the setpoint.

7.7.9 Radiological Protective Clothing

The plant is stocked with a considerable supply of protective clothing for normal personnel use when working in radiologically controlled areas of the plant. Typically, sufficient clothing is available for about 500 people.

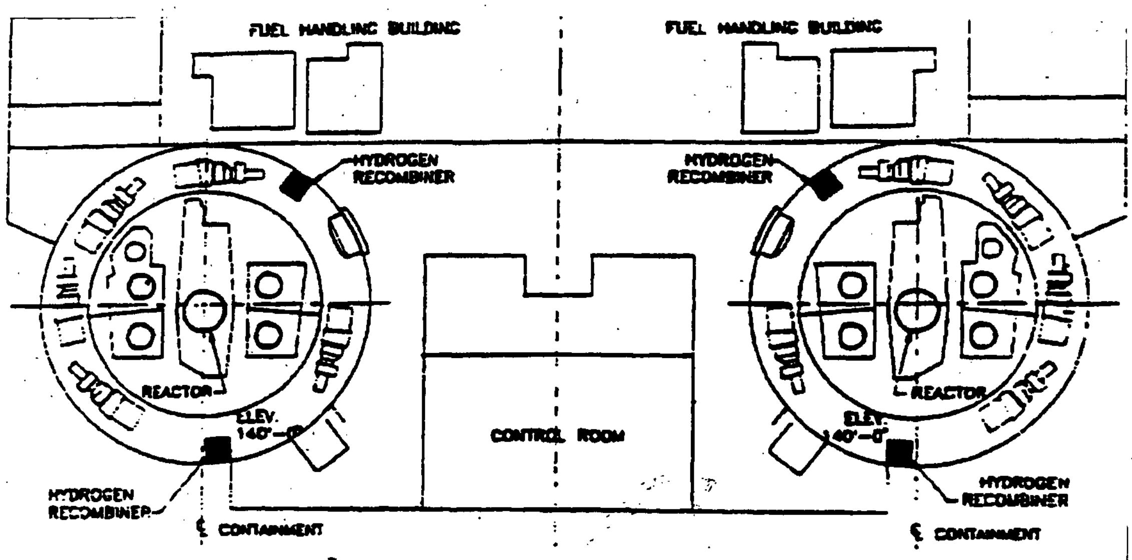
7.7.10 Containment Hydrogen Recombiners

The electric hydrogen recombiner system consists of two completely independent subsystems. Each subsystem is capable of providing the required hydrogen removal capacity.

Each subsystem consists of a recombiner unit which is located in the plant containment building at elevation 140' and a control panel and power supply. The latter two components are located outside the containment in the Auxiliary Building which is an accessible structure following a loss-of-coolant accident (LOCA). Connections external to the containment for removal of hydrogen gas from inside containment by portable recombiner units are provided on the post-accident hydrogen purge system containment penetrations.

Each recombiner unit is a constant volume device with a flow of 100 cfm. It has a minimum hydrogen removal rate equivalent to a removal efficiency of 98 percent with a process gas hydrogen concentration of 4 percent.

The recombiner unit consists of an inlet preheater, a heater-recombination, and an exhaust chamber. The unit is completely enclosed and the internals are protected from impingement by containment spray. The inlet and outlet ports are louvered to minimize moisture entering the unit.



7.7.11 Permanent Lighting Systems

The plant is provided with independent lighting systems:

1) Normal AC Lighting

   Normal lighting is operated at 208/120 volts supplied from the 480-volt system. The DC emergency lighting is supplied at 125 volts from two of the station batteries and is limited to a period of approximately two hours. This lighting is located principally in control and electrical equipment rooms, stairways, exits and entrances, corridors, and passageways.

2) Emergency AC Lighting

   The AC emergency lighting is supplied from two vital 480-volt buses. It is located throughout the plant to provide minimum lighting. The AC emergency lighting is routed in separate conduits from the normal AC lighting.

3) Emergency DC Lighting

   The DC lights are also in separate conduits in vital operating areas of the auxiliary building.

   After the diesel generators start and the single phase AC emergency transformers receive power, the DC emergency lights are automatically relayed off. An overlapping time delay keeps the DC lights operating while the AC emergency lights return to service.

Lighting panels and circuits breakers were tested by the supplier to satisfy seismic requirements. Lighting panel wall attachments have been calculated to be adequate for seismic loads. Conduit supports have adequate seismic supports. The standard fixtures for DC lights are direct mounted to building surfaces, and are therefore subject to building seismic forces. The fixtures will not experience appreciable amplification of seismic forces.

7.7.12 Portable Lighting

Numerous battery-powered lanterns and flashlights are maintained in the storeroom for normal and emergency use. Additional units are stock items in the warehouse.

7.7.13 Transportation

7.7.13.1 Plant Vehicles

There are several types vehicles at the plant site, which can be used to provide emergency transportation. These include:

1) Two-wheel drive vehicles

   There are a variety of two-wheel drive sedans, station wagons as well as quarter ton, half ton, and three quarter ton pickup trucks at the plant site for general use and may be made available for use during an emergency.

2) Four-Wheel Drive Vehicles

   Four-wheel drive vehicles are provided for use at the plant site. These vehicles may be used by the radiological monitoring teams to obtain field measurements in rugged terrain.

7.7.13.2 Ambulance

Prior arrangements exist with local ambulance services in San Luis Obispo for on-site service. An agreement also exists for air ambulance service. The agreements include transportation of contaminated victims if required.

7.7.13.3 Off-Site Company Vehicles

The Company has a fleet of automobiles and trucks in its San Luis Obispo service area which can be mobilized in an emergency if required.

7.7.13.4 Air Transportation

The Company has air transportation capability which is normally based in Oakland, California, and available for emergency use.

The Company also has access to a variety of charter aircraft and helicopter companies for emergency use, including transportation of contaminated victims using an enclosed type litter.

Finally, military helicopters may be available for emergency use. The closest one is at Vandenberg Air Force Base. These helicopters, capable of carrying three litters and four ambulatory persons, can be obtained through the San Luis Obispo County Sheriff's Watch Commander. These helicopters are available around the clock.

7.7.13.5 Other Modes of Transportation

The Company owns and maintains two marine crafts at the Diablo Canyon Power Plant. These vessels are available for personnel transport in the event road access is unavailable.

7.8 FIRST AID AND MEDICAL FACILITIES

The facilities for medical attention and personnel decontamination are located in the Building 102 Medical Facility immediately to the west side of the Turbine Building. Access to the Radiological Controls Areas (RCA) of the plant can be accomplished on foot or with the site emergency vehicle. Off-site ambulances can also readily access the Medical Facility or the RCA. Decontamination effluent generated at the Medical Facility is contained for proper disposal.

The DCPP Medical Facility provides routine occupational medical care as well as emergency care for sick or injured personnel. During normal business hours, a physician's assistant is in charge.

Emergency medical service is available 24 hours a day, seven days a week by registered nurses or trained EMTs.

7.8.1 Personnel Decontamination Facilities

A decontamination room is provided at Access Control on elevation +85' of the Auxiliary Building which has a sink and shower that drain to the liquid radwaste laundry collection tanks. Various decontamination aids are provided, such as brushes, skin decontamination soaps, rubber gloves, creams, wiping tissues, towels, etc. Monitoring instrumentation is readily available from the access control instrument storage.

7.8.2 First Aid Kits and Stretchers

Standard Company first aid kits and basket stretchers are placed at various locations throughout the plant.

7.8.3 Whole Body Counters

The plant has two computer-based whole body counters. The vertical model has an NaI detector. The other is an IGE bed counter, which if circumstances require, can be configured as a NaI bed counter.

7.9 CROSS REFERENCE TO NUREG-0654

NUREG 0654	DCPP Emergency Plan	NUREG 0654	DCPP Emergency Plan
C.1.a to c	7.2, 7.1.6.1	H.6.b	7.6.3, 7.6.4, 7.6.5, 7.6.6
C.3	7.5.5, 7.6.5, 7.6.6	H.6.c	7.5.8, 7.6.5, 7.6.6
C.4	7.6.5	H.7	7.5.6.1, 7.5.8, 7.6.3, 7.6.4
E.6	7.4	H.8	7.5.2
F.1.a	7.2.4, 7.2.7, 7.2.11	H.9	7.1.5.4
F.1.b	7.2.7, 7.2.11	H.10	7.1.5.4
F.1.c	7.2.7, 7.2.9, 7.2.11	H.11	7.1.7.4, 7.5.7, 7.5.8, 7.6.6, 7.7.3, 7.7.5, 7.7.6, 7.7.9, 7.8.1, 7.8.2
F.1.d	7.2.1, 7.2.2, 7.2.4, 7.2.5, 7.2.6, 7.2.11	H.12	7.1.4, 7.1.6, 7.6.6
F.1.e	7.2	I.2	7.5.10, 7.5.10
F.1.f	7.2	I.3.b	7.5.3, 7.5.4, 7.5.6, 7.5.6.1
F.2	7.2.7	I.4	7.5.6.1
F.3	7.1.5.4	I.5	7.5.2, 7.5.6.1
G.3.b	7.1.6	I.7	7.6.3, 7.6.4
H.1	7.1.4, 7.1.5	I.9	7.6.3, 7.5.8
H.2	7.1.6	I.10	7.5.6.1
H.5	7.5	J.10.a	7.5.6.1, 7.6.3, 7.6.4
H.5.a	7.5	J.10.c	7.4
H.5.b	7.5.3, 7.5.4, 7.5.5, 7.5.6, 7.5.6.1, 7.5.8, 7.5.10	K.5.a and b	7.8
H.5.c	7.5.4, 7.5.10	K.7	7.1.7, 7.8.1
H.5.d	7.5.9	L.2	7.8
H.6.a	7.6.1, 7.6.2	L.4	7.7.13

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8. MAINTAINING EMERGENCY PREPAREDNESS

The Diablo Canyon Power Plant emergency preparedness program consists of:

- Personnel training
- Participation in scheduled drills and exercises
- Regular emergency plan review and evaluation by company specialists and PG&E management.

The Senior Vice President, Generation and Chief Nuclear Officer, is the individual charged with overall authority and responsibility for emergency preparedness planning and training of Company emergency response personnel for DCPP. The administrative duties of managing the routine personnel training and updating the Plan, its appendices and the associated procedures is delegated through the Site Vice President to the Emergency Planning Manager. The Emergency Planning Manager is assigned the collateral duty of offsite emergency planning coordination with local government agencies. In this capacity, he provides support to training programs involving non-company support agencies.

8.1 TRAINING OF PLANT STAFF PERSONNEL

Plant personnel participate in a program of lectures, demonstrations, written assignments, Computer Based Training, and drills designed to familiarize them with fire protection and emergency procedures, and radiation protection principles. First aid techniques are offered and made available to those site personnel requesting it. Except as noted, courses are conducted by qualified plant personnel.

8.1.1 General Employee Training

All personnel receive instruction on the fundamentals of radiation protection, emergency signals and response, and site rules and regulations.

Additional instruction, as required based upon level of access, is provided in the basics of quality control, fire protection, industrial safety and contamination control, to those individuals who will be granted plant protected/vital areas access. Requalification is required on an annual basis for those individuals permitted protected/vital area access.

8.1.2 Emergency Plan Training

Emergency Plan training is incorporated into everyone's training program at the appropriate level. This training includes descriptions of the emergency categories, emergency response organization, facilities, and extent of emergency response.

Personnel assigned to specific emergency response organization positions receive additional, specialized radiological emergency plan training. This instruction is advanced, task specific training, designed to provide each site emergency organization position with the skills/knowledge required for the performance of emergency response tasks.

The positions in the site emergency response organization include the following categories:

- Coordinators of the response organization (Site Emergency Coordinators)
- Personnel responsible for accident assessment (Radiological Advisor, Shift Technical Advisor) Radiological monitoring teams and radiological analysis personnel (Health Physics Technicians and Radiological Data Processors)
- Security (Security Advisor, Security Force)
- Repair and damage control/correctional action teams (Maintenance Coordinators, Team Coordinator)
- Personnel responsible for transmission of emergency information and instructions (Communications Advisor, Agency/ENS Communicator, Ops Communicator, Communications Coordinator, Offsite Communicator)

Re-qualification is required on an annual basis for those personnel with specific assignments in the emergency response organization.

8.1.3 Emergency Operating Procedures

All plant operators above the level of Nuclear Operator, plus selected members of the engineering staff, receive training under the provisions of NRC licensing requirements. This instruction, provided in the Licensed Operator and Senior Operator Training and Retraining Programs, includes nuclear power fundamentals, plant systems, operating practice, procedures and experience and simulator training for normal, transient and emergency operation of the plant. Re-qualification is required on an annual basis.

8.1.4 Fire Protection

In addition to the general fire protection training provided to individuals as part of General Employee Training, members of the plant Fire Brigade receive instruction in fire fighting tactics, procedures and practical fire fighting as part of the Fire Brigade and Emergency Response Training Program, in accordance with FSAR Appendix 9.5H requirements.

8.1.5 First Aid and Accident Prevention

The first aid and accident prevention program is an integral part of Company activities at Diablo Canyon. First aid training includes cardiopulmonary resuscitation instruction for requesting plant employees. Accident prevention meetings are conducted on a continuing basis for plant employees. New employees receive an industrial safety orientation program.

8.1.6 Retraining

Retraining is required, on an annual basis, for those personnel with emergency response assignments. Participation in drills and exercise may satisfy some course retraining requirements if a person's performance is satisfactory.

8.2 TRAINING OF OFFSITE PERSONNEL

8.2.1 Corporate Support Personnel

Corporate support personnel are assigned emergency functions which are extensions of their regular duties and minimal training is required. Where persons may be required to fulfill new or unique tasks, special training will be provided as discussed in the PG&E Corporate Emergency Plan.

8.2.2 Non-Company Support Agencies

The development of training programs for individuals from offsite agencies is primarily the responsibility of the agency involved. This is particularly true in their areas of technical competence. The Company provides supplemental training on radiation protection, offsite monitoring, and other topics related to the Emergency Plan for agencies that require and request it. Normally this training will be scheduled on an annual basis. Examples of these programs include:

1) Company personnel contact local support services personnel, such as the County Emergency Services Coordinator, Sheriff, Marian and French Hospitals, San Luis Ambulance, and the CDF/County Fire on an annual basis to reaffirm the arrangements of the Emergency Plan. The need for special training programs is discussed at these meetings.

2) Selected physicians at Marian and French Hospitals, as well as other selected local physicians are sent to the Radiation Emergency Assistance Center/Training Site (REAC/TS) courses in the handling of radiation accidents conducted by the Oak Ridge Associated Universities.

3) Company or contracted personnel conduct periodic training seminars in monitoring, decontamination, and other radiation protection topics for emergency room personnel at Marian and French Hospitals.

4) Department of Forestry fire fighting personnel that respond to emergencies at DCPP involving radiation hazards receive initial and annual training. The initial training includes radiation protection training and a tour to review plant layout and target hazards. Refresher training includes radiation protection training.

5) The Company assists the County Health Department to train personnel in the use of environmental radiological monitoring equipment and radiological dose assessments.

8.2.3 Public Education

Educational information is distributed to the public within the Basic Emergency Planning Zone (BEPZ) regarding warning procedures used in a major emergency and the protective actions that may have to be taken by the public.

1) An emergency planner describing warning procedures and protective actions is distributed annually to persons residing or doing business in the Basic Emergency Planning Zone. The planner contains information about the plant, radiation and its health effects. The planner is revised annually and distributed within the San Luis Obispo County telephone book. A separate pamphlet describing warning procedures and protective actions is included along with the telephone book distribution. The public is advised that the planner is available separately for those without phone books.

2) Siren information stickers are distributed to all businesses, hotels, motels and places of public assembly within the BEPZ. The stickers advise tuning your radio to one of the primary emergency alert system radio stations if a steady siren sounds for 3 to 5 minutes.

3) When requested, trained spokespersons present information on warning procedures and protective actions to civic and other groups throughout the Basic Emergency Planning Zone.

4) Information on warning procedures and protective actions are presented to key groups within the Basic Emergency Planning Zone.

Those businesses and organizations accommodating transient populations are provided with siren information stickers. Signs telling what to do if sirens sound have been posted in parks and scenic locations within the BEPZ.

5) News media staffs (Editors, reporters, newsroom personnel, etc.) are offered special programs annually to acquaint them with the power plant and its characteristics, emergency plans, radiation and radioactive effects and points of contact for public information releases in an emergency.

6) Copies of printed material are available at the PG&E district office in the Basic Emergency Planning Zone, the PG&E Community Center, and the County Office of Emergency Services.

8.3 DRILLS AND EXERCISES

Periodic drills and exercises involving both onsite and offsite emergency organizations are conducted to provide assurance that a high level of familiarity with the Emergency Plan is maintained and to expose any weakness in its execution. Drills are conducted to maintain emergency response skills and to update skills when modifications to the Emergency Plan or equipment require. During drills, if appropriate, on-the-spot correction of incorrect performance is made and demonstration of the proper performance offered by the instructor.

The Site Vice President along with other key supervisory personnel from both the plant staff and other participating offsite organizations, is responsible for planning and scheduling drills and exercises. Following the drill or exercise, observers and principal participants from all agencies involved will meet to critique emergency response. Whenever drills or exercises involve participation of offsite agencies, particular emphasis will be placed on the review of the coordination between the Onsite Emergency Organization and communication links and notification procedures of offsite emergency organizations.

8.3.1 Drills

A drill is a supervised instruction period aimed at testing, developing and maintaining skills in a particular operation. Specifically, drills are intended to test the capability to respond and perform assigned emergency functions. Drills can be combined for actual execution, i.e., a Medical Drill or Health Physics Drill may be combined with a Full-Scale Drill.

1) Communication Drills

   a) Communications with San Luis Obispo County Sheriff and California Office of Emergency Services response centers will be tested at least monthly.

   Routine telecommunication checks from the Control Room are conducted with the County Sheriff's office and State OES on a weekly basis in addition to the communications drill.

   b) Communications between the Plant, Corporate Response Centers and EOF, and field assessment teams will be tested annually.

2) Fire Drills

   Fire drills are conducted in accordance with FSAR Appendix 9.5H requirements.

3) Medical Emergency Drills

   A medical emergency drill involving a simulated contaminated individual who is transported by ambulance to a medical facility will be conducted annually.

4) Radiological Monitoring Drills

A plant environs and radiological monitoring drill will be conducted annually. This drill will include collection and analysis of environmental sample media (e.g., water, vegetation, soil and air) and provisions for communication and record keeping.

5) Health Physics Drills

a) Health Physics drills will be conducted semi-annually which involve response to and analysis of simulated elevated airborne and liquid samples and direct radiation measurements in the environment.

b) Analysis of inplant samples and the use of the post-accident sampling system will be included in the Health Physics drills annually.

6) Ingestion Pathway Zone (IPZ) Drills

At least every twelve years, an IPZ drill will be conducted with California State participation.

8.3.2 Exercises

1) An exercise is an event testing the integrated capability, and a major portion of the basic elements, existing within emergency preparedness plans and organizations.

2) Full-scale team drills that involve County and State participation meet the criteria for an "exercise" as defined in NUREG 0654. As such, they can also be used to meet the six-year requirements for off-hours and unannounced exercises.

3) An evaluated exercise will be conducted biennially. In the graded exercise, County personnel and resources will be mobilized to verify the adequate capability to respond to an accident scenario. The graded exercise will be critiqued by the NRC and FEMA sponsored observers / evaluators.

4) The scenario should be varied from year to year such that all major elements of the plans and preparedness organizations are tested within a six year period. This will include provisions for an off-hours exercise/drill. The exercises/drills will be run under various weather conditions. Some exercises/drills will be unannounced.

8.3.3 Scenario Development

Scenarios for exercises and drills are to be varied to cover all elements of the plans and their implementing procedures and to exercise the preparedness organization. Federal documents provide guidance on the time-of-day, weather, frequency, free-play flexibility and pre-announcement criteria to be imposed. They also require exercise scenarios to be submitted to FEMA (when applicable) and the NRC for prior approval. In any event, these parameters are to be established through concurrence with the reviewers prior to the exercise or drill. As a minimum, each event scenario text shall include the following:

1) The basic objectives of each drill and exercises and appropriate evaluation criteria;

2) The dates, time period, places and participating organizations;

3) The simulated events;

4) A time schedule of real and simulated initiating events;

5) A narrative summary describing the conduct of the exercises or drills to include such things as simulated casualties, offsite fire department assistance, rescue of personnel, use of protective clothing, deployment of radiological monitoring teams, and public information activities and,

6) A description of the arrangements for and advance materials to be provided to official observers.

Scenarios are to be developed by the Company in conjunction with the County authorities. In each case, they are to be developed and administered by third parties to the actual exercise. Scenarios for exercises which are to receive FEMA evaluation and approval require prior approval by the Federal Regional Advisory Committee (RAC), Region IX.

8.4 EMERGENCY PLAN AND PROCEDURES

8.4.1 Review and Updating

The DCPP Emergency Plan is reviewed and updated annually. The intent of the review is to update and improve the plan based on plant changes, drills and exercises results, and assess onsite capabilities. Based on that review, the associated implementing procedures are updated as necessary. Following any revisions to the Emergency Plan, all participating agencies in the total emergency organization which are affected by the revisions are apprised of the changes through the distribution provided by the document control system. The Emergency Planning Supervisor maintains documentation substantiating the annual review and updating.

In addition, independent assessments of the various aspects of the emergency preparedness program are conducted every twelve months in accordance with 50.54(t). The independent assessment includes, but is not limited to, the Emergency Plan, implementing procedures, training, readiness testing, equipment, and interfaces with state and local organizations. The results are considered by management in modifying aspects of the plan.

8.4.2 Document Control

The Emergency Plan is included in the plant manual, Volume 11. Revisions to the plan will be controlled and distributed by the plant staff in accordance with plant document control procedures. This process requires review by the Plant Staff Review Committee (PSRC) and approval of the Site Vice President or Station Director. Changes will be issued periodically in the same manner to incorporate important modifications and information pending the issuance of a new revision.

Documentation of the activities required for emergency plan maintenance shall be in accordance with the standard practices defined by the Administrative Procedures, Volume 1 of the Plant Manual.

Changes and revisions will be issued to holders of Controlled Copies according to the distribution list maintained by Procedure Services.

8.5 IMPLEMENTING PROCEDURES

Each new Emergency Plan implementing procedure or procedure revision shall be reviewed, approved, maintained, and rescinded in accordance with plant administrative procedures.

Approval of new Emergency Plan implementing procedures or revisions shall be by the Station Director, or his designee. If approved by a designee, the designee shall be knowledgeable in the procedure's subject area and shall meet the experience requirements of ANSI/ANS 3.1-1978 for Plant Managers.

8.5.1 Temporary Changes to Procedures

Temporary changes to Emergency Plan implementing procedures may be made in accordance with plant administrative procedures.

8.6 LETTERS-OF-AGREEMENT

Local participating emergency organizations with which a letter-of-agreement has been signed will be contacted on an annual basis to certify continuing support of the agreement. Emergency support organizations that have an approved contract will be contacted prior to contract expiration. Contracts will be reviewed on an annual basis to ensure they are current.

In general, letters-of-agreements are appropriate for organizations that are expected to respond in a declared state of emergency. To be specific, if the DCPP Emergency Response Plan or any of it's appendices identifies an organization from which a particular service or response is required, then documentation must exist that there is prior recognition by both parties that a prompt, effective response will be initiated. Concurrently, documentation must clearly point out the degree of emergency preparedness that response organization must continuously maintain in order to be capable of carrying out the obligation when the mobilization request occurs.

The letter-of-agreement should have as a primary purpose the identification of points of contact and telephone numbers for requesting assistance and verifying requests.

8.7 OFFSITE AGENCY SUPPORT DOCUMENTS

1) Offsite agency support documents shall be maintained. These documents may include Emergency Response Plans from agencies such as:
   - San Luis Obispo County
   - State of California
   - Westinghouse
2) Other documents may include Emergency Procedures or Protocols form agencies such as:
   - French Hospital
   - INPO
   - U.S. Coast Guard

3) Changes to Offsite agency support documents are not controlled by Diablo Canyon and do not necessarily constitute a change to the Diablo Canyon Emergency Plan.

4) Changes to Offsite agency support documents will be reviewed for impact to the Diablo Canyon Emergency Plan.

8.8 EMERGENCY EQUIPMENT AND SUPPLIES

To ensure the availability and operational readiness of emergency equipment and supplies, provisions have been made for performing periodic maintenance, calibration and inventory checks of the equipment. Equipment that is only used in the event of an emergency, such as emergency kits, is inspected on a quarterly basis and after each use in compliance with procedures. Sufficient reserves exist to replace equipment removed from emergency kits for calibration or repair. Other equipment which is used during normal plant operation and may be used in an emergency, such as radiation detection instruments and communications networks, is included in plant surveillance testing and preventive maintenance programs. Additionally, inventories of expendable supplies are maintained in an onsite warehouse and are replenished when minimum levels are reached, thus maintaining an adequate supply at all times.

8.9 CROSS REFERENCE TO NUREG-0654

NUREG 0654	DCPP Emergency Plan	NUREG 0654	DCPP Emergency Plan
A.3	8.6	N.3.a to f	8.3.3
B.9	8.6	N.4	8.3.2
F.3	8.8, 8.3.1	N.5	8.3.2
G.1.a to d	8.2.3	O.1	8.1, 8.2
G.2	8.2.3	O.1.a	8.2.2
G.5	8.2.3	O.2	8.1, 8.3
H.9	8.8	O.3	8.1.5
H.10	8.8	O.4.a to j	8.1, 8.2, 8.2.2
L.2	8.1.5	O.5	8.1, 8.2, 8.2.2
N.1.a and b	8.3.2	P.1	8.1, 8.2, 8.2.2
N.2.a	8.3.1	P.2	8.1
N.2.b	8.1.4, 8.3.1	P.4	8.4
N.2.c	8.3.1	P.5	8.4
N.2.d	8.3.1	P.9	8.4.1
N.2.e	8.3.1	P.10	8.4

9. RECOVERY

This section provides guidance for the operational Recovery Phase following plant emergency declaration and response but prior to the return to normal operations. The emergency phase, in which an Emergency Classification Level (ECL) is declared at the Alert classification or higher, is addressed by the G-series and EF-series Emergency Plan Implementing Procedures. When the emergency is terminated and the plant no longer requires an Alert or higher emergency classification for the purposes of offsite notification, there may still remain recovery tasks to be performed outside of normal operations. This section addresses this transitional Recovery Phase.

9.1 RECOVERY PHASE

The Recovery Phase commences at the time at which an emergency at the Alert or higher level has been terminated (either de-escalated to a UE or to no ECL), but before the plant has returned to normal operations. In order to enter the Recovery Phase, the plant should be in a stable condition and any real or potential radioactive releases limited to less than license limits without the likelihood of again increasing beyond those limits. There should be little probability that the accident will recur or that a new, declarable event (Alert or higher) occur as a result of the previous emergency or response actions. The emergency phase goals of the Emergency Response Organization (ERO) for events at the Alert or higher level include the list below. Upon entering the Recovery Phase, these goals should have been mostly or wholly attained.

1) Bring the plant to a stable condition.

2) Mitigate any real or potential radioactive releases.

3) Minimize hazards to plant personnel.

4) Work with offsite agencies to minimize hazards to the public.

During the Recovery Phase, while the Emergency Director is still in control, management of the event is shifting from Emergency Response Organization as augmented by the incident Technical Review Group, Event Investigation Team, and, as necessary, the Outage Coordination Center. Use is made of these normal programs of widespread familiarity that deal with non-emergency operational problems in order to maximize the efficiency and effectiveness of the transition phase and minimize potential confusion and error.

The Recovery Phase is in turn terminated by declaration of the Emergency Director, generally at the time he determines normal operational programs are sufficient to close out the event, and his continuous personal control is no longer required at or near the site. The Recovery Phase goals for returning the plant to normal operations should include the following:

1) Terminate the emergency declaration.

2) Determine which ERO positions will initially be retained in the Recovery Phase.

3) Initiate the NCR process, including the Event Response Plan as a blueprint for Recovery activities.

4) Activate the Outage Coordination Center for significant plant material restoration.

5) Complete the transition to fully normal plant (non-ERO) management.

6) Assist the State as requested to determine population dose.

9.2 EXTRAORDINARY EVENT RECOVERY

In the more probable accident scenarios, the Recovery Phase would be expected to begin upon the termination of the emergency phase (downgrade of an event from the Alert level or above to an UE or to no ECL).

However, there is a possibility in the case of an extraordinary event that the Recovery Phase start may be appropriate while the plant still technically exceeds the criteria for an Alert or higher ECL, due to plant conditions or release (or potential release) status.

In such cases, the Emergency Director may start the Recovery Phase, either concurrent with or after terminating the emergency phase for offsite notification purposes, as agreed with the County, State and NRC authorities.

9.3 RESPONSIBILITY AND ORGANIZATION

1) The Emergency Director (ED) shall determine when the emergency event shall be terminated (downgraded to a UE or to no ECL). As appropriate in the case of an extraordinary event, the ED shall determine (as agreed by County, State and NRC officials) when the Recovery Phase should be commenced prior to the end of the emergency phase, or when the emergency phase may be terminated for offsite notification purposes and the Recovery Phase begun, though the plant is technically in an ECL at the Alert or higher level. The ED shall then direct the activities of the Recovery organization until he declares the Recovery Phase complete. At this time his position will be deactivated and general site area control will revert to the Plant Manager, and the use of Emergency Plan Implementing Procedures (EPIPs) will no longer be necessary to close out the event.

2) The Recovery Organization will consist of the normal plant management organizations as assisted by the event-related Technical Review Group, Event Investigation Team and Outage Coordination Center and as overlaid by remaining elements of the Emergency Response Organization determined necessary by the Emergency Director. These ERO positions will in time be deactivated prior to the ED declaring the Recovery Phase complete.

3) The structure and function of corporate positions during the Recovery Phase in the PG&E Emergency Operations Center (General Office) are addressed in the Corporate Emergency Plan.

9.4 CROSS REFERENCE TO NUREG-0654

NUREG 0654	DCPP Emergency Plan	NUREG 0654	DCPP Emergency Plan
B.7.a to d	9	M.3	9.1
M.1	9	M.4	9.16)
M.2	9.1		

10. REFERENCES

1) Nuclear Regulatory Commission, Emergency Planning, Final Rule, Title 10, Part 50 of the Code of the Federal Register (10 CFR 50), Appendix E, August 19, 1980.

2) Criteria for Preparation and Evaluation of Radiological Emergency Response Plans and Preparedness in Support of Nuclear Power Plants, NUREG-0654/FEMA-REP-1, Rev. 1, 10/80.

3) Final Safety Analysis Report (FSAR), Diablo Canyon Power plant.

4) Evacuation Times Assessment Transient and Permanent Population from Various Areas Within the Plume Exposure Pathway Emergency Planning Zone, Wilbur Smith and Associates, September 2002.

5) Manual of Protective Action Guides and Protective Actions for Nuclear Accidents, EPA-400-R-92-001, Environmental Protection Agency (EPA), October 1991.

6) Memoranda-of-Understanding (MOU) between the NRC and FEMA, Federal Register Vol. 45, Number 243, April 18, 1985/Notices.

7) Federal Radiological Emergency Response Plan, November 8, 1985.

A. PROCEDURES

This appendix lists:

- Procedures that implement the Diablo Canyon Emergency Plan.
- Cross-reference of Diablo Canyon Emergency Planning Procedures to the NUREG-0654 planning standards.

A.1 EMERGENCY PLAN IMPLEMENTING PROCEDURES

Radiation Protection & Assessment (Immediate Operator Response)

EP R-2	Release of Airborne Radioactive Materials Initial Assessment
EP R-3	Release of Radioactive Liquids

General

EP G-1	Emergency Classification and Emergency Plan Activation
EP G-2	Interim Emergency Response Organization
EP G-3	Emergency Notification of Off-Site Agencies
EP G-4	Assembly and Accountability
EP G-5	Evacuation of Nonessential Site Personnel

Training

OM10.ID1	Maintaining Emergency Preparedness
OM10.DC1	Emergency Preparedness Drills and Exercises

Organization

EP OR-3	Emergency Recovery

Equipment and Facilities

EP EF-1	Activation and Operation of the Technical Support Center
EP EF-2	Activation and Operation of the Operational Support Center
EP EF-3	Activation and Operation of the Emergency Operations Facility
EP EF-4	Activation of the Off-Site Emergency Laboratory
EP EF-9	Back-up Emergency Response Facilities
EP EF-10	Activation and Operation of the Joint Media Center

Radiation Protection & Assessment

EP RB-1	Personnel Dosimetry
EP RB-2	Emergency Exposure Guides
EP RB-3	Stable Iodine Thyroid Blocking
EP RB-4	Access to and Establishment of Controlled Areas Under Emergency Conditions
EP RB-5	Alternate Personnel Decontamination Facilities

Radiation Protection & Assessment (Immediate Operator Response)

RB-8	Instructions for Field Monitoring Teams
RB-9	Calculation of Release Rate
RB-10	Protective Action Guidelines
RB-11	Emergency Offsite Dose Calculations
RB-12	Plant Vent Iodine and Particulate Sampling During Accident Conditions
RB-14	Core Damage Assessment Procedure
RB-14A	Initial Detection of Fuel Damage
RB-16	Operating Instructions for the EARS Computer Program

A.2 CROSS REFERENCE OF PROCEDURES TO NUREG-0654

NUREG 0654	PROCEDURES	NUREG 0654	PROCEDURES
A.1.d	G-2, EF-1, EF-3	F.2	EF-3, M-13
A.4	G-2, EF-1, EF-3	F.3	OM10.DC1
B.1	G-2, G-3	G.1.a to d	OM10
B.2	G-2	G.2	OM10
B.3	G-2, EF-1, EF-3	G.3.a	EF-10
B.4	G-3, EF-1, EF-3, RB-10	G.3.b	EF-3, EF-10
B.5	G-2, G-3, EF-1, EF-2 EF-3, EF-4, EF-10	G.4.a	EF-3, EF-10
B.6	G-2, G-3, EF-1, EF-3	G.4.b	EF-10
B.8	EF-1, EF-3	G.4.c	EF-10
B.9	EF-1, EF-3	G.5	OM10.DC1
C.1.a - c	G-2, EF-1, EF-3	H.1	EF-1, EF-2
C.2.b	EF-3	H.2	EF-3
C.3	EF-4	H.4	G-2, EF-1, EF-2, EF-3
C.4	EF-1, EF-3	H.5	OM10.DC3
D.1 & D.2	G-1	H.5.a	M-4, M-5
E.1	G-3	H.5.b	R-2, R-3, RB-8, RB-12
E.2	G-2, G-3	H.5.c	RB-14, RB-14A
E.3	G-3, RB-10	H.6.a	M-4, M-5
E.4.a to n	G-3, RB-10 EF-1, EF-3	H.6.b	R-2, R-3, RB-1, RB-8, RB-12
E.6	G-3	H.6.c	RB-8, EF-4
E.7	EF-10	H.7	EF-4
F.1.a	G-3	H.8	RB-11
F.1.b	G-3	H.9	EF-2
F.1.c	G-3	H.10	MT-21, 25, 26, 27, 28, 29, 31, 35, 43, 50, RCP EM-16
F.1.d	G-2, G-3, EF-1, EF-3, RB-8	H.11	RB-8, M-13, MT-21
F.1.e	G-2, G-3, EF-1, EF-3	H.12	EF-4, RB-8
F.1.f	G-3, EF-1, EF-3	I.1	G-1

NUREG 0654	PROCEDURES	NUREG 0654	PROCEDURES
I.2	R-2, RB-9, RB-11, RB-14, RB-14A, RB-16	L.1	M-13, MT-21
I.3.a	R-2, R-3, RB-9, RB-11, RB-16	L.2	M-13
I.3.b	RB-9, RB-11	L.4	M-13
I.4	RB-9, RB-11	M.1	OR-3
I.5	EF-1, EF-3, RB-9, RB-11	M.2	OR-3
I.6	R-2, RB-9, RB-11, RB-16	M.3	OR-3
I.7	EF-4, RB-8	M.4	RB-11, OR-3
I.8	EF-4, RB-8	N.1.a and b	OM10.DC1
I.9	EF-4, RB-8	N.2.a	OM10.DC1
I.10	RB-8, RB-9, RB-10, RB-11	N.2.b	OM10.DC1
J.1.a to d	G-2, G-4, EF-1	N.2.c	OM10.DC1
J.2	G-5	N.2.d	OM10.DC1
J.3	G-5	N.2.e	OM10.DC1
J.4	G-5, RB-5	N.3.a to f	OM10.DC1
J.5	G-4	N.4	OM10.DC1
J.6.a	EF-1, EF-2, EF-9	N.5	OM10.DC1
J.6.b	EF-1, EF-2, EF-9	0.1	OM10.DC1
J.6.c	RB-3	0.1.a	OM10.DC1
J.7	G-3, RB-10	0.1.b	OM10.DC1
J.8	RB-10	0.2	OM10.DC1
J.10.a	RB-10, G-5	0.3	OM10.DC1
J.10.b	RB-10	0.4.a to j	OM10.DC1
J.10.c	RB-10, G-4, G-5	0.5	OM10.DC1
J.10.m	RB-10	P.1	OM10
K.1.a to g	RB-2	P.2	OM10
K.2	RB-2	P.3	OM10.ID2
K.3.a and b	RB-1	P.4	OM10.ID2
K.5.a and b	RB-5	P.5	OM10.ID2
K.6.a to c	RB-4, RB-5	P.9	OM10
K.7	RB-5	P.10	MT-25

A.3 CROSS REFERENCE TO NUREG-0654

This appendix implements NUREG-0654 Part II, P.7.

B. OFFSITE AGENCY SUPPORT DOCUMENTS

The following plans and procedures, which relate to response to an emergency at Diablo Canyon, are provided by the associated offsite agencies. These documents are maintained by Emergency Planning. These documents are reviewed for any impact to the Diablo Canyon Emergency Plan.

- San Luis Obispo County / Cities Nuclear Power Plant Emergency Response Plan
- State of California Nuclear Power Plant Emergency Response Plan
- Westinghouse Electric Company.Emergency Response Plan
- Diablo Canyon Power Plant Emergency Response Plan Implementing Procedures Listing
- French Hospital Emergency Response Plan
- Federal Radiological Emergency Response Plan
- Institute of Nuclear Power Operation INPO's Role in an Emergency
- U.S. Coast Guard Procedures to be Followed Upon Notification of an Emergency at Diablo Canyon Nuclear Power Plant

This appendix implements NUREG-0654 Part II, C.1.a, C.1.b, and C.1.c.

The following NUREG-0654 Part II planning standards are not applicable to the Diablo Canyon Emergency Plan.

- A.2.a
- A.2.b
- C.2.a
- D.3
- D.4
- E.5
- H.3
- I.11
- J.9
- J.10. d to l
- J.11
- J.12
- K.4
- L.3
- O.1.b

Diablo Canyon Power Plant Emergency Plan

Appendix D - Emergency Action Level Technical Basis Manual

Table of Contents

1. PURPOSE

This manual provides an explanation and rationale for each Emergency Action Level (EAL) for Diablo Canyon Power Plant (DCPP). It should be used to facilitate review of the DCPP EALs and provide historical documentation for future reference. Decision-makers responsible for implementation of EP G-1, Emergency Classification and Emergency Plan Activation, may use this manual as a technical reference and an aid in EAL interpretation.

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

2. DISCUSSION

2.1 Background

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

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

NEI 99-01 (NUMARC/NESP-007) Revision 4 represented the most recently accepted methodology at the time of DCPP submittal (December 2006). Enhancements over earlier revisions included:

- Consolidating the system malfunction initiating conditions and example emergency action levels which address conditions that may be postulated to occur during plant shutdown conditions.
- Initiating conditions and example emergency action levels that fully address conditions that may be postulated to occur at permanently Defueled Stations and Independent Spent Fuel Storage Installations.
- Simplifying the fission product barrier EAL threshold for a Site Area Emergency.

Using NEI 99-01, Rev. 4, DCPP conducted an EAL implementation upgrade project that produced the EALs and obtained prior NRC approval (reference 3.1.8).

2.2 Fission Product Barriers

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

The primary fission product barriers are:

A. Reactor Fuel Clad (FC): The zircalloy or stainless steel tubes that contain the fuel pellets.

B. Reactor Coolant System (RCS): Includes the RCS primary side and its connections up to and including the pressurizer safety and relief valves, and other connections up to and including the primary isolation valves.

C. Containment (CNMT): he Containment Barrier includes the containment building, its connections up to and including the outermost containment isolation valves. This barrier also includes the main steam, feedwater, and blowdown line extensions outside the containment building up to and including the outermost secondary side isolation valve.

2.3 Emergency Classification Based on Fission Product Barrier Degradation

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

<u>Unusual Event:</u>

Any loss or any potential loss of Containment

<u>Alert:</u>

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

<u>Site Area Emergency:</u>

Loss or potential loss of any two barriers

<u>General Emergency:</u>

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

2.4 EAL Relationship to EOPs and Critical Safety Function Status

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

2.5 Symptom-Based vs. Event-Based Approach

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

2.6 EAL Organization

The DCPP EALs are divided into six broad categories consistent with NEI 99-01:

- EAL Category R – EALs that are radiological in nature
- EAL Category C – EALs related to system malfunctions occurring while in Cold Shutdown or Refueling mode
- EAL Category E – EALs associated with the Independent Spent Fuel Storage Installation (ISFSI)
- EAL Category H – EALs that are related to external hazards such as security events, fires or natural events
- EAL Category S – EALs that are related to system or equipment malfunctions
- EAL Category F – EALs that are related to loss or potential loss of one or more fission product barriers

Some categories are broad and may contain several EALs. Further division of a category into subcategories helps promote association of existing plant conditions to the conditions requiring emergency classification.

The primary tool for determining the emergency classification level is the EAL Wall Chart or Matrix. The user of the EAL Wall Chart or Matrix may (but is not required to) consult the EAL Technical Bases Manual in order to obtain additional information concerning the EALs under classification consideration. Technical Bases Information

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

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

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

<u>EAL Identifier</u>

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

2.6.1 First character (letter): Corresponds to the EAL category as described above (R, C, E, H, S or F)

2.6.2 Second character (letter): The emergency classification (U, A, S or G)

2.6.3 Third character (number): The numerical sequence of the subcategories given in the EAL wall chart or matrix. If the category has only one subcategory, it is given the number one (1). In most cases, the third character also represents the IC number.

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

<u>Classification:</u>

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

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

Exact wording of the EAL as it appears in the EAL wall chart or Matrix

<u>Mode Applicability</u>

One or more of the following plant operating conditions comprise the mode to which each EAL is applicable: 1 - Power Operation, 2 - Startup, 3 - Hot Standby, 4 - Hot Shutdown, 5 - Cold Shutdown, 6 - Refueling, D - Defueled, N/A - Not Applicable or All. (See Section 2.8 for operating mode definitions.)

Basis:

Description of the rationale for the EAL

DCPP Basis Reference(s):

Site-specific source documentation from which the EAL is derived

2.7 Operating Mode Applicability

1 Power Operation

$K_{eff} \geq 0.99$ and rated thermal power > 5%.

2 Startup

$K_{eff} \geq 0.99$ and rated thermal power ≤ 5%.

3 Hot Standby

$K_{eff} < 0.99$ and average reactor coolant temperature (T_{avg}) ≥ 350°F.

4 Hot Shutdown

$K_{eff} < 0.99$ and average reactor coolant temperature 350°F > T_{avg} > 200°F with all reactor vessel head closure bolts fully tensioned.

5 Cold Shutdown

$K_{eff} < 0.99$ and average reactor coolant temperature (T_{avg}) ≤ 200°F with all reactor vessel head closure bolts fully tensioned.

6 Refueling

One or more reactor vessel head closure bolts less than fully tensioned.

D Defueled

Reactor Vessel contains no irradiated fuel (full core off-load during refueling or extended outage).

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

2.8 Classifying Transient Events

For some events, the condition may be corrected before a declaration has been made. For example, an emergency classification is warranted when automatic and manual actions taken within the control room do not result in a required reactor trip. However, it is likely that actions taken outside of the control room will be successful, probably before the Shift Manager/Site Emergency Coordinator/or Emergency Director (SM/SEC/ED) classifies the event. The key consideration in this situation is to determine whether or not further plant damage occurred while the corrective actions were being taken. In some situations, this can be readily determined; in other situations, further analyses (e.g., coolant sampling) may be necessary.

In general, observe the following guidance: Classify the event as indicated and terminate the emergency once assessment shows that there were no consequences from the event and other termination criteria are met. For example, a momentary event, such as an anticipated transients without scram (ATWS) or an earthquake, requires declaration even though the condition may have been resolved by the time the declaration is made.

- An ATWS represents a failure of a front-line safety system (reactor trip system or RTS) designed to protect the health and safety of the public.
- The affect of an earthquake on plant equipment and structures may not be readily apparent until investigations are conducted.

There may be cases in which a plant condition that exceeded an EAL threshold was not recognized at the time of occurrence, but is identified well after the condition has occurred (e.g., as a result of routine log or record review) and the condition no longer exists. In these cases, an emergency should not be declared. Reporting requirements of 10 CFR 50.72 and 10 CFR 50.73 are applicable and the guidance of XI1.ID2, "Regulatory Reporting Requirements and Reporting Process," and NUREG-1022, Rev. 2, Section 3 should be applied.

2.9 Imminent EAL Thresholds

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

3. REFERENCES

3.1.1 NEI 99-01 Revision 4, Methodology for Development of Emergency Action Levels dated January 2003

3.1.2 NUMARC/NESP-007 Rev. 2, Methodology for Development of Emergency Action Levels, "Questions and Answers"

3.1.3 NRC Regulatory Issue Summary (RIS) 2003-18, Use of Nuclear Energy Institute (NEI) 99-01, Methodology for Development of Emergency Action Levels, Revision 4, Dated January 2003, including Supplement 1 (July 13, 2004) and Supplement 2 dated December 15, 2005

3.1.4 NRC Bulletin 2005-02 "Emergency Preparedness and Response Actions for Security-Based Events" dated July 18, 2005

3.1.5 NEI White Paper "Enhancements to Emergency Preparedness Programs for Hostile Action," May 2005 (Revised November 1, 2005)

3.1.6 EP G-1, "Emergency Classification and Emergency Plan Activation"

3.1.7 DCPP Emergency Action Level wall chart or Matrix

3.1.8 NRC letter dated 12/31/07, "Diablo Canyon Power Plant, Unit Nos 1 and 2 – Conversion of Emergency Action Levels based on Nuclear Energy Institute's (NEI) Letter 99-01, "Methodology for Development of Emergency Action Levels" (TAC Nos. MD3924 and MD3925)" and associated Safety Evaluation

4. DEFINITIONS

Adversary

As applied to security EALs, an armed or suspected-to-be-armed intruder whose intent is to commit sabotage, disrupt station operations or otherwise commit a crime on station property.

Affecting Safe Shutdown

Event in progress has adversely affected functions that are necessary to bring the plant to and maintain it in the applicable HOT or COLD SHUTDOWN condition. Plant condition applicability is determined by Technical Specification Limiting Conditions for Operation (LCOs) in effect.

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

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

Alert

Events are in process 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 intentional malicious dedicated efforts of a hostile act. Any releases are expected to be limited to small fractions of the Environmental Protection Agency (EPA) Protective Action Guideline (PAG) exposure levels.

Available

The state or condition of being ready and able to be used (placed into operation) to accomplish the stated (or implied) action or function. As applied to a system, this requires the operability of necessary support systems (electrical power supplies, cooling water, lubrication, etc.).

Bomb

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

Civil Disturbance

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

Close

To position a valve or damper so as to prevent flow of the process fluid. To make an electrical connection to supply power.

Confinement Boundary

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

Confirm / Confirmation

To validate, through visual observation or physical inspection, that an assumed condition is as expected or required, without taking action to alter the "as found" configuration.

Containment Closure

The procedurally defined action taken to secure containment and its associated structures, systems and components as a functional barrier to fission product release under existing plant conditions. Containment Closure is defined by Administrative Procedure AD8.DC54, "Containment Closure."

Contiguous

Being in actual contact; touching along a boundary or at a point

Control

Take action, as necessary, to maintain the value of a specified parameter within applicable limits; to fix or adjust the time, amount, or rate of; to regulate or restrict.

Environment Protection Agency (EPA) Protective Action Guidelines (PAGs)

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

Exceeds

To go or be beyond a stated or implied limit, measure, or degree.

Explosion

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

Extortion

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

Faulted

In a steam generator, the existence of secondary side leakage that results in an uncontrolled decrease in steam generator pressure or the steam generator being completely depressurized.

Failure

A state of inability to perform a normal function.

Fire

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

Fission Product Barriers (FPB)

Multiple physical barriers any one of which, if maintained intact, precludes the release of significant amounts of radioactive fission products to the environment. The FPBs are the Reactor Fuel Clad (FC), Reactor Coolant System (RCS) and Containment (CNMT).

General Emergency

Events are in process or have occurred which involve actual or imminent substantial core degradation or melting with potential for loss of containment integrity or security events that result in an actual loss of physical control of the facility. Releases can be reasonably expected to exceed EPA PAG exposure levels offsite for more than the immediate site area.

Hostage

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

Hostile Action

An act toward DCPP or its personnel that includes the use of violent force to destroy equipment, take hostages, and/or intimidate DCPP to achieve an end. This includes attack by air, land, or water using guns, explosives, projectiles, vehicles, or other devices used to deliver destructive force. Other acts that satisfy the overall intent may be included. Hostile Action should not be construed to include acts of civil disobedience or felonious acts that are not part of a concerted attack on DCPP. Non-terrorism-based EALs should be used to address such activities, (e.g., violent acts between individuals in the owner controlled area).[1]

Hostile Force

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

Inoperable

Not able to perform its intended function.

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1 Definition as provided in NRC Bulletin 2005-02 "Emergency Preparedness and Response Actions for Security-Based Events" and NEI 99-01 draft Revision 5 (July 2007)

Immediately Dangerous to Life and Health (IDLH)

A condition that either poses an immediate threat to life and health or an immediate threat of severe exposure to contaminants which are likely to have adverse delayed effects on health.

Intruder

Person(s) present in a specified area without authorization.

Intrusion

The act of entering without authorization. Discovery of a bomb in a specified area is indication of intrusion into that area by a hostile force.

Lower Flammability Limit (LFL)

The minimum concentration of a combustible substance that is capable of propagating a flame through a homogenous mixture of the combustible and a gaseous oxidizer.

Maintain

Take action, as necessary, to keep the value of the specified parameter within the applicable limits.

Normal Plant Operations

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

Operable

Able to perform its intended function.

Owner Controlled Area

The land area(s) adjacent to the site boundary that are owned and controlled by the licensee, whereby access can be limited by the licensee for any reason. Generally described, the DCPP OCA is the area between the Port San Luis gate and security gate A, bounded by the eastern hills directly adjacent to the site access road and the northern evacuation route, and bounded to the west by the Pacific Ocean. (NOTE: for commitments identified in the Diablo Canyon Physical Security Plan, the "Owner Controlled Area" (OCA) is within the "Site Boundary" as described in UFSAR).

Primary System

The pipes, valves, and other equipment which connect directly to the Reactor Vessel or reactor coolant system such that a reduction in Reactor Vessel pressure will effect a decrease in the steam or water being discharged through an unisolated break in the system.

Protected Area

The Protected Area is within the security isolation zone and is depicted in Dwg. 471124 "Plot Plan."

Reduced Inventory Condition (RIC)

The condition existing whenever RCS water level is lower than 3 feet below the Reactor Vessel flange (below 111 foot elevation) with fuel in the core.

Restore

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

Ruptured

In a steam generator, existence of primary-to-secondary leakage of a magnitude sufficient to require or cause a reactor trip and safety injection.

Sabotage

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

Safe Plant Shutdown

Hot or cold shutdown (reactor subcritical) with control of coolant inventory and decay heat removal.

Security Condition

Any Security Event as listed in the approved security contingency plan that constitutes a threat/compromise to site security, threat/risk to site personnel, or a potential degradation to the level of safety of the plant. A Security Condition does not involve a Hostile Action.

Significant Transient

An unplanned event involving any of the following:

- Automatic turbine runback > 25% thermal reactor power
- Electrical load rejection > 25% full electrical load
- Reactor trip
- Safety injection activation
- Thermal power oscillations > 10%

Site Area Emergency

Events are in process or have occurred which involve an actual or likely major failures of plant functions needed for protection of the public or security events that result in intentional damage or malicious acts; (1) toward site personnel or equipment that could lead to the likely failure of or; (2) prevents effective access to equipment needed for the protection of the public. Any releases are not expected to result in exposure levels which exceed EPA PAG exposure levels beyond the site boundary.

Site Boundary

As depicted in the Final Safety Analysis Report Update (UFSAR), Figure 2.1-2, Site Plan and Gaseous Effluent Release Points.

Strike Action

Work stoppage within the protected area by a body of organized labor workers to enforce compliance with demands made on DCPP. The strike action must threaten to interrupt normal plant operations.

Sustained

Prolonged. Not intermittent or of a transitory nature.

Trip

To deenergize a pump or fan motor; to position a breaker so as to interrupt or prevent the flow of current in the associated circuit; to manually activate a semi-automatic feature.

Unavailable

Not able to perform its intended function.

Uncontrolled

An evolution lacking control, but is not the result of operator action.

Unplanned

A parameter change or an event that is not the result of an intended evolution and requires corrective or mitigative actions.

Unusual Event

Events are in process or have occurred which indicate a potential degradation of the level of safety of the plant or indicate a security threat to facility protection. No releases of radioactive material requiring offsite response or monitoring are expected unless further degradation of safety systems occurs.

Valid

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

Visible Damage

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

Vital Area

Any plant area which contains vital equipment. Any area, normally within the protected area, which contains equipment, systems, components, or material, the failure, destruction, or release of which could directly or indirectly endanger the public health and safety by exposure to radiation.

5. DCPP-TO-NEI 99-01 EAL CROSSREFERENCE

This cross-reference is provided to facilitate association and location of a DCPP EAL within the NEI 99-01 IC/EAL identification scheme.

DCPP	NEI 99-01	
EAL	IC	Example EAL
RU1.1	AU1	1
RU1.2	AU1	2
RU1.3	AU1	3
N/A	AU1	5
RU2.1	AU2	1
RU2.2	AU2	2
RA1.1	AA1	1
RA1.2	AA1	2
RA1.3	AA1	3
N/A	AA1	5
RA2.1	AA2	1
RA2.2	AA2	2
RA2.3	AA3	1
RA2.4	AA3	2
RS1.1	AS1	1
RS1.2	AS1	2
RS1.3	AS1	4

DCPP	NEI 99-01	
EAL	IC	Example EAL
RG1.1	AG1	1
RG1.2	AG1	2
RG1.3	AG1	4
CU1.1	CU3	1
CU1.2	CU7	1
CU2.1	CU2	1
CU2.2	CU2	2
CU3.1	CU4	1
CU3.2	CU4	2
CU4.1	CU6	1, 2
CU5.1	CU1	1, 2
CU6.1	CU8	2
CA1.1	CA3	1
CA2.1	CA1	1, 2
CA3.1	CA4	1, 2, 3
CS2.1	CS1	1
CS2.2	CS1	2
CS2.3	CS2	1
CS2.4	CS2	2

DCPP	NEI 99-01	
EAL	IC	Example EAL
CG2.1	CG1	1
FU1.1	FU1	1
FA1.1	FA1	1
FS1.1	FS1	1
FG1.1	FG1	1
HU1.1	HU1	1
HU1.2	HU1	2
HU1.3	HU1	5
HU1.4	HU1	6
HU1.5	HU1	7
HU2.1	HU2	1
HU2.2	HU1	4
HU3.1	HU3	1
HU3.2	HU3	2
HU4.1	HU4	1, 2, 3
HU6.1	HU5	1
HA1.1	HA1	1
HA1.2	HA1	2
HA1.3	HA1	3

DCPP	NEI 99-01	
EAL	**IC**	**Example EAL**
HA1.4	HA1	4
HA1.5	HA1	5
HA2.1	HA2	1
HA3.1	HA3	1
HA3.2	HA3	2
HA4.1	HA4	1, 2
HA5.1	HA5	1
HA6.1	HA6	1
HS4.1	HS1	1
HS5.1	HS2	1
HS6.1	HS3	1
HG4.1	HG1	1, 2
HG6.1	HG2	1
EU1.1	E-HU1	1 2 3
SU1.1	SU1	1
SU3.1	SU2	1
SU4.1	SU3	1
SU4.2	SU6	1, 2

DCPP	NEI 99-01	
EAL	**IC**	**Example EAL**
SU5.1	SU4	1
SU5.2	SU4	2
SU6.1	SU5	1, 2
SU7.1	SU8	2
SA1.1	SA5	1
SA2.1	SA2	1
SA4.1	SA4	1
SS1.1	SS1	1
SS1.2	SS3	1
SS2.1	SS2	1
SS4.1	SS6	1
SG1.1	SG1	1
SG2.1	SG2	1

Table 1 – Acronyms and Abbreviations

A/ER	Action/Expected Response
A/S	Air Supply
AC	Alternating Current
ACCUM	Accumulator
AFW	Auxiliary Feedwater
ALARA	As Low As Reasonably Achievable
AMP	Ampere
AP	Abnormal Procedure
APRM	Average Power Range Meter
ASW	Auxiliary Saltwater
ATM	Atmosphere
ATO/FC	Air to Open Fail Closed
ATWS	Anticipated Transient Without Scram
AUTO	Automatic
AUX	Auxiliary
AVAIL	Available
AVG	Average
AXS	Auxiliary Steam System
B&W	Babcock and Wilcox
B/D	Blowdown
B/S	Bistable
B/U	Backup
BA	Boric Acid
BAST	Boric Acid Storage Tank
BKR	Breaker

BLDG	Building
BRG	Bearing
BYP	Bypass
C/I	Cut In
C/O	Cut Out
CC	Control Console
CCP	Centrifugal Charging Pump
CCW	Component Cooling Water
CDE	Committed Dose Equivalent
CE	Combustion Engineering
CFCU	Containment Fan Cooler Unit
CFM	Cubic Feet per Minute
CFR	Code of Federal Regulations
CHAN	Channel
CHEM	Chemistry
CHG	Charging
CIRC	Circulator
CKT	Circuit
CL	Cold Leg
CLG	Cooling
CLR	Cooler
CND	Condensate
CO	Control Operator
CONC	Concentrate
CONDT'Y	Conductivity

CONT	Control
CONTMT	Containment
CPM	Counts per minute
CPS	Counts Per Second
CRDM	Control Rod Drive Mechanism
CRIT	Critical
CSD	Cold Shutdown
CSF	Critical Safety Function
CSFST	Critical Safety Function Status Tree
CST	Condensate Storage Tank
CTMT/CNMT	Containment
CVCS	Chemical and Volume Control System
CVI	Containment Ventilation Isolation
CWP	Circulating Water Pump
D/G	Diesel Generator
DC	Direct Current
DEG	Degree
DEH	Digital Electro Hydraulic
DELTA P	Differential Pressure
DELTA	Differential
DEMIN WTR	Demineralized Water
DEMIN	Demineralizer (ized)
DEPT	Department

DET	Detector
DHR	Decay Heat Removal
DISCH	Discharge
DIST	Distribution
DMPR	Damper
DOT	Department of Transportation
DR	Dose Rate
DRN	Drain
DRPI	Digital Rod Position Indication
EAL	Emergency Action Level
ECCS	Emergency Core Cooling System
ECL	Emergency Classification Level
EDR	Equipment Drain Receiver
ELEV	Elevation
EMERG	Emergency
EOF	Emergency Operations Facility
EOP	Emergency Operating Procedure
EPA	Environmental Protection Agency
EPIP	Emergency Plan Implementing Procedure
EPRI	Electric Power Research Institute
EPTSC	Emergency Power Technical Support Center/Unit Selector
EPTSN	Emergency Power Technical Support Nonvital or Vital
EQUIP	Equipment

ERFDS	Emergency Response Facility Data System
ERG	Emergency Response Guideline
ESF	Engineered Safety Feature
ESW	Emergency Service Water
EVAP	Evaporator
EXC	Excore
EXCH	Exchange
EXH	Exhaust
FAA	Federal Aviation Administration
FAI	Fail As Is
FBI	Federal Bureau of Investigation
FC	Fail Closed
FCO	Feature Cutout
FCV	Flow Control Valve
FDR	Feeder
FDWTR	Feedwater
FE	Flow Element
FEMA	Federal Emergency Management Agency
FHB	Fuel Handling Building
FI	Flow Indicator
FLO MTR	Flow Meter
FLTR	Filter
FO	Fail Open

FPB	Fission product barrier
FREQ	Frequency
FSAR	Final Safety Analysis Report
FT	Flow Transmitter
FUNC	Functional
FW HTR	Feedwater Heater
FW	Fire Water
FWP	Feedwater Pump
FWST	Fire Water Storage Tank
GAL	Gallon(s)
GDT	Gas Decay Tank
GE	General Emergency
GEN	Generator
GFFDS	Gross Failed Fuel Detection System
GND	Ground
GOV	Governor
GPD	Gallons per Day
GPH	Gallons per Hour
GPM	Gallons per Minute
GRP	Group
GV	Governor Valve
GW	Gaseous Radwaste
H2	Hydrogen

HCV	Hand Control Valve
HDR	Header
HELB	High Energy Line Break
HEPA	High Efficiency Particulate Airborne (Filter)
HI	High
HORZ	Horizontal
HP	High Pressure
HPSI	High Pressure Safety Injection
HSB	Hot Standby
HSD	Hot Shutdown
HT	Heat
HTG	Heating
HTR	Heater
HV	High Voltage
HVAC	Heating, Ventilation, and Air Conditioning
HX	Heat Exchanger
HYD	Hydraulic
HYDRO	Hydrostatic
HZ	Hertz
IC	Initiating Condition or Inside Containment
IDLH	Immediately Dangerous to Life and Health
IMB	Inside Missile Barrier
In.	Inch

INBD	Inboard
IND	Indication
INFO	Information
INIT	Initial
INJ	Inject(ion)
INSP	Inspection
INST	Instrument
INTLK	Interlock
INTMED	Intermediate
INV	Inverter
ION EX	Ion Exchanger
IPEEE	Individual Plant Examination of External Events (Generic Letter 88-20)
IR	Intermediate Range
ISFSI	Independent Spent Fuel Storage Installation
ISO Θ	Isolated Phase
ISOL	Isolation
ITLK	Interlock

Keff	Effective Neutron Multiplication Factor
KG	Kilogram
KV	Kilovolt
KVA	Kilovolt Ampere

KW	Kilowatt
LC	Level Controller
LCO	Limiting Condition of Operation
LD	Load
LER	Licensee Event Report
LFL	Lower Flammability Limit
LHUT	Liquid Holdup Tank
LIQ	Liquid
LO	Lube Oil
LOA	Low Alarm
LOC	Location
LOCA	Loss of Coolant Accident
LP	Low Pressure
LPSI	Low Pressure Safety Injection
LT	Level Transmitter
LTDN	Letdown
LTG	Lighting
LVDT	Linear Variable Differential Transformer
LVL	Level
LWR	Light Water Reactor
MAINT	Maintenance

MAN	Manual
MATL	Material
MAX	Maximum
MCC	Motor Control Center
MD	Motor Driver
MECH	Mechanical
MEDT	Miscellaneous Equipment Drain Tank
MFWPp	Main Feedwater Pump
MG	Motor Generator
MIN	Minimum
MISC	Miscellaneous
MLOR	Main Lube Oil Reservoir
MN BK	Main Bank
MN	Main
MOD	Motor Operated Disconnect
MOL	Man On Line
MOV	Motor Operated Valve
mR	milliRoentgen
MS	Main Steam
MSIV	Main Steam Isolation Valve
MSL	Main Steam Line
MSR	Moisture Separator/Reheater
MTR	Motor

MU	Makeup
MW	Megawatt
N/A	Not Applicable
N2	Nitrogen
Na	Sodium
NC	Normally Closed
NEG	Negative
NEI	Nuclear Energy Institute
NESP	National Environmental Studies Project
NEUT	Neutral
NIS	Nuclear Instrument System
NO	Nuclear Operator
NORAD	North American Aerospace Defense Command
NORM	Normal
NOUE	Notification Of Unusual Event
NPP	Nuclear Power Plant
NPSH	Net Positive Suction Head
NR	Narrow Range
NRC	Nuclear Regulatory Commission
NSSS	Nuclear Steam Supply System
NUMARC	Nuclear Management and Resources Council

O2	Oxygen
OBE	Operating Basis Earthquake
OC	Outside Containment
OCA	Owner Controlled Area
ODCP	Off-site Dose Calculation Procedure
OL	Overload
OMB	Outside Missile Barrier
OP	Operating Procedure
ORO	Off-site Response Organization
OUTBD	Outboard
OV	Overvoltage
OVBD	Overboard
PA	Protected Area
PBX	Private Business Exchange
PCV	Pressure Control Valve
PDP	Positive Displacement Pump
PEN	Penetration
PERM	Permissive
PI	Pressure Indicator
PK	Main Annunciator Window Group Designator
PKG	Package
PLT	Plant

PNL	Panel
POAH	Point of Adding Heat
PORV	Power Operated Relief Valve (Pressurizer)
POS	Positive
PP	Pump
PPC	Plant Process Computer
PPM	Parts Per Million
PPS	Pumps
PR	Power Range; Pressure Recorder
PRA/PSA	Probabilistic Risk Assessment / Probabilistic Safety Assessment
PRESS	Pressure
PRI WTR	Primary Water
PRI	Primary
PRT	Pressurizer Relief Tank
PS	Pressure Switch
PSAT	Saturation Pressure
PSI	Pounds Per Square Inch
PSIA	Pounds Per Square Inch Absolute
PSID	Pounds Per Square Inch Differential
PSIG	Pounds per Square Inch Gauge
PT	Pressure Transmitter
PWR	Power or Pressurized Water Reactor
PWST	Primary Water Storage Tank

PZR	Pressurizer
R	Roentgen
R/I	Rack(ed) In
R/O	Rack(ed) Out
RAD	Radiation
RC	Reactor Coolant
RCA	Radiological Controlled Area
RCC	Reactor Control Console
RCDT	Reactor Coolant Drain Tank
RCIC	Reactor Core Isolation Cooling
RCO	Relay Cutout
RCP	Reactor Coolant Pump
RCS	Reactor Coolant System
RCVR	Receiver
RECIRC	Recirculation
RECT	Rectifier
REF	Reference
REG	Regulator
REGEN	Regenerative
REL	Relief
REM	Radiation Equivalent Man
RESV	Reservoir

RET	Return
RETS	Radiological Effluent Technical Specifications
REV	Reverse
RHR	Residual Heat Removal
RHTR	Reheater
RIL	Rod Insertion Limit
RLY	Relay
RM	Radiation Monitor
RMS	Radiation Monitoring System
RNGE	Range
RNO	Response Not Obtained
RO	Restricting Orifice
RPI	Rod Position Indication
RPM	Revolution Per Minute
RPS	Reactor Protection system
RPV	Reactor Pressure Vessel
RTB	Reactor Trip Breaker
RTD	Resistance Temperature Detector
RTN	Return
RTP	Rated Thermal Power
RV	Relief Valve
RVLIS	Reactor Vessel Level Instrumentation System
RWST	Refueling Water Storage Tank

RX	Reactor
SAE	Site Area Emergency
SAT	Spray Additive Tank
SBGTS	Stand-By Gas Treatment System
SBO	Station Blackout
SC	Sealed Closed
SCHED	Schedule(d)
SCW	Service Cooling Water
SD	System Dispatcher
S/D	Shutdown
SDM	Shutdown Margin
SE	Shift Engineer
SEC	Secondary
SEL	Selector
SEP	Separate
SEQ	Sequence
SFM	Shift Foreman
SFP	Spent Fuel Pit; Spent Fuel Pool
S/G	Steam Generator
SGBD	Steam Generator Blowdown
SGTR	Steam Generator Tube Rupture
SI	Safety Injection

SIS	Safety Injection Signal/System
SJAE	Steam Jet Air Ejector
SM	Shift Manager
SMP	Sump
SO	Sealed Open
SOL	Solenoid
SPDS	Safety Parameter Display System
SPLY	Supply
SQ	Square
SR	Source Range
SRO	Senior Reactor Operator
SRST	Spent Resin Storage Tank
SS	Steady State
SSE	Safe Shutdown Earthquake
SSPS	Solid State Protection System
STA	Shift Technical Advisor
STBY	Stand By
STD	Standard
STM	Steam
STR	Strainer
SUCT	Suction
SUR	Startup Rate
S/U	Startup

SV	Stop Valve
SW	Switch
SWBD	Switchboard
SWGR	Switchgear
SWYD	Switchyard
SYNC	Synchroscope or Synchronize
SYS	System

TAVG	Average Temperature
TCs	Thermocouples
TCV	Temperature Control Valve
TD	Turbine Driven
TE	Temperature Element
TECH SPEC or TS	Technical Specification
TECH	Technician
TEDE	Total Effective Dose Equivalent
TEMP	Temperature
T/G	Turning Gear
THROT VLV	Throttle Valve
TI	Temperature Indicator
TK	Tank
TM	Technical Maintenance
TOAF	Top of Active Fuel
TREF	Reference Temperature

TRN	Train
TRP	Trap
TSAT	Saturation Temperature
TSC	Technical Support Center
TURB	Turbine
UE	Unusual Event
UF	Under Frequency
UV	Undervoltage
V	Voltage
VAC	Vacuum
VAR	Volt Amperes Reactive
VB	Vertical Board
VCT	Volume Control Tank
VDC	Volts direct current
VENT	Ventilation
VIB	Vibration
VLV	Valve
VM	Voltmeter
VOL	Volume
VOLT REG	Voltage Regulator
VSL	Vessel

WE	Westinghouse Electric
WOG	Westinghouse Owners Group
WR	Wide Range
WT	Weight
WTR	Water
Xe	Xenon
XFER	Transfer
XFMR	Transformer
XMTR	Transmitter
XTIE	Crosstie
Θ	Phase
ºF	Degrees Fahrenheit

1. Category R – Abnormal Rad Levels / Radiological Effluent

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

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

Events of this category pertain to the following subcategories:

1. Offsite Rad Conditions

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

2. Onsite Rad Conditions, Spent Fuel Events

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

Category: R – Abnormal Rad Levels / Radiological Effluent

Sub-category: 1 – Offsite Rad Conditions

Initiating Condition: Any unplanned release of gaseous or liquid radioactivity to the environment that exceeds two times the radiological effluent Radiological Effluent Technical Specifications (RETS) limits for 60 minutes or longer

EAL:

RU1.1	**Unusual Event**
Valid reading on ANY **LIQUID** monitors > Table R-1 column “UE” for ≥ 60 min.	

Table R-1 Effluent Monitor Classification Thresholds

	Release Point	Monitor	GE	SAE	Alert	UE
Gaseous	**Plant Vent**	1(2)-RM-14/14R	-----	-----	Offscale hi	8.0E4 cpm 2.0E-3 μC/cc
		1(2)-RM-24/24R	-----	-----	1.0E-6 μC/cc	1.0E-8 μC/cc
		1(2)-RM-28/28R	-----	-----	1.0E-5 μC/cc	1.0E-7 μC/cc
		1(2)-RM-87	3.0E-9 amps 2.0E+1μC/cc	3.0E-10 amps 2.0E0 μC/cc	-----	-----
	Main Steam	1(2)-RM-71/72/73/74 With Steam Dump or one or more SRVs open on affected SG	3.0E+5 cpm	3.0E+4 cpm	3.0E+3 cpm	3.0E+2 cpm
Liquid	**Oily Water Separator Effluent**	0-RM-3	-----	-----	6.0E+4 cpm *	6.0E+2 cpm *
	Liquid Radwaste Discharge Line Effluent	0-RM-18	-----	-----	Offscale hi *	1.0E+5 cpm *
	SGBD Tank Liquid Effluent	1(2)-RM-23	-----	-----	Offscale hi *	2.0E+4 cpm *

* With effluent discharge not isolated

Mode Applicability:

All

Basis:

Unplanned liquid releases in excess of two times the site technical specifications that continue for 60 minutes or longer represent an uncontrolled situation and hence, a potential degradation in the level of safety. The final integrated dose (which is very low in the Unusual Event emergency class) is not the primary concern here; it is the degradation in plant control implied by the fact that the release was not isolated within 60 minutes. Therefore, it is not intended that the release be averaged over 60 minutes. For example, a release of 4 times Technical Specifications for 30 minutes does not exceed this initiating condition. Further, the SM/SEC/ED should not wait until 60 minutes has elapsed, but should declare the event as soon as it is determined that the release duration has or will likely exceed 60 minutes. Also, if an ongoing release is detected and the starting time for that release is unknown, the SM/SEC/ED should, in the absence of data to the contrary, assume that the release has exceeded 60 minutes.

The liquid release values in Table R-1 are two times the effluent monitor alarm setpoints.

Procedure CAP A-8 describes the methodologies and the basis for establishing monitor high alarm set point (HASPs) to ensure that Technical Specifications limits are not exceeded.

Two times (2x) the current HASP is used as the NUE EAL threshold, which is approximately equal to the EAL initiating condition of exceeding 2x tech specs.

DCPP Basis Reference(s):

1. Plant Manual Volume 9B, "Curves and Misc. Data Sheet; I&C Radiation Monitoring and Allied Systems Data"
2. CAP A-8, "Offsite Dose Calculations"
3. A0643677, AE01, Evaluate RMS setpoints for emergency classifications

Category: R – Abnormal Rad Levels / Radiological Effluent

Sub-category: 1 – Offsite Rad Conditions

Initiating Condition: Any unplanned release of gaseous or liquid radioactivity to the environment that exceeds two times the radiological effluent Radiological Effluent Technical Specifications (RETS) limits for 60 minutes or longer

EAL:

RU1.2 Unusual Event

Valid reading on ANY **GASEOUS** monitors > Table R-1 column "UE" for ≥ 60 min.

Table R-1 Effluent Monitor Classification Thresholds

Release Point		Monitor	GE	SAE	Alert	UE
Gaseous	**Plant Vent**	1(2)-RM-14/14R	-----	-----	Offscale hi	8.0E4 cpm 2.0E-3 μC/cc
		1(2)-RM-24/24R	-----	-----	1.0E-6 μC/cc	1.0E-8 μC/cc
		1(2)-RM-28/28R	-----	-----	1.0E-5 μC/cc	1.0E-7 μC/cc
		1(2)-RM-87	3.0E-9 amps 2.0E+1μC/cc	3.0E-10 amps 2.0E0 μC/cc	-----	-----
	Main Steam	1(2)-RM-71/72/73/74 With Steam Dump or one or more SRVs open on affected SG	3.0E+5 cpm	3.0E+4 cpm	3.0E+3 cpm	3.0E+2 cpm
Liquid	**Oily Water Separator Effluent**	0-RM-3	-----	-----	6.0E+4 cpm *	6.0E+2 cpm *
	Liquid Radwaste Discharge Line Effluent	0-RM-18	-----	-----	Offscale hi *	1.0E+5 cpm *
	SGBD Tank Liquid Effluent	1(2)-RM-23	-----	-----	Offscale hi *	2.0E+4 cpm *

* With effluent discharge not isolated

Mode Applicability:

All

Basis:

Unplanned releases in excess of two times the site technical specifications that continue for 60 minutes or longer represent an uncontrolled situation and hence, a potential degradation in the level of safety. The final integrated dose (which is very low in the Unusual Event emergency class) is not the primary concern here; it is the degradation in plant control implied by the fact that the release was not isolated within 60 minutes. Therefore, it is not intended that the release be averaged over 60 minutes. For example, a release of 4 times Technical Specifications for 30 minutes does not exceed this initiating condition. Further, the SM/SEC/ED should not wait until 60 minutes has elapsed, but should declare the event as soon as it is determined that the release duration has or will likely exceed 60 minutes. Also, if an ongoing release is detected and the starting time for that release is unknown, the SM/SEC/ED should, in the absence of data to the contrary, assume that the release has exceeded 60 minutes.

Procedure CAP A-8 describes the methodologies and the basis for establishing monitor high alarm set point (HASPs) to ensure that Technical Specifications limits are not exceeded.

Two times (2x) the current HASP is used as the NUE EAL threshold, which is approximately equal to the EAL initiating condition of exceeding 2x tech specs.

The liquid release values in Table R-1 are two times the effluent monitor alarm setpoints.

R-87 is not on scale at this level.

DCPP Basis Reference(s):

1. Plant Manual Volume 9B, “Curves and Misc. Data Sheet; I&C Radiation Monitoring and Allied Systems Data”
2. CAP A-8, “Offsite Dose Calculations”
3. A0643677, AE01, Evaluate RMS setpoints for emergency classifications

Category: R – Abnormal Rad Levels / Radiological Effluent

Sub-category: 1 – Offsite Rad Conditions

Initiating Condition: Any unplanned release of gaseous or liquid radioactivity to the environment that exceeds two times the Radiological Effluent Technical Specifications (RETS) limits for 60 minutes or longer

EAL:

RU1.3 Unusual Event

Confirmed sample analyses for gaseous or liquid releases indicate concentrations or release rates > 2 x RETS limits for ≥ 60 min.

Mode Applicability:

All

Basis:

Unplanned releases in excess of two times the site Radiological Effluent Technical Specifications (RETS) limits that continue for 60 minutes or longer represent an uncontrolled situation and hence, a potential degradation in the level of safety. The final integrated dose (which is very low in the Unusual Event emergency class) is not the primary concern here; it is the degradation in plant control implied by the fact that the release was not isolated within 60 minutes. Therefore, it is not intended that the release be averaged over 60 minutes. For example, a release of 4 times RETS for 30 minutes does not exceed this initiating condition. Further, the SM/SEC/ED should not wait until 60 minutes has elapsed, but should declare the event as soon as it is determined that the release duration has or will likely exceed 60 minutes. Also, if an ongoing release is detected and the starting time for that release is unknown, the SM/SEC/ED should, in the absence of data to the contrary, assume that the release has exceeded 60 minutes.

DCPP Basis Reference(s):

1. CAP A-8, "Offsite Dose Calculations"

Category: R – Abnormal Rad Levels / Radiological Effluent

Sub-category: 1 – Offsite Rad Conditions

Initiating Condition: Any unplanned release of gaseous or liquid radioactivity to the environment that exceeds 200 times the Radiological Effluent Technical Specifications (RETS) limits for 15 minutes or longer

EAL:

RA1.1 Alert

Valid reading on ANY **LIQUID** monitors > Table R-1 column "Alert" for ≥ 15 min.

Table R-1 Effluent Monitor Classification Thresholds

Release Point		Monitor	GE	SAE	Alert	UE
Gaseous	**Plant Vent**	1(2)-RM-14/14R	-----	-----	Offscale hi	8.0E4 cpm 2.0E-3 μC/cc
		1(2)-RM-24/24R	-----	-----	1.0E-6 μC/cc	1.0E-8 μC/cc
		1(2)-RM-28/28R	-----	-----	1.0E-5 μC/cc	1.0E-7 μC/cc
		1(2)-RM-87	3.0E-9 amps 2.0E+1μC/cc	3.0E-10 amps 2.0E0 μC/cc	-----	-----
	Main Steam	1(2)-RM-71/72/73/74 With Steam Dump or one or more SRVs open on affected SG	3.0E+5 cpm	3.0E+4 cpm	3.0E+3 cpm	3.0E+2 cpm
Liquid	**Oily Water Separator Effluent**	0-RM-3	-----	-----	6.0E+4 cpm *	6.0E+2 cpm *
	Liquid Radwaste Discharge Line Effluent	0-RM-18	-----	-----	Offscale hi *	1.0E+5 cpm *
	SGBD Tank Liquid Effluent	1(2)-RM-23	-----	-----	Offscale hi *	2.0E+4 cpm *

* With effluent discharge <u>not</u> isolated

Mode Applicability:

All

Basis:

This EAL addresses a potential or actual decrease in the level of safety of the plant as indicated by a radiological release that exceeds, by a factor of 200, regulatory commitments for an extended period of time. DCPP incorporates features intended to control the release of radioactive effluents to the environment. Additionally, there are administrative controls established to prevent unintentional releases, or control and monitor intentional releases. These controls are located in the Radiological Effluent Technical Specifications (RETS). The occurrence of extended, uncontrolled radioactive releases to the environment is indicative of a degradation in these features and/or controls. Releases should not be prorated or averaged.

The SM/SEC/ED should not wait until 15 minutes has elapsed, but should declare the event as soon as it is determined that the release duration has or will likely exceed 15 minutes. Also, if an ongoing release is detected and the starting time for that release is unknown, the SM/SEC/ED should, in the absence of data to the contrary, assume that the release has exceeded 15 minutes.

This event escalates from the Unusual Event by escalating the magnitude of the release by a factor of 100.

The gaseous release values in Table R-1 are 200 times the monitor alarm setpoints. The setpoints are established to ensure the RETS release limits are not exceeded.

Procedure CAP A-8 describes the methodologies and the basis for establishing monitor high alarm set point (HASPs) to ensure that Technical Specifications limits are not exceeded.

If onscale, 200x the HASP is used for the Alert EAL threshold; otherwise an offscale high condition is an indication of an Alert.

DCPP Basis Reference(s):

1. Plant Manual Volume 9B, "Curves and Misc. Data Sheet; I&C Radiation Monitoring and Allied Systems Data"
2. CAP A-8, "Offsite Dose Calculations"
3. A0643677, AE01, Evaluate RMS setpoints for emergency classifications

Category: R – Abnormal Rad Levels / Radiological Effluent

Sub-category: 1 – Offsite Rad Conditions

Initiating Condition: Any unplanned release of gaseous or liquid radioactivity to the environment that exceeds 200 times the Radiological Effluent Technical Specifications (RETS) limits for 15 minutes or longer

EAL:

RA1.2 Alert

Valid reading on ANY GASEOUS monitors > Table R-1 column "Alert" for ≥ 15 min.

Table R-1 Effluent Monitor Classification Thresholds

	Release Point	Monitor	GE	SAE	Alert	UE
Gaseous	**Plant Vent**	1(2)-RM-14/14R	-----	-----	Offscale hi	8.0E4 cpm
						2.0E-3 µC/cc
		1(2)-RM-24/24R	-----	-----	1.0E-6 µC/cc	1.0E-8 µC/cc
		1(2)-RM-28/28R	-----	-----	1.0E-5 µC/cc	1.0E-7 µC/cc
		1(2)-RM-87	3.0E-9 amps	3.0E-10 amps	-----	-----
			2.0E+1µC/cc	2.0E0 µC/cc		
	Main Steam	1(2)-RM-71/72/73/74 With Steam Dump or one or more SRVs open on affected SG	3.0E+5 cpm	3.0E+4 cpm	3.0E+3 cpm	3.0E+2 cpm
Liquid	**Oily Water Separator Effluent**	0-RM-3	-----	-----	6.0E+4 cpm *	6.0E+2 cpm *
	Liquid Radwaste Discharge Line Effluent	0-RM-18	-----	-----	Offscale hi *	1.0E+5 cpm *
	SGBD Tank Liquid Effluent	1(2)-RM-23	-----	-----	Offscale hi *	2.0E+4 cpm *

* With effluent discharge not isolated

Mode Applicability:

All

Basis:

This EAL addresses a potential or actual decrease in the level of safety of the plant as indicated by a radiological release that exceeds, by a factor of 200, regulatory commitments for an extended period of time. DCPP incorporates features intended to control the release of radioactive effluents to the environment. Additionally, there are administrative controls established to prevent unintentional releases, or control and monitor intentional releases. These controls are located in the Radiological Effluent Technical Specifications (RETS). The occurrence of extended, uncontrolled radioactive releases to the environment is indicative of a degradation in these features and/or controls. Releases should not be prorated or averaged.

The SM/SEC/ED should not wait until 15 minutes has elapsed, but should declare the event as soon as it is determined that the release duration has or will likely exceed 15 minutes. Also, if an ongoing release is detected and the starting time for that release is unknown, the SM/SEC/ED should, in the absence of data to the contrary, assume that the release has exceeded 15 minutes.

This event escalates from the Unusual Event by escalating the magnitude of the release by a factor of 100.

The gaseous release values in Table R-1 are 200 times the monitor alarm setpoints. The setpoints are established to ensure the RETS release limits are not exceeded.

Procedure CAP A-8 describes the methodologies and the basis for establishing monitor high alarm set point (HASPs) to ensure that Technical Specifications limits are not exceeded.

If onscale, 200x the HASP is used for the Alert EAL threshold; otherwise an offscale high condition is an indication of an Alert.

DCPP Basis Reference(s):

1. Plant Manual Volume 9B, "Curves and Misc. Data Sheet; I&C Radiation Monitoring and Allied Systems Data"
2. CAP A-8, "Offsite Dose Calculations"
3. A0643677, AE01, Evaluate RMS setpoints for emergency classifications

Category: R – Abnormal Rad Levels / Radiological Effluent

Sub-category: 1 – Offsite Rad Conditions

Initiating Condition: Any unplanned release of gaseous or liquid radioactivity to the environment that exceeds 200 times the Radiological Effluent Technical Specifications (RETS) limits for 15 minutes or longer

EAL:

RA1.3 **Alert**

Confirmed sample analyses for gaseous or liquid releases indicate concentrations or release rates > 200 x RETS limits for ≥ 15 *min.*

Mode Applicability:

All

Basis:

Confirmed sample analyses in excess of two hundred times the site Radiological Effluent Technical Specifications (RETS) limits that continue for 15 minutes or longer represent an uncontrolled situation and hence, a potential degradation in the level of safety. This event escalates from the Unusual Event by raising the magnitude of the release by a factor of 100 over the Unusual Event level (i.e., 200 times RETS). The required release duration was reduced to 15 minutes in recognition of the raised severity. Releases should not be prorated or averaged.

The SM/SEC/ED should not wait until 15 minutes has elapsed, but should declare the event as soon as it is determined that the release duration has or will likely exceed 15 minutes. Also, if an ongoing release is detected and the starting time for that release is unknown, the SM/SEC/ED should, in the absence of data to the contrary, assume that the release has exceeded 15 minutes.

DCPP Basis Reference(s):

1. CAP A-8, "Offsite Dose Calculations"

Category: R – Abnormal Rad Levels / Radiological Effluent

Sub-category: 1 – Offsite Rad Conditions

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

EAL:

RS1.1 Site Area Emergency

Valid reading on ANY radiation monitors that exceeds or is expected to exceed Table R-1 column "SAE" for ≥ 15 min. (Note 1)

Table R-1 Effluent Monitor Classification Thresholds

Release Point		Monitor	GE	SAE	Alert	UE
Gaseous	**Plant Vent**	1(2)-RM-14/14R	-----	-----	Offscale hi	8.0E4 cpm 2.0E-3 μ/cc
Gaseous	**Plant Vent**	1(2)-RM-24/24R	-----	-----	1.0E-6 μC/cc	1.0E-8 μC/cc
Gaseous	**Plant Vent**	1(2)-RM-28/28R	-----	-----	1.0E-5 μC/cc	1.0E-7 μC/cc
Gaseous	**Plant Vent**	1(2)-RM-87	3.0E-9 amps 2.0E+1μC/cc	3.0E-10 amps 2.0E0 μC/cc	-----	-----
Gaseous	**Main Steam**	1(2)-RM-71/72/73/74 With Steam Dump or one or more SRVs open on affected SG	3.0E+5 cpm	3.0E+4 cpm	3.0E+3 cpm	3.0E+2 cpm
Liquid	**Oily Water Separator Effluent**	0-RM-3	-----	-----	6.0E+4 cpm *	6.0E+2 cpm *
Liquid	**Liquid Radwaste Discharge Line Effluent**	0-RM-18	-----	-----	Offscale hi *	1.0E+5 cpm *
Liquid	**SGBD Tank Liquid Effluent**	1(2)-RM-23	-----	-----	Offscale hi *	2.0E+4 cpm *

* With effluent discharge not isolated

Note 1: If dose assessment results are available at the time of declaration, the classification should be based on dose assessment instead of radiation monitor readings. While necessary declarations should not be delayed awaiting results, the dose assessment should be initiated / completed in order to determine if the classification should be subsequently escalated. See EAL RS1.2.

Mode Applicability:

All

Basis:

This EAL addresses radioactivity releases that can result in doses at or beyond the Site Boundary that exceed a fraction (10%) of the EPA Protective Action Guides (PAGs). Releases of this magnitude are associated with the failure of plant systems needed for the protection of the public. While these failures are addressed by other EALs, this EAL provides appropriate diversity and addresses events which may not be able to be classified on the basis of plant status alone (e.g., fuel handling accident in spent fuel building).

The SM/SEC/ED should not wait until 15 minutes has elapsed, but should declare the event as soon as it is determined that the release duration has or will likely exceed 15 minutes.

The EPA PAGs are expressed in terms of the sum of the effective dose equivalent (EDE) and the committed effective dose equivalent (CEDE), or as the thyroid committed dose equivalent (CDE). For the purpose of this EAL, the dose quantity total effective dose equivalent (TEDE), as defined in 10 CFR 20, is used in lieu of "...sum of EDE and CEDE...." The EPA PAG guidance provides for the use adult thyroid dose conversion factors.

Since dose assessment is based on actual meteorology whereas the monitor reading EALs are not, the results from these assessments may indicate that the classification is not warranted, or may indicate that a higher classification is warranted. For this reason, emergency implementing procedures call for the timely performance of dose assessments using actual meteorology and release information. If the results of these dose assessments are available when the classification is made (e.g., initiated at a lower classification level), the dose assessment results override the monitor readings listed in Table R-1.

DCPP Basis Reference(s):

1. EP RB-9, "Calculation of Release Rate"
2. EP RB-11, "Emergency Offsite Dose Calculations"
3. EP R-2, "Release of Airborne Radioactive Materials Initial Assessment"
4. A0643677, AE01, Evaluate RMS setpoints for emergency classifications

Category: R – Abnormal Rad Levels / Radiological Effluent

Sub-category: 1 – Offsite Rad Conditions

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

EAL:

RS1.2 Site Area Emergency

Dose assessment using actual meteorology indicates doses > 100 mRem TEDE or 500 mRem thyroid CDE at or beyond the site boundary

Mode Applicability:

All

Basis:

This EAL addresses radioactivity releases that can result in doses at or beyond the Site Boundary that exceed a fraction (10%) of the EPA PAGs. Releases of this magnitude are associated with the failure of plant systems needed for the protection of the public. While these failures are addressed by other EALs, this EAL provides appropriate diversity and addresses events which may not be able to be classified on the basis of plant status alone (e.g., fuel handling accident).

The specified values also provide a desirable gradient (one order of magnitude) between the Alert, Site Area Emergency, and General Emergency classes. It is deemed that exposures less than this limit are not consistent with the Site Area Emergency class description. The 500 mRem integrated CDE thyroid dose was established in consideration of the 1:5 ratio of the EPA Protective Action Guidelines for TEDE and thyroid exposure. Actual meteorology is specifically identified since it gives the most accurate dose assessment. Actual meteorology (including forecasts) should be used whenever possible.

DCPP Basis Reference(s):

1. EP RB-9, "Calculation of Release Rate"
2. EP RB-11, "Emergency Offsite Dose Calculations"
3. EP R-2, "Release of Airborne Radioactive Materials Initial Assessment"
4. UFSAR, Figure 2.1-2, "Site Plan and Gaseous/Liquid Effluent Release Points"

Category: R – Abnormal Rad Levels / Radiological Effluent

Sub-category: 1 – Offsite Rad Conditions

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

EAL:

RS1.3 Site Area Emergency

Field survey indicates closed window dose rate > 100 mRem/hr that is expected to continue for > 1 hr, at or beyond the site boundary

OR

Field survey sample analysis indicates thyroid CDE of ≥ 500 mRem for 1 hr of inhalation at or beyond the site boundary

Mode Applicability:

All

Basis:

The 100 mRem integrated TEDE dose in this EAL is based on the 10 CFR 20 annual average population exposure. This value also provides a desirable gradient (one order of magnitude) between the Alert, Site Area Emergency, and General Emergency classes. It is deemed that exposures less than this limit are not consistent with the Site Area Emergency class description. The 500 mRem integrated CDE thyroid dose was established in consideration of the 1:5 ratio of the EPA Protective Action Guidelines for TEDE and thyroid exposure. In establishing the dose rate emergency action levels, a duration of one hour is assumed. Therefore, the dose rate EALs are based on a Site Boundary dose rate of 100 mRem/hr TEDE or 500 mRem/hr CDE thyroid, whichever is more limiting.

DCPP Basis Reference(s):

1. UFSAR, Figure 2.1-2, "Site Plan and Gaseous/Liquid Effluent Release Points"
2. EP RB-9, "Calculation of Release Rate"
3. EP RB-11, "Emergency Offsite Dose Calculations"
4. EP R-2, "Release of Airborne Radioactive Materials Initial Assessment"

Category: R – Abnormal Rad Levels / Radiological Effluent

Sub-category: 1 – Offsite Rad Conditions

Initiating Condition: Offsite dose resulting from an actual or imminent release of gaseous radioactivity exceeds 1000 mRem TEDE or 5000 mRem thyroid CDE for the actual or projected duration of the release using actual meteorology

EAL:

RG1.1 General Emergency

Valid reading on ANY monitors that exceeds or is expected to exceed Table R-1 column "GE" for ≥ 15 min. (Note 1)

<table>
<tr><th colspan="7">Table R-1 Effluent Monitor Classification Thresholds</th></tr>
<tr><th colspan="2">Release Point</th><th>Monitor</th><th>GE</th><th>SAE</th><th>Alert</th><th>UE</th></tr>
<tr><td rowspan="7">Gaseous</td><td rowspan="6">Plant Vent</td><td rowspan="2">1(2)-RM-14/14R</td><td rowspan="2">-----</td><td rowspan="2">-----</td><td rowspan="2">Offscale hi</td><td>8.0E4 cpm</td></tr>
<tr><td>2.0E-3 μC/cc</td></tr>
<tr><td>1(2)-RM-24/24R</td><td>-----</td><td>-----</td><td>1.0E-6 μC/cc</td><td>1.0E-8 μC/cc</td></tr>
<tr><td>1(2)-RM-28/28R</td><td>-----</td><td>-----</td><td>1.0E-5 μC/cc</td><td>1.0E-7 μC/cc</td></tr>
<tr><td rowspan="2">1(2)-RM-87</td><td>3.0E-9 amps</td><td>3.0E-10 amps</td><td rowspan="2">-----</td><td rowspan="2">-----</td></tr>
<tr><td>2.0E+1μC/cc</td><td>2.0E0 μC/cc</td></tr>
<tr><td>Main Steam</td><td>1(2)-RM-71/72/73/74 With Steam Dump or one or more SRVs open on affected SG</td><td>3.0E+5 cpm</td><td>3.0E+4 cpm</td><td>3.0E+3 cpm</td><td>3.0E+2 cpm</td></tr>
<tr><td rowspan="3">Liquid</td><td>Oily Water Separator Effluent</td><td>0-RM-3</td><td>-----</td><td>-----</td><td>6.0E+4 cpm *</td><td>6.0E+2 cpm *</td></tr>
<tr><td>Liquid Radwaste Discharge Line Effluent</td><td>0-RM-18</td><td>-----</td><td>-----</td><td>Offscale hi *</td><td>1.0E+5 cpm *</td></tr>
<tr><td>SGBD Tank Liquid Effluent</td><td>1(2)-RM-23</td><td>-----</td><td>-----</td><td>Offscale hi *</td><td>2.0E+4 cpm *</td></tr>
</table>

* With effluent discharge not isolated

Note 1: If dose assessment results are available at the time of declaration, the classification should be based on dose assessment instead of radiation monitor readings. While necessary declarations should not be delayed awaiting results, the dose assessment should be initiated / completed in order to determine if the classification is warranted. See EAL RG1.2.

Mode Applicability:

All

Basis:

This EAL addresses radioactivity releases that can result in doses at or beyond the Site Boundary that exceed the EPA Protective Action Guides (PAGs). Public protective actions will be necessary. Releases of this magnitude are associated with the failure of plant systems needed for the protection of the public and likely involve fuel damage. While these failures are addressed by other EALs, this EAL provides appropriate diversity and addresses events which may not be able to be classified on the basis of plant status alone. It is important to note that, for the more severe accidents, the release may be unmonitored or there may be large uncertainties associated with the source term and/or meteorology.

The SM/SEC/ED should not wait until 15 minutes has elapsed, but should declare the event as soon as it is determined that the release duration has or will likely exceed 15 minutes.

The Table RG-1 effluent monitor readings are one decade greater than the "SAE" values.

The EPA PAGs are expressed in terms of the sum of the effective dose equivalent (EDE) and the committed effective dose equivalent (CEDE), or as the thyroid committed dose equivalent (CDE). For the purpose of this EAL, the dose quantity total effective dose equivalent (TEDE), as defined in 10 CFR 20, is used in lieu of "...sum of EDE and CEDE...." The EPA PAG guidance provides for the use adult thyroid dose conversion factors.

Since dose assessment is based on actual meteorology, whereas the monitor reading EALs are not, the results from these assessments may indicate that the classification is not warranted, or may indicate that a higher classification is warranted. For this reason, emergency implementing procedures call for the timely performance of dose assessments using actual meteorology and release information. If the results of these dose assessments are available when the classification is made (e.g., initiated at a lower classification level), the dose assessment results override the monitor readings listed in Table R-1.

DCPP Basis Reference(s):

1. EP RB-9, "Calculation of Release Rate"
2. EP RB-11, "Emergency Offsite Dose Calculations"
3. EP R-2, "Release of Airborne Radioactive Materials Initial Assessment"
4. A0643677, AE01, Evaluate RMS setpoints for emergency classifications

Category: R – Abnormal Rad Levels / Radiological Effluent

Sub-category: 1 – Offsite Rad Conditions

Initiating Condition: Offsite dose resulting from an actual or imminent release of gaseous radioactivity exceeds 1000 mRem TEDE or 5000 mRem thyroid CDE for the actual or projected duration of the release using actual meteorology

EAL:

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

Mode Applicability:

All

Basis:

The General Emergency values are based on the boundary dose resulting from an actual or imminent release of gaseous radioactivity that exceeds 1000 mRem TEDE or 5000 mRem CDE thyroid for the actual or projected duration of the release. The 1000 mRem TEDE and the 5000 mRem CDE thyroid integrated dose are based on the EPA protective action guidance which indicates that public protective actions are indicated if the dose exceeds 1 Rem TEDE or 5 Rem CDE thyroid. This is consistent with the emergency class description for a General Emergency. This level constitutes the upper level of the desirable gradient for the Site Area Emergency. Actual meteorology is specifically identified since it gives the most accurate dose assessment. Actual meteorology (including forecasts) should be used whenever possible.

DCPP Basis Reference(s):

1. EP RB-9, "Calculation of Release Rate"
2. EP RB-11, "Emergency Offsite Dose Calculations"
3. EP R-2, "Release of Airborne Radioactive Materials Initial Assessment"
4. UFSAR, Figure 2.1-2, "Site Plan and Gaseous/Liquid Effluent Release Points"

Category: R – Abnormal Rad Levels / Radiological Effluent

Sub-category: 1 – Offsite Rad Conditions

Initiating Condition: Offsite dose resulting from an actual or imminent release of gaseous radioactivity exceeds 1000 mRem TEDE or 5000 mRem thyroid CDE for the actual or projected duration of the release using actual meteorology

EAL:

RG1.3 General Emergency

Field survey results indicate closed window dose rates > 1,000 mRem/hr expected to continue for > 1 hr, at or beyond the site boundary

OR

Analyses of field survey samples indicate thyroid CDE of 5,000 mRem for 1 hr of inhalation, at or beyond the site boundary

Mode Applicability:

All

Basis:

The 1,000 mRem integrated TEDE dose in this EAL is based on the 10 CFR 20 annual average population exposure. This value also provides a desirable gradient (one order of magnitude) between the Alert, Site Area Emergency, and General Emergency classes. It is deemed that exposures less than this limit are not consistent with the General Emergency class description. The 5,000 mRem integrated CDE thyroid dose was established in consideration of the 1:5 ratio of the EPA Protective Action Guidelines for TEDE and thyroid exposure. In establishing the dose rate emergency action levels, a duration of one hour is assumed.

DCPP Basis Reference(s):

1. UFSAR, Figure 2.1-2, “Site Plan and Gaseous/Liquid Effluent Release Points”
2. EP RB-9, “Calculation of Release Rate”
3. EP RB-11, “Emergency Offsite Dose Calculations”
4. EP R-2, “Release of Airborne Radioactive Materials Initial Assessment”

Category: R – Abnormal Rad Levels / Radiological Effluent

Sub-category: 2 – Onsite Rad Conditions, Spent Fuel Events

Initiating Condition: Unexpected increase in plant radiation

EAL:

RU2.1 Unusual Event

Valid low water level alarm indicating uncontrolled water level decrease in the reactor refueling cavity, spent fuel pool, or fuel transfer canal with all irradiated fuel assemblies remaining covered by water

AND

Unplanned valid direct area radiation monitor reading increases

Mode Applicability:

All

Basis:

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

The Spent Fuel Pool (SFP) low water level alarm setpoint is 23 ft. 9 in. above the top of irradiated fuel seated in the SFP storage racks or 137 feet 4 inches elevation SFP water level at 136 feet 7 inches elevation is the Technical Specification LCO limit (SR 3.7.15) that requires 23 ft. of water above irradiated fuel seated in the Spent Fuel Pool storage racks. The Refueling Cavity low water level alarm setpoint is at 138 feet elevation as measured on refueling water level indication (RWLI) (i.e., 24 feet above the top of Reactor Vessel flange). A minimum depth of 23 feet of water over the irradiated fuel assemblies in the SFP and 23 feet of water over the Reactor Vessel flange in the refueling cavity is maintained to ensure sufficient iodine activity would be retained to limit offsite doses from the accident to < 25% of 10 CFR 100 limits and to ensure that the offsite dose consequences due to a postulated fuel handling accident are acceptable.

Loss of Spent Fuel Pool water inventory results from either a rupture of the pool or transfer canal liner, or failure of the spent fuel cooling system and the subsequent boil-off. Allowing SFP water level to decrease could result in spent fuel being uncovered, reducing spent fuel decay heat removal and creating an extremely hazardous radiation environment. The movement of irradiated fuel assemblies within containment requires a minimum water level of 23 feet above the top of the Reactor Vessel flange. During Refueling mode, this maintains sufficient water level in the containment, fuel transfer canal, refueling cavity, and spent fuel pool. Therefore, this EAL is applicable for conditions in which irradiated fuel is being transferred to and from the Reactor Vessel and the SFP.

While a radiation monitor (RE-58 or RE-59) could detect an increase in dose due to a drop in the water level, it might not be a reliable indication, in and of itself, of whether or not the fuel is uncovered. For example, the reading on an area radiation monitor (permanently installed or temporary) located near the Reactor Cavity may increase due to planned evolutions such as head lift, or even a fuel assembly being raised in the manipulator mast. Generally, elevated radiation monitor indications will need to be combined with another indicator (or personnel report) of water loss.

This event escalates to an Alert under EAL RA2.1 if irradiated fuel outside the Reactor Vessel is uncovered.

DCPP Basis Reference(s):

1. Technical Specification 3.7.15, "SFP Level"
2. AR PK-11-04 input 2.4.1, Spent Fuel Pool Lvl/Temp
3. Technical Specification 3.9.7, "Refueling Cavity Water Level"
4. AR PK-02-22 input 2.4.3, Rx Vsl Refueling Lvl (red)
5. OP-AP-22, "Spent Fuel Pool Low Level/High Rad"
6. AR PK-11-10, "FHB High Radiation RE-58 and 59"

Category: R – Radioactivity Release / Area Radiation

Sub-category: 2 – Onsite Rad Conditions, Spent Fuel Events

Initiating Condition: Unexpected increase in plant radiation

EAL:

RU2.2 Unusual Event

Unplanned valid direct radiation area monitor reading increases by a factor of 1000 over normal* levels

* Normal levels may be considered as the highest reading in the past twenty-four (24) hours excluding the current peak value

Mode Applicability:

All

Basis:

This EAL addresses unplanned increases in radiation levels inside the plant. These radiation levels represent degradation in the control of radioactive material and a potential degradation in the level of safety of the plant. This EAL escalates to an Alert under EAL RA2.3 or RA2.4 if the elevated radiation levels impair the level of safe plant operation.

"Direct radiation area monitor" - means a monitor located within a radiological controlled area (RCA) that detects and records general area dose rates or airborne radioactivity from direct exposure to a radiation source in an area that may be occupied by personnel.

DCPP Basis Reference(s):

1. RCP D-500, "Routine and Job Coverage Surveys"
2. RCP D-240, "Radiological Posting"

Category: R – Abnormal Rad Levels / Radiological Effluent

Sub-category: 2 – Onsite Rad Conditions, Spent Fuel Events

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

EAL:

RA2.1 **Alert**

Damage to irradiated fuel or loss of water level that has or will result in the uncovering of irradiated fuel outside the reactor vessel resulting in a valid high alarm on ANY of the following radiation monitors:

- New Fuel Storage Area Rad Mon RM-59, Hi Rad (6.5 mR/hr)
- Spent Fuel Pool Area Rad Mon RM-58, Hi Rad (20 mR/hr)
- Contmt Area Mon High Rad RM-2, Hi Rad (21 mR/hr)
- Containment Ventilation Exhaust Radiation Monitor RM-44A/B high alarm (1.35E-4 μC/cc)

Mode Applicability:

All

Basis:

This EAL addresses specific events that have resulted, or may result, in unexpected increases in radiation dose rates within plant buildings and may be a precursor to a radioactivity release to the environment. These events represent a loss of control over radioactive material and represent degradation in the level of safety of the plant. These events escalate from RU2.1 in that fuel activity has been released or is anticipated due to fuel heatup. This EAL applies to spent fuel requiring water coverage and is not intended to address spent fuel which is licensed for dry storage (EAL Category E – ISFSI).

Escalation to a Site Area Emergency or General Emergency, if appropriate, would occur via Subcategory R.1 – Onsite Rad Conditions, Spent Fuel Events, or Subcategory H.6 –Judgment.

When considering escalation, information may come from:

- Radiation monitor readings
- Sampling and surveys
- Dose projections/calculations
- Reports from the scene regarding the extent of damage (e.g., refueling crew, radiation protection technicians)

This EAL is defined by the specific areas where irradiated fuel is located, such as the Refueling Cavity, Reactor Vessel or Spent Fuel Pool (SFP).

The bases for the SFP area radiation high alarms and containment area and ventilation radiation high alarms are a spent fuel handling accident and are, therefore, appropriate for this EAL. In the fuel handling building, a fuel assembly could be dropped in the fuel transfer canal or in the SFP. Should a fuel assembly be dropped in the fuel transfer canal or in the spent fuel pool and release radioactivity above a prescribed level, the area radiation monitors sound an alarm, alerting personnel to the problem. Area radiation monitors in the fuel handling building isolate the normal fuel handling building ventilation system and automatically initiate the recirculation and filtration systems. Elevated background at the monitor due to decreasing water level may mask elevated ventilation exhaust airborne activity and needs to be considered. However, while radiation monitors may detect an increase in dose rate due to a drop in the water level, it might not be a reliable indication of whether or not the fuel is covered. For example, the monitor could in fact be properly responding to a known event involving transfer or relocation of a source stored in or near the Spent Fuel Pool or responding to a planned evolution such as removal of the Reactor Vessel head. Interpretation of these EAL thresholds requires some understanding of the actual radiological conditions present in the vicinity of the monitors.

DCPP Basis Reference(s):

1. OP AP-21, “Irradiated Fuel Damage”
2. OP AP-22, “Spent Fuel Pool Low Level/High Temp/High Rad”
3. Plant Manual Volume 9B, “Curves and Misc. Data Sheet; I&C Radiation Monitoring and Allied Systems Data”

Category: R – Abnormal Rad Levels / Radiological Effluent

Sub-category: 2 – Onsite Rad Conditions, Spent Fuel Events

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

EAL:

RA2.2	**Alert**
A water level drop in the reactor refueling cavity, spent fuel pool or fuel transfer canal that will result in irradiated fuel becoming uncovered	

Mode Applicability:

All

Basis:

This EAL addresses specific events that have resulted, or may result, in unexpected increases in radiation dose rates within plant buildings and may be a precursor to a radioactivity release to the environment. These events represent a loss of control over radioactive material and degradation in the level of safety of the plant. These events escalate from EAL RU2.1 in that fuel activity has been released or is anticipated due to fuel heatup. This EAL applies to spent fuel requiring water coverage and is not intended to address spent fuel which is licensed for dry storage (EAL Category E – ISFSI).

Escalation to a Site Area Emergency or General Emergency, if appropriate, would occur via Subcategory R.1 – Onsite Rad Conditions, Spent Fuel Events, or Subcategory H.6 –Judgment.

When considering escalation, information may come from:

- Radiation monitor readings
- Sampling and surveys
- Dose projections/calculations
- Reports from the scene regarding the extent of damage (e.g., refueling crew, radiation protection technicians)

This EAL is defined by the specific areas where irradiated fuel is located such as the Refueling Cavity, Reactor Vessel or Spent Fuel Pool.

There is no indirect level indication that water level in the Spent Fuel Pool (< 111 feet 11 inches elevation) or Refueling Cavity has dropped to the level of the fuel other than by visual observation. Since there is no level indicating system in the fuel transfer canal, visual observation of loss of water level would also be required. The Fuel Transfer Canal is located at 101 feet 6 inches elevation near the bottom of the SFP. Depending on available level indication, the declared threshold may need to be based on indications of makeup rate or decrease in refueling water storage tank level.

The movement of irradiated fuel assemblies within containment requires a minimum water level of 23 feet above the top of the Reactor Vessel flange. During refueling, this maintains sufficient water level in the containment, fuel transfer canal, refueling cavity, and spent fuel pool. Sufficient water is necessary to retain iodine fission product activity in the water in the event of a fuel handling accident.

Loss of SFP water inventory results from either a rupture of the SFP or fuel transfer canal liner, or failure of the spent fuel cooling system, and the subsequent boil-off. Allowing SFP water level to decrease could result in spent fuel being uncovered reducing spent fuel decay heat removal, and creating an extremely hazardous radiation environment.

DCPP Basis Reference(s):

1. OP AP-21, "Irradiated Fuel Damage"
2. OP AP-22, "Spent Fuel Pool Low Level/High Temp/High Rad"
3. Plant Drawing 57730
4. Operator Information Manual Figure A-1-3

Category: R – Abnormal Rad Levels / Radiological Effluent

Sub-category: 2 – Onsite Rad Conditions, Spent Fuel Events

Initiating Condition: Release of radioactive material or increases in radiation levels within the facility that impedes operation of systems required to maintain safe operations or to establish or maintain cold shutdown

EAL:

RA2.3 Alert

Valid radiation monitor readings > 15 mR/hr in areas requiring continuous occupancy to maintain plant safety functions:

Control Room (0-RM-1)

OR

Central Alarm Station (by survey)

Mode Applicability:

All

Basis:

This EAL addresses elevated radiation levels in areas requiring continuous occupancy to maintain safe plant operation or perform a safe plant shutdown. Areas that meet this threshold include the Control Room and the Central Alarm Station (CAS). The security alarm station is included in this EAL because of its importance to permitting access to areas required to assure safe plant operations. Although the Rad Waste Control Station is continuously manned, Radwaste is not required to maintain safe plant operation or perform a safe plant shutdown.

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

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

This EAL could result in declaration of an Alert at Unit 1(2) due to a radioactivity release or radiation shine resulting from a major accident at Unit 2(1). This is appropriate if the increase impairs operations at the operating unit.

This EAL is not intended to apply to anticipated temporary radiation increases due to planned events (e.g., Radwaste container movement, depleted resin transfers, etc.). Areas requiring infrequent access and radiation levels that may adversely affect access to these areas are addressed in EAL RA2.4.

DCPP Basis Reference(s):

1. Plant Manual Volume 9B, "Curves and Misc. Data Sheet; I&C Radiation Monitoring and Allied Systems Data"

Category: R – Abnormal Rad Levels / Radiological Effluent

Sub-category: 2 – Onsite Rad Conditions, Spent Fuel Events

Initiating Condition: Release of radioactive material or increases in radiation levels within the facility that impedes operation of systems required to maintain safe operations or to establish or maintain cold shutdown

EAL:

RA2.4 Alert

Valid radiation monitor readings > 2 R/hr in the following areas requiring infrequent access to maintain plant safety functions:

- Auxiliary Building
- Fuel Handling Building
- Turbine Building
- Intake Structure

Mode Applicability:

All

Basis:

This EAL addresses elevated radiation levels in areas requiring infrequent access in order to maintain safe plant operation or perform a safe plant shutdown. Area radiation levels above 2 R/hr are indicative of radiation fields which may limit personnel access to equipment the operation of which may be needed to assure adequate core cooling or shutdown the reactor. The basis of the 2 R/hr value is as follows:

The DCPP annual administrative personnel exposure limit is 2 Rem/year. It is assumed that an emergency worker can receive up to his administrative limit without the need for emergency exposure authorization thus unduly restricting access to an area necessary for safe plant shutdown. Assuming that an activity required to be performed in the plant would require a one hour stay time in that area, an area exposure rate of 2 R/hr would not unduly restrict access to areas necessary for safe plant shutdown.

It is the impaired ability to operate the plant that results in the actual or potential degradation of the level of safety of the plant. The cause or magnitude of the increase in radiation levels is not a concern of this EAL. The SM/SEC/ED must consider the source or cause of the elevated radiation levels and determine if any other EAL may be involved. For example, a dose rate of 2 R/hr may be a problem in itself. However, the increase may also be indicative of high dose rates in the containment due to a LOCA. In this latter case, a Site Area Emergency or a General Emergency may be indicated by other EAL categories.

This EAL could result in declaration of an Alert at Unit 1(2) due to a radioactivity release or radiation shine resulting from a major accident at Unit 2(1). This is appropriate if the increase impairs operations at the operating unit.

This EAL is not meant to apply to increases in the containment radiation monitors as these are events that are addressed in other EALs. Nor is it intended to apply to anticipated temporary radiation increases due to planned events (e. g., Radwaste container movement, deplete resin transfers, etc.). Permanently installed radiation monitors are not specified in the EAL wording because portable monitoring devices may be used to determine area accessibility. It would then be possible to erroneously exclude information gained from portable monitor surveys when interpreting the EAL.

DCPP Basis Reference(s):

1. RP1 ID6, "Permissible Levels of Exposure"

Category C – Cold Shutdown / Refueling, System Malfunction

Category C EALs are directly associated with cold shutdown or refueling system safety functions. Given the variability of plant configurations (e.g., systems out-of-service for maintenance, containment open, reduced AC power redundancy, time since shutdown) during these periods, the consequences of any given initiating event can vary greatly. For example, a loss of decay heat removal capability that occurs at the end of an extended outage has less significance than a similar loss occurring during the first week after shutdown. Compounding these events is the likelihood that instrumentation necessary for assessment may also be inoperable. The cold shutdown and refueling system malfunction EALs are based on performance capability to the extent possible with consideration given to RCS integrity, containment closure, and Fuel Clad integrity for the applicable operating modes (5 - Cold Shutdown, 6 - Refueling, D – Defueled).

The events of this category pertain to the following subcategories:

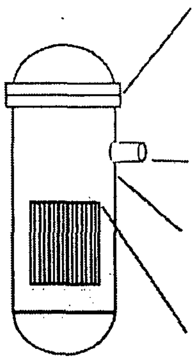
1. Loss of Power

Loss of vital plant electrical power can compromise plant safety system operability including decay heat removal and emergency core cooling systems which may be necessary to ensure fission product barrier integrity. This category involves total losses of vital plant 125 VDC power sources.

2. RCS Level

Reactor Vessel or RCS water level is directly related to the status of adequate core cooling and, therefore, Fuel Clad integrity. The figure below illustrates the Reactor Vessel level thresholds and associated EALs of this subcategory.

Reactor Vessel Level Thresholds



	Elev.	RVLIS
RV flange	114 ft	81.5% upper 83.3% full
Hot leg bottom	105 ft 9 in.	60% upper 63.3% full
6 in. below hot leg	105 ft 3 in.	62.1% full
Top of active fuel	103 ft	56.6% full

3. RCS Temperature

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

4. Communications

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

5. RCS Leakage

The Reactor Vessel provides a volume for the coolant that covers the reactor core. The Reactor Vessel and associated pressure piping (reactor coolant system) together provide a barrier to limit the release of radioactive material should the reactor Fuel Clad integrity fail.

Excessive RCS leakage greater than Technical Specification limits are utilized to indicate potential pipe cracks that may propagate to an extent threatening Fuel Clad, RCS and containment integrity.

6. Inadvertent Criticality

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

Category: C – Cold Shutdown / Refueling, System Malfunction

Sub-category: 1 – Loss of Power

Initiating Condition: AC power capability to vital buses reduced to a single power source for greater than 15 minutes such that any additional single failure would result in station blackout

EAL:

CU1.1	**Unusual Event**
AC power capability to Unit 1(2) Vital 4kV buses F, G and H reduced to a single power source (e.g., one DG, supply line from unaffected unit or one offsite power source) for > 15 min. such that ANY additional single failure would result in loss of ALL AC power to vital buses	

Mode Applicability:

5 - Cold Shutdown, 6 - Refueling

Basis:

The condition indicated by this EAL is the degradation of the offsite and onsite power sources such that any additional single failure would result in a loss of all AC power to the unit Vital 4kV buses. Buses F, G, and H are the Unit 1(2) Vital 4kV (essential) buses. Possible offsite power systems capable of providing power to the Vital 4kV buses include:

- Affected Unit's Aux Power system: The 4 kV system can be powered by backfeeding from the 500 kV Switchyard. The Main Generator MOD is opened. Power is supplied from the switchyard through the generator output breakers 532 (542) and/or 632 (642). The Main Transformer steps down 500 kV to 25 kV and the 25 kV is stepped down to 4 kV by Aux Transformer 1-2.

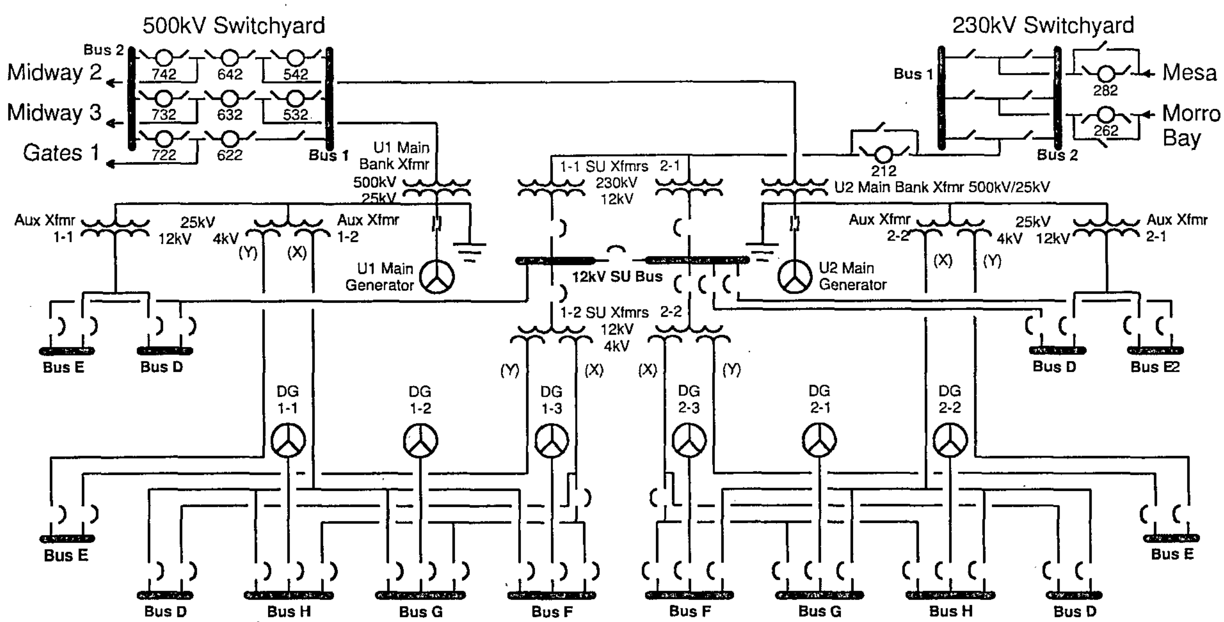
- Affected Unit's SU Power system: The SU bus is normally energized by the 230 kV Switchyard through OCB-212 and SU Transformer 1-1(2-1). The SU bus then supplies SU Transformer 1-2(2-2) which steps the voltage down to 4160 V.

- Other Unit's SU Power system

- Other Unit's Aux Power system

A basic flowpath diagram of the DCPP electrical distribution system is given in Figure C-1.

Figure C-1 DCPP Electrical Distribution System



Several combinations of power failures could therefore satisfy this EAL. The subsequent loss of the remaining power source escalates the event to an Alert under EAL CA1.1.

The 15-minute interval was selected as a threshold to exclude transient power losses. If multiple sources fail to energize the unit 1E 4kV buses within 15 minutes, an Unusual Event is declared under this EAL.

DCPP Basis Reference(s):

1. UFSAR, Section 8.2.1
2. UFSAR, Section 8.3.1
3. OP AP SD 1, "Loss of AC Power"
4. OP AP-26, "Loss of Offsite Power"
5. OP J-2:V, "Backfeeding the Unit From the 500kV System"
6. ECA-0.0, "Loss of All Vital AC Power"
7. ECA-0.3, "Restore 4kV Buses"

Category: C – Cold Shutdown / Refueling, System Malfunction

Sub-category: 1 – Loss of Power

Initiating Condition: Unplanned loss of required DC power for greater than 15 minutes

EAL:

CU1.2 Unusual Event

Unplanned loss of vital DC power to required DC buses based on < 105 VDC bus voltage indications

AND

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

Mode Applicability:

5 - Cold Shutdown, 6 - Refueling

Basis:

The purpose of this EAL is to recognize a loss of DC power compromising the ability to monitor and control the removal of decay heat during cold shutdown or refueling operations. This EAL is intended to be anticipatory in as much as the operating crew may not have necessary indication and control of equipment needed to respond to the loss. The fifteen minute interval is intended to exclude transient or momentary power losses.

The DC Power system is an ungrounded power generation, storage, and distribution system consisting of two subsystems:

- Non-Vital 125/250 VDC
- Vital 125 VDC

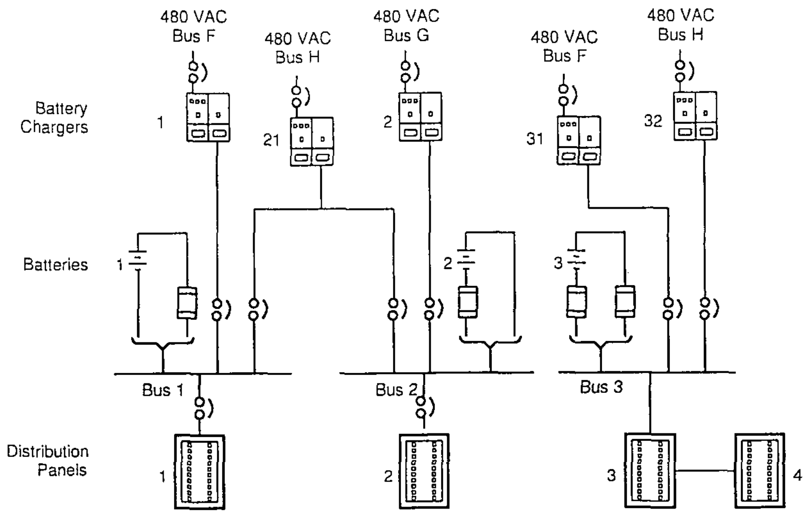
The Non-Vital 125/250 VDC system consists of two 125 VDC distribution networks. The Non-Vital 250 VDC system is a load center supplied by the two Non-Vital 125 VDC buses arranged in series. It supplies power to large DC loads such as backup and emergency lube oil pumps.

The vital 125 VDC system consists of three independent networks. Each network contains the following components:

- Battery
- Battery charger
- Standby battery charger to allow maintenance and/or testing
- Distribution panels
- Ground detector

The vital 125 VDC batteries and distribution panels are located in Area "H" of the 115 feet elevation of the Auxiliary Building. A basic one-line diagram of the Vital 125 VDC System is shown in Figure C-2.

Figure C-2 Vital 125 VDC One-Line Diagram



The vital 125 VDC batteries are Class 1E power supplies. Mean life of the batteries is 20 years. A total of six batteries, 11(21), 12(22), and 13(23) are supplied for Units 1 and 2. The batteries are sized to provide sufficient power to operate the DC loads for the time necessary to safely shut down the unit, should a 480-VAC source to one or more battery chargers be unavailable.

"Unplanned" is included in this EAL to preclude the declaration of an emergency as a result of planned maintenance activities such as maintenance on a train during shutdown periods. This EAL is the cold condition equivalent of the hot condition loss of DC power EAL SS1.2. If this loss results in the inability to maintain cold shutdown, the escalation to an Alert will be per EAL CA3.1.

DCPP Basis Reference(s):

1. UFSAR, Section 8.3.2.2
2. OP AP-23, "Loss of Vital DC Bus"
3. ECA-0.0, "Loss of All Vital AC Power"

Category: C – Cold Shutdown / Refueling, System Malfunction

Sub-category: 1 – Loss of Power

Initiating Condition: Loss of all offsite power and loss of all onsite AC power to vital buses

EAL:

CA1.1	**Alert**
Loss of ALL offsite and onsite AC power to Unit 1(2) Vital 4kV buses F, G and H for > 15 min.	

Mode Applicability:

5 - Cold Shutdown, 6 - Refueling, D - Defueled

Basis:

Loss of all AC power compromises all plant safety systems requiring electric power including RHR, ECCS, Containment Heat Removal, Spent Fuel Heat Removal and the Ultimate Heat Sink. When in cold shutdown, refueling, or defueled mode the event can be classified as an Alert, because of the significantly reduced decay heat, lower temperature and pressure, increasing the time to restore one of the emergency busses, relative to that specified for the Site Area Emergency EAL. Escalating to Site Area Emergency, if appropriate, is by EALs in Subcategory R.1, Offsite Rad Conditions, or Subcategory H.6, Judgment.

Possible offsite power systems capable of providing power to the Vital 4kV buses include:

Affected Unit's Aux Power system: The 4 kV system can be powered by backfeeding from the 500 kV Switchyard. The Main Generator MOD is opened. Power is supplied from the switchyard through the generator output breakers 532 (542) and/or 632 (642). 500 kV is stepped down to 25 kV by the Main Transformer. 25 kV is stepped down to 4 kV by Aux Transformer 1-2.

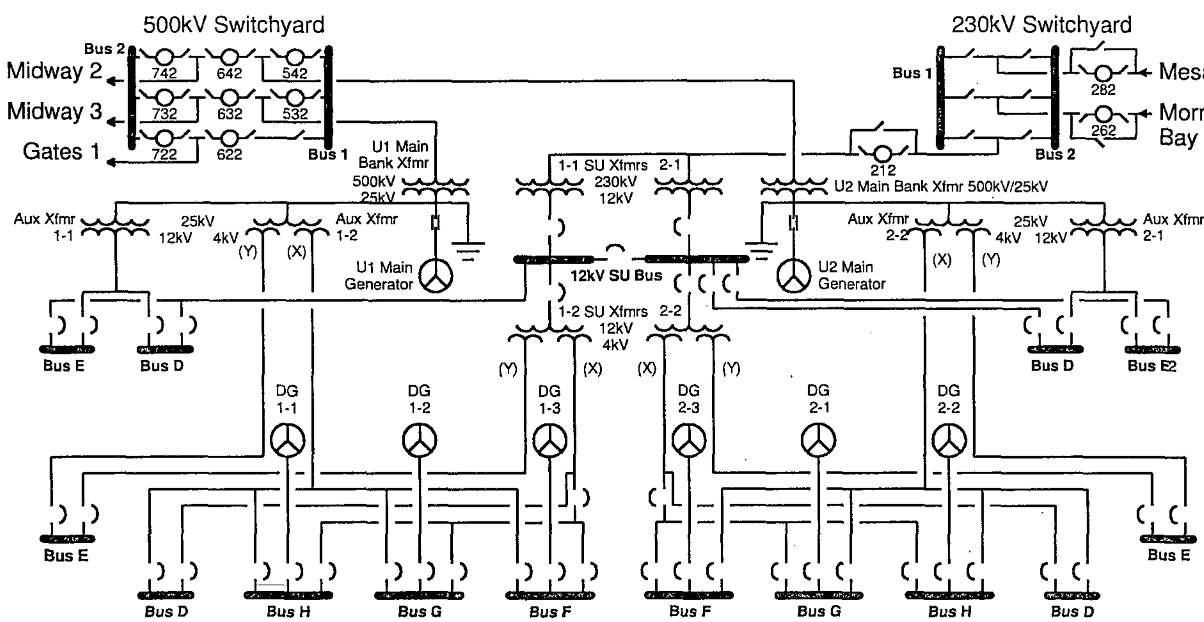
Affected Unit's SU Power system: The SU bus is normally energized by the 230 kV Switchyard through OCB-212 and SU Transformer 1-1(2-1). The SU bus then supplies SU Transformer 1-2(2-2) which steps the voltage down to 4160 V.

Other Unit's SU Power system

Other Unit's Aux Power system

A basic flowpath diagram of the DCPP electrical distribution system is given in Figure C-1.

Figure C-1 DCPP Electrical Distribution System



The emergency diesel generators are the onsite power sources:

- D/G 1-3(2-3) Vital 4kV Bus F
- D/G 1-2(2-2) Vital 4kV Bus G
- D/G 1-1(2-1) Vital 4kV Bus H

Consideration should be given to operable loads necessary to remove decay heat or provide Reactor Vessel makeup capability when evaluating loss of all AC power to Unit 1(2) Vital 4kV buses. Even though a Unit 1(2) Vital 4kV bus may be energized, if necessary loads (i.e., loads that if lost would inhibit decay heat removal capability or Reactor Vessel makeup capability) are not operable on the energized bus then the bus should not be considered operable.

The 15-minute interval was selected as a threshold to exclude transient power losses.

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

DCPP Basis Reference(s):

1. UFSAR, Section 8.2.1
2. UFSAR, Section 8.3.1
3. OP AP SD, "1 Loss of AC Power"
4. OP AP-26, "Loss of Offsite Power"
5. OP J-2:V, "Backfeeding the Unit From the 500kV System"
6. ECA-0.0, "Loss of All Vital AC Power"
7. ECA-0.3, "Restore 4kV Buses"

Category: C – Cold Shutdown / Refueling, System Malfunction

Sub-category: 2 – RCS Level

Initiating Condition: Unplanned loss of RCS inventory with irradiated fuel in the Reactor Vessel

EAL:

CU2.1	**Unusual Event**
Unplanned RCS level decreasing below the Reactor Vessel flange for ≥ 15 min.	

Mode Applicability:

6 - Refueling

Basis:

The Reactor Vessel flange is at 114 feet and can be monitored by:

- RVRLIS (narrow range and wide range)
- LI-400 standpipe (local indication or remote indication provided by use of a video camera and monitor in the control room)
- RVLIS upper range: 81.5%
- RVLIS full range: 83.3%
- Visual observation

The purpose of the Reactor Vessel Refueling Level Instrumentation System (RVRLIS) is to provide Reactor Vessel and refueling cavity level indication during refueling, when the vessel head will be removed, and during drainage to half loop. The system is designed to be used only when the RCS is at near atmospheric pressure or when a vacuum is being established for refill operations. The wide range and narrow range RVRLIS (if required) and the LI-400 standpipe systems remain in service from the time RCS level is lowered below 25% PZR cold cal level until just prior to pressurizing the RCS. Narrow Range RVRLIS is required if reduced inventory conditions (below 111 feet elevation) are planned.

The LI-400 standpipe is a magnetic level indicator (LI-400A, B, C standpipe) and provides local indication of Reactor Vessel refueling level. The indicator is mounted on the outside of the bio-shield and can be viewed from the 91 ft el. of Containment. The indicator is composed of three mechanical flag indicator units.

- Each unit contains a magnetic float that rises and falls with water level.
- As the float passes a flag outside the detector it will flip and display an orange color on the flag.

The purpose of the Reactor Vessel Level Instrumentation System (RVLIS) is to measure the level of the water or the relative void content of the coolant in the Reactor Vessel. RVLIS is not likely to be operable during Refueling mode but are listed here in case it is possible to restore them to service. The RVLIS setpoints corresponding to the elevation of the Reactor Vessel flange were obtained as follows:

- Upper range:
    - Per SC-I-87B, span = 179.4 in., 60% = 105.9536 feet (taps are at 30 deg from centerline)
    - % span/in. = 40/179.4 = 0.223%/in. and vessel flange = 114.0 feet
    - (114.0-105.9536) x 12 x 0.223 + 60 = 81.5%

- Full range:
  - Per SC-I-87B, span = 494.9 in. and 0% = 79.6536 feet
  - % span/in. = 100 / 494.9 = 0.20206%/in. and vessel flange = 114.0 feet
  - (114.0 - 79.6536) x 12 x 0.20206 = 83.3%

This EAL is an Unusual Event because it may be a precursor of more serious conditions and, as result, is considered to be a potential degradation of the level of safety of the plant. Refueling operations that lower RCS water level below the Reactor Vessel flange are carefully planned and procedurally controlled. An unplanned event that results in water level decreasing below the Reactor Vessel flange warrants declaration of an Unusual Event due to the reduced RCS inventory that is available to keep the core covered. The fifteen-minute interval was chosen because it is reasonable to assume that level can be restored within this time frame using one or more of the redundant means of refill that should be available. If level cannot be restored in this time frame, a more serious condition may exist. Continued loss of RCS Inventory will result in escalation to the Alert level via either EAL CA2.1 or EAL CA3.1 (inability to maintain plant in cold shutdown with irradiated fuel in the RPV).

This EAL is not applicable to drops in flooded Refueling Cavity water level (covered by decreasing Spent Fuel Pool water level in EAL RU2.1) until such time as the level decreases to the level of the vessel flange. If level continues to decrease and reaches the bottom inside diameter of the RCS hot leg penetration, escalation to the Alert level under EAL CA2.1 would be appropriate. If the decreasing level is accompanied by RCS heatup, escalation to the Alert level under EAL CA3.1 may also be appropriate.

In Cold Shutdown mode, the RCS will normally be intact and standard RCS inventory and level monitoring means are available. In the Refueling mode, the RCS is not intact and Reactor Vessel level and inventory are monitored by different means. In the Refueling mode, normal means of core temperature indication and RCS level indication may not be available. Redundant means of Reactor Vessel level indication will normally be installed (including the ability to monitor level visually) to assure that the ability to monitor level will not be interrupted.

AP E-55, Equipment Elevations for RCS Flood-Up and Drain-Down, provides a cross-reference of indicated water levels and key plant elevations.

DCPP Basis Reference(s):

1. OP A-2 II, "U1 Reactor Vessel - Draining the RCS to the Vessel Flange - With Fuel in Vessel"
2. OP AP SD 2, "Loss of RCS Inventory"
3. Instrument Scaling Calculation SC-I-87B, "Reactor Vessel Level Instrumentation System and Subcooled Margin Monitor"
4. AP E-55, "Equipment Elevations for RCS Flood-Up and Drain-Down"
5. Operator Information Manual Diagram A-1-3 Equipment Elevations for RCS Flood-Up and Drain-Down

Category: C – Cold Shutdown / Refueling, System Malfunction

Sub-category: 2 – RCS Level

Initiating Condition: Unplanned loss of RCS inventory with irradiated fuel in the Reactor Vessel

EAL:

CU2.2 Unusual Event

Loss of inventory as indicated by unexplained increase in ANY Table C-1 sump/tank level

AND

Reactor Vessel water level cannot be monitored

Table C-1 Sumps/Tanks
• Containment Structure Sump 1-1 (2-1) • Containment Structure Sump 1-2 (2-2) • Reactor Cavity Sump • PRT • RCDT • CCW surge tank • Auxiliary Building sump • RWST

Mode Applicability:

6 - Refueling

Basis:

This EAL is an Unusual Event because it may be a precursor of more serious conditions and, as result, is considered to be a potential degradation of the level of safety of the plant.

In Cold Shutdown mode, the RCS will normally be intact and standard RCS inventory and level monitoring means are available. In the Refueling mode, the RCS is not intact and Reactor Vessel water level and inventory are monitored by different means. In the Refueling mode, normal means of core temperature indication and RCS level indication may not be available. Redundant means of Reactor Vessel level indication will normally be installed (including the ability to monitor level visually) to assure that the ability to monitor level will not be interrupted. Reactor Vessel water level is normally monitored using the following instruments:

- RVRLIS
- LI-400 standpipe
- RVLIS upper range
- RVLIS full range

In this EAL, all water level indication is unavailable, and the Reactor Vessel inventory loss must be detected by sump or tank level changes. Plant design and procedures provide the capability to detect and assess primary system leakage. Sump level increases must be evaluated against other potential sources of leakage such as cooling water sources inside the Containment to ensure they are indicative of RCS leakage.

DCPP Basis Reference(s):

1. UFSAR, "Chapter 6.2"
2. OP AP SD 2, "Loss of RCS Inventory"
3. OP AP-1, "Excessive Reactor Coolant System Leakage"
4. OP A-2 II, "Reactor Vessel - Draining the RCS to the Vessel Flange - With Fuel in Vessel"
5. AP E-55, "Equipment Elevations for RCS Flood-Up and Drain-Down"
6. Operator Information Manual Diagram A-1-3, "Equipment Elevations for RCS Flood-Up and Drain-Down"

Category: C – Cold Shutdown / Refueling, System Malfunction

Sub-category: 2 – RCS Level

Initiating Condition: Loss of RCS/RPV inventory with Irradiated fuel in the RPV

EAL:

CA2.1 **Alert**

Loss of inventory as indicated by **EITHER**:

Reactor Vessel level < bottom of the RCS hot leg as indicated by ANY of the following:

- RVRLIS, LI-400 standpipe or ultrasonic sensor ≤ 105 ft. 9 in.
- RVLIS upper range < 60%
- RVLIS full range < 63.3%

OR

Reactor Vessel level <u>cannot</u> be monitored for > 15 min. with unexplained increase in ANY Table C 1 sump/tank level

Table C-1 Sumps/Tanks
• Containment Structure Sump 1-1 (2-1)
• Containment Structure Sump 1-2 (2-2)
• Reactor Cavity Sump
• PRT
• RCDT
• CCW surge tank
• Auxiliary Building sump
• RWST

Mode Applicability:

5 - Cold Shutdown, 6 - Refueling

Basis:

When Reactor Vessel water level decreases to 105 ft. 9 in. el., the bottom of the RCS hot leg penetration is uncovered. The elevation of the bottom of the RCS hot leg penetration can be monitored by:

- RVRLIS
- LI-400 standpipe
- Ultrasonic sensor
- RVLIS

The purpose of the Reactor Vessel Refueling Level Instrumentation System (RVRLIS) is to provide Reactor Vessel and refueling cavity level indication during refueling, when the vessel head will be removed, and during drainage to half loop. The system is designed to be used only when the RCS is at near atmospheric pressure or when a vacuum is being established for refill operations. The wide range and narrow range RVRLIS (if required) and the LI-400 standpipe systems remain in service from the time RCS level is lowered below 25% PZR cold cal level until just prior to pressurizing the RCS. Narrow Range RVRLIS is required if reduced inventory conditions (below 111 ft. el.) are planned.

The LI-400 standpipe is a magnetic level indicator (LI-400A, B, C standpipe) and provides local indication of Reactor Vessel refueling level. The indicator is mounted on the outside of the bio-shield and can be viewed from the 91 ft. el. of Containment. The indicator is composed of three mechanical flag indicator units.

- Each unit contains a magnetic float that rises and falls with water level.
- As the float passes a flag outside the detector it will flip and display an orange color on the flag.

RVRLIS, LI-400 standpipe and ultrasonic detectors are off-scale low (105 ft 9 in.) when Reactor Vessel water level drops below the elevation of the bottom of the RCS hot leg penetration. The ultrasonic sensor is installed by ISI during an outage and measures level on one of the hot legs; its availability in Cold Shutdown mode is unlikely.

The purpose of the Reactor Vessel Level Instrumentation System (RVLIS) is to measure the level of the water or the relative void content of the coolant in the Reactor Vessel. The RVLIS setpoints corresponding to the elevation of the bottom of the RCS hot leg penetration were obtained as follows:

- Upper range: Per SC-I-87B, the upper range is scaled from hot leg empty to vessel full; therefore, upper range level at the bottom of the hot leg penetration is 60% (hot leg tap is at 30 deg. from centerline and unable to measure below this point).

- Full range:
  - Per SC-I-87B, span = 494.9 in. and 0% = 79.6536 feet
  - % span/in. = 100 / 494.9 = 0.20206%/in. and bottom of the hot leg (from above) = 105.75 feet
  - (105.75 - 79.6536) x 12 x 0.20206 = 63.3%

This EAL serves as a precursor to a loss of ability to adequately cool the fuel. The magnitude of this loss of water indicates that makeup systems have not been effective and may not be capable of preventing further Reactor Vessel water level decrease and potential core uncovery. The inability to restore and maintain level after reaching this setpoint infers a failure of the RCS barrier.

In Cold Shutdown, the decay heat available to raise RCS temperature during a loss of inventory or heat removal event may be significantly greater than in the Refueling mode. Entry into Cold Shutdown mode may be attained within hours of operating at power or hours after refueling is completed. Entry into the Refueling mode procedurally may not occur for many hours after the reactor has been shutdown. Thus, the heatup and the threat to damaging the Fuel Clad may be lower for events that occur in the Refueling mode with irradiated fuel in the Reactor Vessel. Note that the heatup threat could be lower for Cold Shutdown conditions if the entry into Cold Shutdown was following a refueling.

In Cold Shutdown mode, the RCS will normally be intact and standard RCS inventory and level monitoring means are available. In the Refueling mode, the RCS is not intact and Reactor Vessel water level and inventory are monitored by different means. In the Refueling mode, normal means of core temperature indication and RCS level indication may not be available. Redundant means of Reactor Vessel water level indication will normally be installed (including the ability to monitor level visually) to assure that the ability to monitor level will not be interrupted.

AP E-55, Equipment Elevations for RCS Flood-Up and Drain-Down, provides a cross-reference of indicated water levels and key plant elevations.

In the second condition of this EAL, all water level indication would be unavailable, and the Reactor Vessel inventory loss must be detected by sump or tank level changes. Plant design and procedures provide the capability to detect and assess primary system leakage. Sump level increases must be evaluated against other potential sources of leakage such as cooling water sources inside the Containment to ensure they are indicative of RCS leakage.

The 15-minute interval for the loss of level indication was chosen because it is half of the Site Area Emergency EAL duration. The interval allows this EAL to be an effective precursor to the Site Area Emergency EALs CS2.1, CS2.2, CS2.3 and CS2.4 Significant fuel damage is not expected to occur until the core has been uncovered for greater than one hour. Therefore this EAL meets the definition for an Alert emergency.

DCPP Basis Reference(s):

1. Plant Drawing No. 57729
2. UFSAR, Section 5.6.1
3. UFSAR, Section 7.5
4. OP A-2:III, "Reactor Vessel - Draining to Half Loop/Half Loop Operations with Fuel in Vessel"
5. Instrument Scaling Calculation SC-I-87B, "Reactor Vessel Level Instrumentation System and Subcooled Margin Monitor"
6. AP E-55, "Equipment Elevations for RCS Flood-Up and Drain-Down"
7. Operator Information Manual Diagram A-1-3, "Equipment Elevations for RCS Flood-Up and Drain-Down"

Category: C – Cold Shutdown / Refueling, System Malfunction

Sub-category: 2 – RCS Level

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

EAL:

CS2.1 Site Area Emergency

With Containment closure not established:

RVLIS full range < 62.1%

OR

Reactor Vessel level cannot be monitored for > 30 min. with unexplained increase in ANY Table C-1 sump/tank level

Table C-1 Sumps/Tanks
• Containment Structure Sump 1-1 (2-1)
• Containment Structure Sump 1-2 (2-2)
• Reactor Cavity Sump
• PRT
• RCDT
• CCW surge tank
• Auxiliary Building sump
• RWST

Mode Applicability:

5 - Cold Shutdown

Basis:

When Reactor Vessel water level decreases to 105 feet 3 inches elevation, water level is six inches below the elevation of the bottom of the RCS hot leg penetration. When Reactor Vessel water level drops significantly below the elevation of the bottom of the RCS hot leg penetration, all sources of RCS injection have failed or are incapable of making up for the inventory loss. RVLIS full range is the only Reactor Vessel water level monitoring system capable of detecting water level decrease to this level.

The purpose of the Reactor Vessel Level Instrumentation System (RVLIS) is to measure the level of the water or the relative void content of the coolant in the Reactor Vessel. The RVLIS setpoint corresponding to six inches below the elevation of the bottom of the RCS hot leg penetration was obtained as follows:

- Per SC-I-87B, span = 494.9 in. and 0% = 79.6536 feet
- % span/in. = 100 / 494.9 = 0.20206%/in. and bottom of the hot leg (from above) = 105.75 feet
- (105.75 - 6 - 79.6536) x 12 x 0.20206 = 62.1%

Other Reactor Vessel water level monitoring systems (e.g., RVRLIS, LI-400 standpipe, ultrasonic sensor, RVLIS upper range) are downscale-low when water level drops below the elevation of the bottom of the RCS hot leg penetration.

Under the conditions specified by this EAL, continued decrease in Reactor Vessel water level is indicative of a loss of inventory control. Inventory loss may be due to a vessel breach, RCS pressure boundary leakage or continued boiling in the Reactor Vessel. The magnitude of this loss of water indicates that makeup systems have not been effective and may not be capable of preventing further RCS or Reactor Vessel water level decrease and potential core uncovery. The inability to restore and maintain level after reaching this setpoint infers a failure of the RCS barrier and potential loss of the Fuel Clad barrier.

Containment Closure is the procedurally defined action taken to secure containment and its associated structures, systems and components as a functional barrier to fission product release under existing plant conditions. Containment Closure is defined by Administrative Procedure AD8.DC54, “Containment Closure.” Containment closure is tracked if plant conditions change that could raise the risk of a fission product release as a result of a loss of decay heat removal. Effluent release is not expected with closure established.

In Cold Shutdown, the decay heat available to raise RCS temperature during a loss of inventory or heat removal event may be significantly greater than in the Refueling mode. Entry into Cold Shutdown mode may be attained within hours of operating at power or hours after refueling is completed. Entry into the Refueling mode procedurally may not occur for many hours after the reactor has been shutdown. Thus, the heatup and the threat to damaging the Fuel Clad may be lower for events that occur in the Refueling mode with irradiated fuel in the Reactor Vessel. Note that the heatup threat could be lower for Cold Shutdown conditions if the entry into Cold Shutdown was following a refueling. The 30-minute interval associated the inability to monitor Reactor Vessel water level recognizes that the RCS is normally intact while in the Cold Shutdown mode.

If water level monitoring capability is unavailable, the Reactor Vessel inventory loss must be detected by sump or tank level changes. Plant design and procedures provide the capability to detect and assess primary system leakage. Sump level increases must be evaluated against other potential sources of leakage such as cooling water sources inside the Containment to ensure they are indicative of RCS leakage.

Escalation to a General Emergency is via EAL CG2.1 or EALs in Subcategory R.1, Offsite Rad Conditions.

DCPP Basis Reference(s):

1. Plant Drawing No. 57729
2. UFSAR, Section 5.6.1
3. UFSAR, Section 7.5
4. OP A-2:III, "Reactor Vessel - Draining to Half Loop/Half Loop Operations with Fuel in Vessel"
5. AP E-55, "Equipment Elevations for RCS Flood-Up and Drain-Down"
6. Instrument scaling calculation SC-I-87B, "Reactor Vessel Level Instrumentation System and Subcooled Margin Monitor"
7. AD8.DC54, "Containment Closure"

Category: C – Cold Shutdown / Refueling, System Malfunction

Sub-category: 2 – RCS Level

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

EAL:

CS2.2 Site Area Emergency

With Containment closure established:

RVLIS full range < 56.6%

OR

Reactor Vessel level <u>cannot</u> be monitored for > 30 min. with **EITHER**:

- Unexplained increase in ANY Table C-1 sump/tank level
- Erratic source range monitor indication

Table C-1 Sumps/Tanks
• Containment Structure Sump 1-1 (2-1)
• Containment Structure Sump 1-2 (2-2)
• Reactor Cavity Sump
• PRT
• RCDT
• CCW surge tank
• Auxiliary Building sump
• RWST

Mode Applicability:

5 - Cold Shutdown

Basis:

When Reactor Vessel water level drops significantly below the elevation of the bottom of the RCS hot leg penetration, all sources of RCS injection have failed or are incapable of making up for the inventory loss. This level drop can only be remotely monitored by Reactor Vessel Level Instrumentation System (RVLIS). When Reactor Vessel water level drops below RVLIS full range setpoint of 56.6%, core uncovery is about to occur.

The purpose of the RVLIS is to measure the level of the water or the relative void content of the coolant in the Reactor Vessel. The RVLIS setpoint corresponding to the top of the active fuel (TOAF) was obtained as follows:

- Per SC-I-87B, span = 494.9 in. and 0% = 79.6536 feet
- % span/in. = 100 / 494.9 = 0.20206%/in. and top of the core = 103 feet
- (103 - 79.6536) x 12 x 0.20206 = 56.6%

Other Reactor Vessel water level monitoring systems (e.g., RVRLIS, LI-400 standpipe, ultrasonic sensor, RVLIS upper range) are downscale-low when water level drops below the elevation of the bottom of the RCS hot leg penetration.

Under the conditions specified by this EAL, continued decrease in Reactor Vessel water level is indicative of a loss of inventory control. Inventory loss may be due to a vessel breach, RCS pressure boundary leakage or continued boiling in the Reactor Vessel. The magnitude of this loss of water indicates that makeup systems have not been effective and may not be capable of preventing further RCS or Reactor Vessel water level decrease and potential core uncovery. The inability to restore and maintain level after reaching this setpoint infers a failure of the RCS barrier and Potential Loss of the Fuel Clad barrier.

Containment Closure is the procedurally defined action taken to secure containment and its associated structures, systems and components as a functional barrier to fission product release under existing plant conditions. Containment Closure is defined by Administrative Procedure AD8.DC54, "Containment Closure." The status of Containment closure is tracked if plant conditions change that could raise the risk of a fission product release as a result of a loss of decay heat removal. Effluent release is not expected with closure established.

In Cold Shutdown, the decay heat available to raise RCS temperature during a loss of inventory or heat removal event may be significantly greater than in the Refueling mode. Entry into Cold Shutdown mode may be attained within hours of operating at power or hours after refueling is completed. Entry into the Refueling mode procedurally may not occur for many hours after the reactor has been shutdown. Thus, the heatup and the threat to damaging the Fuel Clad may be lower for events that occur in the Refueling mode with irradiated fuel in the Reactor Vessel. Note that the heatup threat could be lower for Cold Shutdown conditions if the entry into Cold Shutdown was following a refueling. The 30-minute interval associated the inability to monitor Reactor Vessel water level recognizes that the RCS is normally intact while in the Cold Shutdown mode.

If water level monitoring capability is unavailable, the Reactor Vessel inventory loss must be detected by sump or tank level changes. Plant design and procedures provide the capability to detect and assess primary system leakage. Sump level increases must be evaluated against other potential sources of leakage such as cooling water sources inside the Containment to ensure they are indicative of RCS leakage. Post-TMI studies indicate that the installed nuclear instrumentation will operates erratically when the core is uncovered and source range monitors can be used as a tool for making such determinations. SRM count rate can be indicated in the Control Room by:

- Individual Start Up Rate meters on CC-1 which receive signals from Startup Channels N-31 and N-32.
- Local meter on the drawer on NIS Rack IV.
- Audio Count Rate (NIS Rack IV)
- NR-45 video recorder located on CC-1

Escalation to a General Emergency is via EAL CG2.1 or EALs in Subcategory R.1, Offsite Rad Conditions.

DCPP Basis Reference(s):

1. Plant Drawing No. 57729
2. UFSAR, Section 5.6.1
3. UFSAR, Section 7.5
4. OP A-2:III, “Reactor Vessel - Draining to Half Loop/Half Loop Operations with Fuel in Vessel”
5. AP E-55, “Equipment Elevations for RCS Flood-Up and Drain-Down”
6. Instrument scaling calculation SC-I-87B, "Reactor Vessel Level Instrumentation System and Subcooled Margin Monitor"
7. AD8.DC54, “Containment Closure”
8. UFSAR, Table 7.5-3
9. OP L-2, “Hot Standby to Startup Mode”
10. STP I-4C, “Calibration of Audio Count Rate/Scaler Timer Channel”

Category: C – Cold Shutdown / Refueling, System Malfunction

Sub-category: 2 – RCS Level

Initiating Condition: Loss of Reactor Vessel inventory affecting core decay heat removal capability with irradiated fuel in the Reactor Vessel

EAL:

CS2.3 Site Area Emergency

With Containment closure not established:

RVLIS full range < 62.1%

OR

Reactor Vessel level cannot be monitored with indication of core uncovery as evidenced by **EITHER**:

- Containment radiation (RE-30 or RE-31) > 20 R/hr
- Erratic source range monitor indication

Mode Applicability:

6 - Refueling

Basis:

When Reactor Vessel water level decreases to 105 feet 3 inches elevation, water level is six inches below the elevation of the bottom of the RCS hot leg penetration. When Reactor Vessel water level drops significantly below the elevation of the bottom of the RCS hot leg penetration, all sources of RCS injection have failed or are incapable of making up for the inventory loss. RVLIS full range is the only Reactor Vessel water level monitoring system capable of detecting water level decrease to this level but is not likely to be in service in Refueling mode.

The purpose of the Reactor Vessel Level Instrumentation System (RVLIS) is to measure the level of the water or the relative void content of the coolant in the Reactor Vessel. The RVLIS setpoint corresponding to six inches below the elevation of the bottom of the RCS hot leg penetration was obtained as follows:

- Per SC-I-87B, span = 494.9 in. and 0% = 79.6536 feet
- % span/in. = 100 / 494.9 = 0.20206%/in. and bottom of the hot leg (from above) = 105.75 feet
- (105.75 - 6 - 79.6536) x 12 x 0.20206 = 62.1%

Other Reactor Vessel water level monitoring systems (e.g., RVRLIS, LI-400 standpipe, ultrasonic sensor, RVLIS upper range) are downscale-low when water level drops below the elevation of the bottom of the RCS hot leg penetration.

Under the conditions specified by this EAL, continued decrease in Reactor Vessel water level is indicative of a loss of inventory control. Inventory loss may be due to a vessel breach, RCS pressure boundary leakage or continued boiling in the Reactor Vessel. The magnitude of this loss of water indicates that makeup systems have not been effective and may not be capable of preventing further RCS or Reactor Vessel water level decrease and potential core uncovery. The inability to restore and maintain level after reaching this setpoint infers a failure of the RCS barrier and Potential Loss of the Fuel Clad barrier.

Containment Closure is the procedurally defined action taken to secure containment and its associated structures, systems and components as a functional barrier to fission product release under existing plant conditions. Containment Closure is defined by Administrative Procedure AD8.DC54, "Containment Closure." Containment closure is tracked if plant conditions change that could raise the risk of a fission product release as a result of a loss of decay heat removal. Effluent release is not expected with closure established.

In the Refueling mode, the decay heat available to raise RCS temperature during a loss of inventory or heat removal event may be significantly less than in the Cold Shutdown mode. Entry into Cold Shutdown mode may be attained within hours of operating at power or hours after refueling is completed. Entry into the Refueling mode procedurally may not occur for many hours after the reactor has been shutdown. The heatup and the threat to damaging the Fuel Clad thus may be lower for events that occur in the Refueling mode with irradiated fuel in the Reactor Vessel than for events that occur in the Cold Shutdown mode. The reduced RCS heatup rate lowers boil-off and may slow the loss of vessel inventory.

This EAL is based on concerns raised by Generic Letter 88-17, "Loss of Decay Heat Removal", SECY 91-283, "Evaluation of Shutdown and Low Power Risk Issues, NUREG-1449, Shutdown and Low-Power Operation at Commercial Nuclear Power Plants in the United States", and, NUMARC 91-06, "Guidelines for Industry Actions to Assess Shutdown Management." A number of variables, (e.g., mid-loop, reduced level/flange level, head in place, or cavity flooded, RCS venting strategy, decay heat removal system design, vortexing pre-disposition, steam generator U-tube draining, etc.) can have a significant impact on heat removal capability challenging the Fuel Clad barrier.

In Refueling mode, Reactor Vessel water level indication from RVLIS is likely unavailable but alternate means of level indication are normally installed (including visual observation) to assure that the ability to monitor water level will not be interrupted. The Reactor Vessel inventory loss may be detected by Containment High Range Area Radiation monitors or erratic Source Range Monitor indication. Valid high alarm (20 R/hr) on Contmt Hi Range Area Mon Hi Rad RE-30 or R-31 is indicative of core uncovery. There are a number of variables governing the projected dose rate from an actual core uncovery. Using the high alarm setpoint on the Containment High Radiation Monitors should provide an anticipatory indication of a reduction in the effectiveness of the shielding provided by the water cover. Post-TMI studies indicate that the installed nuclear instrumentation will operate erratically when the core is uncovered and source range monitors can be used as a tool for making such determinations. SRM count rate can be indicated in the Control Room by:

- Individual SUR meters on CC-1 which receive signals from Startup Channels N-31 and N-32.
- Local meter on the drawer on NIS Rack IV.
- Audio Count Rate (NIS Rack IV)
- NR-45 video recorder located on CC-1

Escalation to a General Emergency is via EAL CG2.1 or EALs in Subcategory R.1, Offsite Rad Conditions.

DCPP Basis Reference(s):

1. Plant Drawing No. 57729
2. UFSAR, Section 5.6.1
3. UFSAR, Section 7.5
4. OP A-2:III, "Reactor Vessel - Draining to Half Loop/Half Loop Operations with Fuel in Vessel"
5. AP E-55, "Equipment Elevations for RCS Flood-Up and Drain-Down"
6. Instrument scaling calculation SC-I-87B, "Reactor Vessel Level Instrumentation System and Subcooled Margin Monitor"
7. AD8.DC54, "Containment Closure"
8. UFSAR, Table 7.5-3
9. OP L-2, "Hot Standby to Startup Mode"
10. STP I-4C, "Calibration of Audio Count Rate/Scaler Timer Channel"

11. 1-AR PK-11-19, "Containment Radiation"
12. 2-AR PK-11-19, "HI LVL RAD MON SYSTEM"
13. Plant Manual Volume 9B, Curves and Misc. Data Sheet; - I&C Radiation Monitoring and Allied Systems Data"

Category: C – Cold Shutdown / Refueling, System Malfunction

Sub-category: 2 – RCS Level

Initiating Condition: Loss of Reactor Vessel inventory affecting core decay heat removal capability with irradiate fuel in the Reactor Vessel

EAL:

CS2.4 Site Area Emergency

With Containment closure established:

RVLIS full range < 56.6%

OR

Reactor Vessel level cannot be monitored with indication of core uncovery as evidenced by **EITHER**:

- Containment radiation (RE-30 or RE-31) > 20 R/hr
- Erratic source range monitor indication

Mode Applicability:

6 - Refueling

Basis:

Under the conditions specified by this EAL, continued decrease in Reactor Vessel water level is indicative of a loss of inventory control. Inventory loss may be due to a vessel breach, RCS pressure boundary leakage or continued boiling in the Reactor Vessel. The magnitude of this loss of water indicates that makeup systems have not been effective and may not be capable of preventing further RCS or Reactor Vessel water level decrease and potential core uncovery. The inability to restore core submergence after reaching this setpoint infers a failure of the RCS barrier and potential loss of the Fuel Clad barrier.

When Reactor Vessel water level drops below RVLIS full range setpoint of 56.6%, core uncovery is about to occur. RVLIS full range is the only Reactor Vessel water level monitoring system capable of detecting water level decrease to this level but is not likely to be in service in Refueling mode.

The purpose of the RVLIS is to measure the level of the water or the relative void content of the coolant in the Reactor Vessel. The RVLIS setpoint corresponding to the top of the active fuel (TOAF) was obtained as follows:

- Per SC-I-87B, span = 494.9 in. and 0% = 79.6536 feet
- % span/in. = 100 / 494.9 = 0.20206%/in. and top of the core = 103 feet
- (103 - 79.6536) x 12 x 0.20206 = 56.6%

Other Reactor Vessel water level monitoring systems (e.g., RVRLIS, LI-400 standpipe, ultrasonic sensor, RVLIS upper range) are downscale-low when water level drops below the elevation of the bottom of the RCS hot leg penetration.

Containment Closure is the procedurally defined action taken to secure containment and its associated structures, systems and components as a functional barrier to fission product release under existing plant conditions. Containment Closure is defined by Administrative Procedure AD8.DC54, “Containment Closure.” Containment closure is tracked if plant conditions change that could raise the risk of a fission product release as a result of a loss of decay heat removal. Effluent release is not expected with closure established.

In the Refueling mode, the decay heat available to raise RCS temperature during a loss of inventory or heat removal event may be significantly less than in the Cold Shutdown mode. Entry into Cold Shutdown mode may be attained within hours of operating at power or hours after refueling is completed. Entry into the Refueling mode procedurally may not occur for many hours after the reactor has been shutdown. The heatup and the threat to damaging the Fuel Clad thus may be lower for events that occur in the Refueling mode with irradiated fuel in the Reactor Vessel than for events that occur in the Cold Shutdown mode. The reduced RCS heatup rate decreases boil-off and may slow the loss of vessel inventory. When in the Refueling mode, the Reactor Vessel inventory loss is therefore allowed to challenge core uncovery before a Site Area Emergency declaration is warranted.

This EAL is based on concerns raised by Generic Letter 88-17, "Loss of Decay Heat Removal", SECY 91-283, "Evaluation of Shutdown and Low Power Risk Issues", NUREG-1449, "Shutdown and Low-Power Operation at Commercial Nuclear Power Plants in the United States", and NUMARC 91-06, "Guidelines for Industry Actions to Assess Shutdown Management." A number of variables (e.g., mid-loop, reduced level/flange level, head in place, or cavity flooded, RCS venting strategy, decay heat removal system design, vortexing pre-disposition, steam generator U-tube draining, etc.) can have a significant impact on heat removal capability challenging the Fuel Clad barrier.

Since in Refueling mode RVLIS is usually inoperable, this EAL threshold must, therefore, also depend on indirect methods of determining if core uncovery has occurred. The Reactor Vessel inventory loss may be detected by Containment High Range Area Radiation monitors or erratic Source Range Monitor indication. Valid high alarm (20 R/hr) on Contmt Hi Range Area Mon Hi Rad RE-30 or R-31 is indicative of core uncovery. There are a number of variables governing the projected dose rate from an actual core uncovery. Using the high alarm setpoint on the Containment High Radiation Monitors should provide an anticipatory indication of a reduction in the effectiveness of the shielding provided by the water cover. Post-TMI studies indicate that the installed nuclear instrumentation will operates erratically when the core is uncovered and source range monitors can be used as a tool for making such determinations. SRM count rate can be indicated in the Control Room by:

- Individual SUR meters on CC-1 which receive signals from Startup Channels N-31 and N-32.
- Local meter on the drawer on NIS Rack IV
- Audio Count Rate (NIS Rack IV)
- NR-45 video recorder located on CC-1

Escalation to a General Emergency is via EAL CG2.1 or EALs in Subcategory R.1, Offsite Rad Conditions.

DCPP Basis Reference(s):

1. Plant Drawing No. 57729
2. UFSAR, Section 5.6.1
3. UFSAR, Section 7.5
4. OP A-2:III, "Reactor Vessel - Draining to Half Loop/Half Loop Operations with Fuel in Vessel"
5. AP E-55, "Equipment Elevations for RCS Flood-Up and Drain-Down"
6. Instrument scaling calculation SC-I-87B, "Reactor Vessel Level Instrumentation System and Subcooled Margin Monitor"
7. AD8.DC54, "Containment Closure"
8. UFSAR, Table 7.5-3
9. OP L-2, "Hot Standby to Startup Mode"
10. STP I-4C, "Calibration of Audio Count Rate/Scaler Timer Channel"

11. 1-AR PK-11-19, "Containment Radiation"
12. 2-AR PK-11-19, "HI LVL RAD MON SYSTEM"
13. Plant Manual Volume 9B "Curves and Misc. Data Sheet; - I&C Radiation Monitoring and Allied Systems Data"

Category: C – Cold Shutdown / Refueling, System Malfunction

Sub-category: 2 – RCS Level

Initiating Condition: Loss of Reactor Vessel inventory affecting Fuel Clad integrity with Containment challenged and irradiated fuel in the Reactor Vessel

EAL:

CG2.1 General Emergency

Containment challenged as indicated by ANY of the following:

- Containment closure not established
- Containment hydrogen concentration > 4%
- Unplanned rise in containment pressure

AND

Core uncovery for > 30 min. as indicated by **EITHER**:

RVLIS full range < 56.6%

OR

Reactor Vessel level cannot be monitored with core uncovery indicated by ANY of the following:

- Containment radiation (RE-30 or RE-31) > 20 R/hr
- Erratic source range monitor indication
- Unexplained increase in ANY Table C-1 sump/tank level

Table C-1 Sumps/Tanks
• Containment Structure Sump 1-1 (2-1) • Containment Structure Sump 1-2 (2-2) • Reactor Cavity Sump • PRT • RCDT • CCW surge tank • Auxiliary Building sump • RWST

Mode Applicability:

5 - Cold Shutdown, 6 - Refueling

Basis:

Three conditions are associated with a challenge to Containment integrity:

- Containment Closure is the procedurally defined action taken to secure containment and its associated structures, systems and components as a functional barrier to fission product release under existing plant conditions. Containment Closure is defined by Administrative Procedure AD8.DC54, "Containment Closure." Containment closure is tracked if plant conditions change that could raise the risk of a fission product release as a result of a loss of decay heat removal.

- The 4% hydrogen concentration threshold is generally considered the lower limit for hydrogen deflagrations. To generate such levels of combustible gas, loss of the Fuel Clad and RCS barriers are likely to have occurred. Operation of the Containment Hydrogen Recombiner with Containment hydrogen concentrations above 4.0% could result in ignition of the hydrogen. Containment hydrogen can be monitored in the Control Room on ANR-82/ANR-83 and PAM1 following local equipment initialization.

- In the operating modes associated with this EAL, Containment pressure is expected to remain very low; thus, an elevated Containment pressure resulting from an unplanned rise above near-atmospheric pressure conditions may be indicative of a challenge to the Containment barrier. The Containment design pressure is 47 psig, while temporary penetration seals employed for Containment Closure during Refueling mode have a maximum pressure capability of 10 psig.

When Reactor Vessel water level drops below the RVLIS full range setpoint of 56.6%, core uncovery is about to occur. RVLIS is the only remotely indicating level monitoring system capable of indicating water level in the Reactor Vessel between the bottom of the RCS hot leg and the top of active fuel. In Refueling mode, however, RVLIS is usually inoperable.

The purpose of the RVLIS is to measure the level of the water or the relative void content of the coolant in the Reactor Vessel. The RVLIS setpoint corresponding to the top of the active fuel (TOAF) was obtained as follows:

- Per SC-I-87B, span = 494.9 in. and 0% = 79.6536 feet
- % span/in. = 100 / 494.9 = 0.20206%/in. and top of the core = 103 feet
- (103 - 79.6536) x 12 x 0.20206 = 56.6%

Other Reactor Vessel water level monitoring systems (e.g., RVRLIS, LI-400 standpipe, ultrasonic sensor, RVLIS upper range) are downscale-low when water level drops below the elevation of the bottom of the RCS hot leg penetration.

This EAL is based on concerns raised by Generic Letter 88-17, "Loss of Decay Heat Removal", SECY 91-283, "Evaluation of Shutdown and Low Power Risk Issues", NUREG-1449, "Shutdown and Low-Power Operation at Commercial Nuclear Power Plants in the United States", and NUMARC 91-06, "Guidelines for Industry Actions to Assess Shutdown Management." A number of variables (e.g., mid-loop, reduced level/flange level, head in place, or cavity flooded, RCS venting strategy, decay heat removal system design, vortexing pre-disposition, steam generator U-tube draining, etc.) can have a significant impact on heat removal capability challenging the Fuel Clad barrier. Analysis in the above references indicates that core damage may occur within an hour following continued core uncovery, therefore, the 30-minute interval was conservatively chosen.

If all means of level monitoring are not available, the Reactor Vessel inventory loss may be detected by the following indirect methods:

- Valid high alarm (20 R/hr) on Contmt Hi Range Area Mon Hi Rad RE-30 or R-31 is indicative of core uncovery. There are a number of variables governing the projected dose rate from an actual core uncovery. Using the high alarm setpoint on the Containment High Radiation Monitors should provide an anticipatory indication of a reduction in the effectiveness of the shielding provided by the water cover.

- Post-TMI studies indicate that the installed nuclear instrumentation will operate erratically when the core is uncovered and source range monitors can be used as a tool for making such determinations. SRM count rate can be indicated in the Control Room by:

  - Individual SUR meters on CC-1 which receive signals from Startup Channels N-31 and N-32.

  - Local meter on the drawer on NIS Rack IV

  - Audio Count Rate (NIS Rack IV)

  - NR-45 video recorder located on CC-1

- Sump or tank level changes may be indicative of a loss of RCS inventory. Plant design and procedures provide the capability to detect and assess primary system leakage. Containment. Sump level increases must be evaluated against other potential sources of leakage such as cooling water sources inside the Containment to ensure they are indicative of RCS leakage.

The General Emergency is declared on the occurrence of the loss or potential loss of the function of all three fission product barriers. Based on the above discussion, RCS barrier failure resulting in core uncovery for 30 minutes or more may cause Fuel Clad failure. With the Containment breached or challenged, the potential for unmonitored fission product release to the environment is high. This is consistent with the definition of a General Emergency.

DCPP Basis Reference(s):

1. Plant Drawing No. 57729
2. UFSAR, Section 5.6.1
3. UFSAR, Section 7.5
4. OP A-2:III, "Reactor Vessel - Draining to Half Loop/Half Loop Operations with Fuel in Vessel"
5. AP E-55, "Equipment Elevations for RCS Flood-Up and Drain-Down"
6. Instrument scaling calculation SC-I-87B, "Reactor Vessel Level Instrumentation System and Subcooled Margin Monitor"
7. AD8.DC54, "Containment Closure"
8. UFSAR, Table 7.5-3
9. OP L-2, "Hot Standby to Startup Mode"
10. STP I-4C, "Calibration of Audio Count Rate/Scaler Timer Channel"
11. 1-AR PK-11-19, "Containment Radiation"
12. 2-AR PK-11-19, "HI LVL RAD MON SYSTEM
13. Plant Manual Volume 9B, "Curves and Misc. Data Sheet; - I&C Radiation Monitoring and Allied Systems Data"
14. UFSAR, Section 6.25
15. Technical Specifications 3.6.3
16. FR-C.1, "Response to Inadequate Core Cooling"
17. OP H-9, "INSIDE CONT H2 RECOMB SYSTEM"
18. CA-3, "Hydrogen Flammability in Containment"
19. F-0, "Critical Safety Function Status Trees, Attachment 6"
20. UFSAR, Section 6.2.2.1
21. DCM T-16, pg 42

Category: C – Cold Shutdown / Refueling, System Malfunction

Sub-category: 3 – RCS Temperature

Initiating Condition: Unplanned loss of decay heat removal capability with irradiated fuel in the reactor vessel

EAL:

CU3.1	**Unusual Event**
An unplanned event results in RCS temperature > 200ºF	

Mode Applicability:

5 - Cold Shutdown, 6 - Refueling

Basis:

This EAL is an Unusual Event because it may be a precursor of more serious conditions and, as a result, is considered to be a potential degradation of the level of safety of the plant. In Cold Shutdown mode, the ability to remove decay heat relies primarily on forced cooling flow. Operation of the systems that provide this forced cooling may be jeopardized due to the unlikely loss of electrical power or RCS inventory. Since the RCS usually remains intact in the Cold Shutdown mode, a large inventory of water is available to keep the core covered. In Cold Shutdown, the decay heat available to raise RCS temperature during a loss of inventory or heat removal event may be significantly greater than in the Refueling mode. Entry into Cold Shutdown conditions may be attained within hours of operating at power. Entry into the Refueling mode procedurally may not occur for many hours after the reactor has been shut down. Thus, the heatup threat and the threat to damaging the Fuel Clad may be lower for events that occur in the Refueling mode with irradiated fuel in the Reactor Vessel. Note that the heatup threat could be lower for Cold Shutdown conditions if the entry into Cold Shutdown was following a refueling. In addition, the operators should be able to monitor RCS temperature and Reactor Vessel level so that escalation to the Alert under EAL CA2.1 or CA3.1 will occur if required.

During refueling operations, the level in the Reactor Vessel will normally be maintained above the vessel flange. Refueling operations that lower water level below the vessel flange are carefully planned and procedurally controlled. Loss of forced decay heat removal at reduced inventory may result in more rapid increases in RCS/Reactor Vessel temperatures depending on the time since shutdown. Escalation directly to the Alert under EAL CA3.1 is provided should an unplanned event result in RCS temperature exceeding the Technical Specification cold shutdown temperature limit with containment closure not established.

Several instruments are capable of providing indication of RCS temperature with respect to the Technical Specification cold shutdown temperature limit (200°F). These include:

- TR413 Loop 1 Wide Range Temp
- TR423 Loop 2 Wide Range Temp
- TR433 Loop 3 Wide Range Temp
- TR443 Loop 4 Wide Range Temp
- Loop Tavg meters 530-630°F
- WR T_{hot} recorder 0-700°F
- WR T_{cold} recorders 0-700°F

The SM/SEC/ED must remain attentive to events or conditions that lead to the conclusion that exceeding the EAL threshold is imminent. If, in the judgment of the SM/SEC/ED, an imminent situation is at hand, the classification should be made as if the threshold has been exceeded.

DCPP Basis Reference(s):

1. UFSAR, Section 7.2.1.1.3
2. OP L-1, "Plant Heatup From Cold Shutdown to Hot Standby", Attachments 11.5 & 11.6

Category: C – Cold Shutdown / Refueling, System Malfunction

Sub-category: 3 – RCS Temperature

Initiating Condition: Unplanned loss of decay heat removal capability with irradiated fuel in the reactor vessel

EAL:

CU3.2	**Unusual Event**
Loss of ALL RCS temperature and Reactor Vessel level indication for > 15 min.	

Mode Applicability:

5 - Cold Shutdown, 6 - Refueling

Basis:

This EAL is an Unusual Event because it may be a precursor of more serious conditions and, as a result, is considered to be a potential degradation of the level of safety of the plant. In Cold Shutdown mode, the ability to remove decay heat relies primarily on forced cooling flow. Operation of the systems that provide this forced cooling may be jeopardized due to the unlikely loss of electrical power or RCS inventory. Since the RCS usually remains intact in the Cold Shutdown mode, a large inventory of water is available to keep the core covered. In Cold Shutdown, the decay heat available to raise RCS temperature during a loss of inventory or heat removal event may be significantly greater than in the Refueling mode. Entry into Cold Shutdown conditions may be attained within hours of operating at power. Entry into the Refueling mode procedurally may not occur for many hours after the reactor has been shut down. Thus, the heatup threat and the threat to damaging the Fuel Clad may be lower for events that occur in the Refueling mode with irradiated fuel in the Reactor Vessel. Note that the heatup threat could be lower for Cold Shutdown conditions if the entry into Cold Shutdown was following a refueling. In addition, the operators should be able to monitor RCS temperature and Reactor Vessel level so that escalation to the Alert under EAL CA2.1 or CA3.1 will occur if required.

During refueling operations, the level in the Reactor Vessel will normally be maintained above the vessel flange. Refueling operations that lower water level below the vessel flange are carefully planned and procedurally controlled. Loss of forced decay heat removal at reduced inventory may result in more rapid increases in RCS/Reactor Vessel temperatures depending on the time since shutdown. Escalation directly to the Alert under EAL CA3.1 is provided should an unplanned event result in RCS temperature exceeding the Technical Specification cold shutdown temperature limit with containment closure not established.

Unlike the Cold Shutdown mode, normal means of RCS temperature indication and Reactor Vessel level indication may not be available in the Refueling mode. Redundant means of Reactor Vessel level indication are therefore procedurally installed to assure that the ability to monitor level will not be interrupted. However, if all level and temperature indication were to be lost in either the Cold Shutdown or Refueling modes, this EAL would result in declaration of an Unusual Event if either temperature or level indication cannot be restored within 15 minutes from the loss of both means of indication. Escalation to Alert under EAL CA3.1 would be based on an inventory loss or exceeding the temperature criterion (200°F).

Reactor Vessel water level is normally monitored using the following instruments:

- RVRLIS
- LI-400 Standpipe
- RVLIS

AP E-55, "Equipment Elevations for RCS Flood-Up and Drain-Down", provides a cross-reference of indicated water levels and key plant elevations.

Several instruments are capable of providing indication of RCS temperature with respect to the Technical Specification cold shutdown temperature limit (200°F). These include:

- TR413 Loop 1 Wide Range Temp
- TR423 Loop 2 Wide Range Temp
- TR433 Loop 3 Wide Range Temp
- TR443 Loop 4 Wide Range Temp
- Loop Tavg meters 530-630°F
- WR T_{hot} recorder 0-700°F
- WR T_{cold} recorders 0-700°F

The SM/SEC/ED must remain attentive to events or conditions that lead to the conclusion that exceeding the EAL threshold is imminent. If, in the judgment of the SM/SEC/ED, an imminent situation is at hand, the classification should be made as if the threshold has been exceeded.

DCPP Basis Reference(s):

1. UFSAR, Section 5.6.1
2. UFSAR, Section 7.5
3. OP A-2 I,I "Reactor Vessel - Draining the RCS to the Vessel Flange - With Fuel in Vessel"
4. E-0.3, "Natural Circulation Cooldown with Steam Void in Vessel with RVLIS"
5. Operator Information Manual Diagram A-1-52
6. OP AP SD 2, "Loss of RCS Inventory"
7. AP E-55, "Equipment Elevations for RCS Flood-Up and Drain-Down"
8. Operator Information Manual Diagram A-1-3 "Equipment Elevations for RCS Flood-Up and Drain-Down"

Category: C – Cold Shutdown / Refueling, System Malfunction

Sub-category: 3 – RCS Temperature

Initiating Condition: Inability to maintain plant in cold shutdown with irradiated fuel in the Reactor Vessel

EAL:

CA3.1 **Alert**

An unplanned event results in RCS temperature > 200°F for > Table C-3 duration

OR

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

Table C-3 RCS Reheat Duration Thresholds	
Containment and RCS Barrier Status	**Duration**
RCS intact (Containment closure N/A)	60 min.*
Containment closure established **AND** RCS not intact	20 min.*
Containment closure not established **AND** RCS not intact	0 min.

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

Mode Applicability:

5 - Cold Shutdown, 6 - Refueling

Basis:

This EAL is based on concerns raised by Generic Letter 88-17, "Loss of Decay Heat Removal." A number of phenomena such as pressurization, vortexing, steam generator U-tube draining, RCS level differences when operating at a mid-loop condition, decay heat removal system design and level instrumentation problems can lead to conditions in which decay heat removal is lost and core uncovery can occur. NRC analyses show that sequences that can cause core uncovery in 15 to 20 minutes and severe core damage within an hour after decay heat removal is lost.

Several instruments are capable of providing indication of RCS temperature with respect to the Technical Specification cold shutdown temperature limit (200°F). These include:

- TR413 Loop 1 Wide Range Temp
- TR423 Loop 2 Wide Range Temp
- TR433 Loop 3 Wide Range Temp
- TR443 Loop 4 Wide Range Temp
- Loop T_{avg} meters 530-630°F
- WR T_{hot} recorder 0-700°F
- WR T_{cold} recorders 0-700°F

The first threshold in Table C-3 addresses complete loss of functions required for core cooling for greater than 60 minutes during Refueling and Cold Shutdown modes when RCS integrity is established (irrespective of the status of containment closure). As in the second and third thresholds, RCS integrity should be considered to be in place when the RCS pressure boundary is in its normal condition for the Cold Shutdown mode of operation (e.g., no freeze seals or nozzle dams). The status of containment closure in this threshold is immaterial given that the RCS is providing a high-pressure barrier to fission product release to the environment. The 60-minute interval should allow sufficient time to restore cooling without a substantial degradation in plant safety.

Containment Closure is the procedurally defined action taken to secure containment and its associated structures, systems and components as a functional barrier to fission product release under existing plant conditions. Containment Closure is defined by Administrative Procedure AD8.DC54, "Containment Closure." Containment closure is tracked if plant conditions change that could raise the risk of a fission product release as a result of a loss of decay heat removal.

The second threshold in Table C-3 addresses the complete loss of functions required for core cooling for greater than 20 minutes during Refueling and Cold Shutdown modes when containment closure is established, but RCS integrity is not established. As in the third threshold, RCS integrity should be assumed to be in place when the RCS pressure boundary is in its normal condition for the Cold Shutdown mode of operation (e.g., no freeze seals or nozzle dams). The allowed 20-minute interval is included to allow operator action to restore the heat removal function, if possible. The allowed time frame is consistent with the guidance provided by Generic Letter 88-17, "Loss of Decay Heat Removal" (discussed later in this basis) and is believed to be conservative given that a low pressure Containment barrier to fission product release is established. The asterisk highlights the note at the bottom of the table. The note indicates that the second threshold is not applicable if actions are successful in restoring an RCS heat removal system to operation and RCS temperature is being reduced within the 20-minute interval.

The third threshold in Table C-3 addresses complete loss of functions required for core cooling during Refueling and Cold Shutdown modes when neither containment closure nor RCS integrity are established. RCS integrity is in place when the RCS pressure boundary is in its normal condition for the Cold Shutdown mode of operation (e.g., no freeze seals or nozzle dams). No delay time is allowed for the third condition because the evaporated reactor coolant that may be released into the Containment during this heatup condition could also be directly released to the environment.

Escalation to a Site Area Emergency would be under EAL CS2.1 or CS2.3 should boiling result in significant Reactor Vessel water level loss leading to core uncovery.

A loss of Technical Specification components alone is not intended to constitute an Alert. The same is true of a momentary unplanned excursion above 200°F when the heat removal function is available.

PI-403A, PI-405 and PI-405A with display on the PPC, SPDS, SCMM, and VB2 are capable of measuring pressure change to less than 10 psig.

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

DCPP Basis Reference(s):

- Technical Specifications Table 1.1-1, Modes Definition for Cold Shutdown
- AD8.DC54, "Containment Closure"
- UFSAR, Section 5

Category: C – Cold Shutdown / Refueling, System Malfunction

Sub-category: 4 – Communications

Initiating Condition: Unplanned loss of all onsite or offsite communications capabilities

EAL:

CU4.1 Unusual Event

Loss of ALL Table C-2 onsite (internal) communications capability affecting the ability to perform routine operations

OR

Loss of ALL Table C-2 offsite (external) communications capability

Table C-2 Communications Systems

System	Onsite (internal)	Offsite (external)
Unit 1, Unit 2 and TSC Radio Consoles	X	X
DCPP Telephone System (PBX)	X	X
Portable radio equipment (handie-talkies)	X	
Operations Radio System	X	X
Security Radio Systems	X	
Central Alarm Station (CAS) and Secondary Alarm Stations (SAS) Consoles	X	X
Fire Radio System	X	
Hot Shutdown Panel Radio Consoles	X	X
Public Address System	X	
NRC FTS		X
Mobile radios	X	
Satellite phones	X	X
Direct line (ATL) to the County and State Office of Emergency Services (OES)		X

Mode Applicability:

5 - Cold Shutdown, 6 - Refueling

Basis:

This EAL addresses loss of communications capability that either prevents the plant operations staff from performing routine tasks necessary for onsite plant operations or inhibits the ability to communicate problems externally to offsite authorities from the Control Room. The loss of offsite communications ability encompasses the loss of all means of communications with offsite authorities and is expected to be significantly more comprehensive than the condition addressed by 10 CFR 50.72.

The availability of one method of ordinary offsite communications is sufficient to inform state and local authorities of plant problems. This should include ENS, FAX transmissions and dedicated phone systems. This EAL is applicable only when extraordinary means are being utilized to make communications possible (e.g., relaying of information from radio transmissions, individuals being sent to offsite locations, etc.).

Onsite/offsite communications include one or more of the systems listed in Table C-2.

DCPP Basis Reference(s):

1. UFSAR, Section 9.5.2
2. AR PK15-23, "Communications"
3. STP I-29, "Emerg. Signals and Communications Sys. Functional Test"
4. OP K-9, "Instructions for Operation of Diablo Canyon Radio Systems"

Category: C – Cold Shutdown / Refueling, System Malfunction

Sub-category: 5 – RCS leakage

Initiating Condition: RCS leakage

EAL:

CU5.1 **Unusual Event**

Unable to restore or maintain **EITHER** of the following due to RCS leakage for > 15 min.:

Pressurizer level > 17%

OR

Above the low end of the target level control band (If pressurizer level was intentionally lowered < 17%)

Mode Applicability:

5 - Cold Shutdown

Basis:

The conditions of this EAL may be a precursor of more serious conditions and, as a result, is considered to be a potential degradation of the level of safety of the plant. When pressurizer level drops to 17%, letdown isolates and pressurizer heaters are deenergized. This condition is signaled by PK05-21 main annunciator "PZR LEVEL HI/LO."

The first threshold applies if pressurizer level is being maintained in the normal band while in cold shutdown. The second threshold applies if pressurizer level is being controlled below 17%, or if level is being maintained in a designated band in the reactor vessel. In either case, it is the inability to maintain level above the low end of the designated control band due to a loss of inventory resulting from a leak in the RCS that is the concern.

This cold condition EAL is the equivalent of the hot condition EAL SU6.1, in which RCS leakage is associated with Technical Specification limits. In cold shutdown conditions, these limits are not applicable; hence, the use of pressurizer low level as the parameter of concern in this EAL. Prolonged loss of RCS Inventory may result in escalation to the Alert level via either EAL CA2.1 (Loss of RCS/RPV Inventory with Irradiated Fuel in the RPV) or EAL CA3.1 (Inability to Maintain Plant in Cold Shutdown with Irradiated Fuel in the RPV)

The difference between this EAL and EAL CU2.1 deals with the RCS conditions that exist between Cold Shutdown and Refueling mode applicability. In Cold Shutdown, the RCS will normally be intact and RCS inventory and level monitoring means (i.e., pressurizer level instruments are normally available). In the Refueling Mode, the RCS is not intact and RPV level and inventory are monitored by different means.

DCPP Basis Reference(s):

1. AR PK05-21, "PZR LEVEL HI/LO"
2. OP AP SD 2, "Loss of RCS Inventory"
3. OP AP-1, "Excessive Reactor Coolant System Leakage"
4. EOP E-0, "Reactor Trip or Safety Injection"

Category: C – Cold Shutdown / Refueling, System Malfunction

Sub-category: 6 – Inadvertent Criticality

Initiating Condition: Inadvertent criticality

EAL:

CU6.1	**Unusual Event**
An unplanned sustained positive startup rate observed on nuclear instrumentation	

Mode Applicability:

5 - Cold Shutdown, 6 - Refueling

Basis:

Sustained is defined as prolonged, not intermittent or of transitory nature.

This EAL addresses criticality events that occur in Cold Shutdown or Refueling modes (NUREG1449, Shutdown and Low-Power Operation at Commercial Nuclear Power Plants in the United States) such as fuel misloading events and inadvertent dilution events. This EAL indicates a potential degradation of the level of safety of the plant, warranting an Unusual Event classification. This EAL excludes inadvertent criticalities that occur during planned reactivity changes associated with reactor startups (e.g., criticality earlier than estimated) which are addressed in the companion EAL SU7.1.

This condition can be identified using:

- NI-31D/32D and NI-35D/36D
- Individual SUR meters on CC-1 that receive signals from Source Range Channels N-31 and N-32 and Intermediate Range Channels NI-35 and NI-36
- Local meter on the drawer on NIS Rack IV
- Audio Count Rate (NIS Rack IV)
- NR-45 is a video recorder located on CC-1 that is used for monitoring and recording source range, Intermediate range and/or power range channel output

The term "sustained" is used in order to allow exclusion of expected short-term positive startup rates from planned fuel bundle or control rod movements during core alteration. These short-term positive startup rates are the result of the increase in neutron population due to subcritical multiplication.

Escalation would be by EALs in Subcategory H.6, Judgment.

DCPP Basis Reference(s):

1. Plant Drawing no. 521130
2. UFSAR, Table 7.5-3
3. OP L-2, "Hot Standby to Startup Mode"
4. STP I-4C, "Calibration of Audio Count Rate/Scaler Timer Channel"

Category H – Hazards

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

The events of this category pertain to the following subcategories:

1. Natural & Destructive Phenomena

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

2. Fire or Explosion

Fires can pose significant hazards to personnel and reactor safety. Appropriate for classification are fires within the site Protected Area or which may affect operability of vital equipment.

3. Toxic and Flammable Gas

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

4. Security

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

5. Control Room Evacuation

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

6. Judgment

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

Category: H – Hazards

Sub-category: 1 – Natural & Destructive Phenomena

Initiating Condition: Natural and destructive phenomena affecting the Protected Area

EAL:

HU1.1 Unusual Event

Seismic event identified by ANY TWO of the following:

- Earthquake felt in plant
- Seismic event confirmed by PK15-24 main annunciator "SEISMIC INSTR SYSTEM"
- U.S. Geological Survey (USGS)

Mode Applicability:

All

Basis:

The method of detection with respect to emergency classification relies on identification of a seismic event by of two of the following three sources:

- The agreement of the shift operators on-duty in the Control Room that the suspected ground motion is a "felt earthquake." Consensus of the Control Room operators with respect to ground motion helps avoid unnecessary classification if the seismic switches inadvertently trip or detect vibrations not related to an earthquake. As defined in the EPRI-sponsored "Guidelines for Nuclear Plant Response to an Earthquake", dated October 1989, a "felt earthquake" is:

  "An earthquake of sufficient intensity such that: (a) the inventory ground motion is felt at the nuclear plant site and recognized as an earthquake based on a consensus of Control Room operators on duty at the time, and (b) for plants with operable seismic instrumentation, the seismic switches of the plant are activated. For most plants with seismic instrumentation, the seismic switches are set at an acceleration of about 0.01g."

  - In the event of an earthquake measuring greater than or equal to 0.01 g, the SEISMIC INSTR SYSTEM annunciator PK15-24 will alert the control room and peak g-force indications will be displayed on the EFM. The control room operator reads the EFM display and can determine if the threshold for an Unusual Event or Alert emergency classification (EAL HA1.1) is reached. The seismic reactor trip system will automatically initiate a reactor trip if two (2) of three (3) unit sensors detect ≥ 0.35 g. in the same axial direction. This is not part of the EFM, but is generated by the reactor protection system.

  - The USGS maintains the National Earthquake Information Center (NEIC), which can be reached at the USGS website link "usgs.gov/regional/neic". The NEIC rapidly determines the location and size of all destructive earthquakes worldwide and immediately disseminates this information to concerned national and international agencies, scientists, and the general public. When the seismic monitoring system alarms or an earthquake is felt, operators take action as defined in CP M-4, "Earthquake," and the seismic instrumentation system engineer is notified to analyze the installed seismic instrumentation. The purpose of this analysis is to determine the exact magnitude of the event. CP M-4 instructs the operator to contact the USGS website.

More than one of the sources listed in this EAL must indicate a seismic event to avoid unnecessary classification if the seismic switches inadvertently actuate or the felt ground motion is the result of a non-seismic source. Each source can be accessed in a timely manner from the Control Room and serves to validate the results of the others.

Damage to some portions of the site may occur as a result of the felt earthquake but it should not affect the ability of safety functions to operate. This event escalates to an Alert under EAL HA1.1 if the earthquake exceeds Operating Basis Earthquake (OBE) levels.

DCPP Basis Reference(s):

1. DCM T-6, "Seismic Analysis of Structures"
2. AR PK-15-24, "SEISMIC INSTR SYSTEM"
3. CP M-4, "Earthquake"

Category: H – Hazards

Sub-category: 1 – Natural & Destructive Phenomena

Initiating Condition: Natural and destructive phenomena affecting the Protected Area

EAL:

HU1.2	**Unusual Event**
Report by plant personnel of tornado or high winds > 80 mph (36.36 m/sec) striking within Protected Area boundary	

Mode Applicability:

All

Basis:

This EAL is based on the assumption that a tornado striking (touching down) or design force winds (> 80 mph) within the Protected Area may have potentially damaged plant structures containing functions or systems required for safe shutdown of the plant. If such damage is confirmed visually or by other in-plant indications, the event may be escalated to an Alert under EAL HA1.2.

Wind speed recorders measure wind speeds up to 100 mph. All structures are designed, however, to withstand a basic wind speed of 80 mph (36.36 m/sec).

The Protected Area is within the security isolation zone and is given in Dwg. 471124 "Plot Plan".

DCPP Basis Reference(s):

1. UFSAR, section 3.3.1
2. Plant Dwg. 471124, "Plot Plan"

Category: H – Hazards

Sub-category: 1 – Natural & Destructive Phenomena

Initiating Condition: Natural and destructive phenomena affecting the Protected Area (turbine)

EAL:

HU1.3	**Unusual Event**
Report of turbine failure resulting in casing penetration or damage to turbine or generator seals	

Mode Applicability:

All

Basis:

This EAL is intended to address main turbine rotating component failures of sufficient magnitude to cause observable damage to the turbine casing or to the seals of the turbine generator. Of major concern is the potential for significant leakage of combustible fluids (lubricating oils) and gases (hydrogen cooling) to the plant environs. It is not the intent of this EAL to classify minor operational leakage. Actual fires and flammable gas build up are appropriately classified through other EALs. This EAL is consistent with the definition of an Unusual Event while maintaining the anticipatory nature desired and recognizing the risk to non-safety related equipment. Escalation of the emergency classification is based on potential damage done by missiles generated by the failure or by the radiological releases in conjunction with a steam generator tube rupture. These latter events would be classified by EALs in Category R or Category F.

DCPP Basis Reference(s):

1. OP AP-29, "Main Turbine Malfunction"
2. AR PK-14-16, "Turbine/Generator Trouble"

Category: H – Hazards

Sub-category: 1 – Natural & Destructive Phenomena

Initiating Condition: Natural and destructive phenomena affecting the Protected Area

EAL:

HU1.4	**Unusual Event**
Uncontrolled flooding in ANY Table H-1 area that has the potential to affect safety related equipment needed for the current operating mode	

Table H-1 Vital Areas
• Containment • Auxiliary Building • Fuel Handling Building • Turbine Building • Intake Structure • RWST • CST

Mode Applicability:

All

Basis:

This EAL addresses flooding caused by internal events (e.g., component failures, circulating water, component cooling or service water line ruptures, equipment misalignment, fire suppression system actuation, outage activity mishaps, etc.) that results in the potential to affect safety related equipment. Uncontrolled internal flooding that degrades safety-related equipment or creates a safety hazard precluding access necessary for the safe operation or monitoring of safety equipment warrants escalation to an Alert emergency classification under EAL HA1.5. The internal flooding areas are important drainage areas and typically contain systems that are:

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

DCPP Basis Reference(s):

1. DCM T-12, "Pipe Break (HELB/MELB), Flooding, and Missiles," Section 5.0, "Internal Flooding Design Criteria"

Category: H – Hazards

Sub-category: 1 – Natural & Destructive Phenomena

Initiating Condition: Natural and destructive phenomena affecting the Protected Area

EAL:

HU1.5	**Unusual Event**
Hurricane warning or tsunami (actual or warning) affecting the Protected Area	

Mode Applicability:

All

Basis:

This EAL covers high and low ocean water level conditions that could be a precursor of more serious events and which may threaten operability of plant cooling systems.

A Tsunami that exceeds design basis may interfere with the normal seawater supply to the circulating water and auxiliary saltwater (ASW) systems. A Tsunami arrives in a series of low water (drawdown) and high water (runup) periods. The time between runup and drawdown may last as long as 10 minutes and may be spaced 30 to 40 minutes apart. The concern during drawdown is that the circulators and ASW pumps could lose suction (cavitate) for a short duration (less than 5 minutes). The concern during runup is that water could crest the intake structure main deck and flood the circulators. Intake structure main deck is at elevation +20 feet MLLW (+17.4 feet MSL). The predicted maximum Tsunami water elevation under a highly unlikely combination of events (high tide coincident with severe storm) would produce a wave crest elevation of +34.6 feet MLLW. (+32.0 feet MSL)[1].

Any of the following offsite authorities may issue a Tsunami or Hurricane Warning:

- California OES, Sacramento, notifies Power Control and DCPP.
- U.S. Coast Guard
- National Weather Service (NWS)
- National Oceanic and Atmospheric Administration (NOAA)

DCPP Basis Reference(s):

1. CP M-5, "Tsunami Warning"

1 Mean Lower Low Water (MLLW) - standard UFSAR reference level that corresponds to 2.6 ft below mean sea level (MSL)

Category: H – Hazards

Sub-category: Natural/Destructive Events

Initiating Condition: Natural and destructive phenomena affecting the plant Vital Area

EAL:

HA1.1	**Alert**
EFM "Alert" alarm or CP M-4, "Earthquake" indicates Operating Basis Earthquake (> 0.2 g) exceeded	

Mode Applicability:

All

Basis:

This EAL addresses 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 this EAL to assess the actual extent of the damage.

In the event of an earthquake measuring greater than or equal to 0.01 g, the SEISMIC INSTR SYSTEM annunciator PK15-24 will alert the control room and peak g-force indications will be displayed on the EFM. The primary means for timely determination of the magnitude of an earthquake, and subsequently assessing emergency action levels, is using the EFM located in the control room. If the EFM indicator shows that the "Alert" alarm (> 0.2 g) has been exceeded an Alert should be declared. If the EFM is not operable, the earthquake magnitude is determined by alternative methods in accordance with CP M-4, "Earthquake." If it is determined that any peak ground acceleration has exceeded 0.2 g, an Alert should be declared.

The seismic reactor trip system will automatically initiate a reactor trip if two (2) of three (3) unit sensors detect ≥ 0.35 g. in the same axial direction. This is not part of the EFM, but is generated by the reactor protection system.

When the seismic monitoring system alarms, Operations takes action as defined in CP M-4, "Earthquake," and the seismic instrumentation system engineer notified to analyze the installed seismic instrumentation. The purpose of this analysis is to determine the exact magnitude of the event.

EPRI-sponsored "Guidelines for Nuclear Plant Response to an Earthquake", dated October 1989, provides information on seismic event categories.

Escalation to a higher classification will be based on EALs in Category S.

DCPP Basis Reference(s):

1. DCM T-6, "Seismic Analysis of Structures"
2. AR PK-15-24, "SEISMIC INSTR SYSTEM"
3. CP M-4, "Earthquake"

Category: H – Hazards

Sub-category: 1 – Natural & Destructive Phenomena

Initiating Condition: Natural and destructive phenomena affecting the plant Vital Area

EAL:

HA1.2	**Alert**
Tornado or high winds > 80 mph (36.36 m/sec) within Protected Area boundary and resulting in visible damage to ANY Table H-1 plant structures / equipment or Control Room indication of degraded performance of those systems	

Table H-1 Vital Areas
• Containment
• Auxiliary Building
• Fuel Handling Building
• Turbine Building
• Intake Structure
• RWST
• CST

Mode Applicability:

All

Basis:

This EAL escalates from EAL HU1.2 in that the tornado or high winds have resulted in visible damage to plant structures or areas containing equipment necessary for a safe shutdown, or has caused damage to the safety systems in those structures evidenced by Control Room indications of degraded system response or performance. The occurrence of visible damage and/or degraded system response is intended to discriminate against lesser events. The initial "report" should not be interpreted as mandating a lengthy damage assessment prior to classification. No attempt is made in this EAL to assess the actual magnitude of the damage. The significance here is not that a particular system or structure was damaged, but rather, that the event was of sufficient magnitude to cause this degradation. Escalation to a higher classification will be based on EALs in Category S. The declaration of an Alert and the activation of the TSC provide the SM/SEC/ED with the resources needed to perform detailed damage assessments.

This EAL is based on the design basis wind speed of 80 mph (36.36 m/sec). Sustained wind loads above this magnitude can cause damage to safety functions.

DCPP Basis Reference(s):

1. UFSAR, section 3.3.1
2. Plant Dwg. 471124, "Plot Plan"
3. CP M-10, “Fire Protection of Safe Shutdown Equipment”
4. DCM T-6, “Seismic Analysis of Structures”

Category: H – Hazards

Sub-category: 1 – Natural & Destructive Phenomena

Initiating Condition: Natural and destructive phenomena affecting the plant Vital area

EAL:

HA1.3 **Alert**

Vehicle crash within Protected Area boundary and resulting in visible damage to ANY Table H-1 plant structures or equipment or control room indication of degraded performance of those systems (Note 2)

Note 2: If vehicle crash is a hostile action, see Subcategory H.4 EALs for possible classification

Table H-1 Vital Areas
• Containment • Auxiliary Building • Fuel Handling Building • Turbine Building • Intake Structure • RWST • CST

Mode Applicability:

All

Basis:

In EAL HA1.3, a vehicle crash has resulted in visible damage to plant structures or areas containing equipment necessary for a safe shutdown, or has caused damage to the safety systems in those structures evidenced by Control Room indications of degraded system response or performance. The occurrence of visible damage and/or degraded system response is intended to discriminate against lesser events. The initial "report" should not be interpreted as mandating a lengthy damage assessment prior to classification. No attempt is made in this EAL to assess the actual magnitude of the damage. The significance here is not that a particular system or structure was damaged, but rather, that the event was of sufficient magnitude to cause this degradation. Escalation to a higher classification will be based on EALs in Category S. The declaration of an Alert and the activation of the TSC provide the SM/SEC/ED with the resources needed to perform detailed damage assessments.

Vital areas house equipment the operation of which may be needed to ensure the reactor reaches and is maintained in its lowest energy state. Personnel access to vital areas may be an important factor in monitoring and controlling equipment operability. This EAL addresses vehicle crashes that preclude personnel access to vital areas or may have resulted in the area being subjected to forces beyond design limits. It is therefore assumed that equipment operability has been challenged or damage has occurred to plant systems necessary for safe shutdown of the plant. The initial report should not be interpreted as mandating a lengthy damage assessment prior to classification. No attempt is made in this EAL to assess the actual magnitude of the damage.

This EAL addresses events such as plane, helicopter, barge, car or truck crashes, or impact of projectiles into a vital area.

Note 2 is a reminder that vehicle crashes may originate from hostile actions and, if they occur within the Protected Area, may require escalation of the classification under the Security EALs in Subcategory H.4.

DCPP Basis Reference(s):

1. DCM T-6, “Seismic Analysis of Structures”
2. CPM-10, “Fire Protection of Safe Shutdown Equipment”
3. Plant Dwg. 471124, "Plot Plan"

Category: H – Hazards

Sub-category: 1 – Natural & Destructive Phenomena

Initiating Condition: Natural and destructive phenomena affecting the plant Vital Area

EAL:

HA1.4 Alert

Turbine failure-generated missiles result in ANY visible damage to or penetration of ANY Table H-1 area

Table H-1 Vital Areas
• Containment • Auxiliary Building • Fuel Handling Building • Turbine Building • Intake Structure • RWST • CST

Mode Applicability:

All

Basis:

This EAL escalates from EAL HU1.3 in that the turbine-generated missiles have resulted in visible damage to plant structures or areas containing equipment necessary for a safe shutdown, or has caused damage to the safety systems in those structures evidenced by Control Room indications of degraded system response or performance. The occurrence of visible damage and/or degraded system response is intended to discriminate against lesser events. The initial "report" should not be interpreted as mandating a lengthy damage assessment prior to classification. No attempt is made in this EAL to assess the actual magnitude of the damage. The significance here is not that a particular system or structure was damaged, but rather, that the event was of sufficient magnitude to cause this degradation. Escalation to a higher classification will be based on EALs in Category S. The declaration of an Alert and the activation of the TSC provide the SM/SEC/ED with the resources needed to perform detailed damage assessments.

The list of areas includes all areas containing 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.

DCPP Basis Reference(s):

1. DCM T-6, "Seismic Analysis of Structures"
2. CPM-10, "Fire Protection of Safe Shutdown Equipment"
3. Plant Dwg. 471124, "Plot Plan"

Category: H – Hazards

Sub-category: 1 – Natural & Destructive Phenomena

Initiating Condition: Natural and destructive phenomena affecting the plant Vital Area

EAL:

HA1.5 **Alert**

Uncontrolled flooding in ANY Table H-1 area that results in degraded safety system performance as indicated in the Control Room or that creates industrial safety hazards (e.g., electric shock) that precludes access necessary to operate or monitor safety equipment

Table H-1 Vital Areas
• Containment • Auxiliary Building • Fuel Handling Building • Turbine Building • Intake Structure • RWST • CST

Mode Applicability:

All

Basis:

This EAL escalates from EAL HU1.4 in that the flooding has resulted in visible damage to plant structures or areas containing equipment necessary for a safe shutdown, or has caused damage to the safety systems in those structures evidenced by Control Room indications of degraded system response or performance. The occurrence of visible damage and/or degraded system response is intended to discriminate against lesser events. The initial "report" should not be interpreted as mandating a lengthy damage assessment prior to classification. No attempt is made in this EAL to assess the actual magnitude of the damage. The significance here is not that a particular system or structure was damaged, but rather, that the event was of sufficient magnitude to cause this degradation. Escalation to a higher classification will be based on EALs in Category S. The declaration of an Alert and the activation of the TSC provide the SM/SEC/ED with the resources needed to perform detailed damage assessments.

This EAL addresses flooding caused by internal events (e.g., component failures, circulating water, component cooling or service water line ruptures, equipment misalignment, fire suppression system actuation, outage activity mishaps, etc.) that results in degraded safety system performance. The internal flooding areas are important drainage areas and typically contain systems that are:

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

Uncontrolled internal flooding that has degraded safety-related equipment or created a safety hazard precluding access necessary for the safe operation or monitoring of safety equipment warrants declaration of an Alert.

DCPP Basis Reference(s):

1. DCM T-12, "Pipe Break (HELB/MELB), Flooding, and Missiles," Section 5.0, "Internal Flooding Design Criteria"

Category: H – Hazards

Sub-category: 2 – Fire or Explosion

Initiating Condition: Fire within Protected Area boundary not extinguished within 15 minutes of detection

EAL:

HU2.1	**Unusual Event**
Fire in buildings or areas contiguous to ANY Table H-1 area not extinguished within 15 min. of Control Room notification or validation of a Control Room alarm (Note 3)	
Note 3: If the fire or explosion is a hostile action, see Subcategory H.4 EALs for possible classification	

Table H-1 Vital Areas
• Containment • Auxiliary Building • Fuel Handling Building • Turbine Building • Intake Structure • RWST • CST

Mode Applicability:

All

Basis:

The purpose of this EAL is to address the magnitude and extent of fires that may be potentially significant precursors to damage to safety systems. As used here, a confirmed fire is a fire that has been identified through visual observation and report by plant personnel or sensor alarm indication. The 15-minute period begins when a credible report is received that a fire is occurring or a valid fire detection system alarm is received. Validation of a fire detection system alarm includes actions that can be taken within the Control Room or other nearby location to ensure that the alarm is not spurious. A validated alarm is assumed to be an indication of a fire unless personnel dispatched to the scene disprove the alarm within the 15-minute period. In other words, a personnel report from the scene may be used to disprove a validated alarm if the report is received within 15 minutes of the alarm. The report, however, shall not be required to validate the alarm.

The intent of the 15-minute period is to size the fire and discriminate against small fires that are readily extinguished (e.g., smoldering waste paper basket). The area lists are limited and apply to buildings and areas contiguous (in actual contact with or immediately adjacent) to vital areas or other significant buildings or areas. The intent of this EAL is not to include buildings or areas that are not contiguous to vital areas. This excludes fires within free-standing support buildings, waste paper basket fires and other small fires of no safety consequence.

EAL HA2.1 provides escalation to the Alert classification.

Note 3 is a reminder that the fire may originate from hostile actions and, if it occurs within the OCA, may require escalation of the classification under the Security EALs in Subcategory H.4.

DCPP Basis Reference(s):

1. DCM T-6, “Seismic Analysis of Structures”
2. CPM-10, “Fire Protection of Safe Shutdown Equipment”
3. Plant Dwg. 471124, "Plot Plan"

Category: H – Hazards

Sub-category: 2 – Fire or Explosion

Initiating Condition: Natural and destructive phenomena affecting the Protected Area

EAL:

HU2.2	**Unusual Event**
Report by plant personnel of an unanticipated explosion within Protected Area boundary resulting in visible damage to permanent structure or equipment (Note 3)	
Note 3: If the fire or explosion is a hostile action, see Subcategory H.4 EALs for possible classification	

Mode Applicability:

All

Basis:

The Protected Area is within the security isolation zone and is given in Dwg. 471124, "Plot Plan".

The Unusual Event is categorized on the basis of the occurrence of an event of sufficient magnitude to be of concern to plant operators. Areas identified in the EALs define the location of the event based on the potential for damage to equipment contained therein. Escalation of the event to an Alert occurs when the magnitude of the event is sufficient to result in damage to equipment contained in the specified location.

For this EAL, only those unanticipated explosions within the Protected Area should be considered. As used here, an explosion is a rapid, violent, unconfined combustion or a catastrophic failure of pressurized equipment that potentially imparts significant energy to nearby structures and materials. No attempt is made in this EAL to assess the actual magnitude of the damage. The occurrence of the explosion with reports of evidence of damage (e.g., deformation, scorching, etc.) is sufficient for declaration. The SM/SEC/ED also needs to consider any security aspects of the explosion.

A steam line break or steam explosion that damages surrounding permanent structures or equipment would be classified under this EAL. This does not mean the emergency is classified simply because the steam line break occurred. The method of damage is not as important as the degradation of plant structures or equipment. The need to classify the steam line break itself is considered in fission product barrier degradation monitoring (EAL Category F).

Note 3 is a reminder that the explosion may originate from hostile actions and, if it occurs within the OCA, may require escalation of the classification under the Security EALs in Subcategory H.4.

DCPP Basis Reference(s):

1. Plant Dwg. 471124, "Plot Plan"

Category: H – Hazards

Sub-category: 2 – Fire or Explosion

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

EAL:

HA2.1 **Alert**

Fire or explosion resulting in EITHER:

- Visible damage to any Table H-1 plant structures containing safety systems or components
- Control Room indication of degraded performance of systems required to establish or maintain safe shutdown (Note 3)

Note 3: If the fire or explosion is a hostile action, see Subcategory H.4 EALs for possible classification

Table H-1 Vital Areas
• Containment • Auxiliary Building • Fuel Handling Building • Turbine Building • Intake Structure • RWST • CST

Mode Applicability:

All

Basis:

The listed areas contain functions and systems required for the safe shutdown of the plant. The DCPP safe shutdown analyses were consulted for equipment and plant areas required for the applicable mode.

The only explosions that should be considered are those of sufficient force to: damage permanent structures or equipment required for safe operation, or result in degraded performance of safety systems within the identified plant areas. An explosion is a rapid, violent, unconfined combustion or a catastrophic failure of pressurized equipment that potentially imparts significant energy to nearby structures and materials. No attempt is made to assess the actual magnitude of the damage. The wording of this EAL does not imply that an assessment of safety system performance should be performed; rather that affected safety system parameter indications show degraded performance as a result of the event. The declaration of an Alert and the activation of the TSC provide the SM/SEC/ED with the resources needed to perform damage assessments. The SM/SEC/ED also needs to consider the security aspects of the explosions.

This situation is not the same as removing equipment for maintenance that is covered by Technical Specifications. Removal of equipment for maintenance is a planned activity controlled in accordance with procedures and, as such, does not constitute a substantial degradation in the level of safety of the plant. A fire/explosion is an unplanned activity and, as such, does constitute a substantial degradation in the level of safety of the plant. In this situation, an Alert classification is warranted.

Escalation to a higher emergency class, if appropriate, will be based on EALs in Category S, Category F, Category R, or Subcategory H.6, Judgment.

Note 3 is a reminder that the explosion may originate from hostile actions and, if it occurs within the OCA, may require escalation of the classification under the Security EALs in Subcategory H.4.

A steam line break or steam explosion that damages permanent structures or equipment would be classified under this EAL. The method of damage is not as important as the degradation of plant structures or equipment. The need to classify the steam line break itself is considered in fission product barrier degradation monitoring (EAL Category F).

DCPP Basis Reference(s):

1. DCM T-6, "Seismic Analysis of Structures"
2. CPM-10, "Fire Protection of Safe Shutdown Equipment"
3. Plant Dwg. 471124, "Plot Plan"

Category: H – Hazards

Sub-category: 3 – Toxic and Flammable Gas

Initiating Condition: Release of toxic, corrosive, asphyxiant or flammable gases deemed detrimental to normal operation of the plant

EAL:

HU3.1	**Unusual Event**
Report or detection of toxic, corrosive, asphyxiant or flammable gases that have entered or could enter the Owner Controlled Area in amounts that can adversely affect normal plant operations	

Mode Applicability:

All

Basis:

This EAL is based on the existence of uncontrolled releases of toxic, corrosive, asphyxiant or flammable gases affecting normal plant operations or the health of plant personnel. The release may have originated within the Owner Controlled Area, or it may have originated offsite and subsequently drifted inside the Owner Controlled Area. Offsite events (e.g., tanker truck accident releasing toxic gases, etc.) resulting in the plant being within the evacuation area should also be considered in this EAL because of the adverse affect on normal plant operations.

It is intended that releases of toxic, corrosive, asphyxiant or flammable gases are of sufficient quantity and the release point of such gases is such that normal plant operations would be affected. This would preclude small or incidental releases, or releases that do not impact structures needed for plant operation. The EAL is not intended to require significant assessment or quantification. The EAL assumes an uncontrolled process that has the potential to affect plant operations or personnel safety. The fact that SCBA may be worn does not eliminate the need to declare the event.

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

Should the release affect plant vital areas, escalation to an Alert would be based on EAL HA3.1 or HA3.2. Should an explosion or fire occur due to flammable gas within an affected plant area, an Alert may be appropriate based on EAL HA2.1.

DCPP Basis Reference(s):

None

Category: H – Hazards

Sub-category: 3 – Toxic and Flammable Gas

Initiating Condition: Release of toxic, corrosive, asphyxiant or flammable gases deemed detrimental to normal operation of the plant

EAL:

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

Mode Applicability:

All

Basis:

This EAL is based on the existence of uncontrolled releases of toxic, corrosive, asphyxiant or flammable gas affecting normal plant operations or the health of plant personnel. The release originated offsite and local, county or state officials have reported the need for evacuation or sheltering of site personnel. Offsite events (e.g., tanker truck accident releasing toxic gases, etc.) are considered in this EAL because they may adversely affect normal plant operations.

The EAL is not intended to require significant assessment or quantification. The EAL assumes an uncontrolled process that has the potential to affect plant operations or personnel safety. The fact that SCBA may be worn does not eliminate the need to declare the event.

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

Should the release affect plant vital areas, escalation to an Alert would be based on EAL HA3.1 or HA3.2. Should an explosion or fire occur due to flammable gas within an affected plant area, an Alert may be appropriate based on EAL HA2.1.

DCPP Basis Reference(s):

None

Category: H – Hazards

Sub-category: 3 – Toxic and Flammable Gas

Initiating Condition: Release of toxic, corrosive, asphyxiant or flammable gases within or contiguous to a Vital Area which jeopardizes operation of systems required to establish or maintain safe shutdown

EAL:

HA3.1 **Alert**

Report or detection of toxic, corrosive or asphyxiant gases within or contiguous to ANY Table H-1 area in concentrations that may result in an atmosphere Immediately Dangerous to Life and Health (IDLH)

Table H-1 Vital Areas
• Containment • Auxiliary Building • Fuel Handling Building • Turbine Building • Intake Structure • RWST • CST

Mode Applicability:

All

Basis:

This EAL is based on toxic, corrosive or asphyxiant gases that have entered a plant structure in concentrations that are unsafe for plant personnel and, therefore, preclude access to equipment necessary for the safe operation of the plant. This EAL applies to buildings and areas contiguous to vital areas or other significant buildings or areas. Vital areas contain systems that are operated to establish or maintain safe shutdown. The intent of this EAL is not to include buildings (i.e., warehouses) or other areas that are not contiguous or immediately adjacent to vital areas. It is appropriate that more frequent monitoring be done to ascertain whether consequential damage has occurred.

The EAL is met if measurement of toxic, corrosive or asphyxiant gas concentration results in an atmosphere that is immediately dangerous to life and health (IDLH) within a vital area or any area or building contiguous to vital area. Exposure to an IDLH atmosphere will result in immediate harm to unprotected personnel, and would preclude access to any such affected areas. The fact that SCBA may be worn does not eliminate the need to declare the event.

Escalation to a higher emergency class, if appropriate, will be based on EALs in Category S, Category F, Category R, or Subcategory H.6, Judgment.

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

DCPP Basis Reference(s):

1. DCM T-6, “Seismic Analysis of Structures”
2. CPM-10, “Fire Protection of Safe Shutdown Equipment”
3. Plant Dwg. 471124, "Plot Plan"

Category: H – Hazards

Sub-category: 3 – Toxic and Flammable Gas

Initiating Condition: Release of toxic, corrosive, asphyxiant or flammable gases within or contiguous to a Vital Area which jeopardizes operation of systems required to establish or maintain safe shutdown

EAL:

HA3.2 Alert

Report or detection of gases in concentration > the Lower Flammability Limit within or contiguous to ANY Table H-1 area

Table H-1 Vital Areas
• Containment • Auxiliary Building • Fuel Handling Building • Turbine Building • Intake Structure • RWST • CST

Mode Applicability:

All

Basis:

This EAL is based on toxic, corrosive, asphyxiant or flammable gases that have entered a plant structure in concentrations that are unsafe for plant personnel and, therefore, preclude access to equipment necessary for the safe operation of the plant. This EAL applies to buildings and areas contiguous to Vital Areas or other significant buildings or areas. Vital Areas contain systems that are operated to establish or maintain safe shutdown. The intent of this EAL is not to include buildings (i.e., warehouses) or other areas that are not contiguous or immediately adjacent to vital areas. It is appropriate that more frequent monitoring be done to ascertain whether consequential damage has occurred.

The EAL is met when the flammable gas concentration in a vital area or any building or area contiguous to a Vital Area exceed the lower flammability limit. Flammable gases, such as hydrogen and acetylene, are routinely used to maintain plant systems (hydrogen) or to repair equipment/components (acetylene - used in welding). This condition addresses concentrations at which gases can ignite/support combustion. An uncontrolled release of flammable gases within a facility structure has the potential to affect safe operation of the plant by limiting either operator or equipment operations due to the potential for ignition and resulting equipment damage/personnel injury. Once it has been determined that an uncontrolled release is occurring, then sampling must be done to determine if the concentration of the released gas is within this range. . The fact that SCBA may be worn does not eliminate the need to declare the event.

Escalation to a higher emergency class, if appropriate, will be based on EALs in Category S, Category F, Category R, or Subcategory H.6, Judgment.

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

DCPP Basis Reference(s):

1. DCM T-6, “Seismic Analysis of Structures”
2. CPM-10, “Fire Protection of Safe Shutdown Equipment”
3. Plant Dwg. 471124, "Plot Plan"

Category: H – Hazards

Sub-category: 4 – Security

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

EAL:

HU4.1 **Unusual Event**

A security condition that does not involve a hostile action as reported by the Security Watch Commander

OR

A credible site-specific security threat notification

OR

A valid notification from NRC providing information of an aircraft threat

Mode Applicability:

All

Basis:

A security event may constitute a risk to site personnel and degrade the level of safety of the plant. It is of utmost importance to ensure available resources are mustered in a timely manner and applied as effectively as possible to combat the threat while at the same time assuring the conduct of safe plant operations. This requires close coordination between Security and Operations.

Reference is made to the Security Watch Commander because these individuals are the designated personnel on-site qualified and trained to confirm that a security event is occurring or has occurred. Training on security event classification confirmation is closely controlled due to the strict secrecy controls placed on the DCPP Security and Safeguards Contingency Plan. The DCPP Security and Safeguards Contingency Plan is based on guidance provided by NEI 03-12.

The first condition of this EAL is based on the DCPP Security and Safeguards Contingency Plan. Security events which do not represent a potential degradation in the level of safety of the plant are reported under 10 CFR 73.71 or in some cases under 10 CFR 50.72. Security events assessed as hostile actions are classifiable under EAL HA4.1, HS4.1 and HG4.1.

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

The intent of the third condition of this EAL is to ensure that notifications for the aircraft threat are made in a timely manner and that Offsite Response Organizations and plant personnel are at a state of heightened awareness regarding the credible threat. Only the plant to which the specific threat is made need declare the Unusual Event. This EAL is met when a plant receives information regarding an aircraft threat from NRC. Should the threat involve an airliner (airliner is meant to be a large aircraft with the potential for causing significant damage to the plant) then escalation to Alert via HA4.1 would be appropriate if the airliner is less than 30 minutes away from the plant. The NRC Headquarters Operations Officer (HOO) will communicate to the licensee if the threat involves an airliner. The status and size of the plane may be provided by NORAD through the NRC. It is not the intent of this EAL to replace existing non-hostile related EALs involving aircraft.

The determination of "credible" is made through use of information found in the DCPP Security and Safeguards Contingency Plan.

A higher initial classification could be made based upon the nature and timing of the security threat and potential consequences. Consideration should be given to upgrading the emergency response status and emergency classification in accordance with the DCPP Security and Safeguards Contingency Plan.

DCPP Basis Reference(s):

1. NRC Safeguards Advisory dated October 6. 2001
2. DCPP Security and Safeguards Contingency Plan
3. Letter from Mr. B. A. Boger (NRC) to Ms. Lynette Hendricks (NEI) dated February 4, 2002
4. NRC Bulletin 2005-02 "Emergency Preparedness and Response Actions for Security-Based Events"
5. NEI White Paper "Enhancements to Emergency Preparedness Programs for Hostile Action," May 2005 (Revised November 1, 2005)

Category: H – Hazards

Sub-category: 4 – Security

Initiating Condition: Hostile action within the Owner Controlled Area or airborne attack threat

EAL:

HA4.1	**Alert**

A hostile action is occurring or has occurred within the Owner Controlled Area as reported by the Security Watch Commander

OR

A valid notification from NRC of an airliner attack threat within 30 min. of the site

Mode Applicability:

All

Basis:

A security event may constitute a risk to site personnel and degrade the level of safety of the plant. It is of utmost importance to ensure available resources are mustered in a timely manner and applied as effectively as possible to combat the threat while at the same time assuring the conduct of safe plant operations. This requires close coordination between Security and Operations.

The first condition addresses the potential for a very rapid progression of events due to a hostile action. This condition is not intended to address incidents that are accidental or acts of civil disobedience, such as hunters or physical disputes between employees within the OCA or PA, which are addressed by other EALs.

The second condition ensures that notifications for the airliner attack threat are made in a timely manner and that Offsite Response Organizations and plant personnel are at a state of heightened awareness regarding the credible threat. Only the plant to which the specific threat is made need declare the Alert. This condition is met when a plant receives information regarding an airliner attack threat from NRC and the airliner is less than 30 minutes away from the plant. This condition addresses the immediacy of an expected threat arrival or impact on the site within a relatively short time. The fact that the site is an identified attack target with minimal time available for further preparation requires a heightened state of readiness and implementation of protective measures that can be effective (onsite evacuation, dispersal or sheltering) before arrival or impact.

These conditions address the contingency for a very rapid progression of events due to an airborne hostile attack (such as that experienced on September 11, 2001) and the possibility for additional attacking aircraft. It is not intended to address accidental small aircraft impact as that initiating condition is adequately addressed by EAL HU4.1. The EAL conditions are not premised solely on the potential for a radiological release. Rather the issue includes the need for assistance due to the possibility for significant and indeterminate damage from additional air, land or water attack elements. Although vulnerability analyses show nuclear plants to be robust, it is appropriate for Offsite Response Organizations to be notified and to activate in order to be better prepared to respond should protective actions become necessary. If not previously notified by NRC that the aircraft impact was intentional, then it would be expected, although not certain, that notification by an appropriate Federal agency would follow. In this case, appropriate federal agency is intended to be NORAD, FBI, FAA or NRC. However, the declaration should not be unduly delayed awaiting Federal notification. Airliner is meant to be a large aircraft with the potential for causing significant damage to the plant. The status and size of the plane may be provided by NORAD through the NRC.

DCPP Basis Reference(s):

1. DCPP Security and Safeguards Contingency Plan
2. NRC Bulletin 2005-02, "Emergency Preparedness and Response Actions for Security-Based Events"
3. NEI White Paper "Enhancements to Emergency Preparedness Programs for Hostile Action," May 2005 (Revised November 1, 2005)

Category: H – Hazards

Sub-category: 4 – Security

Initiating Condition: Hostile action within the Protected Area

EAL:

HS4.1	**Site Area Emergency**
A hostile action is occurring or has occurred within the Protected Area as reported by the Security Watch Commander	

Mode Applicability:

All

Basis:

A security event may constitute a risk to site personnel and degrade the level of safety of the plant. It is of utmost importance to ensure available resources are mustered in a timely manner and applied as effectively as possible to combat the threat while at the same time assuring the conduct of safe plant operations. This requires close coordination between Security and Operations.

This condition represents an escalated threat to plant safety above that contained in the Alert IC in that a hostile force has progressed from the Owner Controlled Area to the Protected Area. Although nuclear plant security officers are well trained and prepared to protect against hostile action, it is appropriate for Offsite Response Organizations to be notified and encouraged to begin preparations for public protective actions (if they do not normally) to be better prepared should it be necessary to consider further actions.

This EAL addresses the potential for a very rapid progression of events due to a dedicated attack. It is not intended to address incidents that are accidental or acts of civil disobedience, such as hunters or physical disputes between employees within the OCA or PA.

In addition to attack by land and water, this EAL addresses events due to an airborne attack such as that experienced on September 11, 2001 and the possibility for additional attacking airliners. It is not intended to address accidental aircraft impact as that initiating condition is adequately addressed by other EALs. This EAL is not premised solely on the potential for a radiological release. Rather the issue includes the need for assistance due to the possibility for significant and indeterminate damage from additional attack elements. Although vulnerability analyses show nuclear plants to be robust, it is appropriate for Offsite Response Organizations to be notified and to activate in order to be better prepared to respond should protective actions become necessary. If not previously notified by NRC that the airborne hostile action was intentional, then it would be expected, although not certain, that notification by an appropriate Federal agency would follow. In this case, appropriate federal agency is intended to be NORAD, FBI, FAA or NRC. However, the declaration should not be unduly delayed awaiting Federal notification. Airliner is meant to be a large aircraft with the potential for causing significant damage to the plant. The status and size of the plane may be provided by NORAD through the NRC.

This EAL addresses the immediacy of a threat to impact site vital areas within a relatively short time. The fact that the site is under serious attack with minimal time available for additional assistance to arrive requires ORO readiness and preparation for the implementation of protective measures.

Consideration should be given to upgrading the classification to a General Emergency based on actual plant status after impact or progression of attack.

DCPP Basis Reference(s):

1. DCPP Security and Safeguards Contingency Plan
2. CPM-10, "Fire Protection of Safe Shutdown Equipment"
3. Plant Dwg. 471124, "Plot Plan"
4. NRC Bulletin 2005-02, "Emergency Preparedness and Response Actions for Security-Based Events"
5. NEI White Paper "Enhancements to Emergency Preparedness Programs for Hostile Action," May 2005 (Revised November 1, 2005)

Category: H – Hazards

Sub-category: 4 – Security

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

EAL:

HG4.1 General Emergency

A hostile action has occurred such that plant personnel are unable to operate equipment required to maintain safety functions

OR

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

Mode Applicability:

All

Basis:

A security event may constitute a risk to site personnel and degrade the level of safety of the plant. It is of utmost importance to ensure available resources are mustered in a timely manner and applied as effectively as possible to combat the threat while at the same time assuring the conduct of safe plant operations. This requires close coordination between Security and Operations.

This EAL encompasses conditions under which a hostile action has resulted in a loss of physical control of vital areas (containing vital equipment or controls of vital equipment) required to maintain safety functions and control of that equipment cannot be transferred to and operated from another location. Typically, these safety functions are reactivity control, RCS inventory, and secondary heat removal. If control of the plant equipment necessary to maintain safety functions can be transferred to another location, then the above initiating condition is not met.

This EAL also addresses failure of spent fuel cooling systems as a result of hostile action if imminent fuel damage is likely (e.g., freshly off-loaded reactor core in pool).

Loss of physical control of the control room or hot/dedicated shutdown capability alone may not prevent the ability to maintain safety functions per se. Design of the remote shutdown capability and the location of the transfer switches should be taken into account. The intent of the EAL is to establish control of important plant equipment and knowledge of important plant parameters in a timely manner. Primary emphasis should be placed on those components and instruments that supply protection for and information about safety functions.

DCPP Basis Reference(s):

1. DCPP Security and Safeguards Contingency Plan
2. NRC Bulletin 2005-02, "Emergency Preparedness and Response Actions for Security-Based Events"
3. NEI White Paper "Enhancements to Emergency Preparedness Programs for Hostile Action," May 2005 (Revised November 1, 2005)

Category: H – Hazards

Sub-category: 5 – Control Room Evacuation

Initiating Condition: Control Room evacuation has been initiated

EAL:

HA5.1	**Alert**
Entry into OP AP-8A, Control Room Inaccessibility, for Control Room evacuation	

Mode Applicability:

All

Basis:

With the Control Room evacuated, additional support, monitoring and direction through the Technical Support Center and/or other emergency operations centers are necessary. OP AP-8A, “Control Room Inaccessibility - Establishing Hot Standby,” provides specific instructions for evacuating the Control Room and establishing plant control at the Hot Shutdown Panel area. Inability to establish plant control from outside the Control Room escalates this event to a Site Area Emergency.

DCPP Basis Reference(s):

1. OP AP-8A, “Control Room Inaccessibility - Establishing Hot Standby”

Category: H – Hazards

Sub-category: 5 – Control Room Evacuation

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

EAL:

HS5.1 Site Area Emergency

Control Room evacuation has been initiated

AND

Control of the plant cannot be established per OP AP-8A, Control Room Inaccessibility, within 15 min.

Mode Applicability:

All

Basis:

This EAL indicates that expeditious transfer of safety systems has not occurred, but fission product barrier damage may not yet be indicated. The intent of this EAL is to capture events in which control of the plant cannot be reestablished in a timely manner. The fifteen minute time for transfer starts when the Control Room begins to be evacuated (not when OP AP-8A is entered). The time interval is based on how quickly control must be reestablished without core uncovery and/or core damage. The determination of whether or not control is established from outside the Control Room is based on SM/SEC/ED judgment. The SM/SEC/ED is expected to make a reasonable, informed judgment that control of the plant from the Hot Shutdown Panel area cannot be established within the fifteen minute interval. Typically, completion of step 12 of OP AP-8A is indication that control of the plant can be established.

Once the Control Room is evacuated, the objective is to establish control of important plant equipment and maintain knowledge of important plant parameters in a timely manner. Primary emphasis should be placed on components and instruments that supply protection for and information about safety functions. Typically, these safety functions are reactivity control (ability to shutdown the reactor and maintain it shutdown), RCS inventory (ability to cool the core), and secondary heat removal (ability to maintain a heat sink). In Cold Shutdown and Refueling modes, operator concern is directed toward maintaining core cooling such as is discussed in Generic Letter 88-17, "Loss of Decay Heat Removal." In Operating, and Hot Standby modes, operator concern is primarily directed toward maintaining critical safety functions and thereby assuring fission product barrier integrity.

OP AP-8A, "Control Room Inaccessibility - Establishing Hot Standby" and OP AP-8B, "Control Room Inaccessibility - Hot Standby to Cold Shutdown," provide specific instructions for evacuating the Control Room and establishing plant control at the Hot Shutdown Panel area.

Escalation of this event, if appropriate, would be by EALs in Category F, Category R or Subcategory H.6, Judgment.

DCPP Basis Reference(s):

1. OP AP-8A, "Control Room Inaccessibility - Establishing Hot Standby"

Category: H – Hazards

Sub-category: 6 –Judgment

Initiating Condition: Other conditions existing which in the judgment of the SM/SEC/ED warrant declaration of a UE

EAL:

HU6.1	**Unusual Event**
Other conditions exist which in the judgment of the SM/SEC/ED indicate that events are in progress or have occurred which indicate a potential degradation of the level of safety of the plant. No releases of radioactive material requiring offsite response or monitoring are expected unless further degradation of safety systems occurs	

Mode Applicability:

All

Basis:

This EAL addresses unanticipated conditions not addressed explicitly elsewhere but that warrant declaration of an emergency because conditions exist which are believed by the SM/SEC/ED to fall under the Unusual Event emergency class. This may include a security threat to facility protection. The responsibilities of the SM/SEC/ED for emergency classification include the following:

- Shift Manager (SM) – The control room Shift Manager is responsible for initial event classification and emergency plan activation. The SM may upgrade the event classification until relieved by either the SEC or ED. In addition, the SM may downgrade an Unusual Event to no emergency classification level.
- Site Emergency Coordinator (SEC) – The SEC may upgrade the classification of an event until relieved by the Emergency Director.
- Emergency Director (ED) – The ED, once staffed, is responsible for upgrading or downgrading emergency classifications, and may direct the SEC to change emergency classification levels.

From a broad perspective, one area that may warrant SM/SEC/ED judgment 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
- Insufficient availability of equipment and/or support personnel

DCPP Basis Reference(s):

1. NRC Bulletin 2005-02, “Emergency Preparedness and Response Actions for Security-Based Events”

Category: H – Hazards

Sub-category: 6 –Judgment

Initiating Condition: Other conditions existing which in the judgment of the SM/SEC/ED warrant declaration of an Alert

EAL:

HA6.1	**Alert**

Other conditions exist which in the judgment of the SM/SEC/ED indicate that are in progress or have occurred which involve an actual or potential substantial degradation of the level of safety of the plant. ANY releases are expected to be limited to small fractions of the EPA Protective Action Guideline exposure levels (1 Rem TEDE and 5 Rem thyroid CDE)

Mode Applicability:

All

Basis:

This EAL addresses unanticipated conditions not addressed explicitly elsewhere but that warrant declaration of an emergency because conditions exist which are believed by the SM/SEC/ED to fall under the Alert emergency class. This may include a security event that involves probable life threatening risk to site personnel or damage to site equipment because of intentional malicious dedicated efforts of a hostile act. The responsibilities of the SM/SEC/ED for emergency classification include the following:

- Shift Manager (SM) – The control room Shift Manager is responsible for initial event classification and emergency plan activation. The SM may upgrade the event classification until relieved by either the SEC or ED. In addition, the SM may downgrade an Unusual Event to no emergency classification level.
- Site Emergency Coordinator (SEC) – The SEC may upgrade the classification of an event until relieved by the Emergency Director.
- Emergency Director (ED) – The ED, once staffed, is responsible for upgrading or downgrading emergency classifications, and may direct the SEC to change emergency classification levels.

EPA PAGs stands for Environmental Protection Agency Protective Action Guidelines.

DCPP Basis Reference(s):

1. NRC Bulletin 2005-02, “Emergency Preparedness and Response Actions for Security-Based Events”

Category: H – Hazards

Sub-category: 6 –Judgment

Initiating Condition: Other conditions existing which in the judgment of the SM/SEC/ED warrant declaration of Site Area Emergency

EAL:

HS6.1 Site Area Emergency

Other conditions exist which in the judgment of the SM/SEC/ED indicate that events are in progress or have occurred which involve an actual or likely major failures of plant functions needed for protection of the public. ANY releases are not expected to result in exposure levels which exceed EPA Protective Action Guideline exposure levels (1 Rem TEDE and 5 Rem thyroid CDE) beyond the site boundary

Mode Applicability:

All

Basis:

This EAL addresses unanticipated conditions not addressed explicitly elsewhere, but that warrant declaration of an emergency because conditions exist which are believed by the SM/SEC/ED to fall under the emergency class description for Site Area Emergency. This may include security events that result in intentional damage or malicious acts; (1) toward site personnel or equipment that could lead to the likely failure of or; (2) prevents effective access to equipment needed for the protection of the public. The responsibilities of the SM/SEC/ED for emergency classification include the following:

- Shift Manager (SM) – The control room Shift Manager is responsible for initial event classification and emergency plan activation. The SM may upgrade the event classification until relieved by either the SEC or ED. In addition, the SM may downgrade an Unusual Event to no emergency classification level.
- Site Emergency Coordinator (SEC) – The SEC may upgrade the classification of an event until relieved by the Emergency Director.
- Emergency Director (ED) – The ED, once staffed, is responsible for upgrading or downgrading emergency classifications, and may direct the SEC to change emergency classification levels.

EPA PAGs stands for Environmental Protection Agency Protective Action Guidelines.

DCPP Basis Reference(s):

1. NRC Bulletin 2005-02, "Emergency Preparedness and Response Actions for Security-Based Events"

Category: H – Hazards

Sub-category: 6 –Judgment

Initiating Condition: Other conditions existing which in the judgment of the SM/SEC/ED warrant declaration of General Emergency

EAL:

HG6.1 General Emergency

Other conditions exist which in the judgment of the SM/SEC/ED indicate that events are in process or have occurred which involve actual or imminent substantial core degradation or melting with potential for loss of containment integrity or security events that result in an actual loss of physical control of the facility. Releases can be reasonably expected to exceed EPA Protective Action Guideline exposure levels (1 Rem TEDE and 5 Rem thyroid CDE) offsite for more than the site boundary

Mode Applicability:

All

Basis:

This EAL addresses unanticipated conditions not addressed explicitly elsewhere, but that warrant declaration of an emergency because conditions exist which are believed by the SM/SEC/ED to fall under the General Emergency class. This may include security events that result in an actual loss of physical control of the facility. The responsibilities of the SM/SEC/ED for emergency classification include the following:

- Shift Manager (SM) – The control room Shift Manager is responsible for initial event classification and emergency plan activation. The SM may upgrade the event classification until relieved by either the SEC or ED. In addition, the SM may downgrade an Unusual Event to no emergency classification level.
- Site Emergency Coordinator (SEC) – The SEC may upgrade the classification of an event until relieved by the Emergency Director.
- Emergency Director (ED) – The ED, once staffed, is responsible for upgrading or downgrading emergency classifications, and may direct the SEC to change emergency classification levels.

Releases can reasonably be expected to exceed EPA PAG plume exposure levels outside the Exclusion Area Boundary. EPA PAGs stands for Environmental Protection Agency Protective Action Guidelines.

DCPP Basis Reference(s):

1. NRC Bulletin 2005-02, “Emergency Preparedness and Response Actions for Security-Based Events”

Category S – System Malfunction

Numerous system-related equipment failure events that warrant emergency classification have been identified in this category. They may pose actual or potential threats to plant safety.

The events of this category pertain to the following subcategories:

1. Loss of Power

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

2. RTS Failure

Events related to failure of the Reactor Protection System (RPS) to initiate and complete reactor trips. In the plant licensing basis, postulated failures of the RPS to complete a reactor trip comprise a specific set of analyzed events referred to as Anticipated Transient Without Scram (ATWS) events. For EAL classification however, ATWS is intended to mean any trip failure event that does not achieve reactor shutdown. If RPS actuation fails to assure reactor shutdown, positive control of reactivity is at risk and could cause a threat to Fuel Clad, RCS and Containment integrity.

3. Inability to Reach or Maintain Shutdown Conditions

System malfunctions may lead to loss of capability to remove heat removal the reactor core and RCS.

Only one EAL falls into this subcategory. It is related to the failure of the plant to be brought to the required plant operating condition required by technical specifications if a limiting condition for operation (LCO) is not met.

4. Instrumentation / Communications

Certain events that degrade plant operator ability to effectively assess plant conditions within the plant warrant emergency classification. Losses of annunciators are in this subcategory.

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

5. Fuel Clad Degradation

During normal operation, reactor coolant fission product activity is very low. Small concentrations of fission products in the coolant are primarily from the fission of tramp uranium in the Fuel Clad or minor perforations in the clad itself. Any significant increase from these base-line levels (2% - 5% clad failures) is indicative of fuel failures and is covered under the Fission Product Barriers category. However, lesser amounts of clad damage may result in coolant activity exceeding Technical Specification limits. These fission products will be circulated with the reactor coolant and can be detected by coolant sampling.

6. RCS Leakage

The Reactor Vessel provides a volume for the coolant that covers the reactor core. The Reactor Vessel and associated pressure piping (reactor coolant system) together provide a barrier to limit the release of radioactive material should the reactor Fuel Clad integrity fail.

Excessive RCS leakage greater than Technical Specification limits are utilized to indicate potential pipe cracks that may propagate to an extent threatening Fuel Clad, RCS and Containment integrity.

7. Inadvertent Criticality

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

Category: S – System Malfunction

Sub-category: 1 – Loss of Power

Initiating Condition: Loss of all offsite power to vital buses for greater than 15 minutes

EAL:

SU1.1	**Unusual Event**
Loss of ALL offsite AC power to Unit 1(2) Vital 4kV buses F, G and H for > 15 min.	

Mode Applicability:

1 - Power Operation, 2 - Startup, 3 - Hot Standby, 4 - Hot Shutdown

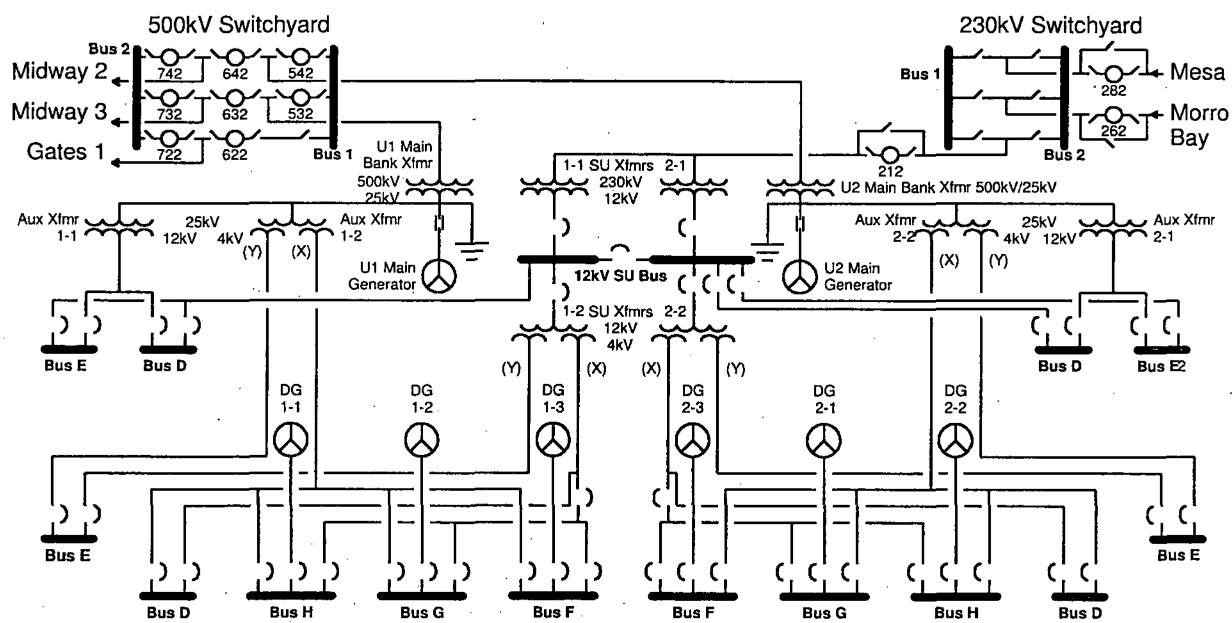
Basis:

Prolonged loss of all offsite 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). Buses F, G and H are the Unit 1(2) Vital 4kV (essential) buses. Possible offsite power systems capable of providing power to the Vital 4kV buses include:

- Affected Unit's Aux Power system: The 4 kV system can be powered by backfeeding from the 500 kV Switchyard. The Main Generator MOD is opened. Power is supplied from the switchyard through the generator output breakers 532 (542) and/or 632 (642). The Main Transformer steps down 500 kV to 25 kV and the 25 kV is stepped down to 4 kV by Aux Transformer 1-2.
- Affected Unit's SU Power system: The SU bus is normally energized by the 230 kV Switchyard through OCB-212 and SU Transformer 1-1(2-1). The SU bus then supplies SU Transformer 1-2(2-2) which steps the voltage down to 4160 V.
- Other Unit's SU Power system
- Other Unit's Aux Power system

A basic flowpath diagram of the DCPP electrical distribution system is given in Figure S-1.

Figure S-1 DCPP Electrical Distribution System



The 15-minute interval was selected as a threshold to exclude transient power losses. If neither unit 1E 4kV bus is energized by an offsite source within 15 minutes, an Unusual Event is declared under this EAL.

DCPP Basis Reference(s):

1. UFSAR, Section 8.2.1
2. UFSAR, Section 8.3.1
3. OP AP SD 1, "Loss of AC Power"
4. OP AP-26, "Loss of Offsite Power"
5. OP J-2:V, "Backfeeding the Unit From the 500kV System"
6. ECA-0.0, "Loss of All Vital AC Power"
7. ECA-0.3, "Restore 4kV Buses"

Category: S – System Malfunction

Sub-category: 1 – Loss of Power

Initiating Condition: AC power capability to vital buses reduced to a single power source for greater than 15 minutes such that any additional single failure would result in station blackout

EAL:

SA1.1	**Alert**
AC power capability to Unit 1(2) Vital 4kV buses F, G and H reduced to a single power source (i.e., one DG or one offsite power source) for > 15 min.	

Mode Applicability:

1 - Power Operation, 2 - Startup, 3 - Hot Standby, 4 - Hot Shutdown

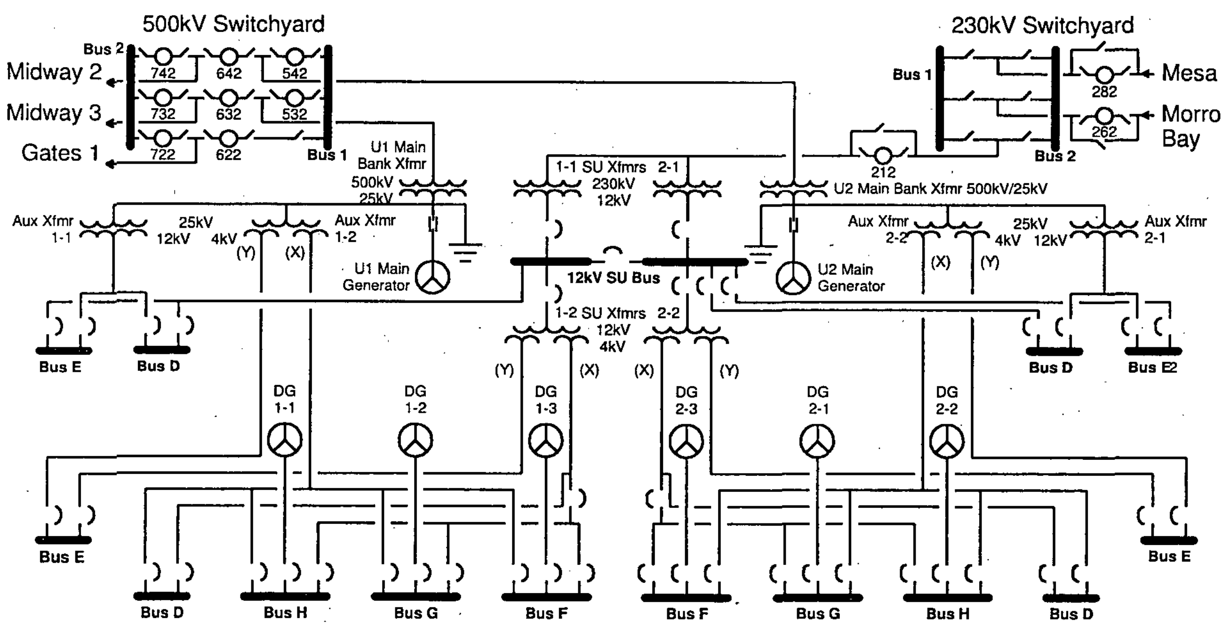
Basis:

The condition indicated by this EAL is the degradation of the offsite and onsite power sources such that any additional single failure would result in a loss of all AC power to the unit Vital 4kV buses. Buses F, G and H are the Unit 1(2) Vital 4kV (essential) buses. Possible offsite power systems capable of providing power to the Vital 4kV buses include:

- Affected Unit's Aux Power system: The 4 kV system can be powered by backfeeding from the 500 kV Switchyard. The Main Generator MOD is opened. Power is supplied from the switchyard through the generator output breakers 532 (542) and/or 632 (642). The Main Transformer steps down 500 kV to 25 kV and the 25 kV is stepped down to 4 kV by Aux Transformer 1-2.
- Affected Unit's SU Power system: The SU bus is normally energized by the 230 kV Switchyard through OCB-212 and SU Transformer 1-1(2-1). The SU bus then supplies SU Transformer 1-2(2-2) which steps the voltage down to 4160 V.
- Other Unit's SU Power system
- Other Unit's Aux Power system

A basic flowpath diagram of the DCPP electrical distribution system is given in Figure S-1.

Figure S-1 DCPP Electrical Distribution System



The emergency diesel generators are the onsite power sources:

- D/G 1-3(2-3) Vital 4kV Bus F
- D/G 1-2(2-2) Vital 4kV Bus G
- D/G 1-1(2-1) Vital 4kV Bus H

Several combinations of power failures could therefore satisfy this EAL. The subsequent loss of the remaining power source escalates the event to a Site Area Emergency under EAL SS1.1.

The 15-minute interval was selected as a threshold to exclude transient power losses. If multiple sources fail to energize the unit 1E 4kV buses within 15 minutes, an Alert is declared under this EAL.

DCPP Basis Reference(s):

1. UFSAR, Section 8.2.1
2. UFSAR, Section 8.3.1.6
3. OP AP SD 1, "Loss of AC Power"
4. OP AP-2, "Loss of Offsite Power"
5. OP J-2:V, "Backfeeding the Unit From the 500kV System"
6. ECA-0.0, "Loss of All Vital AC Power"
7. ECA-0.3, "Restore 4kV Buses"
8. DCM T-42, "Station Blackout"

Category: S – System Malfunction

Sub-category: 1 – Loss of Power

Initiating Condition: Loss of all offsite power and loss of all onsite AC power to vital buses

EAL

SS1.1	**Site Area Emergency**
Loss of ALL offsite and onsite AC power to Unit 1(2) Vital 4kV buses F, G and H for > 15 min.	

Mode Applicability:

1 - Power Operation, 2 - Startup, 3 - Hot Standby, 4 - Hot Shutdown

Basis:

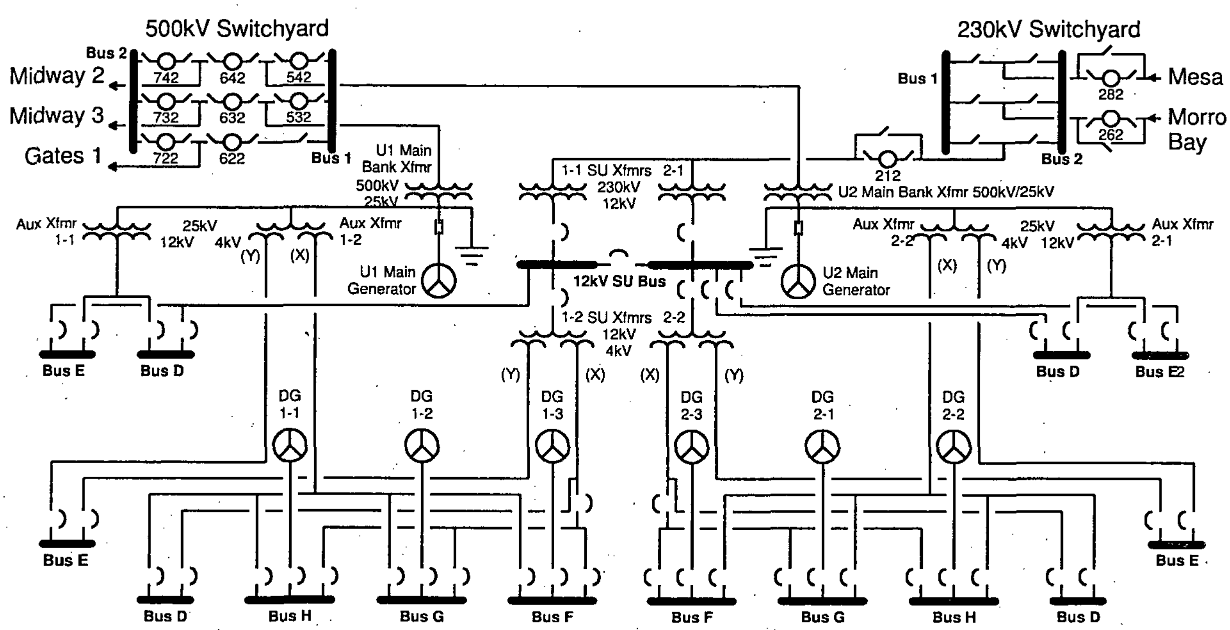
Consideration should be given to operable loads necessary to remove decay heat or provide RCS makeup capability when evaluating loss of AC power to the vital 4kV buses. Even though a vital 4kV bus may be energized, if necessary loads (i.e., loads that if lost would inhibit decay heat removal capability or RCS makeup capability) are not operable on the energized bus, the bus should not be considered operable.

Loss of offsite and onsite AC power compromises all plant safety systems requiring electrical power. Buses F, G and H are the Unit 1(2) Vital 4kV (essential) buses. Possible offsite power systems capable of providing power to the Vital 4kV buses include:

- Affected Unit's Aux Power system: The 4 kV system can be powered by backfeeding from the 500 kV Switchyard. The Main Generator MOD is opened. Power is supplied from the switchyard through the generator output breakers 532 (542) and/or 632 (642). The Main Transformer steps down 500 kV to 25 kV and the 25 kV is stepped down to 4 kV by Aux Transformer 1-2.
- Affected Unit's SU Power system: The SU bus is normally energized by the 230 kV Switchyard through OCB-212 and SU Transformer 1-1(2-1). The SU bus then supplies SU Transformer 1-2(2-2) which steps the voltage down to 4160 V.
- Other Unit's SU Power system
- Other Unit's Aux Power system

A basic flowpath diagram of the DCPP electrical distribution system is given in Figure S-1.

Figure S-1 DCPP Electrical Distribution System



The emergency diesel generators are the onsite power sources:

- D/G 1-3(2-3) Vital 4kV Bus F
- D/G 1-2(2-2) Vital 4kV Bus G
- D/G 1-1(2-1) Vital 4kV Bus H

Prolonged loss of all AC power will cause core uncovery and loss of containment integrity; thus, this event can escalate to a General Emergency under EAL FG1.1 or EAL SG1.1. The 15-minute interval was selected as a threshold to exclude transient power losses.

DCPP Basis Reference(s):

1. UFSAR, Section 8.2.1
2. UFSAR, Section 8.3.1.6
3. OP AP SD 1, "Loss of AC Power"
4. OP AP-26, "Loss of Offsite Power"
5. OP J-2:V, "Backfeeding the Unit From the 500kV System"
6. ECA-0.0, "Loss of All Vital AC Power"
7. ECA-0.3, "Restore 4kV Buses"
8. DCM T-42, "Station Blackout"

Category: S – System Malfunction

Sub-category: 1 – Loss of Power

Initiating Condition: Loss of all vital DC power

EAL:

SS1.2	**Site Area Emergency**
Loss of ALL vital DC power based on < 105VDC bus voltage indications for > 15 min.	

Mode Applicability:

1 - Power Operation, 2 - Startup, 3 - Hot Standby, 4 - Hot Shutdown

Basis:

Loss of all DC power compromises the ability to monitor and control plant safety functions. Prolonged loss of all DC power will cause core uncovery and loss of containment integrity when there is significant decay heat and sensible heat in the reactor system. Fifteen minutes was selected as a threshold to exclude transient or momentary power losses.

The DC Power system is an ungrounded power generation, storage, and distribution system consisting of two subsystems:

- Non-Vital 125/250 VDC
- Vital 125 VDC

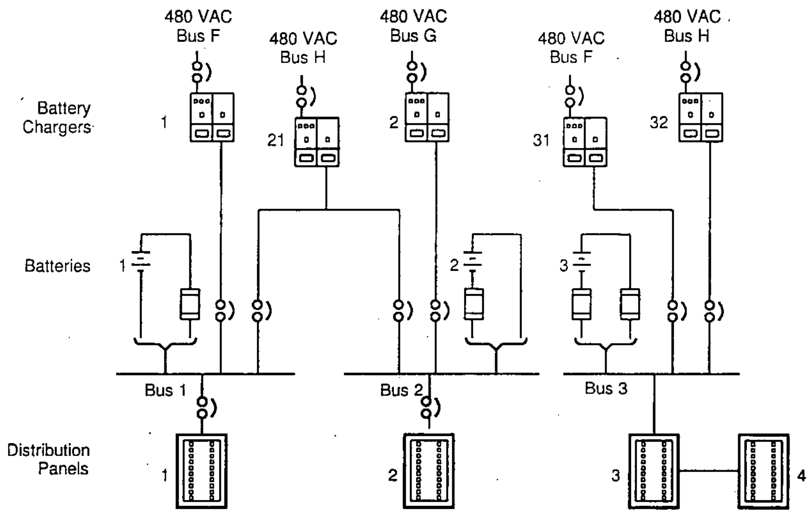
The Non-Vital 125/250 VDC system consists of two 125 VDC distribution networks. The Non-Vital 250 VDC system is a load center supplied by the two Non-Vital 125 VDC buses arranged in series. It supplies power to large DC loads such as backup and emergency lube oil pumps.

The vital 125 VDC system consists of three independent networks. Each network contains the following components:

- Battery
- Battery charger
- Standby battery charger to allow maintenance and/or testing
- Distribution panels
- Ground detector

The vital 125 VDC batteries and distribution panels are located in Area "H" of the 115 feet elevation of the Auxiliary Building. A basic one-line diagram of the Vital 125 VDC System is shown in Figure D-2.

Figure D-2 Vital 125 VDC One-Line Diagram



The vital 125 VDC batteries are Class 1E power supplies. Mean life of the batteries is 20 years. A total of six batteries, 11(21), 12(22), and 13(23) are supplied for Units 1 and 2. The batteries are sized to provide sufficient power to operate the DC loads for the time necessary to safely shut down the unit, should a 480-VAC source to one or more battery chargers be unavailable. Each battery is rated at 2320 ampere-hours (8-hour rate).

Escalation to a General Emergency would occur by EALs in Category R, EAL FG1.1 or EAL HG6.1.

This EAL is the hot condition equivalent of the cold condition loss of DC power
EAL CU1.2.

DCPP Basis Reference(s):

1. UFSAR, Section 8.3.2.2
2. OP AP-23, "Loss of Vital DC Bus"
3. ECA-0.0, "Loss of All Vital AC Power"

Category: S – System Malfunction

Sub-category: 1 – Loss of Power

Initiating Condition: Prolonged loss of all offsite power and prolonged loss of all onsite AC power to vital buses

EAL:

SG1.1 General Emergency

Loss of ALL offsite and onsite AC power to Unit 1(2) Vital 4kV buses F, G and H

AND EITHER:

Restoration of ANY Vital 4kV bus within 4 hours is <u>not</u> likely

OR

CSFST Core Cooling-RED or MAGENTA path

Mode Applicability:

1 - Power Operation, 2 - Startup, 3 - Hot Standby, 4 - Hot Shutdown

Basis:

Loss of all AC power compromises all plant safety systems requiring electrical power including RHR, ECCS, Containment heat removal and secondary heat removal. Prolonged loss of all AC power leads to loss of Fuel Clad, RCS and Containment barriers. 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, "Station Blackout." Although this EAL may be viewed as redundant to the Fission Product Barrier EALs, its inclusion is necessary to better assure timely recognition and emergency response.

The likelihood of restoring at least one Vital 4kV bus to a unit should be based on a realistic appraisal of the situation since a delay in an upgrade decision based on only a chance of mitigating the event could result in a loss of valuable time in preparing and implementing public protective actions. Buses F, G and H are the Unit 1(2) Vital 4kV (essential) buses. Possible offsite power systems capable of providing power to the Vital 4kV buses include:

- Affected Unit's Aux Power system: The 4 kV system can be powered by backfeeding from the 500 kV Switchyard. The Main Generator MOD is opened. Power is supplied from the switchyard through the generator output breakers 532 (542) and/or 632 (642). The Main Transformer steps down 500 kV to 25 kV and the 25 kV is stepped down to 4 kV by Aux Transformer 1-2.

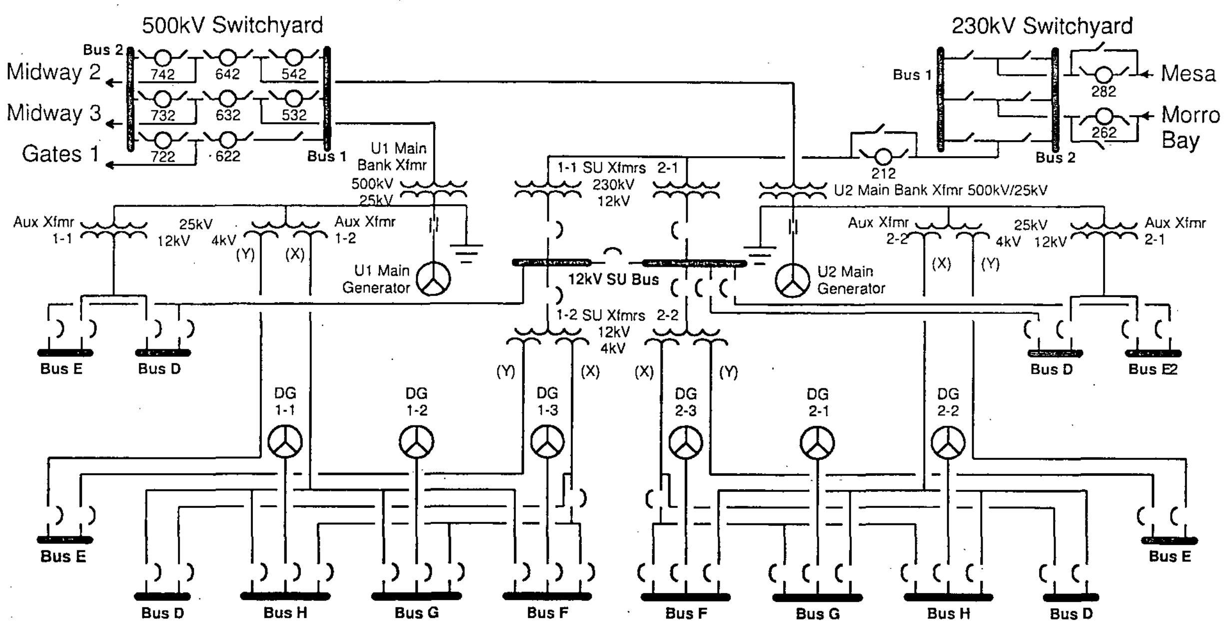
- Affected Unit's SU Power system: The SU bus is normally energized by the 230 kV Switchyard through OCB-212 and SU Transformer 1-1(2-1). The SU bus then supplies SU Transformer 1-2(2-2) which steps the voltage down to 4160 V.

- Other Unit's SU Power system

- Other Unit's Aux Power system

A basic flowpath diagram of the DCPP electrical distribution system is given in Figure S-1.

Figure S-1 DCPP Electrical Distribution System



The emergency diesel generators are the onsite power sources:

- D/G 1-3(2-3) Vital 4kV Bus F
- D/G 1-2(2-2) Vital 4kV Bus G
- D/G 1-1(2-1) Vital 4kV Bus H

In addition, under these conditions, fission product barrier monitoring capability may be degraded. Although it may be difficult to predict when power can be restored, it is necessary to give the SM/SEC/ED a reasonable idea of how quickly to declare a General Emergency based on two major considerations:

1. Are there any present indications that core cooling is already degraded to the point that loss or potential loss of fission product barriers is imminent?
2. If there are no present indications of such core cooling degradation, how likely is it that power can be restored in time to assure that a loss of two barriers with a potential loss of the third barrier can be prevented?

Thus, indication of continuing core cooling degradation must be based on fission product barrier monitoring with particular emphasis on SM/SEC/ED judgment as it relates to imminent loss or potential loss of fission product barriers and degraded ability to monitor fission product barriers. Indication of continuing core cooling degradation is manifested by entry to Critical Safety Function Status Tree Core Cooling-MAGENTA and RED paths CSFST Core Cooling-MAGENTA path is entered if core exit thermocouples (TCs) are less than 1200ºF, RCS core exit TCs subcooling margin (SCM) is less than 20ºF and any of the following:

- No RCPs are running and either: core exit TCs are less than 700ºF and RVLIS full range is less than or equal to 32%, or core exit TCs are greater than or equal to 700ºF and RVLIS full range is greater than 32%.
- At least one RCP is running and Reactor Vessel water level is less than or equal to RVLIS dynamic head readings in Table F-2.

These conditions indicate subcooling has been lost and that some Fuel Clad damage may potentially occur.

CSFST Core Cooling-RED path is entered if either:

- Core exit TCs are greater than or equal to 1200ºF, or
- Core exit TCs are greater than or equal to 700ºF with RCS subcooling margin (SCM) less than 20ºF, no RCPs are running, and RVLIS full range is less than or equal to 32%.

Either set of conditions indicates significant core exit superheating or core uncovery.

DCPP Basis Reference(s):

1. UFSAR, Section 8.2.1
2. UFSAR, Section 8.3.1.6
3. OP AP SD 1, "Loss of AC Power"
4. OP AP-26, "Loss of Offsite Power"
5. OP J-2:V, "Backfeeding the Unit From the 500kV System"
6. ECA-0.0, "Loss of All Vital AC Power"
7. ECA-0.3, "Restore 4kV Buses"
8. DCM T-42, "Station Blackout"
9. F-0, "Critical Safety Function Status Trees," Attachment 2, "Core Cooling"
10. FR-C.1, "Response to Inadequate Core Cooling"
11. FR-C.2, "Response to Degraded Core Cooling"

Category: S – System Malfunction

Sub-category: 2 – RTS Failure

Initiating Condition: Automatic trip fails to shut down the reactor and the manual actions taken from the reactor control console are successful in shutting down the reactor

EAL:

SA2.1 Alert

An automatic trip failed to shut down the reactor

AND

Manual actions taken at the reactor control console successfully shut down the reactor as indicated by reactor power < 5%

Mode Applicability:

1 - Power Operation, 2 - Startup

Basis:

This condition indicates failure of the automatic Reactor Trip System to trip the reactor. This condition is more than a potential degradation of a safety system in that a front line automatic protection system did not function in response to a plant transient. Thus, plant safety has been compromised because design limits of the fuel may have been exceeded. An Alert is indicated because conditions may exist that lead to potential loss of fuel clad or RCS and because of the failure of the Reactor Protection System to automatically shut down the plant.

A reactor trip is automatically initiated by the Reactor Trip System (RTS) when certain continuously monitored parameters exceed predetermined setpoints. A reactor trip may be the result of automatic action initiated by the RTS in response to any of the following parameters:

- Power Range Neutron Flux (high/low, P-8, P-9, P-10)
- Power Range Neutron Flux Rate (high/low)
- Intermediate Range Neutron Flux (P-6)
- Source Range Neutron Flux
- Low Power Reactor Trips Block (P-7)
- Turbine Impulse Chamber Pressure (P-13)
- Overtemperature ΔT
- Overpower ΔT
- Pressurizer Pressure (high/low)
- Pressurizer Water Level (high)
- Reactor Coolant Flow (low)
- Reactor Coolant Pump (RCP) Breaker Position
- Undervoltage RCPs
- Underfrequency RCPs
- Steam Generator (SG) Water Level (low-low)

- SG Water Level - Low Low Trip Time Delay (TTD)
- Turbine Trip (Low Auto-Stop Oil Pressure / Turbine Stop Valve Closure)
- SI Input from ESFAS
- Reactor Trip Breakers (RTBs)
- Reactor Trip Breaker Undervoltage and Shunt Trip Mechanisms
- Automatic Trip Logic
- Seismic Trip

Following a successful reactor trip, rapid insertion of the control rods occurs. Nuclear power promptly drops to a few percent of the original power level and then decays to a level some 8 decades less at a startup rate of about -1/3 DPM. The reactor power drop continues until reactor power reaches the point at which the influence of source neutrons on reactor power starts to be observable. A predictable post-trip response from an automatic reactor trip signal should therefore consist of a prompt drop in reactor power as sensed by the nuclear instrumentation and a negative startup rate as nuclear power drops into the source range.

Per E-0 U1(2), "Reactor Trip or Safety Injection," the operator ensures that the reactor has tripped by verifying the reactor trip and bypass breakers are open, reactor power is decreasing and control rods are fully inserted. If these responses cannot be verified, as part of contingency actions, the operator deenergizes 480V Buses 13D and 13E (52-HD-13 and 52-HE-4). The manual actions taken at the reactor control console in the Control Room to trip the reactor cause rapid control rod insertion. For the purposes of evaluating this EAL, any of the following manual actions are considered successful if reactor power is reduced below 5%:

- Actuation of the reactor trip switches.
- Deenergization of 480V Buses 13D and 13E (52-HD-13 and 52-HE-4) at the Control Room vertical board; local deenergization of these buses requires actions away from the Control Room and is therefore not considered a "successful" manual reactor trip.
- Actuation of the turbine trip switch

The AMSAC system has no manual actuation but automatically trips the turbine, starts auxiliary feedwater, and isolates steam generator blowdown on coincidence of low-low steam generator water level in three out of four steam generators.

The reactor should be considered shutdown when it is producing less heat than the maximum decay heat load for which the safety systems are designed. The reactor power level of 5% is the minimum level of power production as specified in CSFST Subcriticality – Red. Reactor power above this level is indicative of continued power production at a rate that may exceed the decay heat removal capability of safety systems.

The Alert emergency classification is required whenever the Shift Manager determines that a required automatic reactor trip did not occur. It is recognized that E-0 instructs the operator to insert a manual reactor trip whether or not a required automatic reactor trip actually occurred. However, the failure of the automatic RTS trip signal to complete a reactor trip following receipt of an automatic trip signal meets the Alert classification threshold of potential substantial degradation in the level of safety of the plant. This is true even if no radiation alarms indicate fuel problems.

In the event that the operator identifies a reactor trip is imminent and successfully initiates a manual reactor trip before the automatic trip setpoint is reached, no declaration is required. The successful manual trip of the reactor before it reaches its automatic trip setpoint or reactor trip signals caused by instrumentation channel failures does not lead to a potential fission product barrier loss. If manual reactor trip actions at the reactor control console (following an unsuccessful automatic reactor trip) fail to reduce reactor power below 5%, the event escalates to the Site Area Emergency under EAL SS2.1.

DCPP Basis Reference(s):

1. UFSAR, Table 7.2-1
2. Technical Specifications, Table 3.3.1-1
3. PK04-14, "REACTOR TRIP ACTUATED"
4. E-0 U1(2)," Reactor Trip or Safety Injection"
5. F-0, "Critical Safety Function Status Trees," Attachment 1, "Subcriticality"
6. FR-S.1 U1(2), "Response to Nuclear Power Generation – ATWS"

Category: S – System Malfunction

Sub-category: 2 – RTS Failure

Initiating Condition: Automatic trip fails to shut down the reactor and manual actions taken from the reactor control console are NOT successful in shutting down the reactor

EAL:

SS2.1 Site Area Emergency

An automatic trip failed to shut down the reactor

AND

Manual actions taken at the reactor control console do not shut down the reactor as indicated by reactor power > 5%

Mode Applicability:

1 - Power Operation, 2 - Startup

Basis:

This EAL addresses any automatic reactor trip signal followed by a manual trip that fails to shut down the reactor to an extent the reactor is producing energy (> 5%) in excess of the heat load for which the safety systems were designed. The manual actions taken at the reactor control console in the Control Room to trip the reactor cause rapid control rod insertion. For the purposes of evaluating this EAL, any of the following manual actions are considered successful if reactor power is reduced below 5%:

- Actuation of the reactor trip switches.
- Deenergization of 480V Buses 13D and 13E (52-HD-13 and 52-HE-4) at the Control Room vertical board; local deenergization of these buses requires actions away from the Control Room and is therefore not considered a "successful" manual reactor trip.
- Actuation of the turbine trip switch

The AMSAC system has no manual actuation but automatically trips the turbine, starts auxiliary feedwater, and isolates steam generator blowdown on coincidence of low-low steam generator water level in three out of four steam generators.

Automatic and manual trips are not considered successful if action away from the Control Room is required to trip the reactor. If any of the alternate recovery actions for emergency boration of the RCS listed in EOPs are required to reduce reactor power below 5%, the reactor trips have been unsuccessful. Start up rate (SUR) on the Intermediate Range or Source Range is used as an indicator of decreasing power and should be observed following any reactor trip from power.

This EAL is still applicable even if actions taken away from the reactor control console are successful in shutting the reactor down because the design limits of the fuel may have been exceeded or because of the gross failure of the Reactor Protection System to shut down the plant.

Escalation of this event to a General Emergency would be under EAL SG2.1 or Shift Manager judgment.

DCPP Basis Reference(s):

1. UFSAR, Table 7.2-1
2. Technical Specifications, Table 3.3.1-1
3. PK04-14, "REACTOR TRIP ACTUATED"
4. E-0 U1(2), "Reactor Trip or Safety Injection"
5. F-0, "Critical Safety Function Status Trees," Attachment 1 "Subcriticality"
6. FR-S.1 U1(2), "Response to Nuclear Power Generation – ATWS"

Category:	S – System Malfunction
Sub-category:	2 – RTS Failure
Initiating Condition:	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

EAL:

SG2.1 General Emergency

Automatic trip and ALL manual actions fail to shut down the reactor

AND EITHER of the following have occurred due to continued power generation:

CSFST Core Cooling-RED

OR

CSFST Heat Sink-RED

Mode Applicability:

1 - Power Operation, 2 - Startup

Basis:

Under the conditions of this EAL, the efforts to bring the reactor subcritical are unsuccessful. This includes automatic trips and all manual actions to shut down the reactor. Unlike EALs SA2.1 and SS2.1, all manual actions to shut down the reactor may include actions that are conducted away from the reactor control console and outside the Control Room. For example, if local opening of the reactor trip breakers terminates power production and shuts down the reactor, this threshold is not met.

Continued power generation under the conditions of this EAL means the reactor may be producing more heat than the maximum decay heat load for which the safety systems were designed. This situation could be a precursor for a core melt sequence. This combination of failures of both front line and backup protection systems to function in response to a plant transient, along with the continued production of heat, poses a direct threat to the Fuel Clad and RCS barriers.

Indication that core cooling is extremely challenged is manifested by entry to Critical Safety Function Status Tree (CSFST) Core Cooling-RED path. CSFST Core Cooling-RED path is entered if either:

- Core exit TCs are greater than or equal to 1200ºF, or
- Core exit TCs are greater than or equal to 700ºF with RCS subcooling margin (SCM) less than 20ºF, no RCPs are running, and RVLIS full range is less than or equal to 32%.

Either set of conditions indicates significant core exit superheating and core uncovery.

Indication that heat removal is extremely challenged is manifested by entry to CSFST Heat Sink-RED path. CSFST Heat Sink-RED path is entered if all steam generator narrow range levels are less than or equal to the values indicated in EOP F-0, "Critical Safety Function Status Tree" (ref. 9) and total AFW flow is less than or equal to 435 gpm. The combination of these conditions when heat sink is required indicates the ultimate heat sink function is under extreme challenge. This condition addresses loss of functions required for hot shutdown with the reactor at pressure and temperature. CSFST setpoints enclosed in brackets are used under adverse Containment conditions. Adverse Containment condition thresholds apply when Containment pressure is greater than or equal to 3 psig or Containment radiation is greater than or equal to:

- 10^5 R/hr (RI-30/31 PAM 2)
- 10^6 R (RR-30/31 PAM 1)

In the event the challenge to either core cooling or heat removal occurs at a time when reactor trip failure fails to terminate power generation, a core melt sequence may exist and rapid degradation of the Fuel Clad could begin. To permit maximum offsite intervention time, the General Emergency declaration is therefore appropriate in anticipation of an inevitable General Emergency declaration due to loss and Potential Loss of fission product barriers.

DCPP Basis Reference(s):

1. UFSAR, Table 7.2-1
2. Technical Specifications, Table 3.3.1-1
3. PK04-14, "REACTOR TRIP ACTUATED"
4. E-0 U1(2), "Reactor Trip or Safety Injection"
5. F-0, "Critical Safety Function Status Trees," Attachment 1 "Subcriticality"
6. FR-S.1 U1(2), "Response to Nuclear Power Generation – ATWS"
7. F-0, "Critical Safety Function Status Trees," Attachment 2, "Core Cooling"
8. FR-C.1, "Response to Inadequate Core Cooling"
9. F-0, "Critical Safety Function Status Trees," Attachment 3, "Heat Sink"
10. FR-H.1, "Response to Loss of Secondary Heat Sink"

Category: S – System Malfunction

Sub-category: 3 – Inability to Reach or Maintain Shutdown Conditions

Initiating Condition: Inability to reach required shutdown within Technical Specification limits

EAL:

SU3.1	**Unusual Event**
Plant is not brought to required operating mode within Technical Specifications LCO action statement time	

Mode Applicability:

1 - Power Operation, 2 - Startup, 3 - Hot Standby, 4 - Hot Shutdown

Basis:

Limiting Conditions of Operation (LCOs) require the plant to be brought to a prescribed shutdown mode when the Technical Specification configuration cannot be restored. Depending on the circumstances, this may or may not be an emergency or precursor to a more severe condition. In any case, the initiation of plant shutdown required by the Technical Specification requires a one-hour report under 10 CFR 50.72 (b) non-emergency events. The plant is within its safety envelope when being shut down within the allowable action statement time in the Technical Specifications. An immediate declaration of an Unusual Event is required when the plant is not brought to the required operating mode within the allowable action statement time in the Technical Specifications. Declaration of an Unusual Event is based on the time at which the LCO-specified action completion period elapses under Technical Specifications and is not related to how long a condition may have existed. Other Technical Specification shutdowns that involve precursors to more serious events are addressed by other EALs.

DCPP Basis Reference(s):

1. DCPP Operating License and Technical Specifications
2. UFSAR, Chapter 16, Technical Specifications and Equipment Control Guidelines

Category: S – System Malfunction

Sub-category: 4 – Instrumentation / Communications

Initiating Condition: Unplanned loss of most or all safety system annunciation or indication in the control room for greater than 15 minutes

EAL:

SU4.1	**Unusual Event**
Unplanned loss of greater than approximately 75% of the annunciators or indicators associated with safety systems on Vertical Boards 1 through 5, and Control Consoles 1, 2 and 3 for > 15 min.	

Mode Applicability:

1 - Power Operation, 2 - Startup, 3 - Hot Standby, 4 - Hot Shutdown

Basis:

The listed vertical boards and control consoles are all of the front panels in the Control Room. Each panel contains some annunciators or indicators that would be desirable to facilitate conduct of safe plant operations. Annunciators or indicators for this EAL include those identified in the Abnormal Operating Instructions, the Emergency Operating Instructions and other EALs (e.g., area, process, and/or effluent rad monitors, etc.).

This EAL recognizes the difficulty associated with monitoring changing plant conditions without the use of a major portion of the annunciation or indication equipment. The availability of computer-based monitoring capability (i.e., PPC or SPDS) is not a factor at the Unusual Event emergency classification level.

"Unplanned" loss of annunciators or indicators excludes scheduled maintenance and testing activities.

If approximately 75% of the safety system annunciators or indications are lost, an elevated risk exists that a degraded plant condition may be undetected. A detailed count of the lost instrumentation is not required. The judgment of the Shift Manager, however, should be used as the threshold for determining the severity of the plant conditions.

Plant design provides redundant safety system indication powered from separate uninterruptible power supplies. While failure of a large portion of annunciators is more likely than a failure of a large portion of indications, failure of indications is included in this EAL due to difficulty associated with assessment of plant conditions when indications are not available. The loss of several safety system indicators should remain a function of the specific system or component operability status and is addressed by the applicable Technical Specification. The initiation of a Technical Specification imposed plant shutdown related to instrument loss must be reported via 10 CFR 50.72. If the shutdown is not in compliance with the Technical Specification action time, EAL SU2.1 ensures declaration of an Unusual Event.

The 15-minute interval offers time to recover from transient or momentary power losses. Due to the limited number of safety systems in operation during Cold Shutdown, Refueling and Defueled modes, this EAL is not applicable during these modes of operation.

If all computer-driven monitoring capability is unavailable or a significant transient is in progress during the loss of annunciation or indication, the event escalates to an Alert classification under EAL SA4.1.

DCPP Basis Reference(s):

1. UFSAR, Section 7
2. Plant Drawing no. 521121
3. Plant Drawing no. 521122
4. Plant Drawing no. 521133
5. Plant Drawing no. 521134
6. Plant Drawing no. 521135
7. Plant Drawing no. 521130
8. AR PK15-22 U1(2), "Main Annunciator System Trouble"
9. OP1.DC11, "Control of Main Annunciator System Problems"

Category: S – System Malfunction

Sub-category: 4 – Instrumentation / Communications

Initiating Condition: Unplanned loss of all onsite or offsite communications capabilities

EAL:

SU4.2 Unusual Event

Loss of ALL Table C-2 onsite (internal) communications capability affecting the ability to perform routine operations

OR

Loss of ALL Table C-2 offsite (external) communications capability

Table C-2 Communications Systems

System	**Onsite** (internal)	**Offsite** (external)
Unit 1, Unit 2 and TSC Radio Consoles	X	X
DCPP Telephone System (PBX)	X	X
Portable radio equipment (handie-talkies)	X	
Operations Radio System	X	X
Security Radio Systems	X	
Central Alarm Station (CAS) and Secondary Alarm Stations (SAS) Consoles	X	X
Fire Radio System	X	
Hot Shutdown Panel Radio Consoles	X	X
Public Address System	X	
NRC FTS		X
Mobile radios	X	
Satellite phones	X	X
Direct line (ATL) to the County and State Office of Emergency Services (OES)		X

Mode Applicability:

1 - Power Operation, 2 - Startup, 3 - Hot Standby, 4 - Hot Shutdown

Basis:

This EAL addresses loss of communications capability that either prevents the plant operations staff from performing routine tasks necessary for onsite plant operations or inhibits the ability to communicate problems externally to offsite authorities from the Control Room. The loss of offsite communications ability encompasses the loss of all means of communications with offsite authorities and is expected to be significantly more comprehensive than the condition addressed by 10 CFR 50.72.

The availability of one method of ordinary offsite communications is sufficient to inform state and local authorities of plant problems. This should include ENS, FAX transmissions and dedicated phone systems. This EAL is applicable only when extraordinary means are being utilized to make communications possible (e.g., relaying of information from radio transmissions, individuals being sent to offsite locations, etc.).

Onsite/offsite communications include one or more of the systems listed in Table C-2.

DCPP Basis Reference(s):

1. UFSAR, Section 9.5.2
2. AR PK15-23, "Communications"
3. STP I-29, "Emerg. Signals and Communications Sys. Functional Test"
4. OP K-9, "Instructions for Operation of Diablo Canyon Radio Systems"

Category: S – System Malfunction

Sub-category: 4 – Instrumentation / Communications

Initiating Condition: Unplanned loss of most or all safety system annunciation or indication in control room with either (1) a significant transient in progress, or (2) compensatory non-alarming indicators are unavailable

EAL:

SA4.1 **Alert**

Unplanned loss of greater than approximately 75% of the annunciators or indicators associated with safety systems on Vertical Boards 1 through 5, and Control Consoles 1, 2 and 3 for > 15 min.
AND EITHER:

A significant transient is in progress

OR

Compensatory non-alarming indications (PPC, SPDS) are unavailable

Mode Applicability:

1 - Power Operation, 2 - Startup, 3 - Hot Standby, 4 - Hot Shutdown

Basis:

The listed vertical boards and control consoles are all of the front panels in the Control Room. Each panel contains some annunciators or indicators that would be desirable to facilitate conduct of safe plant operations. Annunciators or indicators for this EAL include those identified in the Abnormal Operating Instructions, the Emergency Operating Instructions and other EALs (e.g., area, process, and/or effluent rad monitors, etc.).

This EAL recognizes the difficulty associated with monitoring changing plant conditions without the use of a major portion of the annunciation or indication equipment during a transient.

"Unplanned" loss of annunciators or indicators does not include scheduled maintenance and testing activities.

If approximately 75% of the safety system annunciators or indications are lost, an elevated risk exists that a degraded plant condition may be undetected. A detailed count of the lost instrumentation is not required. The judgment of the Shift Manager, however, should be used as the threshold for determining the severity of the plant conditions.

Plant design provides redundant safety system indication powered from separate uninterruptible power supplies. While failure of a large portion of annunciators is more likely than a failure of a large portion of indications, failure of indications is included in this EAL due to difficulty associated with assessment of plant conditions. The loss of several safety system indicators should remain a function of the specific system or component operability status and will be addressed by the applicable Technical Specification. The initiation of a Technical Specification imposed plant shutdown related to instrument loss must be reported via 10 CFR 50.72.

"Significant transient" includes response to automatic or manually initiated functions such as reactor trips, safety injection activation, runbacks involving greater than 25% thermal power change, electrical load rejection greater than 25% full electrical load, or thermal power oscillations of 10% or greater.

If both a major portion of the annunciation system and all computer monitoring capability (i.e., PPC and SPDS) are unavailable, the Alert declaration is required. It is not intended that the Shift Manager be tasked with making a judgment decision as to whether additional personnel are required to provide increased monitoring of system operation. The compensatory indications include:

- Plant Process Computer (PPC): The purpose of the PPC system is primarily to serve as a data collection device and calculation processor for Nuclear Steam Supply System (NSSS) parameters. The PPC allows the display and trend of both current and historical plant operating data. The PPC updates, alarm processes, and stores current values of plant operating data.
- Safety Parameter Display System (SPDS): The principle purpose of the SPDS system is to provide a display of plant parameters from which the safety status of operation may be assessed. It aids the Control Room personnel during abnormal and emergency conditions in determining the safety status of the plant and assessing whether abnormal conditions warrant corrective actions.

Due to the limited number of safety systems in operation during Cold Shutdown, Refueling and Defueled modes, this EAL is not applicable during these modes of operation. If the operating crew cannot monitor the transient in progress, the Alert escalates to a Site Area Emergency under EAL SS4.1.

DCPP Basis Reference(s):

1. UFSAR, Section 7.5.1.10

Category: S – System Malfunction

Sub-category: 4 – Instrumentation / Communications

Initiating Condition: Inability to monitor a significant transient in progress

EAL:

SS4.1 Site Area Emergency

Loss of greater than approximately 75% of the annunciators or indicators associated with safety systems on Vertical Boards 1 through 5, and Control Consoles 1, 2 and 3

AND

Significant transient is in progress

AND

Compensatory non-alarming indications (PPC, SPDS) are unavailable

Mode Applicability:

1 - Power Operation, 2 - Startup, 3 - Hot Standby, 4 - Hot Shutdown

Basis:

The listed vertical boards and control consoles are all of the front panels in the Control Room. Each panel contains some annunciators or indicators that would be desirable to facilitate conduct of safe plant operations. Annunciators or indicators for this EAL include those identified in the Abnormal Operating Instructions, the Emergency Operating Instructions and other EALs (e.g., area, process, and/or effluent rad monitors, etc.).

This EAL recognizes the inability of the Control Room staff to monitor the plant response to a significant transient. A Site Area Emergency exists if the Control Room staff cannot monitor safety functions needed for protection of the public.

If approximately 75% of the safety system annunciators or indications are lost, an elevated risk exists that a degraded plant condition may be undetected. A detailed count of the lost instrumentation is not required. The judgment of the Shift Manager, however, should be used as the threshold for determining the severity of the plant conditions.

EOPs are entered if a significant transient is in progress. The hierarchy on controlling and maintaining safety functions within acceptance criteria are specified therein and include the following:

- Reactivity control (ability to shut down the reactor and keep it shutdown)
- RCS inventory (ability to cool the core)
- Secondary heat removal (ability to maintain a heat sink)
- Spent Fuel Pool cooling (ability to remove decay heat from irradiated fuel in storage)

"Significant transient" includes response to automatic or manually initiated functions such as reactor trips, safety injection activation, runbacks involving greater than 25% thermal power change, electrical load rejection greater than 25% full electrical load, or thermal power oscillations of 10% or greater.

Indications needed to monitor safety functions necessary for protection of the public must include Control Room indications, computer generated indications (i.e., PPC and SPDS) and dedicated annunciation capability. The specific indications should be those used to determine such functions as the ability to shut down the reactor, maintain the core cooled and in a coolable geometry, remove heat from the core, and maintain the reactor coolant system and containment intact.

The compensatory indications include:

- Plant Process Computer (PPC): The purpose of the PPC system is primarily to serve as a data collection device and calculation processor for Nuclear Steam Supply System (NSSS) parameters. The PPC allows the display and trend of both current and historical plant operating data. The PPC updates, alarm processes, and stores current values of plant operating data.

- Safety Parameter Display System (SPDS): The principle purpose of the SPDS system is to provide a display of plant parameters from which the safety status of operation may be assessed. It aids the Control Room personnel during abnormal and emergency conditions in determining the safety status of the plant and assessing whether abnormal conditions warrant corrective actions.

Planned actions are included in the EAL since a loss of instrumentation of this magnitude is of such significance during a transient that the cause of the loss is not an ameliorating factor.

DCPP Basis Reference(s):

1. UFSAR, Section 7.5.1.10
2. F-0, "Critical Safety Function Status Trees", Attachments 1-7

Category: S – System Malfunction

Sub-category: 5 – Fuel Clad Degradation

Initiating Condition: Fuel Clad degradation

EAL:

SU5.1	**Unusual Event**
With letdown in service, EP-RB-14A Dose Point radiation > 3 R/hr	

Mode Applicability:

1 - Power Operation, 2 - Startup, 3 - Hot Standby, 4 - Hot Shutdown

Basis:

Initial indication of Fuel Clad degradation can be determined by measuring the external radiation dose rate at a distance of one foot from the center of the letdown line in the letdown heat exchanger room using the technique described in Attachment 7.1 of EP RB-14A, Initial Detection of Core Damage. An external radiation dose rate exceeding 3 R/hr indicates Fuel Clad degradation greater than Technical Specification allowable limits. This value was determined by ratioing 15 R/hr which corresponds to coolant activity at 300 µCi/gm (see FPB FC Loss C.4) to the Technical Specification LCO coolant activity of 60 µCi/gm which includes iodine spike (see EAL SU5.1), or 15 R/hr x 60/300 = 3 R/hr.

Escalation of this EAL to the Alert level is via the EALs in Category F.

DCPP Basis Reference(s):

1. EP RB-14A, "Initial Detection of Fuel Damage - RCS Letdown"
2. PG&E Calculation EP 95-02 Rev. 0, "Letdown Heat Exchanger Rom Dose Rates Corresponding to EP G-1, Alert No. 2 RCS Activity"
3. Technical Specifications 3.4.16

Category: S – System Malfunction

Sub-category: 5 – Fuel Clad Degradation

Initiating Condition: Fuel Clad degradation

EAL:

SU5.2	**Unusual Event**

Coolant activity > 60 µCi/gm Dose Equivalent I-131

OR

Coolant activity > 600.0 µCi/gm Dose Equivalent Xe-133

Mode Applicability:

1 - Power Operation, 2 - Startup, 3 - Hot Standby, 4 - Hot Shutdown

Basis:

Elevated reactor coolant activity represents a potential degradation in the level of safety of the plant and a potential precursor of more serious problems. This EAL addresses reactor coolant samples exceeding coolant Technical Specifications. The Technical Specification LCO limits are established to minimize the offsite radiological dose consequences in the event of a steam generator tube rupture (SGTR) accident. The allowable level is intended to limit the two-hour dose at the Exclusion Area Boundary to a small fraction of the 10 CFR 100 dose guideline limits. Permitting power operation to continue for limited time periods with the primary coolant specific activity greater than 1.0 µCi/gm Dose Equivalent I-131, but not in excess of either 60 µCi/gm Dose Equivalent I-131 or 600.0 µCi/gm Dose Equivalent Xe-133, accommodates the possible iodine spiking phenomenon which may occur following changes in thermal power.

Escalation of this EAL to the Alert level is via the EALs in Category F.

DCPP Basis Reference(s):

1. Technical Specifications 3.4.16

Category: S – System Malfunction

Sub-category: 6 – RCS Leakage

Initiating Condition: RCS leakage

EAL:

SU6.1 **Unusual Event**

Unidentified or pressure boundary leakage > 10 gpm

OR

Identified leakage > 25 gpm

Mode Applicability:

1 - Power Operation, 2 - Startup, 3 - Hot Standby, 4 - Hot Shutdown

Basis:

The conditions of this EAL may be a precursor of more serious conditions and, as a result, is considered to be a potential degradation of the level of safety of the plant. Manual or computer-calculated water balance inventory methods are normally used to determine RCS leakage. Symptoms associated with reactor coolant leakage include one or more of the following:

- Air Binding of the operating RHR pump as indicated by any of the following:
  - Motor current oscillations
  - Erratic flow oscillations
  - Excessive pump noise
  - Pump cavitation
- Increasing level indications in the PRT
- Increasing level indications in the RCDT
- Increasing level indications in the CCW surge tank
- Possible annunciator alarms:
  - CCW surge tanks
  - RHR System
  - RHR pumps
  - High radiation
  - Reactor Vessel refueling level
  - PRT tank pressure, level, temperature
  - Containment Structure sump
  - Reactor Cavity sump
- Decreasing VCT level
- Decreasing RVRLIS indication
- Local indications of abnormal flows to building sumps

Identified leakage is defined in Technical Specifications as:

- Leakage from pump seals or valve packing (except reactor coolant pump seal water injection or leakoff) that is captured and conducted to collection systems or a sump or collecting tank.
- Leakage into the containment atmosphere from sources that are both specifically located and known either not to interfere with the operation of leakage detection systems or not to be pressure boundary leakage.
- RCS leakage through a steam generator to the Secondary System.

Unidentified leakage is all leakage (except RCP seal water injection or leakoff) that is not identified leakage. Pressure Boundary leakage is leakage (except SG leakage) through a nonisolable fault in an RCS component body, pipe wall, or vessel wall.

Generally, leakage into closed systems, or leakage into the containment atmosphere from sources that are both specifically located and known either not to interfere with the operation of the unidentified leakage monitoring systems or not to be from a flow in the reactor coolant pressure boundary, are called identified leakages.

Uncontained leakage to the containment atmosphere may be the result of a variety of possible leakages that are generally classified as unidentified leakages. Unidentified leakage is eventually collected in tanks or sumps where the flowrate can be established and monitored during operation.

The 10 gpm value for the unidentified leakage and pressure boundary leakage was selected because it is quantifiable with normal Control Room leak detection methods. STP R-10C, Reactor Coolant System Water Inventory Balance, is performed to determine the source and flow rate of the leakage. Steam Generator leakage is also considered when evaluating unidentified leakage. The 25 gpm value for identified leakage is set at a higher value because of the significance of identified leakage in comparison to unidentified or pressure boundary leakage.

Escalation of this EAL to the Alert level is via the EALs in Category F.

DCPP Basis Reference(s):

1. UFSAR, Section 5.2.7
2. Technical Specifications Section 1.1
3. OP AP-1, "Excessive Reactor Coolant System Leakage"
4. STP R-10, "Reactor Coolant System Leakage Evaluation"
5. STP I-1A, "Routine Shift Checks Required by Licenses"
6. STP I-1B, "Routine Daily Checks Required by Licenses"
7. STP R-10C, "Reactor Coolant System Water Inventory Balance"
8. STP R-10D, "RCP #3 Seal LEAKAGE Evaluation"

Category:	S – System Malfunction
Sub-category:	7 – Inadvertent Criticality
Initiating Condition:	Inadvertent criticality

EAL:

SU7.1	**Unusual Event**
An unplanned sustained positive startup rate observed on nuclear instrumentation	

Mode Applicability:

3 - Hot Standby, 4 - Hot Shutdown

Basis:

Sustained is defined as prolonged, not intermittent or of transitory nature.

This IC addresses inadvertent criticality events. While the primary concern of this IC is criticality events that occur in Cold Shutdown or Refueling modes (NUREG 1449, "Shutdown and Low-Power Operation at Commercial Nuclear Power Plants in the United States"), the IC is applicable in other modes in which inadvertent criticalities are possible. This IC indicates a potential degradation of the level of safety of the plant, warranting an Unusual Event classification. This IC excludes inadvertent criticalities that occur during planned reactivity changes associated with reactor startups (e.g., criticality earlier than estimated). The Cold Shutdown/Refueling IC is CU6.

This condition can be identified using:

- NI-31D/32D and NI-35D/36D
- Individual SUR meters on CC-1 that receive signals from Source Range Channels N-31 and N-32 and Intermediate Range Channels NI-35 and NI-36
- Local meters on the drawer on NIS Rack IV
- Audio Count Rate (NIS Rack IV)
- NR-45 is a video recorder located on CC-1 that is used for monitoring and recording source range, Intermediate range and/or power range channel output.

The term "sustained" is used in order to allow exclusion of expected short-term positive startup rates from planned fuel bundle or control rod movements during core alteration. These short-term positive startup rates are the result of the increase in neutron population due to subcritical multiplication.

DCPP Basis Reference(s):

1. Plant Drawing no. 521130
2. UFSAR, Table 7.5-3
3. OP L-2, "Hot Standby to Startup Mode"
4. STP I-4C, "Calibration of Audio Count Rate/Scaler Timer Channel"

Category F – Fission Product Barriers

EALs in this category represent threats to the defense in depth design concept that precludes the release of highly radioactive fission products to the environment. This concept relies on multiple physical barriers any one of which, if maintained intact, precludes the release of significant amounts of radioactive fission products to the environment. The primary fission product barriers are:

A. Reactor Fuel Clad (FC): The zircalloy or stainless steel tubes that contain the fuel pellets.

B. Reactor Coolant System (RCS): Includes the RCS primary side and its connections up to and including the pressurizer safety and relief valves, and other connections up to and including the primary isolation valves.

C. Containment (CNMT): The Containment Barrier includes the containment building, its connections up to and including the outermost containment isolation valves. This barrier also includes the main steam, feedwater, and blowdown line extensions outside the containment building up to and including the outermost secondary side isolation valve..

The EALs in this category require evaluation of the loss and Potential Loss thresholds listed in the fission product barrier matrix of Table F-1 (Attachment 2). "Loss" and "Potential Loss" signify the relative damage and threat of damage to the barrier. "Loss" means the barrier no longer assures containment of radioactive materials. "Potential Loss" means integrity of the barrier is threatened and could be lost if conditions continue to degrade. The number of barriers that are lost or potentially lost and the following criteria determine the appropriate emergency classification level:

Unusual Event:

Any loss or any potential loss of Containment

Alert:

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

Site Area Emergency:

Loss or potential loss of any two barriers

General Emergency:

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

Category: Fission Product Barriers

Sub-category: N/A

Initiating Condition: Any loss or any potential loss of Containment

EAL:

FU1.1	**Unusual Event**
ANY loss or ANY potential loss of Containment (Table F-1) (Note 5)	

Note 5: 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. UE EALs associated with RCS and Fuel Clad barriers are addressed under System Malfunction EALs.
- 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 offsite 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 SM/SEC/ED would have more assurance that there was no immediate need to escalate to a General Emergency.
- The ability to escalate to higher emergency classes as an event deteriorates must be maintained. For example, RCS leakage steadily increasing would represent an increasing risk to public health and safety.

Mode Applicability:

1 - Power Operation, 2 - Startup, 3 - Hot Standby, 4 - Hot Shutdown

Basis:

Fuel Clad, RCS and Containment comprise the fission product barriers. Table F-1 (Attachment 2) lists the fission product barrier thresholds, bases and references.

Fuel Clad and RCS barriers are weighted more heavily than the Containment barrier. Unlike the Fuel Clad and RCS barriers, the loss of either of which results in an Alert (EAL FA1.1), loss of the Containment barrier in and of itself does not result in the relocation of radioactive materials or the potential for degradation of core cooling capability. However, loss or potential loss of the Containment barrier in combination with the loss or potential loss of either the Fuel Clad or RCS barrier results in declaration of a Site Area Emergency under EAL FS1.1.

DCPP Basis Reference(s):

None

Category: Fission Product Barriers

Sub-category: N/A

Initiating Condition: Any loss or any potential loss of either Fuel Clad or RCS

EAL:

FA1.1	**Alert**
ANY loss or ANY potential loss of either Fuel Clad or RCS (Table F-1) (Note 5)	

Note 5: 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. UE EALs associated with RCS and Fuel Clad barriers are addressed under System Malfunction EALs.
- 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 offsite 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 SM/SEC/ED would have more assurance that there was no immediate need to escalate to a General Emergency.
- The ability to escalate to higher emergency classes as an event deteriorates must be maintained. For example, RCS leakage steadily increasing would represent an increasing risk to public health and safety.

Mode Applicability:

1 - Power Operation, 2 - Startup, 3 - Hot Standby, 4 - Hot Shutdown

Basis:

Fuel Clad, RCS and Containment comprise the fission product barriers. Table F-1 (Attachment 2) lists the fission product barrier thresholds, bases and references.

At the Alert classification level, Fuel Clad and RCS barriers are weighted more heavily than the Containment barrier. Unlike the Containment barrier, loss or potential loss of either the Fuel Clad or RCS barrier may result in the relocation of radioactive materials or degradation of core cooling capability. Note that the loss or potential loss of Containment barrier in combination with loss or potential loss of either Fuel Clad or RCS barrier results in declaration of a Site Area Emergency under EAL FS1.

DCPP Basis Reference(s):

None

Category: Fission Product Barriers

Sub-category: N/A

Initiating Condition: Loss or potential loss of any two barriers

EAL:

FS1.1	**Site Area Emergency**
Loss or potential loss of ANY two barriers (Table F-1) (Note 5)	

Note 5: 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. UE EALs associated with RCS and Fuel Clad barriers are addressed under System Malfunction EALs.
- 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 offsite 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 SM/SEC/ED would have more assurance that there was no immediate need to escalate to a General Emergency.
- The ability to escalate to higher emergency classes as an event deteriorates must be maintained. For example, RCS leakage steadily increasing would represent an increasing risk to public health and safety.

Mode Applicability:

1 - Power Operation, 2 - Startup, 3 - Hot Standby, 4 - Hot Shutdown

Basis:

Fuel Clad, RCS and Containment comprise the fission product barriers. Table F-1 (Attachment 2) lists the fission product barrier thresholds, bases and references.

At the Site Area Emergency classification level, each barrier is weighted equally. A Site Area Emergency is therefore appropriate for any combination of the following conditions:

- One barrier loss and a second barrier loss (i.e., loss - loss)
- One barrier loss and a second barrier potential loss (i.e., loss - potential loss)
- One barrier potential loss and a second barrier potential loss (i.e., potential loss - potential loss)

At the Site Area Emergency classification level, the ability to dynamically assess the proximity of present conditions with respect to the threshold for a General Emergency is important. For example, the existence of Fuel Clad and RCS Barrier loss thresholds in addition to offsite dose assessments would require continual assessments of radioactive inventory and Containment integrity in anticipation of reaching a General Emergency classification. Alternatively, if both Fuel Clad and RCS potential loss thresholds existed, the SM/SEC/ED would have greater assurance that escalation to a General Emergency is less imminent.

DCPP Basis Reference(s):

None

Category: Fission Product Barriers

Sub-category: N/A

Initiating Condition: Loss of any two barriers and loss or potential loss of third barrier

EAL:

FG1.1 General Emergency

Loss of ANY two barriers

AND

Loss or potential loss of third barrier (Table F-1) (Note 5)

Note 5: 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. UE EALs associated with RCS and Fuel Clad barriers are addressed under System Malfunction EALs.
- 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 offsite 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 SM/SEC/ED would have more assurance that there was no immediate need to escalate to a General Emergency.
- The ability to escalate to higher emergency classes as an event deteriorates must be maintained. For example, RCS leakage steadily increasing would represent an increasing risk to public health and safety.

Mode Applicability:

1 - Power Operation, 2 - Startup, 3 - Hot Standby, 4 - Hot Shutdown

Basis:

Fuel Clad, RCS and Containment comprise the fission product barriers. Table F-1 (Attachment 2) lists the fission product barrier thresholds, bases and references.

At the General Emergency classification level each barrier is weighted equally. A General Emergency is therefore appropriate for any combination of the following conditions:

- Loss of Fuel Clad, RCS and Containment barriers
- Loss of Fuel Clad and RCS barriers with potential loss of Containment barrier
- Loss of RCS and Containment barriers with potential loss of Fuel Clad barrier
- Loss of Fuel Clad and Containment barriers with potential loss of RCS barrier

DCPP Basis Reference(s):

None

Introduction

Table F-1 lists the threshold conditions that define the Loss and Potential Loss of the three fission product barriers (Fuel Clad, Reactor Coolant System, and Containment). The table is structured so that each of the three barriers occupies adjacent columns. Each fission product barrier column is further divided into two columns; one for Loss thresholds and one for Potential Loss thresholds.

The first column of the table (to the left of the Fuel Clad Loss column) lists the categories (types) of fission product barrier thresholds. The fission product barrier categories are:

A. Critical Safety Function Status

B. Core Exit TCs

C. Radiation

D. Inventory

E. Other

F. Judgment

Each category occupies a row in Table F-1 thus forming a matrix defined by the category rows and the Loss/Potential Loss columns. The intersection of each category row with each Loss/Potential Loss column forms a cell in which one or more fission product barrier thresholds appear. If NEI 99-01 does not define a threshold for a barrier Loss/Potential Loss, the word "None" is entered in the cell.

Thresholds are assigned sequential numbers within each Loss and Potential Loss column beginning with number one. In this manner, a threshold can be identified by its category title and number. For example, the first Fuel Clad barrier Loss in Category A would be assigned "FC Loss A.1," the third Containment barrier Potential Loss would be assigned "CNMT P-Loss B.3," etc.

If a cell in Table F-1 contains more than one numbered threshold, each of the numbered thresholds, if exceeded, signifies a Loss or Potential Loss of the barrier. It is not necessary to exceed all of the thresholds in a category before declaring a barrier Loss/Potential Loss.

Subdivision of Table F-1 by category facilitates association of plant conditions to the applicable fission product barrier Loss and Potential Loss thresholds. This structure promotes a systematic approach to assessing the classification status of the fission product barriers.

When equipped with knowledge of plant conditions related to the fission product barriers, the EAL-user first scans down the category column of Table F-1, locates the likely category and then reads across the row of fission product barrier Loss and Potential Loss thresholds in that category to determine if any threshold has been exceeded. If a threshold has not been exceeded in that category row, the EAL-user proceeds to the next likely category and continues review of the row of thresholds in the new category.

If the EAL-user determines that a Loss threshold has been exceeded, a check mark is placed in the box at the top of the Loss column for the given barrier. This signifies that the barrier is lost. Similarly, this is done for a Potential Loss threshold that has been exceeded. The EAL-user must examine each of the three fission product barriers to determine if other barrier thresholds in the category are lost or potentially lost. For example, if Containment radiation is sufficiently high (i.e., greater than 80 R/hr), a Loss of the Fuel Clad and RCS barriers and a Potential Loss of the Containment barrier exist. Barrier Losses and Potential Losses are then applied to the algorithms given in EALs FG1.1, FS1.1, FA1.1 and FU1.1 to determine the appropriate emergency classification. The algorithms are illustrated in the following flow diagram.

In the remainder of this Attachment, the Fuel Clad barrier threshold bases appear first, followed by the RCS barrier and finally the Containment barrier threshold bases.

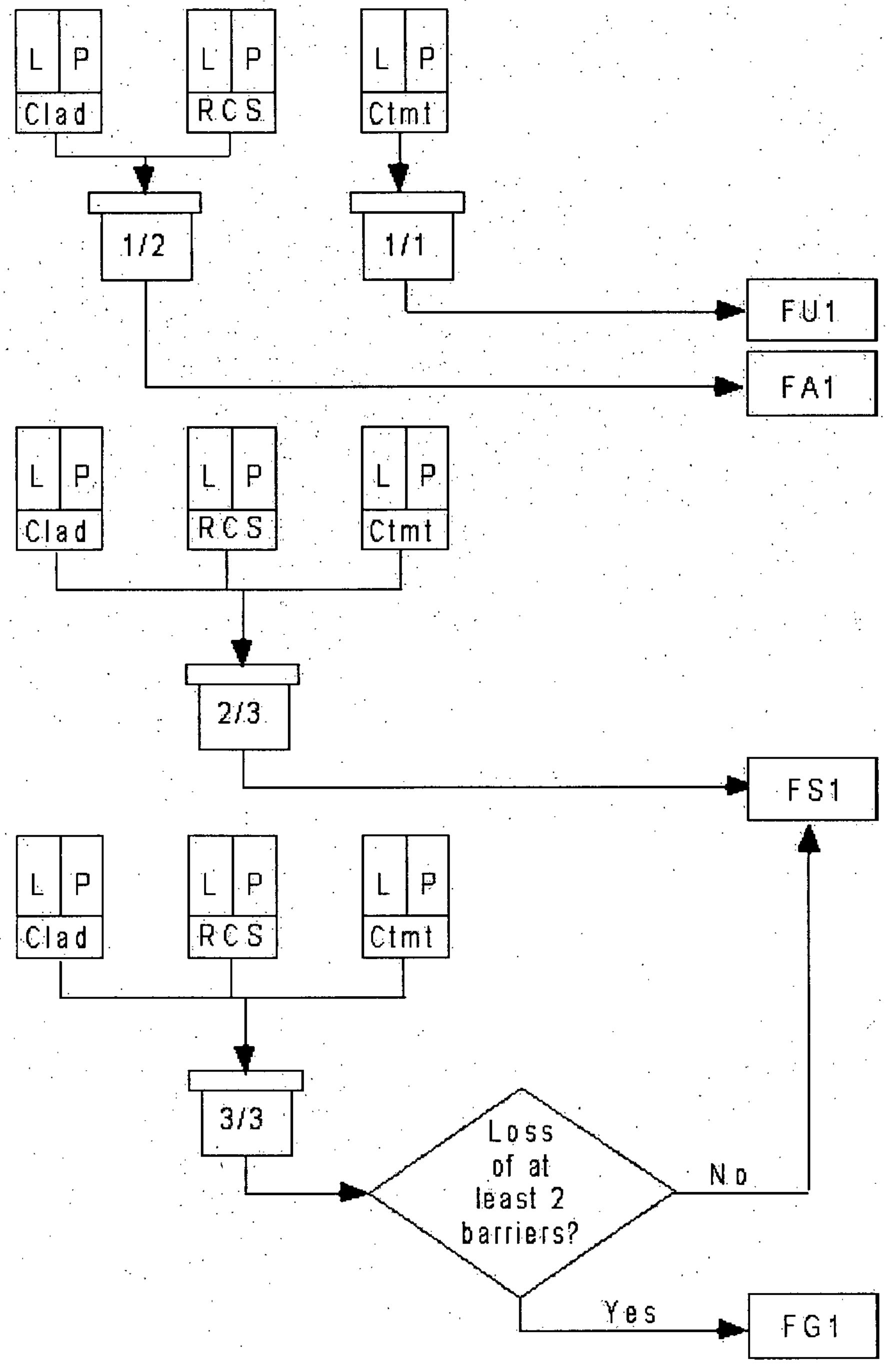

L
P
Clad
L
P
RCS
L
P
Ctmt
1/2
1/1
FU1
FA1
L
P
Clad
L
P
RCS
L
P
Ctmt
2/3
FS1
L
P
Clad
L
P
RCS
L
P
Ctmt
3/3
Loss of at least 2 barriers?
No
Yes
FG1

Table F-1 Fission Product Barrier Matrix

	Fuel Cladding Barrier		Reactor Coolant System Barrier		Containment Barrier	
	☐ Loss	☐ Potential Loss	☐ Loss	☐ Potential Loss	☐ Loss	☐ Potential Loss
A.CSFST	☐ 1. CSFST Core Cooling-RED	☐ 1. CSFST Core Cooling-MAGENTA OR CSFST Heat Sink-RED and heat sink required	None	☐ 1. CSFST RCS Integrity-RED OR CSFST Heat Sink-RED and heat sink required	None	☐ 1. CSFST Containment-RED
B.Core Exit TCs	☐ 2. Core exit TCs > 1,200°F	☐ 2. Core exit TCs > 700°F	None	None	None	☐ 2. Core exit TCs > 1,200°F AND Restoration procedures <u>not</u> effective within 15 min. ☐ 3. ALL of the following: - Core exit TCs > 700°F - Reactor Vessel water level < Table F-2 thresholds - Restoration procedures <u>not</u> effective within 15 min.
C.Radiation	☐ 3. Containment radiation (RM-30 or RM-31) > 20 R/hr ☐ 4. With letdown in service, EP-RB-14A Dose Point radiation > 15 R/hr	None	☐ 1. Containment radiation (RM-30 or RM-31) > 6 R/hr	None	None	☐ 4. Containment radiation (RM-30 or RM-31) > 80 R/hr
D.Inventory	☐ 5. SGTR in progress AND MSL radiation (RM-71, 72, 73 or 74) > 5.0E4 cpm (> 5 min. after reactor shutdown)	☐ 3. Reactor Vessel water level < Table F-2 thresholds	☐ 2. RCS leak rate > available makeup capacity as indicated by a loss of RCS subcooling ☐ 3. SGTR that results in an ECCS (SI) actuation	☐ 2. Unisolable RCS leak exceeding the capacity of one charging pump in the normal charging mode (160 gpm)	☐ 1. Rapid unexplained Containment pressure drop following initial increase ☐ 2. Following LOCA, Containment pressure or sump level response <u>not</u> consistent with LOCA conditions ☐ 3. Ruptured S/G is also faulted outside of Containment ☐ 4. *Primary-to-secondary leakage > 10 gpm with* non-isolable steam release from affected S/G to the environment	☐ 5. Containment pressure 47 psig and increasing ☐ 6. Containment hydrogen concentration > 4% ☐ 7. Containment pressure > 22 psig with < one full train of depressurization equipment operating Note: One Containment Spray pump and two CFCUs comprise one full train of depressurization equipment
E.Other	☐ 6. Coolant activity > 300 μCi/gm Dose Equivalent I-131	None	None	None	☐ 5. Valve(s) <u>not</u> closed AND Direct pathway to the environment exists after Containment isolation signal	None
F. Judgment	☐ 7. ANY condition in the opinion of the ISEC/SEC/RM that indicates loss of the Fuel Clad barrier	☐ 4. ANY condition in the opinion of the ISEC/SEC/RM that indicates potential loss of the Fuel Clad barrier	☐ 4. ANY condition in the opinion of the ISEC/SEC/RM that indicates loss of the RCS barrier	☐ 3. ANY condition in the opinion of the ISEC/SEC/RM that indicates potential loss of the RCS barrier	☐ 6. ANY condition in the opinion of the ISEC/SEC/RM that indicates loss of the Containment barrier	☐ 8. ANY condition in the opinion of the ISEC/SEC/RM that indicates potential loss of the Containment barrier

Table F-2 Reactor Vessel Water Level Thresholds

RVLIS	No. RCPs	Level
Full Range	None	32%
Dynamic Head	4	46%
	3	35%
	2	26%
	1	22%

Bases

Fuel Clad Barrier Potential Loss

1. CSFST Core Cooling-Magenta

OR

CSFST Heat Sink-Red and heat sink required

Critical Safety Function Status Tree (CSFST) Core Cooling-MAGENTA path is entered if core exit thermocouples (TCs) are less than 1200ºF, RCS core exit TCs subcooling margin (SCM} is less than 20ºF and any of the following:

- No RCPs are running and either: core exit TCs are less than 700ºF and RVLIS full range is less than or equal to 32%, or core exit TCs are greater than or equal to 700ºF and RVLIS full range is greater than 32%.
- At least one RCP is running and Reactor Vessel water level is less than or equal to RVLIS dynamic head readings in Table F-2.

These conditions indicate subcooling has been lost and that some Fuel Clad damage may potentially occur.

CSFST Heat Sink-RED path is entered if all steam generator narrow range levels are less than or equal to the values indicated in EOP F-0, "Critical Safety Function Status Tree" (ref. 2) and total AFW flow is less than or equal to 435 gpm. The phrase "and heat sink required" precludes over-classification for conditions in which RCS pressure is less than SG pressure or Heat Sink-RED path entry was created by operator action directed by an Emergency Operating Procedure.

The combination of these conditions when heat sink is required indicates the ultimate heat sink function is under extreme challenge. This condition addresses loss of functions required for hot shutdown with the reactor at pressure and temperature and thus is a challenge of the Fuel Clad barrier. CSFST setpoints enclosed in brackets are used under adverse Containment conditions. Adverse Containment condition thresholds apply when Containment pressure is greater than or equal to 3 psig or Containment radiation is greater than or equal to:

- 10^5 R/hr (RI-30/31 PAM 2)
- 10^6 R (RR-30/31 PAM 1)

1.1 DCPP References

1.1.1 F-0, "Critical Safety Function Status Trees," Attachment 2, "Core Cooling"

1.1.2 F-0, "Critical Safety Function Status Trees," Attachment 3, "Heat Sink"

1.1.3 FR-C.2, "Response to Degraded Core Cooling"

1.1.4 FR-H.1, "Response to Loss of Secondary Heat Sink"

2. Core exit TCs > 700ºF

The core exit TC value corresponds to the temperature in the Core Cooling Critical Safety Function Status Tree (CSFST) MAGENTA path but is evaluated separately because the CSFST considers the degree of subcooling prior to status determination. This threshold is an explicit Fuel Clad potential loss to address conditions when the CSFSTs may not be in use (initiation after SI is blocked). This temperature indicates subcooling has been lost and that some Fuel Clad damage may occur.

The following criteria should be used to determine if five TCs have exceeded 700°F or 1200°F. Any TCs listed on PAMS 3/4 or SPDS may be used to determine an inadequate core cooling condition (ref. 2).

2.1 DCPP References:

2.1.1 F-0, "Critical Safety Function Status Trees," Attachment 2, "Core Cooling"

2.1.2 FR-C.1, "Response to Inadequate Core Cooling"

3. Reactor Vessel water level < Table F-2 thresholds

The Reactor Vessel water levels listed in Table F-2 are used in the EOPs to signal core uncovery and are, therefore, indications of inadequate coolant inventory. According to the Core Cooling-MAGENATA path, these water levels indicate subcooling has been lost and that some Fuel Clad damage may occur.

3.1 DCPP References:

3.1.1 F-0, "Critical Safety Function Status Trees," Attachment 2, "Core Cooling"

3.1.2 FR-C.1, "Response to Inadequate Core Cooling"

4. <u>ANY condition in the opinion of the SM/SEC/ED that indicates potential loss of the Fuel Clad barrier</u>

The SM/SEC/ED judgment threshold addresses any other factors relevant to determining if the Fuel Clad barrier is potentially lost. Such a determination should include imminent barrier degradation, barrier monitoring capability and dominant accident sequences.

- <u>Imminent barrier degradation</u> exists if the degradation will likely occur within two hours based on a projection of current safety system performance. The term "imminent" refers to recognition of the inability to reach safety acceptance criteria before completion of all checks.
- <u>Barrier monitoring</u> capability is decreased if there is a loss or lack of reliable indicators. This assessment should include instrumentation operability concerns, readings from portable instrumentation and consideration of offsite monitoring results.
- <u>Dominant accident sequences</u> lead to degradation of all fission product barriers and likely entry to the EOPs. The SM/SEC/ED should be mindful of the Loss of AC power (Station Blackout) and ATWS EALs to assure timely emergency classification declarations.

4.1 DCPP References:

4.1.1 E-0.1 U1(2), "Reactor Trip Response"

Fuel Clad Barrier Loss

1. CSFST Core-Cooling RED

Critical Safety Function Status Tree (CSFST) Core Cooling-RED path is entered if either:

- Core exit TCs are greater than or equal to 1200ºF, or
- Core exit TCs are greater than or equal to 700ºF with RCS subcooling margin (SCM) less than 20ºF, no RCPs are running, and RVLIS full range is less than or equal to 32%.

Either set of conditions indicates significant core exit superheating and core uncovery. This is considered a loss of the Fuel Clad barrier.

1.1 DCPP References:

1.1.1 F-0, "Critical Safety Function Status Trees," Attachment 2, "Core Cooling"

1.1.2 FR-C.1, "Response to Inadequate Core Cooling"

2. Core exit TCs > 1,200ºF

Core exit TC readings greater than 1200°F indicate significant core exit superheating and core uncovery. This is considered a loss of the Fuel Clad barrier.

The following criteria should be used to determine if five TCs have exceeded 700°F or 1200°F. Any TCs listed on PAMS 3/4 or SPDS may be used to determine an inadequate core cooling condition (ref. 2).

2.1 DCPP References:

2.1.1 F-0, "Critical Safety Function Status Trees," Attachment 2, "Core Cooling"

2.1.2 FR-C.1, "Response to Inadequate Core Cooling"

3. Containment radiation (RM-30 or RM-31) > 20 R/hr

Containment radiation monitor readings greater than 20 R/hr indicates the release of reactor coolant, with elevated activity indicative of fuel damage, into the Containment. The reading is derived assuming the instantaneous release and dispersal of the reactor coolant noble gas and iodine inventory associated with a concentration of 300 µCi/cc dose equivalent I-131 into the Containment atmosphere. Reactor coolant concentrations of this magnitude are several times larger than the maximum concentrations (including iodine spiking) allowed within technical specifications and are therefore indicative of fuel damage (approximately 5% clad failure depending on core inventory and RCS volume).

Monitors used for this fission product barrier loss threshold are the Containment high-range area monitors RM-30 and RM-31. These monitors provide indication in the Control Room on PAM 2 with a range of 1R/hr to 1E7 R/hr.

3.1 DCPP References:

3.1.1 AR, A0643677, AE40, "Evaluate EAL threshold for the potential loss of containment".

4. <u>With letdown in service, EP-RB-14A Dose Point radiation > 15 R/hr</u>

Initial indication of fuel damage can be determined by measuring the external radiation dose rate at a distance of one foot from the center of the letdown line in the letdown heat exchanger room in accordance with EP-RB-14A, Initial Detection of Core Damage. An external radiation dose rate exceeding 15 R/hr indicates fuel damage meeting the Alert emergency action level criterion of 300 μCi/cc I-131 dose equivalent RCS activity.

4.1 DCPP References:

4.1.1 EP RB-14A, "Initial Detection of Fuel Damage - RCS Letdown"

4.1.2 PG&E Calculation EP 95-02, Rev. 0

5. <u>SGTR in progress</u>

AND

MSL radiation (RM-71, 72, 73 or 74) > 5.0E4 cpm (> 5 min. after reactor shutdown)

In the special case of a steam generator tube rupture (SGTR) with a monitored steam release, indication of fuel damage can be determined by a sudden increase on the main steam line (MSL) monitor (RM-71 to RM-74). A main steam line monitor reading exceeding 5.0E4 cpm indicates Fuel Clad damage. A notable increase in dose rate on the Containment low range monitor (RE 2) should also be observed due to RCS piping shine. The monitor reading should be measured 5 minutes after reactor shutdown to eliminate MSL monitor response caused by N-16 (nitrogen-16).

If the SGTR causes an ECCS (SI) actuation (RCS loss #2) or the ruptured S/G is also faulted outside of Containment (Containment barrier loss #3), a Site Area Emergency declaration is warranted.

5.1 DCPP References:

5.1.1 EP RB-14A, "Initial Detection of Fuel Damage - RCS Letdown"

5.1.2 PG&E Calculation EP 95-02, Rev. 0

6. <u>Coolant activity > 300 µCi/gm Dose Equivalent I-131</u>

Elevated reactor coolant activity represents a potential degradation in the level of safety of the plant and a potential precursor of more serious problems. The threshold Dose Equivalent (DEQ) I-131 concentration is well above that expected for iodine spikes and corresponds to less than 5% Fuel Clad damage. When reactor coolant activity reaches this level, significant clad heating has occurred and thus the Fuel Clad barrier is considered lost.

6.1 DCPP References:

None

7. <u>ANY condition in the opinion of the SM/SEC/ED that indicates loss of the Fuel Clad barrier</u>

The SM/SEC/ED judgment threshold addresses any other factors relevant to determining if the Fuel Clad barrier is lost. Such a determination should include imminent barrier degradation, barrier monitoring capability and dominant accident sequences.

- <u>Imminent barrier degradation</u> exists if the degradation will likely occur within two hours based on a projection of current safety system performance. The term "imminent" refers to recognition of the inability to reach safety acceptance criteria before completion of all checks.
- <u>Barrier monitoring</u> capability is decreased if there is a loss or lack of reliable indicators. This assessment should include instrumentation operability concerns, readings from portable instrumentation and consideration of offsite monitoring results.
- <u>Dominant accident sequences</u> lead to degradation of all fission product barriers and likely entry to the EOPs. The SM/SEC/ED should be mindful of the Loss of AC power (Station Blackout) and ATWS EALs to assure timely emergency classification declarations.

7.1 DCPP References:

7.1.1 E-0.1 U1(2), "Reactor Trip Response"

Reactor Coolant System Barrier Potential Loss

8. CSFST RCS Integrity-RED

OR

CSFST Heat Sink-RED and heat sink required

Critical Safety Function Status Tree (CSFST) RCS Integrity-RED path is entered if no cold leg temperature decreases less than 100ºF in the last 60 minutes and any RCS pressure/cold leg temperature is to the left of the Reactor Vessel Pressure-Temperature Limit Curve A.

CSFST Heat Sink-RED path is entered if all S/G NR LVLs are less than or equal to the values indicated in EOP F-0, "Critical Safety Function Status Tree" (ref. 2) and total AFW flow is less than or equal to 435 gpm. The phrase "and heat sink required" precludes over-classification for conditions in which RCS pressure is less than SG pressure or Heat Sink-RED path entry was created by operator action directed by an Emergency Operating Procedure.

The combination of these conditions indicates the RCS barrier is under significant challenge. CSFST setpoints enclosed in brackets (e.g., [500 psig], etc.) are used under adverse containment conditions. CSFST setpoints enclosed in brackets are used under adverse Containment conditions. Adverse Containment condition thresholds apply when Containment pressure is greater than or equal to 3 psig or Containment radiation is greater than or equal to:

- 10^5 R/hr (RI-30/31 PAM 2)
- 10^6 R (RR-30/31 PAM 1)

8.1 DCPP References:

8.1.1 F-0, "Critical Safety Function Status Trees," Attachment 4, "RCS Integrity"

8.1.2 F-0, "Critical Safety Function Status Trees," Attachment 3, "Heat Sink"

8.1.3 FR-P.1, "Response to Imminent Pressurized Thermal Shock Condition"

8.1.4 FR-H.1, "Response to Loss of Secondary Heat Sink"

9. <u>**Unisolable RCS leak exceeding the capacity of one charging pump in the normal charging mode (150 gpm)**</u>

This threshold is based on the inability to maintain normal liquid inventory within the RCS by normal operation of the Chemical and Volume Control System (CVCS). The CVCS includes three charging pumps with a nominal flow capacity of 150 gpm. Charging flowrate is determined from a pressurizer level signal. The charging flow control is accomplished by a modulating valve on the discharge side of the centrifugal pumps. The charging pumps also serve as safety injection pumps in the ECCS.

The need for a second or third charging pump to make up leakage in excess of letdown flow would be indicative of substantial RCS leakage.

9.1 DCPP References:

9.1.1 UFSAR, Section 9.3.4.2

9.1.2 UFSAR, Table 9.3-6

10. <u>**ANY condition in the opinion of the SM/SEC/ED that indicates potential loss of the RCS barrier**</u>

The SM/SEC/ED judgment threshold addresses any other factors relevant to determining if the RCS barrier is potentially lost. Such a determination should include imminent barrier degradation, barrier monitoring capability and dominant accident sequences.

- <u>Imminent barrier degradation</u> exists if the degradation will likely occur within two hours based on a projection of current safety system performance. The term "imminent" refers to the inability to reach final safety acceptance criteria before completing all checks.
- <u>Barrier monitoring</u> capability is decreased if there is a loss or lack of reliable indicators. This assessment should include instrumentation operability concerns, readings from portable instrumentation and consideration of offsite monitoring results.
- <u>Dominant accident sequences</u> lead to degradation of all fission product barriers and likely entry to the EOPs. The SM/SEC/ED should be mindful of the Loss of AC power (Station Blackout) and ATWS EALs to assure timely emergency classification declarations.

10.1 DCPP References:

10.1.1 E-0.1 U1(2), "Reactor Trip Response"

Reactor Coolant System Barrier Loss

11. Containment radiation (RM-30 or RM-31) > 6 R/hr

The 6 R/hr reading is a value which indicates the release of reactor coolant to the Containment. The lowest indicated reading on RM-30/RM-31 is 1 R/hr. If the 6 R/hr threshold is exceeded, it must be assumed that the instantaneous release and uniform dispersal of the reactor coolant noble gas and iodine inventory into the Containment atmosphere at the Technical Specifications limit has occurred due to a LOCA. This reading is less than that specified for Fuel Clad barrier Loss C.3. Thus, this threshold would be indicative of an RCS leak only. If the radiation monitor reading increased to that specified by Fuel Clad Barrier Loss C.3, fuel damage would also be indicated.

There is no RCS Potential Loss threshold associated with Containment radiation.

Monitors used for this fission product barrier loss threshold are the Containment high-range area monitors RM-30 and RM-31. These monitors provide indication in the Control Room on PAM 2 with a range of 1R/hr to 1E7 R/hr.

11.1 DCPP References:

11.1.1 AR, A0643677, AE 40, "Evaluate EAL threshold for the potential loss of containment"

12. RCS leak rate > available makeup capacity as indicated by a loss of RCS subcooling

F-0, "Critical Safety Function Status Trees," Attachment 2, "Core Cooling," indicates that if subcooling margin based on core exit TCs is less than or equal to 20°F, a loss of RCS subcooling has occurred. The loss of subcooling is the fundamental indication that the inventory control systems are inadequate in maintaining RCS pressure and inventory against the mass loss through the leak. This threshold addresses conditions in which leakage from the RCS is greater than available inventory control capacity such that a loss of subcooling has occurred. OP AP-1, Excessive Reactor Coolant System 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 greater than 50°F. Subcooling margin greater than 20°F ensures the fluid surrounding the core is sufficiently cooled and provides margin for reestablishing flow should subcooling deteriorate when SI flow is secured. 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.

The loss of subcooling as a result of inability to establish RCS heat transfer to the ultimate heat sink is indicative of Potential Losses of the Fuel Clad and RCS barriers.

12.1 DCPP References:

12.1.1 F-0, "Critical Safety Function Status Trees," Attachment 2, "Core Cooling"

12.1.2 OP AP-1, "Excessive Reactor Coolant System Leakage"

13. SGTR that results in an ECCS (SI) actuation

In conjunction with Containment barrier Loss #3 and the Fuel Clad barrier thresholds, this threshold addresses the full spectrum of Steam Generator Tube Rupture (SGTR) events. To meet this threshold, the leakage must be large enough to cause actuation of ECCS (SI). ECCS (SI) actuation is caused by:

- Pressurizer Low pressure (<1850 psig)
- Low steam line pressure (<600 psig)
- Containment hi pressure (>3 psig)

This condition is described by "entry into E-3 required by EOPs". By itself, this RCS barrier Loss will result in the declaration of an Alert. However, if the SG is also faulted (i.e., two barriers failed), the declaration escalates to a Site Area Emergency per Containment barrier Loss D.3 or D.4.

13.1 DCPP References:

13.1.1 AR PK02-02, "SAFETY INJECTION INITIATE (RED)"

13.1.2 UFSAR, Section 6.3.1

13.1.3 Technical Specifications Table 3.3.2-1

14. ANY condition in the opinion of the SM/SEC/ED that indicates loss of the RCS barrier

The SM/SEC/ED judgment threshold addresses any other factors relevant to determining if the RCS barrier is lost. Such a determination should include imminent barrier degradation, barrier monitoring capability and dominant accident sequences.

- Imminent barrier degradation exists if the degradation will likely occur within two hours based on a projection of current safety system performance. The term "imminent" refers to the recognition of the inability to reach safety acceptance criteria before completion of all checks.
- Barrier monitoring capability is decreased if there is a loss or lack of reliable indicators. This assessment should include instrumentation operability concerns, readings from portable instrumentation and consideration of offsite monitoring results.
- Dominant accident sequences lead to degradation of all fission product barriers and likely entry to the EOPs. The SM/SEC/ED should be mindful of the Loss of AC power (Station Blackout) and ATWS EALs to assure timely emergency classification declarations.

14.1 DCPP References:

14.1.1 E-0.1 U1(2), "Reactor Trip Response"

Containment Barrier Potential Loss

15. CSFST Containment-RED

Critical Safety Function Status Tree (CSFST) Containment-Red path is entered if Containment pressure is equal to or greater than 47 psig. This pressure is the containment design pressure and is in excess of that expected from the design basis loss of coolant accident. This threshold is indicative of a loss of both RCS and Fuel Clad barriers in that it is not possible to reach this condition without severe core degradation (metal-water reaction) or failure to trip in combination with RCS breach. This combination of conditions would be expected to require the declaration of a General Emergency.

15.1 DCPP References:

15.1.1 F-0, "Critical Safety Function Status Trees", Attachment 6, "Containment"

15.1.2 UFSAR, Section 6.2.2.1

16. Core exit TCs > 1,200°F

AND

Restoration procedures <u>not</u> effective within 15 min.

This threshold indicates significant core exit superheating and core uncovery. If core exit thermocouple (TC) readings are greater than 1200°F, Fuel Clad barrier is lost. Core exit TCs provide an indirect indication of Fuel Clad temperature by measuring the temperature of the primary coolant that leaves the core region. Although clad rupture due to high temperature is not expected for core exit TC readings less than the threshold, temperatures of this magnitude signal significant superheating of the reactor coolant and core uncovery. Events that result in core exit TC readings above the loss threshold are severe accidents and are a severe accident Management "Badly Damaged (BD)" condition. The BD descriptor signifies possible core overheating to the point that clad ballooning/collapse may occur and portions of the core may have melted.

It must also be assumed the loss of RCS inventory is a result of a loss of the RCS barrier. These conditions, if not mitigated, can lead to core melt which in turn may result in a loss of Containment. Severe accident analyses (e.g., NUREG-1150) have concluded that function restoration procedures can arrest core degradation within the Reactor Vessel in a significant fraction of the core damage scenarios, and the likelihood of Containment failure is very small in these events. Given this, it is appropriate to provide a reasonable period to allow function restoration procedures to arrest the core melt sequence. The procedure is considered effective if the temperature is decreasing or if the vessel water level is increasing. The 15-minute period allows implementation of procedural guidance to restore RCS inventory. The SM/SEC/ED should make the declaration as soon as it is determined the guidance has not been or will not be effective.

The conditions in this potential loss threshold represent an imminent core melt sequence which, if not corrected, could lead to vessel failure and an increased potential for Containment failure. In conjunction with the Core Cooling and Heat Sink criteria in the Fuel Clad and RCS barrier thresholds, this threshold would result in the declaration of a General Emergency - loss of two barriers and the potential loss of a third. If the function restoration procedures are ineffective, there is no "success" path.

The following criteria should be used to determine if five TCs have exceeded 700°F or 1200°F. Any TCs listed on PAMS 3/4 or SPDS may be used to determine an inadequate core cooling condition (ref. 2).

16.1 DCPP References:

16.1.1 F-0, "Critical Safety Function Status Trees," Attachment 2, "Core Cooling"

16.1.2 FR-C.1, "Response to Inadequate Core Cooling"

17. <u>ALL of the following:</u>

- Core exit TCs > 700°F
- Reactor Vessel water level < Table F-2 thresholds
- Restoration procedures <u>not</u> effective within 15 min.

This threshold indicates significant core exit superheating (core exit TC readings >700°F) and core uncovery. It must be assumed that the loss of RCS inventory is a result of a loss of the RCS barrier. If RVLIS is reading greater than or equal to the Table F-2 thresholds, safety injection has been successful in restoring RCS inventory and core cooling. In the event that RVLIS reads less than Table F-2 thresholds, core cooling continues to be degraded. These conditions, if not mitigated, will likely lead to core melt which will in turn result in a challenge of Containment.

Severe accident analyses (e.g., NUREG-1150) have concluded that function restoration procedures can arrest core degradation within the Reactor Vessel in a significant fraction of the core damage scenarios, and that the likelihood of Containment failure is very small in these events. Given this, it is appropriate to provide a reasonable period to allow function restoration procedures to arrest the core melt sequence. The procedure is considered effective if the temperature is decreasing or if the vessel water level is increasing. Whether or not procedures will be effective should be apparent within 15 minutes. The SM/SEC/ED should make the declaration as soon as it is determined that the procedures have not been, or will not be effective.

The conditions in this potential loss threshold represent an imminent core melt sequence which, if not corrected, could lead to vessel failure and an increased potential for Containment failure. In conjunction with the Core Cooling and Heat Sink criteria in the Fuel Clad and RCS barrier thresholds, this threshold would result in the declaration of a General Emergency - loss of two barriers and the potential loss of a third. If the function restoration procedures are ineffective, there is no "success" path.

The following criteria should be used to determine if five TCs have exceeded 700°F or 1200°F. Any TCs listed on PAMS 3/4 or SPDS may be used to determine an inadequate core cooling condition (ref. 2).

17.1 DCPP References:

17.1.1 F-0, "Critical Safety Function Status Trees," Attachment 2, "Core Cooling"

17.1.2 FR-C.2, "Response to Inadequate Core Cooling"

18. <u>Containment radiation (RM-30 or RM-31) > 80 R/hr</u>

Containment radiation monitor readings greater than 80 R/hr indicates significant fuel damage well in excess of that required for the Fuel Clad barrier. NUREG-1228, "Source Estimations During Incident Response to Severe Nuclear Power Plant Accidents" states that such readings do not exist when the amount of clad damage is less than 20%. A major release of radioactivity requiring offsite protective actions from core damage is not possible unless a major failure into the reactor coolant has occurred. Regardless of whether the Containment barrier itself is challenged, this amount of activity in containment, if released, could have such severe consequences that it is prudent to treat this as a potential loss of Containment, such that a General Emergency declaration is warranted. It is, therefore, prudent to treat this as a Potential Loss of the Containment barrier. The readings are higher than that specified for Fuel Clad barrier Loss C.3. Containment radiation readings at or above this Containment barrier Potential Loss threshold, therefore, signify a loss of two fission product barriers and a potential loss of a third.

Monitors used for this fission product barrier loss threshold are the Containment high-range area monitors RM-30 and RM-31. These monitors provide indication in the Control Room on PAM 2 with a range of 1R/hr to 1E7 R/hr.

18.1 DCPP References:

18.1.1 AR, A0643677, AE 40, "Evaluate EAL threshold for the potential loss of containment"

19. <u>Containment pressure 47 psig and increasing</u>

This threshold is the containment design pressure and is in excess of that expected from the design basis loss of coolant accident (LOCA). Proper actuation and operation of the Containment heat removal system when required should maintain containment pressure well below the design pressure. The Containment response for the spectrum of LOCAs considered in the plant design basis is described in Section 6 of the UFSAR. The threshold is therefore indicative of a loss of both RCS and Fuel Clad barriers in that it should not be reached without severe core degradation (metal-water reaction) or failure to trip in combination with RCS breach. This condition would be expected to require the declaration of a General Emergency.

19.1 DCPP References:

19.1.1 F-0, "Critical Safety Function Status Trees," Attachment 6, "Containment"

19.1.2 UFSAR, Section 6.2.2.1

20. Containment hydrogen concentration > 4%

After a LOCA, the containment atmosphere is a homogeneous mixture of steam, air, solid and gaseous fission products, hydrogen, and water droplets. During and following a LOCA, the hydrogen concentration in the containment results from radiolytic decomposition of water and metal-water reaction. If hydrogen concentration exceeds the lower flammability limit (4%) in an oxygen rich environment, a potentially explosive mixture exists. Operation of the Containment Hydrogen Recombiner with Containment hydrogen concentrations greater than 4% could result in ignition of the hydrogen. If the combustible mixture ignites inside containment, loss of the Containment barrier could occur. To generate such levels of combustible gas, loss of the Fuel Clad and RCS barriers must also have occurred. Since this threshold is also indicative of loss of both Fuel Clad and RCS barriers with the Potential Loss of the Containment barrier, it therefore will likely warrant declaration of a General Emergency.

Containment hydrogen concentration is indicated in the Control Room on ANR-82/ANR-83 PAM1, (range: 1 - 10%).

20.1 DCPP References:

20.1.1 UFSAR, Section 6.25

20.1.2 Technical Specifications 3.6.3

20.1.3 FR-C.1, "Response to Inadequate Core Cooling"

20.1.4 OP H-9, "INSIDE CONT H2 RECOMB SYSTEM"

20.1.5 CA-3, "Hydrogen Flammability in Containment"

21. Containment pressure > 22 psig with less than one full train of depressurization equipment operating

This threshold represents a Potential Loss of the Containment barrier because the Containment heat removal and depressurization equipment (but not including Containment venting strategies) is either lost or degraded. The Containment Spray System and the Containment Cooling System are designed to rapidly reduce the Containment temperature and pressure and to maintain these parameters at acceptably low levels.

The Containment Spray System consists of two separate trains of equal capacity, each capable of meeting the design bases. Each train includes a Containment spray pump, spray headers, nozzles, valves, and piping. Each train is powered from a separate ESF bus. The refueling water storage tank (RWST) supplies borated water to the Containment Spray System during the injection phase of operation. In the recirculation mode of operation, Containment spray is supplied by manual realignment of the residual heat removal (RHR) pumps after the RWST is empty. Each train of the Containment Spray System provides adequate spray coverage to meet the system design requirements for Containment atmospheric heat removal.

There are two trains of cooling in the Containment Cooling System, each consisting of two Containment Fan Coil Units (CFCUs) and one shared CFCU for a total of five. The five CFCUs are powered from three separate vital buses, with two CFCUs on each of two vital buses and the remaining CFCU from the third vital bus. Each CFCU is supplied with cooling water from one of two separate loops of component cooling water (CCW). During normal operation, three CFCUs are operating. The CFCUs are designed to limit the ambient containment air temperature during normal unit operation to less than the limit specified in Technical Specifications LCO 3.6.5, Containment Air Temperature. This temperature limitation ensures that the containment temperature does not exceed the initial temperature conditions assumed for the design basis accidents.

The containment pressure setpoint (22 psig) is the pressure at which the equipment should actuate and begin performing its function. The design basis accident analyses and evaluations assume the operation of one Containment Spray train and two CFCUs. One Containment Spray train (with one pump running) and one CFCU train (with two CFCUs running) is defined to be "one full train of depressurization equipment." If less than this equipment is operating and containment pressure is above the actuation setpoint, the threshold is met.

21.1 DCPP References:

21.1.1 AR PK01-18, "CONTMT SPRAY ACTUATION red"

21.1.2 F-0, "Critical Safety Function Status Trees," Attachment 6, "Containment"

21.1.3 FR-Z.1, "Response to High Containment Pressure"

21.1.4 Technical Specifications Table 3.3.2-1

21.1.5 Technical Specifications B 3.6.6

21.1.6 Technical Specifications LCO 3.6.5

22. <u>ANY condition in the opinion of the SM/SEC/ED that indicates potential loss of the Containment barrier</u>

The SM/SEC/ED judgment threshold addresses any other factors relevant to determining if the Containment barrier is potentially lost. Such a determination should include imminent barrier degradation, barrier monitoring capability and dominant accident sequences.

- <u>Imminent barrier degradation</u> exists if the degradation will likely occur within two hours based on a projection of current safety system performance. The term "imminent" refers to recognition of the inability to reach safety acceptance criteria before completion of all checks.

- <u>Barrier monitoring</u> capability is decreased if there is a loss or lack of reliable indicators. This assessment should include instrumentation operability concerns, readings from portable instrumentation and consideration of offsite monitoring results.

- Dominant accident sequences lead to degradation of all fission product barriers and likely entry to the EOPs. The SM/SEC/ED should be mindful of the Loss of AC power (Station Blackout) and ATWS EALs to assure timely emergency classification declarations.

22.1 DCPP References:

22.1.1 E-0.1 U1(2), "Reactor Trip Response"

Containment Barrier Loss

23. Rapid unexplained Containment pressure drop following initial increase

Rapid unexplained loss of pressure (i.e., not attributable to Containment spray operation, running Containment Fan Cooling Units or condensation effects) following an initial pressure increase indicates a loss of both RCS and Containment integrity. UFSAR Section 6 describes Containment pressure response for LOCA events.

23.1 DCPP References:

23.1.1 UFSAR, Section 6.2.1

23.1.2 UFSAR, Figure 6.2-1

24. Following LOCA, Containment pressure or sump level response not consistent with LOCA conditions

This threshold addresses unexpected changes occurring in Containment pressure or sump level that is not explainable due to operator actions or automatic system actions. Containment pressure and sump levels should increase as a result of the mass and energy release into Containment from a LOCA. Thus, Containment pressure or sump levels not increasing indicate Containment bypass and a loss of Containment integrity. UFSAR, Section 6 describes containment pressure response for LOCA events.

24.1 DCPP References:

24.1.1 UFSAR, Section 6.2.1

24.1.2 UFSAR, Figure 6.2-1

24.1.3 UFSAR, Section 6.2.2

24.1.4 UFSAR, Figure 6.2-15

25. Ruptured S/G is also faulted outside of Containment

Steam Generator (S/G) tube leakage can represent a bypass of the Containment barrier as well as a loss of the RCS barrier. This threshold addresses the condition in which a ruptured S/G is also faulted and represents a bypass of the RCS and Containment barriers. A faulted S/G means the existence of secondary side leakage that results in an uncontrolled decrease in steam generator pressure or the steam generator being completely depressurized. A ruptured S/G means the existence of primary-to-secondary leakage of a magnitude sufficient to require or cause a reactor trip and safety injection. In conjunction with RCS barrier Loss C.3, this threshold would always result in the declaration of a Site Area Emergency.

25.1 DCPP References:

25.1.1 E-2 U1(2), "Faulted Steam Generator Isolation"

25.1.2 E-3 U1(2), "Steam Generator Tube Rupture"

26. Primary-to-secondary leakrate > 10 gpm with non-isolable steam release from affected S/G to the environment

Steam Generator (S/G) tube leakage can represent a bypass of the Containment barrier as well as a loss of the RCS barrier. This threshold represents a bypass of the RCS and Containment barriers. In conjunction with RCS barrier Loss C.3, this would always result in the declaration of a Site Area Emergency.

The threshold for establishing the non-isolable secondary side release is intended to be a prolonged release of radioactivity from the affected steam generator directly to the environment. This could be expected to occur when the main condenser is unavailable to accept the contaminated steam (i.e., SGTR with concurrent loss of offsite power and the ruptured steam generator is required for plant cooldown or has a stuck open relief valve). If the main condenser is available, there may be releases through the air ejectors, gland seal exhausters and other similarly controlled and monitored pathways. These pathways do not meet the intent of a non-isolable release path to the environment. These minor releases are assessed using radiological effluent EAL thresholds.

A pressure boundary leakage of 10 gpm is also used as the threshold in RCS Leakage EAL SU6.1. For smaller breaks, not exceeding the normal charging capacity threshold in RCS barrier Potential Loss D.2 or not resulting in ECCS actuation in RCS barrier Loss D.2, this threshold results in the declaration of an Unusual Event. For larger breaks, RCS barrier Potential Loss D.2 and RCS barrier Loss C.3 would result in an Alert. For S/G tube ruptures (SGTRs) which may involve more than one steam generator or unisolable secondary line breaks, this threshold would occur in conjunction with RCS barrier Loss C.3 and would result in a Site Area Emergency. Escalation to General Emergency would be based on the Potential Loss of the Fuel Clad barrier.

There is some redundancy in the Containment loss thresholds D.3 and D.4. This was recognized during the NEI EAL development process.

26.1 DCPP References:

26.1.1 E-2 U1(2), "Faulted Steam Generator Isolation"

26.1.2 E-3 U1(2), "Steam Generator Tube Rupture"

27. Valve(s) not closed

AND

Direct pathway to the environment exists after Containment isolation signal

This threshold addresses incomplete Containment (CNMT) isolation that allows direct release to the environment. It represents a loss of both the RCS and Containment barriers and therefore warrants declaration of a Site Area Emergency. Failure of Containment isolation or Containment ventilation isolation valves to isolate when required addresses incomplete Containment isolation that allows direct release to the environment.

The use of the modifier "direct" in defining the release path discriminates against release paths through interfacing liquid systems. The existence of an in-line charcoal filter does not make a release path indirect since the filter is not effective at removing fission noble gases. Typical filters have an efficiency of 95-99% removal of iodine. Given the magnitude of the core inventory of iodine, significant releases could still occur. In addition, since the fission product release would be driven by boiling in the reactor vessel, the high humidity in the release stream can be expected to render the filters ineffective in a short period.

27.1 DCPP References:

27.1.1 ECA-1.2 U1(2), "LOCA Outside Containment"

28. <u>ANY condition in the opinion of the SM/SEC/ED that indicates loss of the Containment barrier</u>

The SM/SEC/ED judgment threshold addresses any other factors relevant to determining if the Containment barrier is lost. Such a determination should include imminent barrier degradation, barrier monitoring capability and dominant accident sequences.

- <u>Imminent barrier degradation</u> exists if the degradation will likely occur within two hours based on a projection of current safety system performance. The term "imminent" refers to recognition of the inability to reach safety acceptance criteria before completion of all checks.
- <u>Barrier monitoring</u> capability is decreased if there is a loss or lack of reliable indicators. This assessment should include instrumentation operability concerns, readings from portable instrumentation and consideration of offsite monitoring results.
- <u>Dominant accident sequences</u> lead to degradation of all fission product barriers and likely entry to the EOPs. The SM/SEC/ED should be mindful of the Loss of AC power (Station Blackout) and ATWS EALs to assure timely emergency classification declarations.

28.1 DCPP References:

28.1.1 E-0.1 U1(2), "Reactor Trip Response"

Category E – ISFSI

Category: ISFSI

Sub-category: None

Initiating Condition: Damage to a loaded cask confinement boundary

EAL:

EU1.1 Unusual Event

Damage to a loaded cask confinement boundary as indicated by **EITHER**:

Occurrence of ANY Table E-1 event/condition

OR

Beyond normal ISFSI radiation reading

Table E-1 ISFSI Events/Conditions

Natural Phenomena	Accidents	
- High wind	- Cask drop	- Vehicle impact
- Tornado	- Cask tip-over	- Airborne missile
- Earthquake	- Cask air inlet/outlet blockage	- Transmission tower collapse
- Lightning strike	- Fire	- 500kV transmission line drop
	- Explosion	

Mode Applicability:

N/A

Basis:

Confinement boundary is the barrier(s) between areas containing radioactive substances and the environment. For the purpose of this EAL, the cask confinement boundary is the Multi-Purpose Canister (MPC).

In addition to the ISFSI FSAR scenarios, other hypothetical scenarios are considered as the basis for developing the emergency plan, specifically the classification levels. Most of the accidents which constitute emergency events at the ISFSI are not considered credible, but they are included in the planning basis so as not to restrict the scope of the emergency planning and to assure adequate emergency response to a wide range of postulated scenarios.

The ISFSI includes the dry-cask storage system, the cask transfer facility, onsite transporter, and the storage pads. The dry-cask storage system is the HI-STORM 100 System. This is a canister-based storage system that stores spent nuclear fuel in a vertical orientation. It consists of three discrete components: the MPC, the HI-TRAC 125 Transfer Cask, and the HI-STORM 100 System Overpack. The MPC provides the confinement boundary for the stored fuel. The HI-TRAC 125 Transfer Cask provides radiation shielding and structural protection of the MPC during transfer operations, while the storage overpack provides radiation shielding and structural protection of the MPC during storage. The HI-STORM 100 System is passive and does not rely on any active cooling systems to remove spent fuel decay heat. After the storage casks are placed on the storage pad, the ISFSI Technical Specifications require that the casks be inspected periodically to ensure that the air vents are not blocked. Security personnel control access to the storage area and identify and assess off-normal and emergency events. Health physics personnel perform dose rate and contamination surveys to ensure that the appropriate regulatory limits are maintained. Maintenance personnel maintain the facilities including the storage casks, emergency equipment, and transport systems.

An Unusual Event in this EAL is declared on the basis of the occurrence of any event of sufficient magnitude that a loaded cask confinement boundary is damaged or violated. Examples of natural phenomena events and accident conditions considered in the ISFSI design are listed in Table E-1. This EAL includes classification based on a loaded fuel storage cask confinement boundary loss leading to the degradation of the fuel during storage, or posing an operational safety problem with respect to its removal from storage.

DCPP Basis Reference(s):

1. NRC Materials License No. SNM-2511, LICENSE FOR INDEPENDENT STORAGE OF SPENT NUCLEAR FUEL AND HIGH-LEVEL RADIOACTIVE WASTE, Safety Evaluation Report, Sections 1.1.3, 2.1, 3.1.1, 15.1, Table 4.5.