DIABLO CANYON POWER PLANT
DETAILED CONTROL ROOM DESIGN REVIEW
SUPPLEMENTAL SUMMARY REPORT

VOLUME 1

Pacific Gas and Electric Company
October 1987

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VOLUME I

<u>Appendix</u>

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VOLUME II

ACRONYMS

ATWS	Anticipated Transient Without Scram
DCP	Design Change Package
DCPP	Diablo Canyon Power Plant
DCRDR	Detailed Control Room Design Review
EOP	Emergency Operating Procedure
ERG	Emergency Response Guidelines
FC	Field Change
FCT	Field Change Transmittal
FSAR	Final Safety Analysis Report
GPC	General Physics Corporation
HED	Human Engineering Discrepancy
HFS	Human Factors Specialist
HSDP	Hot Shutdown Panel
HVAC	Heating, Ventilation and Air Conditioning
I&C	Instrumentation and Controls
ICA	Interim Compensatory Action
INPO	Institute of Nuclear Power Operations
JCO	Justification for Continued Operation
LER	Licensee Event Report
LOCA	Loss-of-Coolant Accident
NECS	Nuclear Engineering and Construction Services (PG&E department)
NOS	Nuclear Operations Support (PG&E department)
NPG	Nuclear Power Generation (PG&E department)
NRC	Nuclear Regulatory Commission
NUTAC	Nuclear Utility Technical Assistance Committee (Westinghouse Owners Group)

ACRONYMS (CONT'D)

PAM	Postaccident Monitoring
PAMS	Postaccident Monitoring System
PG&E	Pacific Gas and Electric Company
QC	Quality Control
SFRTA	System Function Review and Task Analysis
SGTR	Steam Generator Tube Rupture
SPDS	Safety Parameter Display System
STA	Shift Technical Advisor
TAW	Task Analysis Worksheet
WOG	Westinghouse Owners Group

1 INTRODUCTION

1.1 PG&E'S COMMITMENT

Pacific Gas and Electric Company (PG&E) is committed to the safe operation of Diablo Canyon Power Plant (DCPP) Units 1 and 2. The Diablo Canyon detailed control room design review (DCRDR) implements this philosophy by identifying weaknesses in the man-machine interface between control room operators and equipment, and by taking steps to resolve or mitigate those weaknesses. The goal of the review is to provide an enhanced control room which will promote error-free operation during emergency as well as normal operating conditions. PG&E believes that the DCRDR meets or exceeds nuclear regulatory requirements.

1.2 HISTORY OF THE DIABLO CANYON DETAILED CONTROL ROOM DESIGN REVIEW

On December 17, 1982, the NRC issued Supplement 1 to NUREG-0737 (Generic Letter No. 82-33), the purpose of which was to provide additional clarification of the requirements for emergency response capabilities and, in particular, for performing a detailed control room design review. In August 1983, PG&E submitted to the NRC the DCRDR Program Plan (Reference 1) in response to Supplement 1 to NUREG-0737. The Program Plan was intended as a reference document and initiated Phase 1 of the DCRDR. PG&E issued the DCRDR Summary Report in December 1984 (Reference 2). The NRC conducted an in-progress audit in February 1985 and issued the results in September 1985 (Reference 3). The audit findings indicated that Phase 1 of the DCRDR did not fully meet the requirements of Supplement 1 to NUREG-0737. As a condition of the operating license for DCPP Unit 2, PG&E was required to comply with the requirements of Supplement 1 to NUREG-0737, for the conduct of the DCRDR (Reference 4).

In November 1985, PG&E met with the NRC Staff in Bethesda, Maryland, to discuss the nine elements of the DCRDR evaluated in the February 1985 audit. It was agreed that PG&E would upgrade the Diablo Canyon DCRDR program to be in full compliance with Supplement 1 to NUREG-0737 (Reference 5). This upgrade initiated Phase II of the Diablo Canyon DCRDR.

In January 1986, PG&E again met with NRC Staff in Bethesda and agreed to modify the system function review and task analysis (SFRTA) to meet the intent of Supplement 1 to NUREG-0737. The NRC Staff and PG&E agreed on the methodology to be employed (Reference 6).

In February 1986, PG&E submitted to the NRC the DCRDR Phase II program plan (Reference 7). That submittal provided responses to each open issue of the nine DCRDR elements addressed in the February 1985 audit and the November 1985 and January 1986 meetings. This report is consistent with the program plan.

PG&E submitted to the NRC the Phase II schedule plan in April 1986 and a DCRDR status report in June 1987 (References 8 and 9).

1.3 CONTENTS OF THIS REPORT

The Diablo Canyon Detailed Control Room Design Review Supplemental Summary Report has been prepared to meet the requirements of Supplement 1 to NUREG-0737.

This report consists of two volumes. Volume 1 describes how the nine elements of the DCRDR have been accomplished:

1. Establishment of a qualified multidisciplinary review team - Section 2
2. Function and task analysis to identify control room operator tasks and information control requirements during emergency operations - Section 3
3. Comparison of display and control requirements with an inventory of control room characteristics - Section 4
4. A control room survey to identify deviations from accepted human factors principles - Section 5
5. Assessment of HEDs to determine which are significant and should be corrected - Section 6
6. Selection of design improvements - Section 7

7 and 8. Verification that selected improvements will provide the necessary correction, and validation that improvements will not introduce new HEDs - Section 8

9. Coordination of control room improvements with changes resulting from other programs - Section 9

Section 10 provides conclusions regarding the DCRDR program, including a summary of activities required to be performed prior to the completion of the DCRDR. Section 11 describes the HED numerical listing, HED categories, and HED summary forms which are provided in Volume 2 (Appendices E and F).

Volume 2 of this report contains the entire database of DCPP Units 1 and 2 human engineering discrepancies (HEDs) and consists of Appendices E and F. Appendix E provides an HED numerical listing, and Appendix F provides a summary form for each HED identified during the DCRDR process. These summary forms provide a description of the HED, the assessed priority rating, recommended corrections, proposed schedules for implementation of changes, HED status, and justification for HEDs to be left uncorrected or partially corrected.

2 QUALIFICATIONS AND STRUCTURE OF THE REVIEW TEAM

2.1 INTRODUCTION

This section of the report demonstrates that the structure, qualifications, and management of the DCRDR team meet the requirements of Supplement 1 to NUREG-0737. The team consists of individuals with broad experience and uses a project management concept that has been successful within PG&E. The project manager has direct access to PG&E upper management, which further strengthens DCRDR project management.

2.2 DESCRIPTION OF MULTIDISCIPLINARY TEAM

As indicated on the organization chart (Figure 2-1), the DCRDR project team is composed of a project manager, a management team, and a review team. The management team consists of supervisory personnel from Nuclear Engineering and Construction Services (NECS), Nuclear Operations Support (NOS), and DCPP. The review team consists of representatives from NECS Engineering, NOS Engineering, DCPP Operations, and DCPP Training. A representative group of the review team acts as a core review team (Figure 2-1), which is responsible for the performance of the DCRDR. This core review team is supplemented by other members of the project team as needed. The project team has included two human factors consultants: General Physics Corporation (GPC) and an independent consultant. GPC performed the system function review and task analysis (SFRTA). The independent consultant has been involved with the DCRDR since its inception and participates in all elements of the review as a member of the core review team.

Table 2-1 summarizes the qualifications of the DCRDR team. Appendix A contains the detailed resumes of the team members. The team is composed of individuals with strong nuclear experience and diverse academic backgrounds, including nuclear engineering, instrumentation and controls engineering, electrical engineering, operations, and human factors.

2.3 DCRDR TRAINING COURSES

During Phase I of the DCRDR program, the human factors specialists provided an intensive five-day training course for seven other members of the core review team, from both the Diablo Canyon plant and General Office. The agenda for this training program is provided as Table 2-2. The objective of this course was to give engineering and operations personnel on the core review team a basic understanding of human factors engineering as it applies to control room reviews and enhancement. The course included the history, content, and methodology of the human factors discipline. It combined formal lectures on control room design issues with hands-on training and data-gathering experience in the Diablo Canyon control room.

During Phase II of the DCRDR program, this training course was repeated with less emphasis on data-gathering techniques and more emphasis on enhancement methods and options for corrections of generic human factors issues. This second course was administered to about 20 individuals from corporate headquarters and one operator from Diablo Canyon. The trainees included new members of the DCRDR team as well as engineers and draftsmen who were assigned to work on design change packages required to implement control room enhancements.

Also during Phase II of the program, an abbreviated half-day overview human factors course was given to an assemblage of approximately 25 plant personnel who were to interface with the DCRDR program. This course covered human factors issues and the PG&E DCRDR program content and schedule. The group that received this training included training personnel, operations supervisors, members of DCPP management staff, and personnel responsible for procedures. This course helped to establish cooperative working relationships between plant and General Office ogranizations that interacted on the DCRDR program.

2.4 PG&E MANAGEMENT OVERVIEW

The DCRDR was identified as a project that would require formal PG&E project management because of its licensing impact, complex tasks, interfacing organizations, and specific end-product.

In August 1985, the Vice President, Nuclear Power Generation, appointed a project manager for Phase II of the DCRDR to be responsible for all activities necessary for successful project completion. The project manager has direct access to the Vice President, Nuclear Power Generation, to resolve any problems such as organization and funding. The Vice President is kept apprised of the project's status by periodic status reports.

2.5 RESPONSIBILITIES OF TEAM MEMBERS

Table 2-3 indicates the responsibilities of the DCRDR team members in accomplishing Phase II activities. The roles of DCRDR members are categorized as follows: approval authority, lead responsibility, and support responsibility.

2.6 ORGANIZATIONAL RELATIONSHIPS

The DCRDR organization is composed of members from three departments within Nuclear Power Generation (NPG):

- Nuclear Operations Support (NOS)
- Nuclear Engineering and Construction Services (NECS)
- Diablo Canyon Power Plant (DCPP)

These departments in varying degrees are also responsible for the successful integration of changes resulting from the following programs:

- Safety parameter display system (SPDS) enhancement
- Emergency operating procedures (EOPs) upgrading
- Regulatory Guide 1.97 instrumentation

Section 9 of this report details the relationships of these programs with the DCRDR.

Table 2-1

PHASE II
DETAILED CONTROL ROOM DESIGN REVIEW TEAM

Title	Person	Education/Degree	Registrations/ Licenses Held	Human Factors/ Nuclear Experience
Project Manager	P. E. Beckham	BSCE, BSNE	SRO (certified) (BWR)	17 yrs.
Management Team Leader	B. W. Giffin	BSEE	-	20 yrs.
Review Team Leader	F. J. Cucco	BSME	-	12 yrs.
Management Team (NECS-EE)	S. Auer	BSEE	PE (EE)	5 yrs.
Management Team (NECS-I&C)	K. L. Herman	BSEE,BSNS, MBA	PE (EE)	14 yrs.
Management Team (DCPP)	L. F. Womack	BS Physics, MSME	SRO (PWR)	9 yrs.
Management Team (DCPP)	J. A. Sexton	BS Engineering	SRO (PWR), RO (BWR)	24 yrs.
NECS Engineering	B. M. Grosse	BSEE	PE (EE)	6 yrs.
NECS Engineering	J. J. Lisboa	MBA, MSEE, BSEE	PE (EE), PE (I&C)	20 yrs.
NECS Engineering	N. G. Seshagiri	MArch, MArchE	Arch (Calif)	10 yrs.
NECS Engineering	S. L. Wong	BSEE, MSEE	-	2 yrs.
NECS Engineering	J. R. Parris	BSIE, MSIE, MSOE	EIT	3 yrs.
Human Factors Consultant	J. L. Seminara	MA Exp. Psych.	-	32 yrs.
Human Factors Consultant	D. C. Burgy	MA Exp. Psych.	-	8 yrs.
Human Factors Consultant	R. Danna	MS Env. Eng., MA Physics	PE (ME)	11 yrs.
Human Factors Consultant	M. W. Dawson	MSNE (Candidate)	SRO (PWR)	9 yrs.

Table 2-1 (continued)

Title	Person	Education/Degree	Registrations/ Licenses Held	Human Factors/ Nuclear Experience
Human Factors Consultant	M. E. Jennex	MBA, BA Chem. & Physics	-	9 yrs.
Human Factors Consultant	L. R. Schroeder	PhD Exp.Psych., BS Gen.Eng.	-	7 yrs.
NOS Engineering	J. J. Vranicar	BSME	PE (ME)	16 yrs.
NOS Engineering	R. C. Washington	BSEE	-	7 yrs.
NOS Engineering	J. B. Neale	BSNE	SRO (PWR)	6 yrs.
NOS Engineering	S. A. Schaen	BSME	-	5 yrs.
DCPP Operations	R. L. Fisher	BSNE	SRO (PWR)	20 yrs.
DCPP Operations	T. W. Pellisero	BSNE	RO, PE (ME)	10 yrs.
DCPP Operations	S. R. Fridley	-	SRO (PWR)	17 yrs.
DCPP Operations	R. L. Ewing	-	SRO (PWR), RO (BWR)	22 yrs.
DCPP Operations	C. G. Smith	-	RO	16 yrs.
DCPP Operations	L. R. Waters	BSAE	SRO (PWR)	7 yrs.
DCPP Training	R. L. Jett	-	SRO (PWR)	20 yrs.
DCPP Training	J. D. Lodge	BSES, MSNE	-	28 yrs.

DAY 1	DAY 2	DAY 3	DAY 4	DAY 5
GENERAL INTRODUCTION HISTORY AND SCOPE OF HUMAN FACTORS GENERIC CONTROL ROOM HF ISSUES OVERVIEW OF EPRI HUMAN FACTORS STUDIES o EPRI NP 309 o EPRI NP 1118 o EPRI NP 1567	ANTHROPOMETRICS DISPLAYS o INDICATORS o METERS o CHART RECORDERS o CRTS o PRINTOUTS CONTROLS o SELECTOR SWITCHES o PUSHBUTTONS o SWITCH/LIGHTS o COUNTERS	ANNUCIATOR/WARNING SYS o DISPLAYS o LEGENDS o CONTROLS o LAMP REPLACEMENT o FALSE ANNUNCIATORS o SENSORY OVERLOAD COMMUNICATIONS PROCEDURES ACCIDENTAL CONTROL ACTIVATION	NUREG 0700 ENHANCEMENT ISSUES SYSTEMS REVIEW FUNCTIONS ANALYSES TASK ANALYSES WALK THROUGHS PROCEDURES REVIEW	INTERVIEWING EXPERIENCE ANALYSES EXPERIENCE SURVEY EXPERIENCE
CONTROL ROOM ARRANGEMENT CONSOLE DESIGN ENVIRONMENTAL FACTORS HABITABILITY OPERATOR STATIONS SUPERVISOR STATIONS INGRESS/EGRESS PATTERNS VISITOR CONTROL	BOARD LAYOUT RULES o FUNCTIONAL GROUPS o SEQUENTIAL o MIMICS o IMPORTANCE o HANDEDNESS LABELS CODING PRACTICES o COLOR o SHAPE o LOCATION	OPERATOR ERRORS AND NEAR ERRORS ORGANIZATIONAL EFFECTIVENESS WATCH TURNOVER OVERTIME SHIFT SCHEDULES SHIFT COMPOSITION OPERATIONAL MANPOWER TRAINING	INTERVIEWING TECHNIQUES CONDUCTING SURVEYS o NOISE o ILLUMINATION o COMMUNICATIONS o PROTECTIVE GEAR o DESIGN CONVENTIONS CHECKLIST APPLICATION CRITICAL INCIDENTS LERs	DOCUMENTATION EXPERIENCE (HEDs) PRIORITIZING HEDs TYPICAL CONTROL ROOM REVIEW CASE HISTORY QUESTIONS AND ANSWERS

Table 2-2 Agenda for Core Team Human Factors Training Program

Table 2-3

DCRDR TEAM RESPONSIBILITIES

DCRDR Activity	Project Manager	Management Team	Review Team Leader	Human Factors Specialist	NECS Engineering	DCPP Operations
Function and Task Analyses	A			S	L	S
Control Room Inventory	A			S	L	S
Control Room Survey	A		L	S	S	S
Assessment of HEDs	A	A	L	S	S	S
Selection of Design Improvements	A	A	S	S	L	S
Verification and Validation	A		L	S	S	S
Interfaces						
- Training	A			S	L	
- Emergency Operating Procedures	A		S	L		
- Reg. Guide 1.97	A		S	S	L	
- Safety Parameter Display System	A/L	A		S	S	S
PGandE Enhancement Guidelines Document	A	A	S	S	L	
Supplemental Summary Report	A/L	A	S	S	S	

A = Approval Authority L = Lead Responsibility S = Support Responsibility

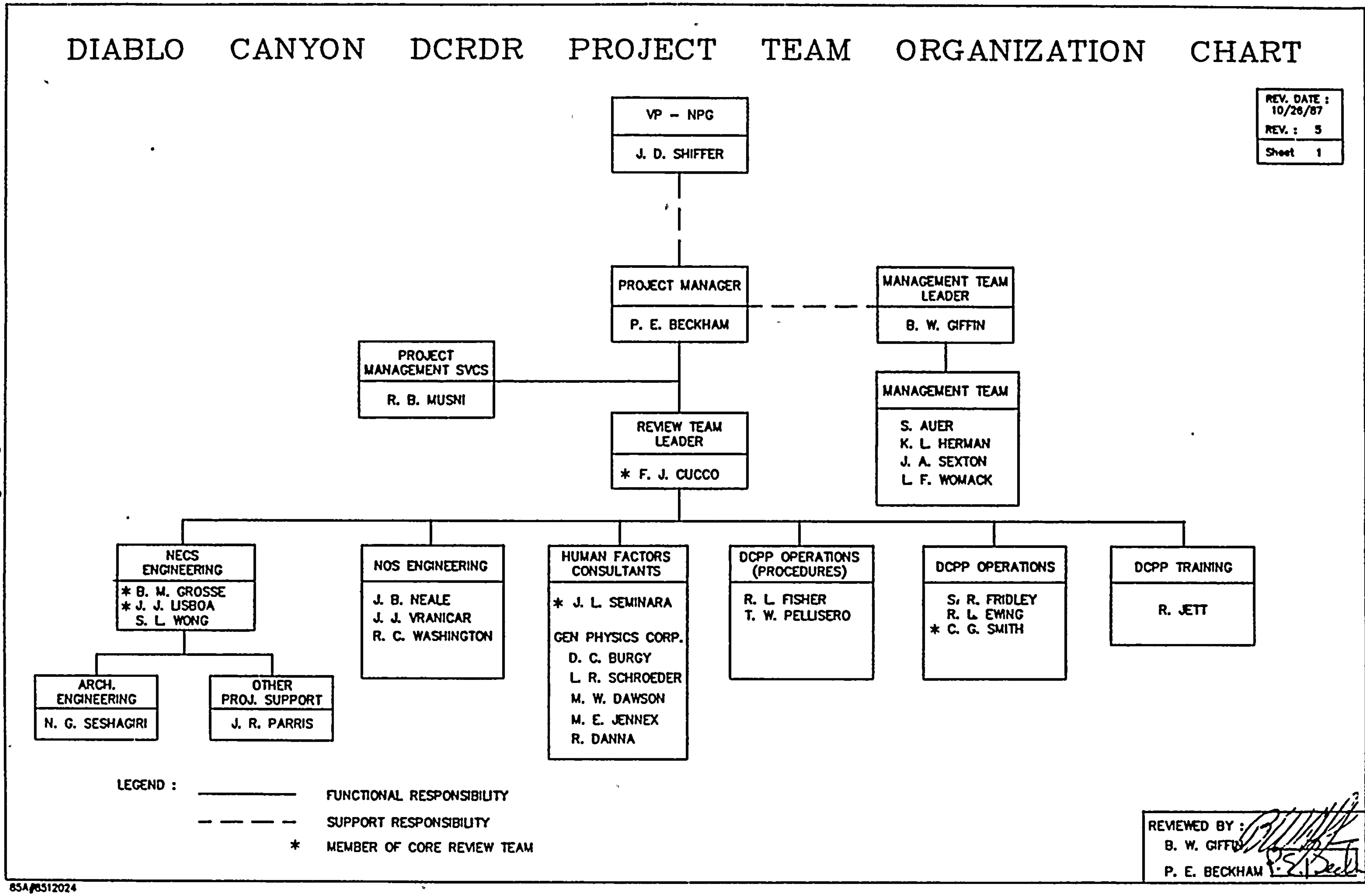

Figure 2-1 Diablo Canyon DCRDR Project Team Organization Chart

3 SYSTEM FUNCTION REVIEW AND TASK ANALYSIS

3.1 INTRODUCTION

This section describes the methods used by General Physics Corporation (GPC) for the system function review and task analysis (SFRTA) and the results obtained. The methods used by GPC were agreed on by PG&E and the NRC Staff (Reference 6).

3.2 PURPOSE

The purpose of the SFRTA was to provide a complete set of plant-specific information and control characteristics required to support operator tasks during emergency operations and to ensure that required systems can be efficiently and reliably operated under emergency conditions.

The SFRTA also generated information and controls characteristics required to conduct the DCPP remote shutdown procedure.

3.3 METHODS

The SFRTA activities are diagrammed in Figure 3-1. The methods used for each activity are descibed below.

3.3.1 Identification of Plant-Specific Systems and Functions Required During Emergency Operations

Plant systems and subsystems in the DCPP control room and remote shutdown area that the operator must access during emergency operations were listed. This list was comparable to the safety-related systems identified in the emergency operating procedures (EOPs) and the DCPP remote shutdown procedure. For each of the systems identified, a description was prepared of the functions of the system and the conditions when the system is used. This description of system functions served as a reference for subsequent task analysis and was also used in the selection of operating scenarios.

The DCPP Final Safety Analysis Report (FSAR) Update was the primary source of information for the system descriptions and was supplemented, as necessary, with other plant information and documentation.

An example of a system functions description is provided as Figure 3-2.

3.3.2 Analysis of System Functions for Indentification of Scenarios and Residual Tasks

The list of DCPP safety-related systems was used to define a set of scenarios that serve as adequate samples of various emergency conditions and the plant systems and functions exercised in those conditions. The related DCPP EOP (and remote shutdown procedure) steps were also identified.

A check was performed to ensure that the desired systems and functions would be exercised in the scenarios chosen. The scenarios selected ensured the establishment of those tasks applicable to the systems.

A brief narrative description of each scenario was prepared that established the limits and conditions of the events analyzed. The descriptions included:

- Procedures used
- Initial conditions
- Scenario sequence
- Expected response
- Termination criteria

The five scenarios developed were:

- Anticipated transient without scram (ATWS) and loss of reactor coolant
- Large-break loss-of-coolant accident (LOCA)
- Steam generator tube rupture (SGTR) with cooldown using backfill
- Secondary break inside containment with loss of spray capability
- Loss of secondary heat sink

The scenario descriptions are provided in Appendix B.

Residual operator tasks from the plant-specific EOPs and remote shutdown procedure not covered in the scenarios were analyzed independently for information and control requirements. The analysis of residual tasks was performed to ensure that all operator interfaces had been examined even if those interfaces are not exercised in the sample of emergency scenarios selected for validation. Verification of equipment availability and suitability was performed for these residual tasks as well as for tasks included in the emergency scenarios.

3.3.3 Development of Task Analysis Worksheets and Identification of Information and Control Requirements

A Task Analysis Worksheet (TAW) (Figure 3-3) was developed to collect task performance data and other information needed for the DCRDR. The TAW indicates the operational steps required in each scenario, along with the appropriate information and control requirements, means of operation, and instrumentation and controls (I&C) present on the control boards. The operator tasks were analyzed with the selected plant-specific EOPs as a starting basis and documented in the following manner:

1. The discrete steps in the plant-specific EOPs, in order of performance, were recorded in the PROC NO. STEP NO. column of the TAW.

2. A brief description of the operator's tasks (in order of procedural steps) was recorded in the TASK/SUBTASK column of the TAW. All tasks, both explicit and implicit, were documented.

3. The operator decisions and actions required for task performance were recorded in the TASK DECISION REQUIREMENTS and TASK ACTION REQUIREMENTS columns, respectively. System functional response was described, when appropriate, in these columns. This set of data also included branching points in the EOPs that determined the outcome of the operating sequence.

4. Input and output requirements for successful task performance were recorded in the SYSTEM COMP PARAM and RELEVANT CHARACTERISTICS columns. These were system component and parameter, relevant characteristics, and procedural information necessary for operators to assess plant conditions or system status (e.g., hot leg temperature, reactor coolant system flow, pressurizer pressure, etc.). Specific values for parameter readings or control characteristics (e.g., close-open, off-auto-on) were recorded.

TAW steps 1 through 4 were completed using independent sources of data other than the actual instruments and controls present in the control room.

The remaining columns of the TAW were completed during the verification and validation steps, which are described below:

5. Once the tasks, decision requirements, and information and control requirements were specified, the existing I&C that the operator uses or can use for each procedural step were documented. Listed in the MEANS CHARACTERISTICS column were all I&C needed or available:
   - To bring a system into service, maintain it in service, or remove it from service
   - To confirm that an appropriate system response has or has not occurred (feedback)
   - To make a decision regarding plant or system status

   The identification number of the control or instrument was listed in the I&C NO. column. The panel on which the control or instrument is located was indicated on the PANEL column.

6. Verification was documented in the AVAIL and SUIT columns as follows:

   - Availability of the I&C required for successful operator task performance was noted by a YES or NO in the AVAIL column.

- Suitability of the existing I&C for meeting the postulated information and control requirements for operator tasks was noted by a YES or NO in the SUIT column.

7. The presence or absence of information and control requirements on the SPDS, postaccident monitoring (PAM) panel, or hot shutdown panel (HSDP) was noted by a YES or NO in the SPDS, PAMP, and HSDP columns.

8. Candidate HEDs were noted in the COMMENTS column during any step of the task analysis. Data for HEDs were entered on an HED form for assessment and input into the computerized HED database.

The TAW, developed through the steps described above, serves as the complete record of: operator tasks; decision, information, and control requirements; and I&C availability and suitability.

3.3.4 Data Entry

All data in the TAWs were entered into a DCRDR database using standard database software and personal computers. Data entry and modification were performed continuously throughout the project. The task statements and information on the task analysis worksheets, including information and controls characteristics, were continually updated to reflect changes, additions, and deletions.

3.3.5 Control Room Inventory

The list of plant-specific I&C requirements for successful task performance was compared with the control room I&C inventory to verify the availability and suitability of controls and displays in the control room required to support operator decisions and actions.

Section 4 of this report provides a detailed description of this activity.

3.3.6 Validation of Control Room Functions

Utilizing the TAWs (operator tasks, information and control requirements, etc.), a full complement of DCPP control room operators walked through each scenario in the DCPP simulator.

The walkthroughs were first performed in slow time. Operators were instructed to describe their actions as they performed them.

Following each slow-time walkthrough of a scenario, the operators performed the walkthrough in real time to evaluate the operational aspects of the control room design in terms of control/display grouping, control feedback, manning levels, and traffic patterns.

In a debriefing after the walkthroughs, operators were asked to describe errors or problems that they encountered and the source of the errors or problems. These errors or problems were documented as HEDs.

The walkthroughs were videotaped to fully document the tasks involved for all crew members. Link analyses, which trace the movement patterns of the operating crew in the control room, were developed by examining the videotapes to assess whether the control room layout hinders operator movement.

The link analyses were reviewed with regard to three aspects of control room operation:

- Control room staffing
- Traffic patterns
- I&C distribution

Control room staffing refers to the adequacy of the number of personnel provided for operation in the control room. Traffic patterns, the routes that each person traverses during the scenario, were examined to ascertain how far and how frequently each person must travel to complete his tasks. The distribution of instruments and controls was assessed in terms of the relative distances between them.

The results of the review of the link analyses are as follows:

- Control room staffing - The control room staffing appears to be adequate. The control operator typically mans control consoles CC-1, CC-2, and CC-3. The assistant control operator works mainly at the vertical boards. The monitoring of displays and manipulation of controls are more or less evenly divided between the two.

- Traffic patterns - The link analyses revealed some repeated traversal of long paths. Path intersection and overlapping were not excessive.

- I&C distribution - The majority of the instruments and controls used during the scenario walkthroughs were distributed evenly among the vertical panels and consoles in the control room. The number of times widely separated I&C were used was minimal.

In summary, a review of the link analyses indicated that there were no significant problems that hindered operating crew movement or control access during performance of the scenarios.

Any dynamic performance problems that were uncovered during this phase of the SFRTA process were documented as HEDs.

3.4 RESULTS

All findings from the SFRTA were documented on HED forms, which include a description of the finding, the source of the finding, the panel on which the finding occurred, and the components found discrepant.

The HEDs were assessed by the DCRDR team in accordance with the assessment procedure described in Section 6. Recommended resolutions for each discrepancy were developed and documented on the HED assessment form (Figure 6-2).

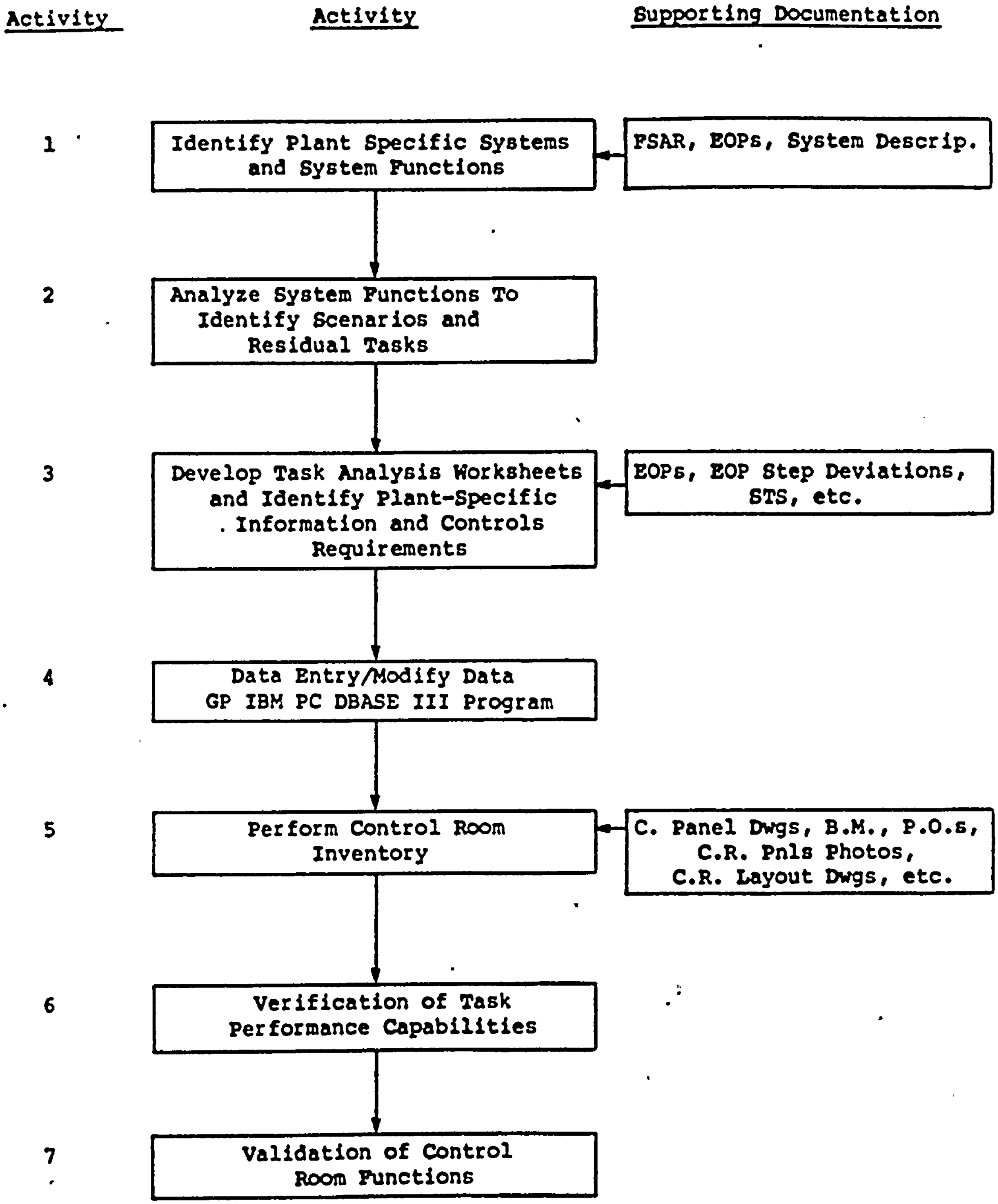

Figure 3-1 Flow Diagram of Major Activities Involved in the Generation of Plant-Specific SFRTA I&C Requirements

PLANT SYSTEM FUNCTION DESCRIPTION
DCNPP

Plant System Name: Residual Heat Removal System

System Abbreviation: RHR

System Number: B-2

System Function(s):

Used to transfer decay heat from the core and RCS during shutdown and refueling operations also used to transfer water from the RWST to the RCS when filling the refueling cavity and to transfer water back to the RWST following refueling operation. Sections of the RHR System are used for injection and recirculation as part of the ECCS. RHR system supplies containment spray during recirculation phase of LOCA.

Conditions for System Use:

Plant heatup and cooldown. Also used to remove decay and residual heat during cold shutdown and refueling operations.

Reviewer: B. Drane **Date:** 5/86

Figure 3-2 Sample Plant System Function Description

PACIFIC GAS & ELECTRIC
TASK ANALYSIS WORKSHEET
PAGE 1

REC NO	PROC NO. STEP NO.	TASK/SUBTASK	TASK DECISION REQUIREMENTS	TASK ACTION REQUIREMENTS
100	E-O- 1A	VERIFY REACTOR TRIP: (VB2/CC1). REACTOR TRIP AND BYPASS BREAKERS - OPEN, ROD BOTTOM LIGHTS - LIT, NEUTRON FLUX - DECREASING	TO DETERMINE IF THE REACTOR TRIPPED AND THE BYPASS BREAKERS ARE OPEN. (CONTROL ROD BOTTOM LIGHTS AND NEUTRON FLUX DECREASING) . TO DETERMINE IF THE REACTOR WILL NOT TRIP.	IF REACTOR IS TRIPPED, GO TO NEXT TASK. IF REACTOR IS NOT TRIPPED, MANUALLY TRIP REACTOR AND GO TO NEXT TASK. IF REACTOR WILL NOT TRIP, GO TO FR-S.1, RESPONSE TO NUCLEAR POWER GENERATOR/ATWS
200	E-O- 2A	VERIFY TURBINE TRIP: (CC3,VB2). ALL FOUR SVS CLOSED.	TO DETERMINE IF THE TURBINE IS TRIPPED (FOUR SVS CLOSED). TO DETERMINE IF THE TURBINE WILL NOT TRIP.	IF TURBINE IS TRIPPED (FOUR SVS CLOSED) GO TO NEXT TASK. IF TURBINE IS NOT TRIPPED MANUALLY TRIP TURBINE, IF TURBINE WILL NOT TRIP, THEN CLOSE MSIVS AND BYPASSES.
300	E-O- 3A	VERIFY VITAL 4KV BUS STATUS: (VB4). VERIFY ALL THREE VITAL 4KV. BUSES F,G, AND H - ENERGIZED.	TO DETERMINE IF 4KV BUSES F.G, AND H ARE ENERGIZED	IF ALL THREE BUSES ENERGIZED, GO TO NEXT TASK. IF TWO VITAL BUSSES ARE ENERGIZED, THEN REFER TO ECA-0.3, RESTORE VITAL BUS WHILE CONTINUEING IN THIS PROCEDURE. IF ONLY ONE VITAL BUS IS ENERGIZED, THEN GO TO ECA-0.3, RESTORE VITAL BUS. IF ALL VITAL BUSSES ARE DE-ENERGIZED, THEN GO TO ECA-0.0, LOSS OF ALL AC POWER.
400	E-O- 4A	CHECK IF SI IS ACTUATED: (VB3). PK08-21, SAFETY INJECTION ACTUATION ANNUNCIATOR - ON	TO DETERMINE IF SI IS ACTUATED OR IS REQUIRED	IF SI IS ACTUATED, GO TO NEXT TASK, IF SI REQUIRED, AND NOT ACTUATED THEN MANUALLY ACTUATE. IF SI NOT REQUIRED, THEN GO TO E-0.2, REACTOR TRIP RESPONSE.
500	E-O- 5A	VERIFY CONTAINMENT ISOLATION PHASE A: (VB1). PHASE A PORTION OF MONITOR LIGHT BOX B: RED ACTIVATED LIGHT - ON. WHITE STATUS LIGHTS - OFF	TO DETERMINE IF CONTAINM ENT ISOLATION, PHASE A ACTUATED	IF PHASE A ACTUATED, GO TO NEXT TASK, IF PHASES A DID NOT ACTUATED, THEN MANUALLY ACTUATE PHASE A , OR, MANUALLY CLOSE PHASE A ISO VLVS AS NECESSARY.
600	E-O- 6A	VERIFY CONTAINMENT VENT ISO: (VB1). CONTAINMENT VENT ISO PORTION OF MONITOR LIGHT BOX B: RED ACTIVATED LIGHT - ON. WHITE STATUS LIGHTS - OFF.	TO DETERMINE IF CONTAINM ENT VENT ISOLATION (CVI) ACTUATED	IF CVI ACTUATED, GO TO NEXT TASK. IF CVI DID NOT ACTUATE MANUALLY ACTUATE OR MANUALLY CLOSE CVI VLVS, AS NECESSARY.
700	E-O- 7A	VERIFY ESF PUMP AND VALVE STATUS: (VB1). SI PORTION OF MONITOR LIGHT BOX C: RED ACTUATED LIGHT - ON. WHITE STATUS LIGHTS - OFF.	TO DETERMINE STATUS OF ESF PUMP AND VALVE STATUS.	IF ESF PUMP IS RUNNING AND VLVS ARE ALIGNED PROPERLY, GO TO NEXT TASK. IF ESF PUMP NOT RUNNING, MANUALLY START PUMPS, IF ESF PUMP VLVES NOT ALIGNED PROPERLY, ALIGN VALVES AS NECESSARY.

Figure 3-3 Sample Task Analysis Worksheet

PACIFIC GAS & ELECTRIC
TASK ANALYSIS WORKSHEET
PAGE 2

REC NO	PROC NO. STEP NO.	SYSTEM. COMP PARAM	RELEVANT CHARACTERISTICS	MEANS CHARACTERISTICS	I&C NO.	PANEL	AVAL	SUIT	SPDS	PAMP	HSDP	COMMENTS
100	E-0- 1A	ROD BOTTOM POSITION INDICATION	RED BOTTOM LIGHTS, LIT WHEN ROD ON BOTTOM. METER, 0-230 STEPS, LINEAR, ANALOG	MATRIX W/ROD BOTTOM RED LIGHTS METER 0-228 STEPS		VB-1	YES'	YES				
108	E-0- 1A	ROD BOTTOM POSITION INDICATION	RED BOTTOM LIGHTS, LIT WHEN ROD ON BOTTOM. METER, 0-230 STEPS, LINEAR, ANALOG	MATRIX W/ROD BOTTOM RED LIGHTS METER 0-228 STEPS		VB-1	YES	YES				
110	E-0- 1A	REACTOR TRIP BREAKER POSITION INDICATION	BREAKER OPEN AND CLOSE LIGHTS	2 SETS OF 2 IND. LIGHTS TRAINS A/B RED-CLOSED GREEN-OPEN	52/RT1, 52/RT B	VB-2	YES	YES	YES			
	E-0- 1A	NUETRON FLUX INDICATION A - SOURCE RANGE, B - INTERMEDIATE, C - POWER	A. METER, RECORDER, LOG SCALES, ANALOG 0-10 E6 CPS RANGE, B. METER, RECORDER, LOG SCALES, ANALOG 10E-11 TO 10E-3 AMPS, C. METER, RECORDER, LINEAR, ANALOG 0-120%, INC = 2	2 VER LOG SR MTRS, RNG 10E0-10E6 CPS/4 LOG MTRS, RNG: 10E-11 -10E3 AMPS:4 LIN PR MTRS, 0-120%, 10 MJ, 2 MN	NI 31B, NI 32B, NI 34B, NI 36B, NI 41B, NI 42B, NI 43B, NI 44B	CC-1	YES	YES	YES	YES		
114	E-0- 1A	REACTOR TRIP ACTUATION CONTROL	TWO POSITION, NORMAL/TRIP	VB-2:2 POS T-HANDLE SWITCH, NEUTRAL/TRIP, CC:1-3 POS T-HANDLE SWITCH TRIP/NEUTRAL/RESET		VB-2, CC-1	YES	NO				HED #644
200	E-0- 2A	MAIN TURBINE STOP VALVE POSITION INDICATION	VALVE OPEN AND CLOSE LIGHTS	4-2 POS IND LIGHT BOXES GREEN-CLOSED, RED-OPEN	FCV-146, FCV-143, FCV-144, FCV-145	CC-3	YES	YES				

Figure 3-3 Sample Task Analysis Worksheet (continued)

4 CONTROL ROOM INVENTORY

4.1 INTRODUCTION

An inventory of control room and hot shutdown panel instrumentation was conducted for Units 1 and 2. The purpose of the inventory was to compare available controls and displays with instrumentation requirements identified through task analysis (Section 3). Missing or faulty instrumentation revealed through this comparison resulted in HEDs, reported as described in Section 11.

4.2 METHODS

A written inventory of the existing instrumentation and controls in the control room was developed. This was compared with the instrumentation requirements to verify task performance capabilities.

4.2.1 Development of a Control Room Inventory

Source documents for this inventory included photos and layout drawings of the control panels, as well as purchase orders, bills of materials, etc. Equipment characteristics were listed on a form (Figure 4-1) and then entered into a computerized database.

The equipment characteristics form included the following data, listed by column heading:

- I&C DESCRIPTION AND PARAMETER - The description of the instrument or control as it appears on the panel. The parameter measured was included where applicable.

- I&C NUMBER - The alphanumeric identification code given to an instrument or control.

- PANEL - The alphanumeric panel identification code.

- INSTRUMENT TYPE - Switch, meter, recorder, controller, potentiometer, pushbutton, indicator light, etc.

- RANGE - The meter range from minimum to maximum on the scale.

- UNITS - The units of measurement: gpm, amps, inches, rpm, etc.

- DIVISIONS AND SCALE - Divisions were listed as major, minor, and intermediate graduations. Scale is either log or linear.

- CTRL and LTS - For a control, a list of all the switch positions (e.g., open, normal, closed). For a light, the color and its meaning when illuminated.

4.2.2 Verification of Task Performance Capabilities

Task performance capabilities were systematically verified in two phases to ensure that the instruments and controls identified in the task analysis as being required by the operator are:

- Present in the control room or hot shutdown panel

- Effectively designed to support correct task performance

4.2.2.1 Phase 1 - I&C Availability

In the first phase, the presence or absence of the required instruments and controls was confirmed. This was done by comparing the requirements from the task analysis, listed in the SYSTEM COMP PARAM and RELEVANT CHARACTERISTICS columns of the TAW, to the actual control room I&C.

The presence or absence of a required instrument or control was noted by a YES or NO in the AVAIL column of the TAW. Any required instrument or control discovered to be unavailable was identified as an HED and documented on an HED form.

4.2.2.2 Phase 2 - I&C Suitability

The second phase addressed the human engineering suitability of the required instruments and controls by evaluating them according to the criteria shown on Figure 4-2. For example, if a meter used in a particular procedural step was present in the control room, it was examined to determine whether or not it has the range and scaling appropriate to that procedural step. The suitability of the range and scaling was noted with a YES or NO in the SUIT column of the TAW. Any instrument or control found unsuitable was identified as an HED and documented on an HED form. Figure 4-2 charts the decision process of the suitability review.

EQUIPMENT CHARACTERISTICS

I&C DESCRIPTION AND PARAMETER	I&C NUMBER	PANEL	INST. TYPE: SW/METER/ RECORDER/CONTROLLER	RANGE	UNITS	DIVS: MAJOR/MINOR SCALE: LOG/LINEAR	CTRL: SW POS LTS: CLR/MEANING
CONTAINMENT W/R LEVEL LR 942 1-88	LR942	PAM1	CONT. REC	0-100 (SCALE) 65-98 (PAPER)	%/ FEET	20/2 LINEAR 5/1 LINEAR	N/A
CONTAINMENT W/R LEVEL LR943 1-89	LR943	PAM1	CONT. REC.	0-100 (SCALE)/65 -98 (PAPER)	%/ FEET	20/2 LINEAR 5/1 LINEAR	N/A
CONTAINMENT GROSS ACTIVITY RR30 1-90	RR30	PAM1	CONT. REC.	10E0 - 10E7		10E1/10%	N/A
CONTAINMENT GROSS ACTIVITY RR31 1-91	RR31	PAM1	CONT. REC.	10E0-10E7		10E1/10% LOG	N/A
RE-30 CONTAINMENT	1-R30	PAM2	CIRC METER	1-10E7	R/HR	10E1/20%	RET SW: OFF,A-11
HI RNGE AREA MONITOR	HRCM	PAM2	ROT SW 4PB SW	1-10E7	R/HR	10E1/20% LOG	ROT SW OFF, ALL,/ PB:3-RESET 1-TEST
RE 31 CONTAINMENT	1-R31	PAM2	ROT SW 4 PB SW	1-10	R/HR	10E1/20% LOG	ROT SW OFF, ALL,/ PB: 3-RESET 1-TEST
RE-35 PLT. VENT ALARA	1-R35	PAM2	CIRC METER	0.1 - 10E7	MR/HR	10E1/10%	ROT SW OFF, ALL
85' SAMPLE AREA MON.	ALARM	PAM2	ROT SW 3PB SW	0.1 - 10E7	MR/HR	10E1/10% LOG	GREEN - FAIL
RE-29 PLT. VENT 150' HI	1-R29	PAM2	ROT SW 3 PB SW	0.1 - 10E7	MR/HR	10E/10% LOG	GREEN - FAIL
RNGE GROSS GAMMA PROCESS MON.	PV GGM	PAM2	ROT SW 3 PB SW	0.1 - 10E7	MR/ HR	10E1/10% LOG	GREEN- FAIL
RE-33 PLT. VENT 115'	1-R33	PAM2	CIRC METER	10 - 10E7	CPM	10E1/ 20% LOG	ROT SW OFF,CAL,H.V., OPER. PB ALL RESET & BKLT

Figure 4-1 Sample Equipment Characteristics Form

CRITERIA FOR DECISIONS

HFE/OPS

- Information displayed to appropriate modality (visual vs. auditory)
- Appropriate parameter displayed
- Display of quantitative and/ or qualitative information appropriate for task
- Discrete/continuous control functions appropriate
- Display of trend information available when appropriate

I&C

- Actual system/equipment status information is provided rather than indirect information (e.g., demand vs. valve position for controllers, direct vs.indirect measure of flow in system loop)

HFE/OPS/I&C

- Equipment provides appropriate precision and range of control
- Scale units are consistent with the degree of precision needed
- Scale range spans the expected range of operational parameters
- Values displayed are in a form immediately usable w/o conversion

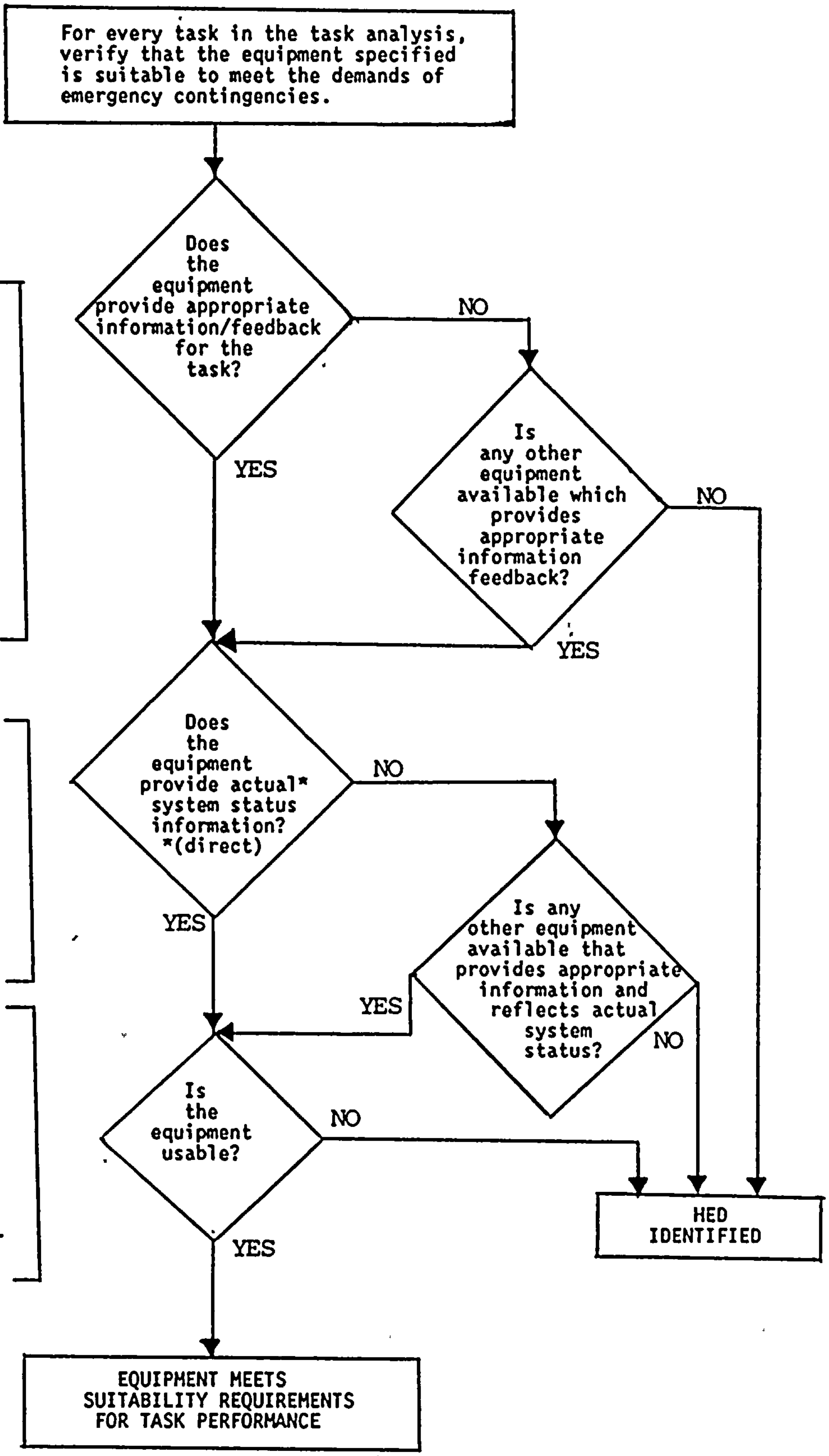

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Figure 4-2 Flow Chart of Decision Process for Verifying Equipment Suitability

5 CONTROL ROOM SURVEY

5.1 INTRODUCTION

PG&E performed a thorough, detailed control room survey to identify deviations from accepted human factors principles. The first phase of data collection began in late 1983 and was completed in mid-1984. The results from this phase were reported in the December 1984 DCRDR Summary Report (Reference 2). The second phase of data collection began in 1986, and its results are in this report.

5.2 METHODS

5.2.1 Phase I

In the first phase, the control room and control panels, including the hot shutdown panel, were reviewed according to standard human factors guidelines (checklist and environmental surveys); operators were interviewed; operability walkthroughs were performed; and historical operator experience was reviewed.

5.2.1.1 Human Factors Guidelines Surveys

The core review team performed the initial human factors survey in late 1983, using the Nuclear Utility Technical Assistance Committee (NUTAC) checklists as base documents. The NUTAC checklists are a series of descriptions of acceptable design characteristics. Two or more core review team members reviewed each panel or area within the control room. The checklists included sections on labeling, mimics, demarcation, general panel design including controls and displays, and the process computer, as well as overview and operator-assisted checklists. The checklists were supplemented by a general survey of the control room by the human factors specialists, who have extensive experience in the field of human factors in nuclear plant control rooms.

The checklist surveys were initially performed on the Unit 1 side of the control room. To complete the review for the Unit 2 side while ensuring consistency between Units 1 and 2, half-size photos of the Unit 1 main control boards were used. Members of the review team compared the half-size photos to the equivalent sections of the Unit 2 control boards. Any differences were noted and later recorded as HEDs. The quality and size of the photos permitted a complete and systematic comparison of the two units. This comparison allowed the review team to identify problem areas that were associated with Unit 2 only, as well as to identify differences between the units.

During this same period, NUTAC criteria were used to survey control room lighting and noise. It was recognized at the time of these surveys that because construction of the control room was incomplete, these surveys would have to be reperformed after the plant became operational (see Section 5.2.2.1).

5.2.1.2 Operator Interviews

The survey was supplemented with operator interviews performed by the human factors specialists and members of the review team. The interview questions were initially drafted by the consultant human factors specialists to supplement the NUTAC checklist surveys. The interview questions were reviewed and modified by the DCRDR review team, including the human factors specialists. During the interviews the questions were further modified, and redundant or otherwise unproductive questions were eliminated. Sixteen operators with various levels of experience were interviewed.

5.2.1.3 Operability Walkthroughs

The survey was further supplemented with operability walkthroughs of selected emergency operating procedures (EOPs). These reviews used the interim EOPs, which were based on the Westinghouse Owners Group (WOG) Emergency Response Guidelines (ERGs), Rev. 0. (The Phase II systems function review and task analysis performed by the General Physics Corporation used the Rev. 0 version

of the EOPs, which were based on the Rev. 1 version of the WOG ERGs.) During the walkthroughs, members of the review team, including the human factors specialists, read the procedure steps while a control room operator performed the associated tasks. A reviewer observed and recorded the operator's actions and noted any apparent difficulties or irregularities on an operability walkthrough form. (The operability walkthrough form and the corresponding procedure are described fully in Appendix C of the December 1984 DCRDR Summary Report [Reference 2].) The reviewer/observer questioned the operator concerning the reasons for his actions if the reasons were not clear.

5.2.1.4 Historical Experience Review

The historical experience review consisted of two separate activities. The first was a review of historical documentation pertaining to plant-specific and generic control room occurrences, including operators' logs and Licensee Event Reports (LERs). This review is described more fully in the December 1984 DCRDR Summary Report (Reference 2).

The second step was to survey the operating personnel using a structured interview format. This step was included in the interviews discussed in Section 5.2.1.2.

5.2.2 Phase II

After an in-progress audit of the PG&E DCRDR in February 1985, the NRC identified additional survey requirements (Reference 3). These included gathering more quantitative and qualitative information on the operators' work space, reperforming various environmental surveys to reflect the differences between the operating plant and the plant under construction, and determining the full extent of various generic HEDs. This additional survey work was based on the human factors guidelines in NUREG-0700.

5.2.2.1 Environmental and Workspace Survey

A systematic survey of the control room working environment was performed in 1986 after both DCPP units went into commercial operation. It included

reviews of the lighting, noise, HVAC, furnishings, and anthropometric factors for Units 1 and 2 panel areas and staff work spaces. NUREG-0700, Section 6.1, was used as the base human factors guideline.

Survey tasks included:

- Development of a work plan, which was reviewed with the human factors specialist

- Preparation of a checklist and forms for data collection

- Field observations, interviews, questionnaires, and measurements covering NUREG-0700 Section 6.1 criteria

- Analysis of data and documentation of findings

- Identification and assessment of HEDs and inclusion in the HED database

A complete report, including the operator questionnaires, interview results, quantitative survey results, and findings, is on file at PG&E.

5.2.2.2 Supplemental Surveys and Investigations

Various I&C and electrical engineering investigations were performed for HEDs that required more information prior to their assessment. These included investigations of generic HEDs to determine their full extent and investigations of various specific HEDs to determine their actual significance. These investigations were documented and later used by the assessment team in their analysis of the subject HEDs.

5.3 IDENTIFICATION OF HEDs

After completion of each data gathering stage of the DCRDR, the results of the checklists, surveys, interviews, and operability walkthroughs were analyzed to

identify any departures from human engineering guidelines. These were recorded on the upper half of the Human Engineering Discrepancy form (Figure 5-1). The bottom half of the form was originally intended to be used for assessment as described in the draft version of NUREG-0801. When the assessment process was revised during Phase II, the bottom half of the HED form was abandoned in favor of the HED assessment form (Figure 6-2).

A computer database was set up during Phase II of the DCRDR to permit easy tracking of the status of each HED. HEDs can also be sorted on any field, including location, type (category), and priority.

5.4 RESULTS

Nearly 900 HEDs have been recorded. Volume 2 of this report contains a numerical listing of all HEDs (Appendix E) and summary forms (Appendix F) which include HED descriptions, priority ratings, assessments, recommendations, actual corrections to date, and implementation schedules. The summary forms are arranged by category.

HUMAN ENGINEERING DISCREPANCY CHRON ______ LOC ______

UNIT 1 - 2 (CIRCLE WHICH) ANALYST ______

ITEM/LOCATION(S)
CHECKLIST CODE(S)
DISCREPANCY (MEASUREMENTS AS APPROPRIATE)

PRIORITIZATION

PRELIMINARY SIGNIFICANCE ESTIMATE:
PRELIMINARY BACKFIT FEASIBILITY ESTIMATE:
PRELIMINARY RECOMMENDATION

Figure 5-1 Human Engineering Discrepancy Form

6 ASSESSMENT OF HEDS

6.1 INTRODUCTION

Each HED identified during the data collection phases of the DCRDR is assessed to determine its significance for plant safety or operability. The assessment takes into account the safety function of the subject of the HED, the potential for error as a result of the HED, and the consequence should that error occur.

6.2 METHODS

HEDs affecting equipment that is safety-related or used in an emergency operating procedure are assessed for safety significance. HEDs that affect the balance-of-plant equipment or do not directly affect plant equipment are assessed for plant operability.

The assessment and prioritization of an HED follows the flow chart in Figure 6-1.

6.2.1 Review Process

The human factors consultants and plant operators have the lead role in determining the potential for error due to an HED. Their combined knowledge of plant operation and extensive experience are supplemented with a list of questions derived from Exhibit 2-2 of the Standard Review Plan, Section 18.0, NUREG-0800.

The HED is reviewed for the consequences of a potential error. Significant safety consequences include:

- Unsafe operation or the violation of a Technical Specification, safety limit, or limiting condition of operation

- Unavailability of a safety-related system needed to mitigate transients or to safely shut down the plant

- A challenge to the safety-related systems in shutting down the plant (e.g., a reactor trip or a safety injection)

HEDs judged not to lead to errors of significant safety consequence are further reviewed for operability concerns, including plant availability, plant efficiency, and plant reliability.

When the review team cannot reach consensus, the majority opinion is used. The rationale is recorded when the assessment may not be clear. Dissent is recorded separately on the assessment form (Figure 6-2), directly below the rationale section.

HEDs are assessed during meetings of the DCRDR team by members of the core review team, including human factors consultant(s), plant operator(s), and specialists as required to supplement discussions of particular HEDs, such as HEDs concerning the fire alarm system and HEDs generated during the performance of the environmental and workspace survey and the SFRTA. A typical HED begins with one member of the review team reading the HED. Typically, the HED originator then explains the HED further so that all review team members clearly understand the deficiency. A number of visual aids and references are routinely available, including plant systems drawings, instrument drawings, electrical schematics, a copy of the EOPs, a half-size photomosaic mockup, and other photographs. If the nature or extent of a particular HED is not clear, a volunteer from the review team researches it and reports his findings.

After an HED is assessed by the multidisciplinary review team, potential corrective actions are discussed. This method takes advantage of the expertise of engineering, plant operations, and human factors personnel who have a full and common understanding of the HED. The team explores immediately required compensatory corrections and longer-term optimal remedial options. These are documented on the assessment form as recommended

corrections. Optimal corrections are recommended without a cost-benefit analysis. Review team members take special care to ensure the independence of the HED assessments and recommendations.

6.2.2 HED Priorities

HEDs assessed as having safety significance are rated as Priorities I and II, while those with plant operability concerns only are rated as Priorities III and IV. HEDs with high potential for error and significant safety consequences receive a Priority I rating. HEDs with significant safety consequences but low potential for error receive a Priority II rating. HEDs with little or no safety consequences but with operability concerns receive a Priority III rating if the potential for error is high and a Priority IV rating if the potential for error is low. HEDs with no safety consequence or operability concern are rated N/A and are not addressed for correction. HEDs already corrected at the time of assessment receive a Priority C rating. The acceptability of the correction for Priority C HEDs is verified during the verification and validation task (Section 8).

For Priority 1 HEDs, an interim compensatory action (ICA) or summary justification for continued operation (JCO) is included under the Recommended Correction Plan section of the HED assessment form (Figure 6-2). The ICA serves to reduce the HED's safety significance or potential for error until a final correction can be implemented. A JCO is documented when it is impractical to implement an interim fix or when no correction is planned.

6.2.3 HED Assessment Packages

In order to make the HED database more manageable and to promote consistency of assessments, similar HEDs are grouped into assessment packages. For example, HEDs that are duplicates, nearly identical, or complementary are grouped together, and specific HEDs of the same type are grouped with a related generic HED. The assessment package includes the HED, the completed assessment form numbered to correspond to that HED, and all related HEDs that

are included in that assessment. Since a conservative approach is taken, some lower priority HEDs are included in assessment packages that have been given a higher priority. In no case is a higher priority HED included in a lower priority assessment package. Combining related HEDs into assessment packages not only facilitates the HED assessment process, it also ensures that related HEDs are corrected similarly and fully.

Completed HED assessment packages are forwarded to members of the DCRDR management team for concurrence.

6.3 RESULTS

To date, more than 880 individual HEDs have been recorded and grouped into assessment packages. Their priority status is as follows:

	PRIORITY	NUMBER OF HED ASSESSMENT PACKAGES
1	High safety significance	57
2	Some safety significance	28
3	High operability concern	89
4	Low operability concern	154
N/A	Invalid or no concern	156
C	HED corrected before assessment	92

212 HEDs are either duplicates or are included in the above assessment packages.

Approximately 100 HEDs have not been formally assessed or require further study.

A complete listing of all HEDs, with priorities, is included in Volume 2 of this report.

6.4 CUMULATIVE EFFECTS OF LOW PRIORITY HEDs

Uncorrected low priority HEDs have been reviewed for interactive effects with other uncorrected HEDs affecting the control room equipment and panels as described below. Where interactive effects between two or more HEDs result in a higher potential for operator error, the HEDs are given the higher priority.

To facilitate the review of interactive effects, the HEDs were first sorted by location for specific panels or consoles. HEDs not related to a particular panel or console were considered generically for all panels. Within each location grouping, the HEDs were further sorted to eliminate from consideration corrected HEDs and HEDs assessed as N/A. Since many of the Priority I and II HEDs had been previously corrected, the remainder were mostly uncorrected, low priority HEDs. At this point, a human factors specialist reviewed the low priority HEDs for a given console along with the generic HEDs to find clusters of HEDs that in unison would appear to pose higher priority problems. Several clusters of potentially interactive HEDs were identified. These clusters will be reviewed by the review team for possible priority changes.

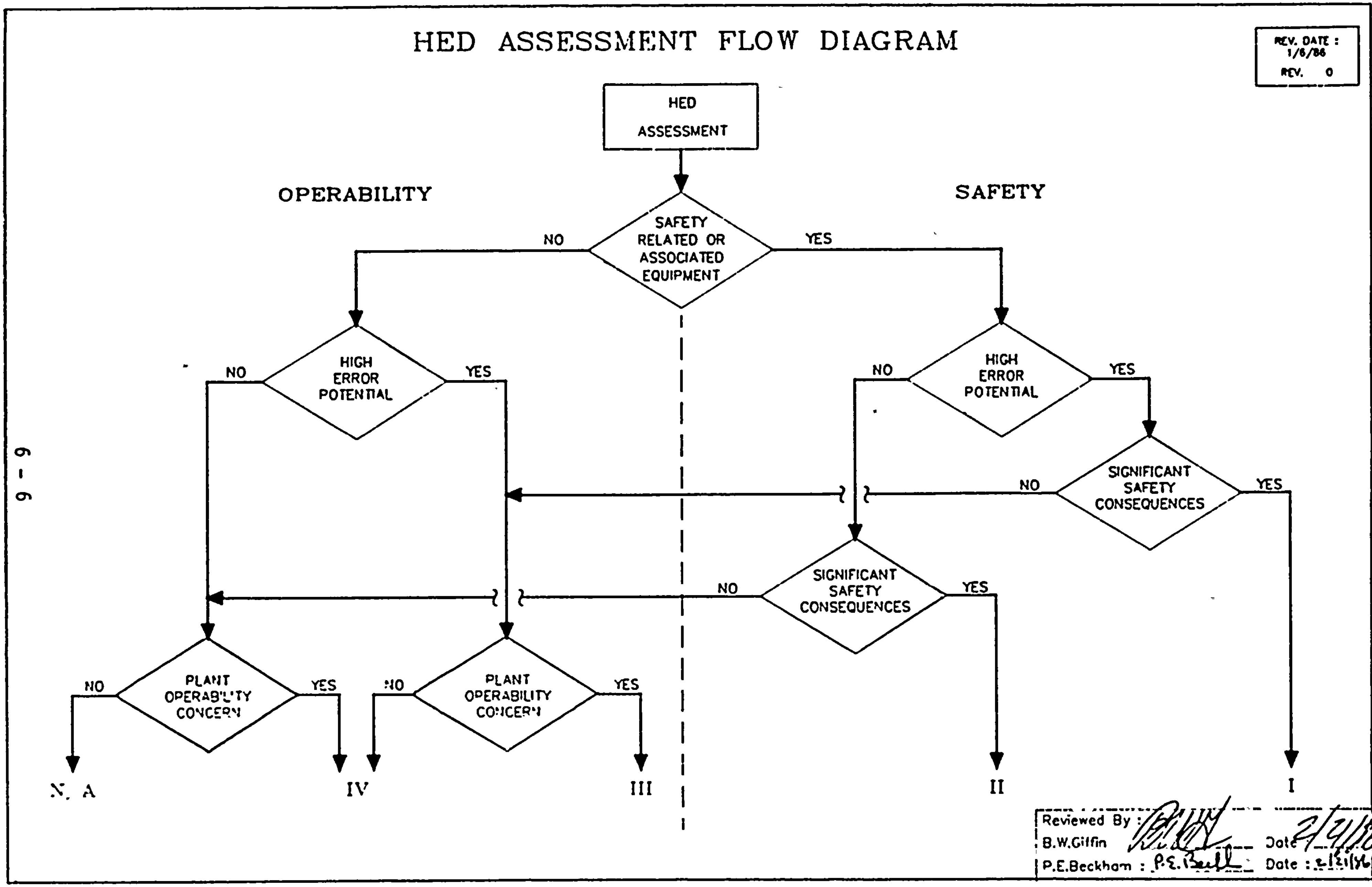


Figure 6-1 HED Assessment Flow Diagram

CHRON # __________

CATEGORY __________

HUMAN ENGINEERING DISCREPANCY ASSESSMENT

TITLE/SUBJECT ____
(Complete description is on attached HED)

ASSESSMENT:

Reviewers					
______ ______	SAFETY RELATED:	YES	NO	
______ ______	HIGH ERROR POTENTIAL:	YES	NO	
______ ______	SIGNIFICANT SAFETY CONSEQUENCE:	YES	NO	N/A
______ ______	OPERABILITY CONCERN:	YES	NO	N/A

____
Date

PRIORITY: ______

RATIONALE:

DISSENT:

RECOMMENDED CORRECTION PLAN:

A. Interim Compensatory Actions/Justification for Continued Operation:

B. Suggested Optimal Correction:

CONCURRENCE: ______________________________ __________
Management Team Representative Date

REJECTED BY: ________________ Reason:

Figure 6-2 HED Assessment Form

7 SELECTION OF DESIGN IMPROVEMENTS

7.1 INTRODUCTION

The main objective of the design improvement process is the correction of the HEDs. The evaluation and preimplementation phases of the design improvement are performed systematically and integrate all relevant HEDs associated with one or more deficiencies of an apparatus and/or system. All design improvements meet the rigorous requirements of PG&E DCPP seismic analysis, safety evaluation, and quality assurance.

7.2 DESIGN ENHANCEMENT PROCESS

As noted in Section 6, the initial resolutions of HEDs are developed during the assessment meetings immediately after each HED is prioritized. This is just the first step of the rigorous development and review process for design improvements.

7.2.1 Development of Guidelines

To ensure that design improvements are consistent and fully resolve the identified human factors deficiencies, PG&E developed Human Factors Enhancement Guidelines. These guidelines provide human factors guidance in the areas of operator workspace, panel layout, controls, displays, panel labeling, and annunciators. These guidelines have been used by the designers of the DCRDR-related changes. The human factors enhancement guidelines will be finalized and incorporated into permanent engineering guidelines to provide design guidance for future changes.

7.2.2 Determination of Design Improvements

Surface panel enhancements and physical modifications were typically developed using a half-size photomosaic mockup. Proposed modifications were depicted on clear overlays placed over the mockup. The enhancement designs were reviewed by the core review team and refined as necessary. The completed changes on

the mockup were then presented to plant operators for comments during training at the plant simulator. Comments from operators were reviewed for possible incorporation.

Human factors studies and I&C investigations were conducted to resolve special issues. For example, size-graduated control room labels with differently colored backgrounds were investigated for readability and operator preferences. In another study, controls located at the edge of the benchboard were reviewed for vulnerability to accidental activation and the consequences of potential accidents.

7.3 DESIGN CHANGE PROCESS

Verified panel enhancement designs, which have been reviewed by a human factors specialist (HFS) and corrected for compliance with sound human engineering principles, are prepared as design change packages (DCPs).

NECS Engineering is responsible for preparation of the DCPs using Nuclear Engineering Manual Procedure 3.6 ON. This procedure ensures that the enhancement design changes are traceable and properly documented in revised plant drawings, and are implemented systematically in conformance with nuclear industry standards and PG&E's Quality Assurance and Quality Control procedures for implementing changes. DCPs are reviewed for safety by NECS Engineering in accordance with 10 CFR 50.59 and then are reviewed and approved by the DCPP Plant Staff Review Committee (PSRC).

The following major steps delineate the process of implementing DCPs:

- The DCRDR team establishes an HED corrective action and recommends a change to a control room panel.

- DCPs are prepared by NECS Engineering, which includes several DCRDR members. The DCPs also receive a safety evaluation review.

- The DCPP PSRC accepts the safety evaluation of the DCPs.

- NECS Construction completes the work and submits as-built drawings. DCRDR members are involved in this process by checking work and assisting construction. Therefore, changes developed at this stage are reviewed by the DCRDR team members and documented. DCPs for panel surface enhancements (labeling, demarcation, etc.) follow the same procedural requirements as DCPs for physical modification, except that their implementation is performed by plant personnel as directed by a DCRDR team member, with support from the HFS, DCPP Operations, and NECS Construction.

- DCPP Operations accepts the completed work.

- NECS Engineering closes the DCPs and updates documentation. Several DCRDR members are involved in this process.

- The DCRDR team verifies the completed work and documentation.

As an example, Unit 1 DCP No. J-37348, "RHR Pump Control Switches Relocation at Board VB1 Mimic," is provided in Appendix C.

7.4 IMPLEMENTATION

Many HEDs have already been corrected in either one or both units at DCPP. The changes to date have included physical modifications as well as surface panel enhancements. Resolutions of individual HEDs, along with their implementation schedules, are included on the HED summary sheets in Volume 2, Appendix F.

7.4.1 Surface Panel Enhancements

Surface panel enhancements to the Unit 2 main control boards and PAM panels consisted of new size-graduated, hierarchical, black-on-light-background labeling and lines of demarcation to show functional groupings of related instruments by plant systems. The black borders on the control switch modules were painted out to reduce visual clutter and to improve the visibility of the

functional demarcation lines and board mimics. These enhancements were completed during the Unit 2 first refueling outage to allow for complete repainting of the panels and incorporation of physical changes also implemented. The labeling and demarcation will be fully completed during the second refueling outage for Unit 1 for similar reasons. An example of a DCPP Unit 2 labeling and demarcation DCP, No. J-38114, is provided in Appendix D.

7.4.2 Physical Modifications

Various controls and displays on the Unit 2 main control boards were rearranged during the Unit 2 first refueling outage to eliminate problems of mirror-imaging with Unit 1, to provide a more logical order of components, to improve functional groupings, to improve readability of high-placed indications, and to prevent the accidental actuation of selected devices. Other changes included the replacement of obsolete multipoint recorders and the revision of various control operations. Modifications to the Unit 2 PAM panels included the complete replacement of all the strip chart recorders and relocation of indications and controls to improve functional groupings. Corresponding Unit 1 modifications are scheduled during the Unit 1 second refueling outage.

8 VERIFICATION AND VALIDATION

8.1 INTRODUCTION

Verification and validation ensures that HED corrections are implemented correctly and do not create new human factors concerns. This iterative process begins during the selection of design improvements and continues through the completion of design implementation. During the verification and validation process, HED corrections are reviewed to ensure that: (1) the corrections meet the intent of the recommendations made by the multidisciplinary DCRDR team, (2) the corrections are implemented in accordance with the design, and (3) the corrections do not create any new human factors concerns.

8.2 METHODS

8.2.1 Verification That Correction Meets Intent of Recommendation

In most cases, the design engineers responsible for developing the design change packages (DCPs) for the DCRDR-related changes are either members of the DCRDR core review team or are directly supervised by core team members. Where this is not the case, the designs are subsequently reviewed by members of the core review team during DCP coordination activities.

Occasionally during the selection of a design improvement, it is determined that the final correction should be different from what was originally recommended. The reason for this may be that the original recommendation did not adequately resolve the HED, was not cost effective, or was not practical; or a simpler, more effective resolution may have been found. Before a design different from the original recommendation is issued, it is discussed with other members of the core review team, including a human factors specialist and a DCPP Operations representative.

The DCPs for the HED corrections are written in accordance with Nuclear Engineering Manual Procedure 3.6 ON. This procedure contains a list of plant

design concerns and commitments, including the Control Room Design Review, that must be considered prior to the final issuance of the design. A DCP affecting the control room must contain documentation that a proper review has been performed before it receives final approval.

8.2.2 Verification That Corrections Were Implemented Correctly

The implementation of HED-related DCPs, like all design changes at DCPP, is verified according to standard PG&E practices and procedures which are overlapping and complementary. Typically, a PG&E inspector oversees work performed by in-house construction forces or contractor personnel. The inspector mediates between the constructing personnel and the discipline engineer to resolve any problems in implementing the DCP. When the change cannot be implemented in exact accordance with the DCP, a field change (FC) may be issued to resolve the conflict. The FC must be coordinated with the responsible discipline engineer and cannot be used to change the intent of the original design. Minor differences between the actual implementation and the design may be documented on a field change transmittal (FCT) (as-built) drawing.

Members of the DCRDR core review team, including those from NECS Engineering, NOS, and DCPP Operations, oversaw the implementation of the HED-related changes and participated in the approval of any field changes.

Plant Quality Control (QC) personnel review plant changes prior to their acceptance. They inspect the completed work and perform inspections at any intermediate hold points required. The inspections are based on the design drawings. The QC personnel also ensure that any differences between the design and the actual field implementation are properly documented with the appropriate FC(s) or FCT(s).

After a change is physically completed, plant personnel perform a functional and/or loop test to verify that the change was implemented correctly and that it meets the intent of the design. For typical HED-related changes involving relocation or changeout of devices, tests are made to verify that wiring has

been reterminated correctly and that instrument loops are within the required limits for accuracy.

Final closeout of the DCP is performed by the discipline engineer who reviews the closeout documentation, including all FC(s) or FCT(s) attached to the DCP. If the documentation shows that the change was implemented per the intent of the original design, the DCP is forwarded to NECS Design Drafting to update the applicable design drawings. If the change as installed does not meet the intent of the original design, the discipline engineer issues a new DCP to correct the deficiency.

8.2.3 Verification That No New Human Factors Concerns Were Created

After a change is implemented, the new plant configuration is reviewed in light of the original HED to ensure that the HED is corrected and that no additional human factors concerns have been created. The human factors specialist is one of the reviewers, and control room operators are available to assist as needed.

When most of the changes to the simulator are completed, the review team, including human factors specialist(s), will walk through the EOPs to determine if any human factors problems still exist. This walkthrough will also verify that the changes in wording of equipment descriptions that were made during the enhancement process have been completely incorporated into the EOPs. A procedure for conducting these validation exercises will be formalized.

8.3 RESULTS

The results of the verification and validation process are recorded on the form shown in Figure 8-1. The verification and validation form includes the correction implemented to resolve the HED, justification for any differences from the original review team recommendations, reference to the implementing document(s), record of verification and validation acceptability, the reviewers' initials with date of review, and DCRDR management team concurrence.

The verification results are entered into the computerized HED database, which offers an effective mechanism for tracking the status of HEDs.

To date, approximately 200 HED assessment packages have been verified for Unit 2, and approximately 50 have been verified for Unit 1.

CHRON # ________

CATEGORY ________

HUMAN ENGINEERING DESCREPANCY CORRECTION

TITLE/SUBJECT: ______________________________
(Complete description is on attached HED)

CORRECTION:

IMPLEMENTATION:

	SIMULATOR	UNIT 1	UNIT 2
Design Change Package No.:			
Other Request (AR, Memo):			
Issue Date:			
Completion Date:			

VERIFICATION AND VALIDATION:

	SIMULATOR 1st rev/2nd rev	UNIT 1 1st rev/2nd rev	UNIT 2 1st rev/2nd rev
Acceptable (YES or NO):	/	/	/
Date:	/	/	/
Reviewer(s):	/	/	/

Comments:

CONCURRENCE: ______________________________ ________
Management Team Representative   Date

Figure 8-1 HED Correction Form

9 COORDINATION OF DCRDR WITH OTHER IMPROVEMENT PROGRAMS

9.1 INTRODUCTION

PG&E has integrated DCRDR program activities with safety parameter display system (SPDS) enhancement, Regulatory Guide 1.97 instrumentation, upgrading of emergency operating procedures (EOPs), and operator training in order to enhance the operators' ability to comprehend plant conditions and cope with emergencies.

The functional relationships of the integrated programs are shown in Figure 9-1.

9.2 PURPOSE

The purpose of this program integration is to ensure that the design changes recommended in these programs are approached systematically to optimize the man-machine interface in the control room.

9.3 TRAINING PROGRAMS

The DCRDR team has been supported by the Diablo Canyon training organization from the inception of the program. During Phase I of the DCRDR program, the simulator was made available to allow early experimentation and evaluation of candidate approaches for surface enhancement by functional demarcation and hierarchical labeling. One entire console in the simulator (vertical board No. 2) was functionally demarcated and relabeled after initial trial efforts were conducted to establish enhancement standards. This enhanced console was reviewed by operators and management with the assistance of training personnel. Training personnel have also advised the DCRDR team on the relative merits of alternative enhancement approaches.

The training organization has reviewed all DCPs relating to control room enhancement, and operators have been trained on these changes during simulator training or classroom training, using formal, structured lesson plans.

The training organization plans additional training on the DCRDR program to include:

- A short historical perspective
- A review of the guidelines for demarcation and labeling of components
- A review of any additional modifications that have not previously been addressed or are in the planning stage

Subsequent to the additional classroom training program, the operators and STAs will be required to complete a handout that lists all panel changes on Units 1 and 2. This handout requires the trainee to review each change made to the boards and provide initials to indicate his awareness of the changes. The completed handout forms will become part of the training record. Operators are also invited to provide the DCRDR team with comments on the quality and completeness of panel enhancements.

9.4 EMERGENCY OPERATING PROCEDURES (EOPS)

The DCRDR team has maintained close contact with the Diablo Canyon procedure writers and review group. HEDs with procedural implications have been discussed with this group during meetings at both the plant site and at PG&E headquarters. HED resolutions that involved procedure modifications were incorporated into the latest version of the EOPs as part of the normal two-year review cycle for EOPs. As part of the Unit 2 relabeling program to introduce size-graduated, hierarchical labeling with improved label contrast, the new labeling was reviewed to ensure that any modified component, subsystem, and system labeling was reflected in the procedures. The standard abbreviation list developed by the DCRDR team was coordinated with procedure writers to ensure consistency. Members of the DCRDR team are invited to attend meetings of the Diablo Canyon Operation Procedure Review Board (OPRB) when human factors issues are to be discussed.

The overlapping interests of the DCRDR team and the procedure writers led to a special human factors review of Diablo Canyon EOPs. This review consisted of

structured interviews with a sample of 13 operators and STAs and a checklist review of EOPs regarding format requirements. The interview and checklist were based on NUREG-0899, "Guidelines for the Preparation of Emergency Operating Procedures," and NUREG-0700.

The interview posed 26 questions regarding the content, format, readability, usability, and location of EOPs. The checklist contained over 150 evaluation criteria against which the EOPs were reviewed. The results of the interviews are summarized in Figure 9-2. Potential improvements were translated into specific HEDs, and remedial options were discussed with the procedure writers for resolution. The DCRDR team plans to verify the adequacy of procedural HED resolutions by using the verification and validation approach described in Section 8. When control room enhancements are completed in the simulator, EOP walkthroughs will be conducted to perform simultaneous human factors reviews of the latest version of the EOPs and of the adequacy of enhancement measures.

9.5 REGULATORY GUIDE 1.97 POSTACCIDENT MONITORING SYSTEM (PAMS) INSTRUMENTATION

9.5.1 Introduction

During Phase II of the DCRDR, a comprehensive review of the requirements for the Reg Guide 1.97 instrumentation was performed. This review consisted of operator interviews specifically designed for the PAMS instrumentation, a system function review and task analysis to determine informational needs, and a survey of the PAMS instrumentation to verify that the informational needs are met with suitable instrumentation. Comparison of instrumentation to human factors guidelines was included as part of the overall DCRDR effort.

HEDs developed in the PAMS review were included in the DCRDR database for review and assessment by the DCRDR review team.

9.5.2 Review of PAMS Instrumentation

9.5.2.1 Structured Operator Interviews

Structured operator interviews relating specfically to the PAMS were performed

by the DCRDR review team, including a human factors specialist. Questions considered the availability, usability, and identification of the PAMS instrumentation.

9.5.2.2 System Function Review and Task Analysis (SFRTA)

The PAMS instrumentation, including that which is located on the PAM panels, was reviewed during the SFRTA performed by General Physics Corporation, as described in Section 3.

9.5.2.3 PAMS Instrumentation Survey

A survey was made of the PAMS instrumentation requirements and availability in the control room. Considered in the survey were instrument range, accuracy, panel location, number of redundant channels, and usability of information format. This survey included a comparison of PAMS and normal operating requirements for cases where the PAMS function has been incorporated into an instrument used for normal plant operation.

9.5.3 Results

HEDs generated during the Phase I and Phase II reviews of the PAMS were included in the DCRDR database. The assessment and correction of these HEDs are being performed as part of the overall DCRDR.

Changes implemented to date include the rearrangement of instrumentation on Unit 2 PAM panels 1 and 2 to improve functional groupings, and the replacement of all PAM panel recorders with improved models. Hierarchical, size-graduated labeling and lines of demarcation were applied to all the PAM panels in accordance with the design guidelines developed for the main control boards.

9.6 SAFETY PARAMETER DISPLAY SYSTEM (SPDS)

9.6.1 Introduction

The SPDS displays have been reviewed and are being enhanced to incorporate

human factors guidelines, the latest revision to the Westinghouse Owners Group (WOG) Emergency Response Guidelines (ERGs), and SPDS-specific enhancement guidelines. These enhancements have been coordinated with the human factors specialist, DCPP Operations, NOS, and NECS Engineering. The DCRDR Phase II Project Manager is also the Project Manager for any SPDS upgrade programs.

9.6.2 Interviews with Operations Personnel

A principal step in the reviews of the SPDS displays for human factors was the preparation and administration of structured interviews with DCPP Operations personnel to take advantage of the experience and insights of SPDS users, primarily the STAs. The interviews were conducted in the DCPP control room by the DCRDR team members including a human factors specialist. All comments were examined by the DCRDR team members, and deficiencies were written up as HEDs to be assessed and incorporated into plans for SPDS improvements.

9.6.3 SPDS Displays Survey

The DCRDR team members, including a human factors specialist, conducted a survey of the SPDS in the DCPP main control room, at the simulator, and at the emergency operating facility to determine the availability of SPDS displays. Considered in this investigation were the instruments' range, accuracy, types of displays, usability of information format, and compatibility of these displays with the instrumentation and alarms on the main control room boards and panels. The survey revealed the need for replacement of some parameters, regrouping and highlighting of some parameters, and SPDS display enhancements based on human factors enhancement guidelines.

9.6.4 SPDS Improvement Design

The DCRDR team members developed human factors enhancement guidelines for revision of SPDS diplays. Key criteria included in the guidelines were:

- Labeling
- Color coding

- Highlighting
- Parameter displays
- Response times

Based on these enhancement guidelines, structured operator interviews, and analysis of SPDS instrumentation, the following SPDS display modifications are planned:

- Complete revision of critical safety function (CSF) status trees in accordance with latest revision of the WOG ERGs

- Revision of SPDS primary displays

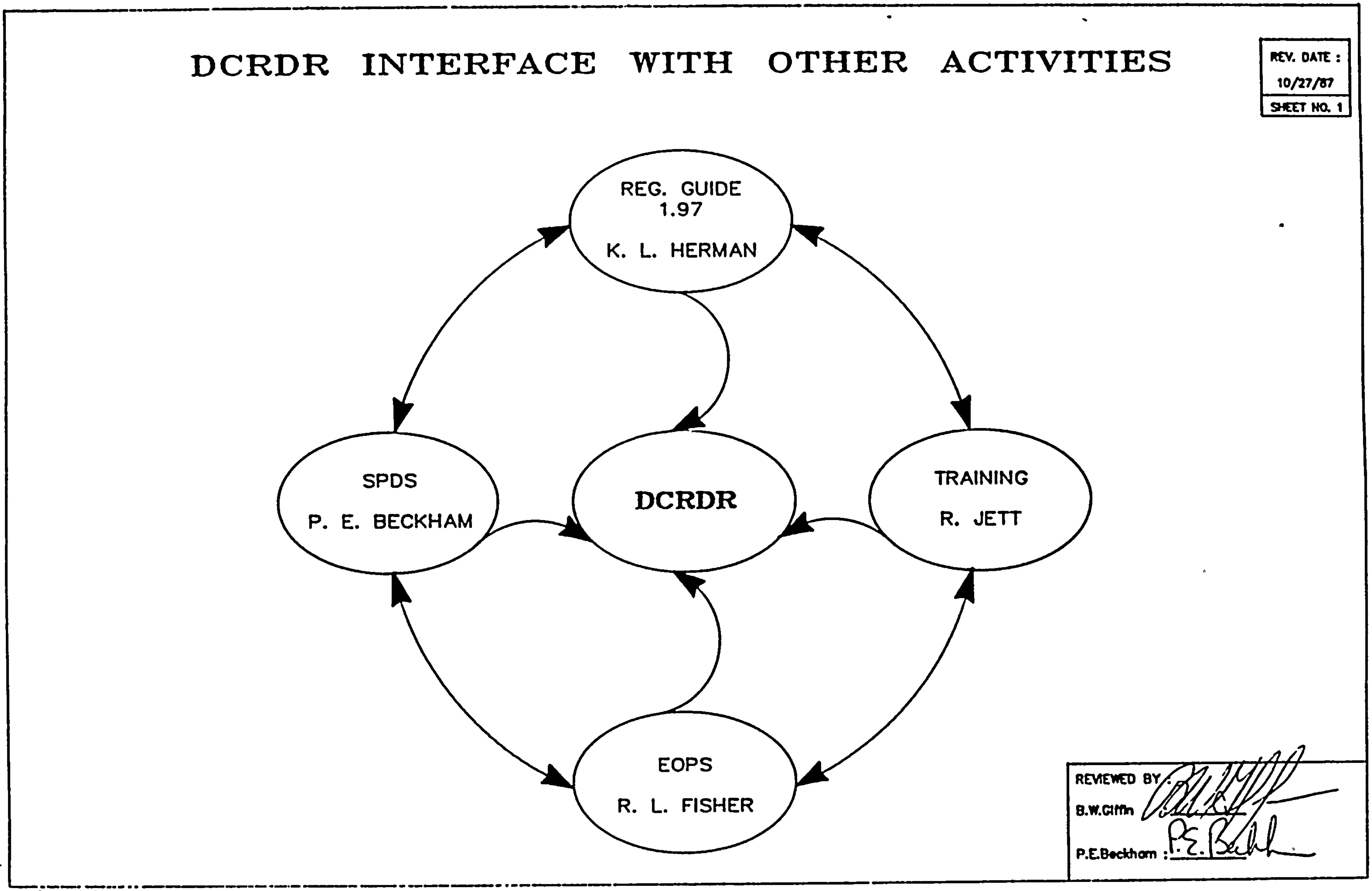
- Revision of SPDS secondary displays (EO, E1, etc.)

- Revision of SPDS pressure-temperature curves displays (heatup etc.)

- Revision of incore thermocouple displays

- Revision of radiation monitoring tabulation display

Consistency between the SPDS displays and corresponding control room displays is considered in the SPDS modification plans.



85A#6512044

Figure 9-1 DCRDR Interface with Other Activities

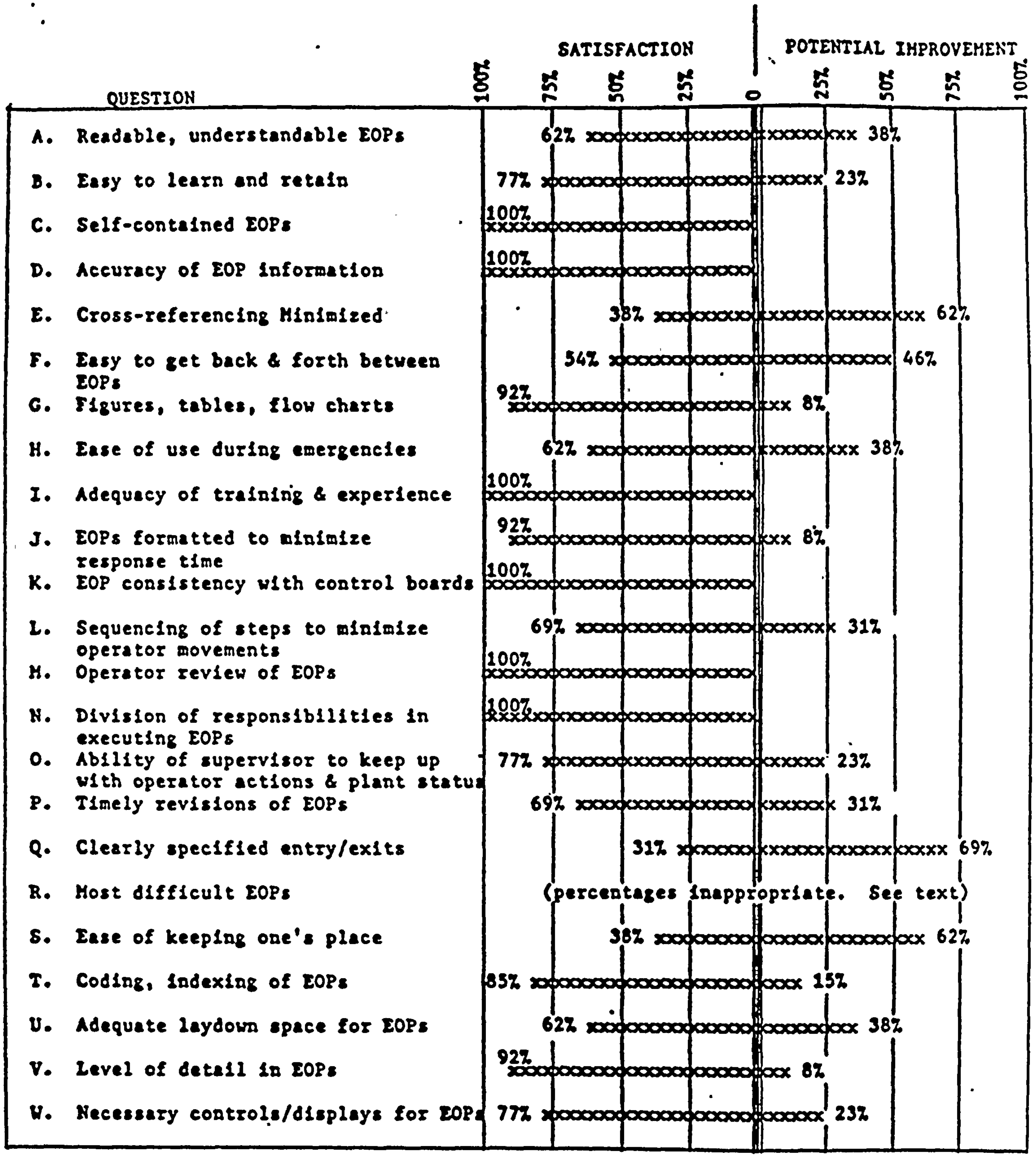


Figure 9-2 Results of EOP Interviews

10 CONCLUSION

Since meeting with the NRC in January 1986 (Reference 6), PG&E has made substantial progress in performing the DCRDR. The progress is evident in all nine DCRDR elements discussed in this report. Special attention has been given to increasing the role of the human factors specialists, improving documentation, and developing additional programs to further increase the DCRDR's effectiveness (e.g., training, enhancement guidelines). Furthermore, PG&E has followed the NRC's recommendations for the conduct of the system function review and task analysis. PG&E believes that the management, methods, and commitment expended on the DCRDR have enabled the project to fully comply with the requirements of Supplement 1 to NUREG-0737.

In addition to the ongoing process of selecting design improvements, implementing corrections, and verifying HED solutions, performance of the following activities is required for completion of the DCRDR:

- Complete formal assessment of approximately 100 residual HEDs (see Section 6.3)
- Reassess HEDs that may assume greater significance as a result of the cumulative effects review (see Section 6.4)
- Finalize the PG&E Human Factors Enhancement Guidelines (see Section 7.2.1)
- Formalize procedure for conducting validation exercises to simultaneously evaluate control room enhancements and updated EOPs (see Section 8.2.3)
- Provide additional classroom training for operators on the DCRDR program (see Section 9.3)

Enhancement of SPDS displays is planned to fully integrate the SPDS with other DCRDR program activities (see Section 9.6).

11 HUMAN ENGINEERING DISCREPANICIES

11.1 INTRODUCTION

This section describes the HED numerical listing, HED categories, and HED summary sheets. Volume 2 of this report provides the numerical listing of HEDs (Appendix E) and all HED summary sheets (Appendix F).

11.2 HED NUMERICAL LISTING

The numerical listing of HEDs in Appendix E provides the location of the HED, the type (category) of HED, the corresponding NUREG-0700 guideline (where applicable), the assessed priority of the HED as described in Section 6 of this report, and a summary description of the HED. The HED category should be used to cross-reference between the HED summary sheets and the HED list. HED categories are described in Section 11.3.

11.3 HED CATEGORIES

HED categories and their subgroups are briefly described below. The categories roughly parallel the guidelines provided in NUREG-0700 and were first introduced in the December 1984 Summary Report.

A - Control Room Design

HEDs in this category deal with environmental factors, work station location, passageways, control of access to the control room, and habitability.

A.1 TRAFFIC FLOW
Accessibility and freedom of movement within the control room.

A.2 EXCESSIVE PERSONNEL IN THE CONTROL ROOM
Access into the control room and the control board area.

A.3 SHIFT FOREMAN'S OFFICE
Accessibility of the Shift Foreman.

A.4 CONTROL OPERATOR WORKSTATION
The Control Operator's desk and visibility of the vertical boards.

A.5 CONTROL BOARD ACCESS LADDERS
Suitability of the existing large ladder.

A.6 RESTROOM ACCESS
Restroom usage and accessibility.

A.7 KITCHEN FACILITIES
Kitchen area usage and accessibility.

A.8 CONTROL ROOM DECOR
General decor of the control room including the carpeting and wall color.

A.9 STORAGE
General storage of plant and personal gear within the control room.

A.10 ILLUMINATION
Normal and backup lighting within the control room.

A.11 NOISE
Sources of undesirable noise within the control room.

A.12 TEMPERATURE
General environment including temperature, humidity, and air flow.

B - Panel Design

HEDs in this category deal with anthropometric considerations, distribution of system controls and displays, laydown space for procedures, visibility, and reachability.

B.1 ANTHROPOMETRIC FACTORS

Identification of controls and displays located outside the anthropometric limits for the fifth to ninety-fifth percentile operator.

B.2 CONTROL OPERATOR STATION

Anthropometric and desk-top availability of the CO desk.

B.3 CONCENTRIC CONSOLE CONFIGURATION

General problems created due to the concentric arrangement of the control console. More specific cases are covered in categories C.3 (control-display separations) and F.4 (labels sized for actual reading distances).

B.4 BACKRACK EQUIPMENT

Concerns with information and controls located behind the main control panels that may be required for diagnostic purposes, status monitoring, and control functions during both routine and emergency operations.

B.5 PROCEDURE LAYDOWN PROVISIONS

Adequacy of top surface of control console and rolling carts.

B.6 ALARM RESPONSES

Interrelationship of locations of CO desk, annunciator acknowledge buttons, and annunciator typewriter.

B.7 SPDS

Manning and use of the SPDS.

C - Panel Layout

HEDs in this category deal with organization of components, mimic arrangements, and functional groupings.

C.1 PANEL ORGANIZATION AND GROUPING

General lack of sufficient demarcation between systems and subsystems. General logic of existing groupings.

C.2 MIMIC PANEL ARRANGEMENTS

Standardization and adequacy of component identification on the mimics.

C.3 CONTROL-DISPLAY SEPARATIONS

Required displays located in a different area than the associated controls.

C.4 CARDOX AND DELUGE SYSTEM PANEL LAYOUT

General arrangement of the components within this panel.

C.5 MISCELLANEOUS PANEL ARRANGEMENT PROBLEMS

This includes abnormal numerical progressions of components, mirror-imaging between units, and apparent misplacing of panel elements with respect to related panel components.

D - Controls

HEDs in this category deal with ease of identification, ease of operation, direction of motion in relation to population stereotypes, and potential for accidental activation.

D.1 CONTROL ACTUATION FORCE

Specific controls that are too easy or too hard to operate.

D.2 CONTROL MOVEMENT DIRECTION

Control whose movement does not conform with population stereotypes, e.g., a clockwise rotation is associated with an increase in value or function actuation.

D.3 AMBIGUOUS CONTROL OPERATION CHARACTERISTICS

Lack of information or cues as to the operation of a control. Examples are spring return to neutral vs. fixed position, and switches requiring extended hold in open position to actuate.

D.4 CONTROL OPTION LABELING AND MARKINGS

Control options not labeled or more control markings than control options.

D.5 NONDISTINCT CONTROLS

Controls whose operation is not clear.

D.6 CONTROLLER ACTUATION FEEDBACK

Controls that do not have direct feedback.

D.7 CONTROLS THAT COULD BE RELOCATED TO SECONDARY LOCATIONS

Controls that operators have identified as not being required in the primary control area.

D.8 SPECIFIC CONTROL OPERATION CASES

Errors caused by misunderstanding or misoperation of controls.

D.9 ACCIDENTAL ACTIVATION OF CONTROLS

Controls that are near the edge of benchboards or project into aisleways.

D.10 GUARDING CRITICAL CONTROLS

Controls that require guarding to prevent inadvertent actuation.

E - Displays

HEDs in this category deal with the availability, usability, and readability of displays.

E.1 CONTROL/DISPLAY INFORMATION ADEQUACY

Information that has been identified as needed or desired by control room operators.

E.2 USABILITY OF DISPLAYED VALUES

Indicator units or range that are different from the parameter of interest.

E.3 DISPLAY CONTRAST/READABILITY

Indicators that are difficult to read due to contrast of scale or aging/dirtying of face.

E.4.a SCALE CHARACTERISTICS - GRADUATIONS

Scales that have more than the recommended nine graduations between numbered markings.

E.4.b SCALE CHARACTERISTICS - SUBDIVISIONS

Displays that use unorthodox scale progressions.

E.4.c SCALE CHARACTERISTICS - USE OF NONLINEAR SCALES

Indicators and recorders that use square root or logarithmic scales.

E.4.d SCALE CHARACTERISTICS - SCALE COMPATIBILITY

Adjacent indicators with identical or related information that do not have identical scales.

E.5 DIRECTIONALITY OF MOVEMENT AND NUMBERING

Value increases that are not in the appropriate direction.

E.6.a POINTERS - SEPARATION

Separation of pointer from associated scale for indicators and recorders.

E.6.b POINTERS - METER OBSCURATION

Meter pointers that may obscure numbered markings.

E.6.c POINTERS - OBSCURED

Meter or recorder pointers that are obscured from view.

E.7 BANDING

Meter banding material, range, and associated color coding.

E.8 MARKINGS ORIENTATION

Meter markings that are not on a horizontal plane.

E.9.a RECORDERS - IDENTIFICATION OF PENS, SCALES, CHANNELS

Recorder pens, scales, channels, etc. that are not fully identified by appropriate labeling or other means.

E.9.b RECORDERS - CHART PAPER
Recorder chart paper that does not match the scale.

E.9.c RECORDERS - PRINTED DIGITAL
General difficulty of reading printed digital multipoint recorders.

E.9.d RECORDERS - REQUIRED MAINTENANCE
Recorders that have misaligned pens or pointers, worn scales, etc.

E.10 COUNTERS
General human factors concerns with display counters.

E.11 MULTIBAND METERS
Indicators that have scales used for more than one range or purpose.

E.12 LEGEND PUSHBUTTONS/DISPLAYS
General cases where legend pushbuttons and/or indications do not meet guidelines.

E.13 LIGHT DISPLAY CODING
Inconsistencies in use of color coding for indicator light displays.

F - Labeling

HEDs in this category deal with accuracy, readability, and visibility of labels, including sizing for reading distances and ease of component identification.

F.1 COMPLETENESS OF LABELING
Panel elements that are not fully described by an appropriate label.

F.2 LABEL VISIBILITY
Labels that are obscured by large projecting knobs, operator's hand while operating, etc.

F.3 LABELING TERMINOLOGY

Labels that are not clear and concise.

F.4 LABEL SIZING FOR ACTUAL READING DISTANCE

Primarily cases where the control operator must monitor the vertical boards from the control console or his desk area.

F.5 LABEL CONTRAST

Labels that are hard to read due to poor letter-to-background contrast.

F.6 LABEL ORIENTATION

Labels that are not oriented in a horizontal plane.

F.7 LABEL MOUNTING

Labels that are coming off of panels.

F.8 COVERED LABELS

Labels that are covered by hanging tags.

F.9 LABEL PLACEMENT

Labels that are not clearly associated with their associated panel elements.

F.10 HIERARCHICAL, SIZE-GRADUATED LABELING

General lack of use of hierarchical, size-graduated labeling in the control room.

F.11 INCONSISTENT LABELING PRACTICES

Use of inconsistent letter heights, stroke widths, or label material.

G - Main Annunciator

HEDs in this category deal with annunciator window identification and labeling, the typewriter/printer, input identification and prioritization, and alarms. Related HEDs may also be found in category B.6 (Alarm Responses).

G.1 MAIN ANNUNCIATOR WINDOWS

Coding and identification of the main annunciator windows.

G.2 MAIN ANNUNCIATOR SYSTEM SUMMARY

Includes areas of alarm acknowledge buttons, audible alarms, systems shared between Unit 1 and Unit 2, multiple inputs, and prioritization of alarm inputs.

G.3 AUXILIARY ANNUNCIATORS

Interpretation and accessibility of remote annunciator readouts.

H - Plant Communication System

HEDS in this category deal with quality, usage, and coverage of the plant communication system.

H.1 PHONETIC REPRODUCTION

Quality of voice signal on telephones, radios, etc.

H.2 USAGE

Availability of phones, accessibility, cord lengths, etc.

H.3 COVERAGE

Code calls, PA system.

J - Emergency Operating Procedure (EOPs)

HEDs in this category deal with EOP availability and identity, clarity, completeness, operator familiarity, and correspondence to component control board identification.

J.1 PROCEDURE AVAILABILITY/IDENTITY

General availability of the EOPs and their indexing.

J.2 PROCEDURE CLARITY/GUIDANCE

Cases where procedure steps are not clear or where more guidance (specific criteria) may be required for the operator.

J.3 PROCEDURE COMPLETENESS

Information that is thought to be necessary is not included in the procedure.

J.4 TRAINING

Cases where operators may not be sufficiently trained to fully perform the EOPs.

J.5 PROCEDURE/LABEL CORRESPONDENCE

EOPs that use terminology different from the component label identification or different from common operator usage.

M - Maintenance Related

HEDs in this category deal with components that require maintenance, as well as designs of components or boards that hinder proper maintenance.

P - Miscellaneous

HEDs in this category do not conveniently fit into any other category.

S - Unit 1 vs. Unit 2 Differences

HEDs in this category deal with differences between the two units in terms of displayed information, information and controls availability, and plant operation.

11.4 HED SUMMARY SHEETS

Appendix F contains the HED summary sheets that make up the HED database for Diablo Canyon Power Plant Units 1 and 2. They are arranged by category for easy review of similar types of HEDs. They are also arranged in ascending numerical order within the categories. A complete numerical listing of HEDs is included in Appendix E.

Due to space limitations, some of the HED summary sheets contain abbreviated descriptions and comments. Complete information is available in the HED

assessment packages. The following is a key to the terms used on the summary sheets:

DATE - Last date that an entry has been either edited or viewed on a personal computer. Dates for assessment and verification are included on the HED assessment packages.

TRACKING STATUS - This field designates the current status of the HED. The entries for this field are defined below.

IDENTIFICATION: This is the period between the time when the HED is first documented and when it is assessed.

ASSESSMENT: This is the period between when the HED is assessed and when the assessment receives management team concurrence.

CORRECTION: This is the period between Management Team concurrence with the HED assessment and the actual issuance of a design change or alternative request for correction.

IMPLEMENTATION: This is the period between the actual issuance of a design change or request for correction and the closeout of the HED. Included in this time frame is the performance of the actual correction and the verification process.

COMPLETE: The HED has been corrected and verified, and the correction has received management team concurrence. HEDs that are assessed as N/A do not require correction and are considered complete as soon as the management team approves the assessment.

SOURCE - This field designates the process used to identify the HED. In some cases, the HED is identified through two or more processes. The entries for this field are defined below.

SURVEY: The HED was identified during the historical experience review, the original panel surveys to NUTAC guidelines, the supplementary surveys to NUREG-0700 guidelines, or the environmental and workspace survey.

INTERVIEW: The HED was identified either during the original operator interviews performed by human factors specialists from Lockheed Missiles and Space Corporation, or during the supplementary operator interviews concerning the EOPs format and the PAM panels (Reg Guide 1.97 instrumentation).

EOP VALIDATION: The HED was identified either during the original operator walkthroughs of the interim EOPs performed by the Lockheed human factors specialists and the core review team members, or during the system function review and task analysis performed by General Physics Corporation.

VERIFICATION: The HED was identified during the verification of HED corrections. A new HED is documented during this phase if it is different from the original HED that motivated the change.

DCRDR VERIFICATION: The HED was identified during the walkthroughs of the simulator and cannot be accounted for with an existing open HED.

CATEGORY - This field designates the type of HED. The HED categories follow the descriptions in the December 1984 DCRDR Summary Report. The groupings of HEDs roughly parallel those in NUREG-0700. A complete list of the categories used and a summary description of each is included in Section 11.3.

LOCATION - This field denotes the physical location of the device or subject of the HED. The location codes are:

B	-	General backrack area (behind the main control panels)
CC1, CC2, CC3	-	Control consoles 1, 2, and 3, respectively

CC4	-	The secondary P-250 computer console
DSP	-	Dedicated shutdown panel (supplementary panel to the remote shutdown panel)
EP	-	Operating procedure used by control room operators (EOPs, AOPs, etc.)
ES	-	Hot shutdown panel (remote shutdown panel)
F	-	Front (west) wall of the control room
G	-	Generic, nonspecific location (sometimes used interchangeably with location OP)
INCR	-	Incore instrumentation racks (within backrack area)
KV	-	500 kV panel (within backrack area)
NIS	-	Nuclear instrumentation system panels (within backrack area)
OP	-	Central operating area
PAM, PAM1, PAM2	-	Postaccident monitoring panels
VB1, VB2, VB3, VB4, VB5	-	Main control boards (vertical boards 1, 2, 3, 4, and 5 respectively)

COMPONENT - Device, procedure, equipment, etc. affected by the HED.

UNIT AFFECTED - HED applicable to DCPP Unit 1, Unit 2, or both.

DESCRIPTION OF HED - Summary description of the HED. Full descriptions are contained on the original HED identification forms.

RECOMMENDATIONS - Recommended corrective action for HED as determined by the multidisciplinary HED assessment team. For Priority 1 HEDs (those assessed as having high safety significance), recommendations are provided for (a) interim compensatory actions (ICAs) or justifications for continued operation (JCOs) and (b) optimal long-term resolutions.

CORRECTION - Actual correction used to resolve HED. If different from the recommended correction, justification for difference is provided.

Justification is also provided for HEDs that are not planned to be corrected or are only partially corrected.

VERIFICATION/VALIDATION - Verification and validation (V/V) of the HED corrections. These are performed jointly. Applicable terms for this field are:

YES	-	V/V is complete and acceptable for both units
YES U1	-	V/V is complete and acceptable for Unit 1 only
YES U2	-	V/V is complete and acceptable for Unit 2 only
PART FIX	-	HED is only partially corrected at the time of this review. This partial fix is not an acceptable final resolution.
NO	-	Correction of HED is complete but is not acceptable. Lack of acceptance may be due to correction not adequately resolving the HED, correction not implemented correctly, or correction creating a new human factors concern.

UNIT 1 DCR - Design change package used to correct HED for Unit 1.

UNIT 2 DCR - Design change package used to correct HED for Unit 2.

OTHER - Request other than a design change package used to correct an HED. This may be an Action Request number, letter number, etc.

PRIORITY - This is the priority assigned to the associated assessment package. The priority codes are:

1 - HED having high safety significance due to high potential for error and significant safety consequences should an error occur.

2 - HED having some safety significance due to significant safety consequences should an error occur but low potential for that error.

3 - HED having high operability concern due to high error potential but no safety significance should an error occur.

4 - HED having some operability concern but no safety significance.

N/A - HED having no safety significance or operability concerns of consequence.

C - HED that has been corrected or is in the process of being corrected at the time of assessment. Priority C HEDs require a verification of their corrections prior to being closed out.

INC - HED that is included or incorporated into the assessment for another HED. Priority INC HEDs include duplicate HEDs and specific HEDs that are included with generic HEDs for assessment and correction purposes. In the latter case, the specific HED is attached to the associated assessment package to ensure that each individual concern is included in the resolution.

IMPLEMENTATION PLAN - This is a general goal by which the HED resolution should be implemented. Specific schedule dates are not established until after a design change is initiated. Refueling outages are counted from commercial operation of the respective unit.

COMMENTS - This section is generally used to record the rationale of HED assessments and correction verifications. Cross-references to associated HEDs are also included here.

REFERENCES

1. PG&E letter to NRC dated August 2, 1983.
   Subject: DCPP Units 1 and 2 Control Room Design Review Program Plan.

2. PG&E Letter to NRC (DCL-84-392) dated December 28, 1984.
   Subject: DCPP Units 1 and 2 Control Room Design Review Summary Report.

3. NRC Letter to PG&E dated September 16, 1985.
   Subject: In-Progress Audit of DCRDR for DCPP Units 1 and 2.

4. NRC Letter to PG&E dated August 26, 1985.
   Subject: Issuance of DCPP Unit 2 Operating License DPR-82.

5. NRC Letter to PG&E dated December 9, 1985.
   Subject: November 6, 1985, Meeting Summary, DCPP Units 1 and 2 DCRDR.

6. NRC Letter to PG&E dated January 29, 1986.
   Subject: January 9, 1986, Meeting Summary, DCPP Units 1 and 2 DCRDR.

7. PG&E Letter to NRC (DCL-86-045) dated February 21, 1986.
   Subject: DCPP Units 1 and 2 DCRDR, Response to NRC Meeting Summaries dated December 9, 1985, and January 29, 1986.

8. PG&E Letter to NRC (DCL-86-096) dated April 10, 1986.
   Subject: DCPP Units 1 and 2 DCRDR Phase II Schedule Plan.

9. PG&E Letter to NRC (DCL-87-125) dated June 2, 1987.
   Subject: DCPP Units 1 and 2 DCRDR Status Report.

APPENDIX A

RESUMES OF DCRDR PROJECT PERSONNEL

Person	Title	Person	Title
P.E.Beckham	Project Manager	M.W.Dawson	Human Factors Consultant
B.W.Giffin	Management Team Leader	M.E.Jennex	Human Factors Consultant
F.J.Cucco	Review Team Leader	L.R.Schroeder	Human Factors Consultant
S.Auer	Management Team (NECS)	J.J.Vranicar	NOS Engineering
K.Herman	Management Team (NECS)	R.C.Washington	NOS Engineering
L.Womack	Management Team (DCPP)	J.B.Neale	NOS Engineering
J.A.Sexton	Management Team (DCPP)	S.A.Schaen	NOS Engineering
B.M.Grosse	NECS Engineering	R.L.Fisher	DCPP Engineering
J.J.Lisboa	NECS Engineering	T.W.Pellisero	DCPP Engineering
G.Seshagiri	NECS Engineering	S.R.Fridley	DCPP Operations
S.Wong	NECS Engineering	R.L.Ewing	DCPP Operations
J.Parris	NECS Engineering	C.G.Smith	DCPP Operations
J.L.Seminara	Human Factors Consultant	L.Waters	DCPP Operations
D.C.Burgy	Human Factors Consultant	R.Jett	DCPP Training
R.Danna	Human Factors Consultant	J.Lodge	DCPP Training

PETER E. BECKHAM
286 Redwood Road
San Anselmo, CA 94960
(415) 456-6571 (Home)
(415) 972-3815 (Work)

EXPERIENCE

4/67 to 4/71	UNITED STATES NAVY
	Operated and maintained the components associated with Naval Tactical Data System including Honeywell digital computer, Univac A/D Converter and telecommunication transmission equipment and radar control consoles.
6/75 to 9/75	U. C. BERKELEY
	Prepared and performed nuclear radiation/chemistry experiments on the TRIGA MK II test reactor.
5/77 to 11/77	GENERAL ELECTRIC COMPANY
	First quarter devoted to Nuclear Steam Supply and Balance of Plant familiarization (reactor vessel and internals, fuel design, emergency core cooling system, plant computer and instrumentation and balance of plant systems).
	Wrote detailed startup Test Instruction for domestic and foreign nuclear plants.
11/77 to 12/77	GARIGLIANO, ITALY NUCLEAR POWER PLANT
	Supervised plant personnel on the installation of nuclear instrumentation necessary for the acquisition of reactivity data for determination of control rod worth.
2/78 to 5/80	COARSO NUCLEAR POWER STATION, ITALY
	Prepared a detailed test instruction and a means of analysis on a system by system bases.
	Fuel inspection/loading and initial critical checks.
	Directed and supervised plant personnel during all phases of power and major transient testing. Verified ECCS design adequacy, SRV (Safety Relief Valve) capacity.
	Participated in primary containment load/stress evaluation and reactor core internal vibration programs.
	Wrote detailed test reports and analysis.
	Operated/maintained a 360-channel Hewelett Packard data acquisition system and peripherals.
	Performed reactor engineering related duties.

JAMES A. FITZPATRICK NUCLEAR POWER STATION
Oswego, NY

Reactor analyst

7/80 to 8/80 — Duties include: setting control rod pattern, thermal limits evaluation, fuel preconditioning and plant surveillance compliance.

Supervised in determination and fuel exposure accounting and daily plant operations.

9/80 to 2/81 — Successfully completed the Senior Reactor Operator training course in Tulsa, OK and obtained my Senior Reactor Operators Certification (BWR 6-Advanced Control Room).

2/81 to 5/81 — Prepared, verified and documented a Fortran program (Fortran 77) to calculate reactor core flow for any model of BWR. This program is available domestically and internationally via the GE Mark III Data System.

5/81 to 6/81 — Susquehanna Units 1 & II Steam Electric Station (PP&L).

Prepared the static and dynamic test cases for the plant process computer (Honeywell) including input signal calibration, nuclear instrumentation, software routines, alarm verification, accounting of fuel burnup and plant efficiency.

PACIFIC GAS AND ELECTRIC - NUCLEAR PLANT GENERATION

7/81 to 7/84 — Nuclear Generation Engineer

Evaluated nuclear industry operating experience for applicability to PGandE nuclear units. Researched and prepared comments on NRC proposed rule makings. Performed technical reviews on PGandE original licensing submittals. Prepared 10 CFR 50.59 safety evaluation for plant modifications. Prepared and performed verification and validation test program on the software associated with the Emergency Response Facility Data System at DCPP. The work has followed with a detailed system operation procedure and a written safety analysis for NRC submittal.

7/84 to 7/85 — Acting Senior Safety Review Engineer

In this capacity, I supervised eight professionals. I was responsible for directing the group's work activities, scheduling and monitoring employee performance. This position also required allocating resources for staff augmentation, contract negotiations and budgeting.

Additional responsibilities have included:

Diablo Canyon Probabilistic Risk Assessment
Scenario Development of the Diablo Canyon Emergency Exercise
Functional Testing and Upgrade of the Diablo Canyon S. G. Snubbers
Programming and Upgrading of the Emergency Response Facility Data System Color Graphic Displays
PGandE Representative for the Utility Group on the Station Blackout Issue

8/85 to Present — <u>Senior Nuclear Generation Engineer of the Plant Modification and Improvement Group (PM&I)</u>

Responsible for the activities of eight professional engineers and one clerical.

Activities include:
- Preparation of safety evaluations per 10CFR50.59
- Project Manager Control Room Design Review
- Safety Parameter Display System (SPDS) Upgrade
- Reliability Centered Maintenance (RCM) Pilot Program
- Plant Outage Support
- Expert System & Artificial Intelligence Pilot Program
- Plant Life Extension Salem ATWS 8328 Responses
- General Safety Issues
- Supporting the Plant Process Computer Upgrade
- Plant Availability Studies
- Probabilistic Risk Assessment (PRA)

Members of the PM&I Group represent PGandE in various industry groups, i.e. Station Blackout and EPRI. One group member is presently assigned to the Institute of Nuclear Power Operation (INPO).

Also responsible for budget preparation, employee performance evaluation and procedure development.

EDUCATION

United States Navy
- Electronics and Computer School

University of California, Berkeley, California
- Bachelor of Science in Nuclear and Civil Engineering

General Electric Company
- Boiling Water Reactor Technology
- Boiling Water Reactor Nuclear Instrumentation
- Station Nuclear Engineer
- Core Management Engineer
- Senior Reactor Operation (BWR 6 Certified)
- Professional Development

Pacific Gas and Electric Company

- Introduction to Nuclear Power Plant Operation - PWR Simulator
- Verification of Emergency Operating Procedures - PWR Simulator
- System Modeling Methods - Fault Tree and Event Tree Analysis - EPRI
- Data Acquisition System Training - Validyne Engineering Corporation
- Supervisory Skills Development - PGandE
- Zinger-Miller - PGandE
- Supervisory Leadership and Business Skills - PGandE

RESUME
OF
BRYANT W. GIFFIN

EDUCATION

North Carolina State University
- B. S., Electrical Engineering (1967)

U. S. Navy
- Nuclear Power Training

EMPLOYMENT HISTORY

1959-80	U. S. NAVY Retired as Lieutenant Commander
(1967-80)	Assigned to U. S. Navy nuclear ships and facilities.
1980-81	WESTINGHOUSE ELECTRIC COMPANY Senior Engineer, Nuclear Training Section of the Nuclear Service Division Responsible for preparation of nuclear-related training material.
	PACIFIC GAS AND ELECTRIC COMPANY
1981-84	Senior Nuclear Generation Engineer. Responsible for the review of industry operating experience, preparation of safety evaluations for plant modifications and the detailed control room design review.
1984-85	Instrument and Controls Maintenance Manager, Diablo Canyon Nuclear Power Plant
1985-Present	Supervising Nuclear Generation Engineer, Nuclear Operations Support
Present	Acting Manager, Nuclear Operations Support

F. JOSEPH CUCCO, JR.
Nuclear Generation Engineer

EDUCATION: B. S., Mechanical Engineering
University of California at Davis, 1975

SUPPLEMENTAL EDUCATION: Three-day extension course "Human Factors in Control Displays, UCLA, 1985.

One-day course "Human Factors Concepts for Nuclear Facilities", IEEE Symposium, Washington, DC, 1986

EXPERIENCE: PACIFIC GAS AND ELECTRIC COMPANY
Worked exclusively on the Diablo Canyon Nuclear Power Plant in various areas of construction, design and support. An active participant in the Detailed Control Room Design Review (DCRDR) since its inception in 1982.

June 1985 - Present: Currently assigned to the Nuclear Operations Support Department.

Major area of responsibility is with the DCRDR Project. Current DCRDR Review Team Leader.

April 1980 - June 1985: Assigned to the Instrumentation and Controls Section of the Mechanical and Nuclear Engineering Department.

As an engineer, responsible for I&C design and procurement for several plant systems including the plant auxiliary water supply, plant air, liquid and gaseous radwaste, fire protection and plant nitrogen and hydrogen gas.

Also Engineering Department's I&C coordinator for ALARA concerns and for spare parts procurement.

Member of the DCRDR Review Team.

July 1975 - April 1980: General Construction Department, Onsite Field Engineer.

Actively participated in the testing, trouble shooting and repair of plant instruments including those used in the main control room and emergency shutdown panels.

Responsible for writing test procedures for instrumentation and control loops for several plant systems and actively participated in startup testing.

SIEGFRIED AUER

EDUCATION

B. S., Electrical Engineering, University of Austria, 1949
Postgraduate studies, University of California, 1959-64

EXPERIENCE

1951 - 1959	CROWN ZELLERBACH CORPORATION Electrical Superintendent In charge of general maintenance of electrically powered equipment.
1961 - 1964	STANDARD OIL Electrical Engineer Involved in the construction (electrical parts) of chemical and oil refineries.
	PACIFIC GAS AND ELECTRIC COMPANY Electrical Engineer
1964 - 1966	Transmission Design Services. Built 500-kv system.
1967 - 1982	Electrical Distribution Engineering. Worked various positions for power supply to customers as electrical engineer, senior electrical engineer and electrical specialist.
1983 - Present	Diablo Canyon Nuclear Power Plant Project. Involved in all electrical aspects as supervising electrical engineer.

PROFESSIONAL DATA

Registered Professional Electrical Engineer in California.
Senior Member, Institute of Electrical and Electronics Engineers.

KLEMME L. HERMAN

RESUME

427 Montford Avenue
Mill Valley, California 94941

Telephone: (415) 383-5834 (Home)
(415) 943-4789 (Work)

PROFESSIONAL OBJECTIVE:

A challenging position in project management or technical management with good advancement opportunities.

EDUCATION:

B.S. Electrical Engineering (with honors), University of Wisconsin
B.S. Naval Science (with honors), University of Wisconsin
M.B.A. Portland State University (all academic requirements completed)
U.S. Naval Nuclear Power Training

EDUCATIONAL HONORS:

Awarded Sophomore and Senior honors; elected to Tau Beta Pi, Eta Kappa Nu, Phi Kappa Phi, Delta Epsilon and Scabbard and Blade honorary societies.

PROFESSIONAL EXPERIENCE:

Over twenty years experience in the operation, design, construction, and start-up of large electric generating stations. This experience includes project management, power plant operations, operational design review, preoperational testing, licensing, quality assurance, plant modification and administrative procedure development. This experience has been gained through a consulting firm, a utility and the U.S. Navy Nuclear Program.

1976 to Present - Impell Corporation (formerly EDS Nuclear), Walnut Creek, California. Presently CLIENT SPONSOR and PROJECT MANAGER for Pacific Gas and Electric Company and Portland General Electric Company. Responsible for coordination of all sales and marketing with these two clients and project manager of all work. Some recent projects include emergency planning (five years), plant information management system data base development, quality assurance contracting, pressurizer surge line analysis, reactor coolant loop hot leg analysis, relaxation of snubber testing criteria and IE bulletin 83-28 response.

Previously DIVISION MANAGER reporting to Western Region Regional Manager, San Francisco. Responsible as profit center for revenues, cost of operations, overhead, hiring and personnel assignments. Responsible for overall direction of assigned services including maintenance programs, operations support, start-up, quality assurance, emergency planning, cogeneration services, spare parts programs and management systems. Managed up to sixty professional employees.

1970 to 1976 - Portland General Electric Company, Portland, Oregon. INSTRUMENT AND CONTROL SUPERVISOR at the Trojan Nuclear Plant. Responsibilities included hiring and training of department personnel; planning and directing start-up of the instruments, computer and electrical controls; testing and maintenance of the computer and instruments; and establishing the administrative organization including the preparation of administrative procedures, surveillance test procedures, maintenance procedures and test equipment calibration procedures. To aid in this effort and to provide an orderly transition into power operations, planned and implemented a computer program for maintenance planning, spare parts control, surveillance testing and instrument calibration.

1963 to 1970 - OFFICER in the U.S. Navy Nuclear Submarine Force. Assignments included engineering and operations duties aboard a polaris submarine during five sixty-day patrols and aboard a fast attack submarine. In addition, served two years as Department Head and Instructor at the Naval Nuclear Power School, Vallejo, California.

PROFESSIONAL QUALIFICATIONS:

Registered Professional Engineer - Electrical, State of Oregon

PUBLICATIONS:

"Trojan Protection Systems and Testability," presented to the Instrument Society of America.

ACTIVITIES AND INTERESTS:

College - Several, highlighted by House President, Vice President and Social Chairman; Signa Phi Fraternity Rush Chairman; Dormitory Association Director of Intramural Sports; Flying Club and Freshman track.

Post College - Swimming (presently swim 4,000 yds. a week), youth program leader for YMCA and Boy Scouts, ski instructor, hiking, fishing, hunting, and private pilot.

Naval Reserve - Captain (06) in Naval Reserve. Positions held have included Commanding Officer, Executive Officer. Training Offi-

REFERENCES:

Available upon request.

LAWRENCE F. WOMACK
Engineering Manager

EMPLOYMENT HISTORY

July 1972 - June 1976	NUCLEAR PHYSICS GROUP, STANFORD UNIVERSITY Laboratory Assistant on a half-time basis in the Physics Department.
July 1976 - Feb. 1978	WESTINGHOUSE HANFORD COMPANY Employed in the Fast Flux Test Facility (FFTF) Operations Department in preparation for startup of the FFTF. Included a nine-week assignment at EBR-II at the Idaho National Engineering Laboratory.
	PACIFIC GAS AND ELECTRIC COMPANY
Mar. 1978 - Sept. 1979	Power Production Nuclear Engineer assigned to Diablo Canyon Power Plant.
Oct. 1979 - Oct. 1981	Promoted to Senior Power Production Nuclear Engineer.
Nov. 1981 - Feb. 1983	Transferred to position as Senior Power Production Engineer (Computer Support).
Mar. 1983 - Nov. 1983	Promoted to Assistant Power Plant Engineer
Dec. 1983 - Present	Promoted to Engineering Manger.

NUCLEAR EXPERIENCE

FFTF - Successfully completed operator's training course. Responsible for presodium fill testing and operation of plant systems as a shift Operations Engineer. Prepared operating procedures and training material for use by the Operations Department. Completed requirements for and received an Operation's Engineer Operating License for the FFTF.

EBR-II - Assigned to EBR-II from April 1977 to June 1977. Participated in accelerated operator training program and shift reactor operations. Completed written and oral qualification on reactor and plant systems before departure.

Diablo Canyon - Assistant Power Plant Engineer engaged in the supervision of procedure preparation, startup testing of plant systems and equipment and design change review.

Diablo Canyon - Successfully completed NRC Senior Reactor Operator Cold License Examination. License Number SOP-4276, issued March 12, 1982.

EDUCATION

B.S., Physics, Stanford University, 1975
M.S., Mechanical Engineering, Stanford University, 1976

FORMAL TRAINING COURSES

FFTF Operator Training Course, Fall 1976

Industrial Thermometry - Fundamentals, Calibration and Time Response, short course at the University of Tennessee Department of Nuclear Engineering, Fall 1978.

PWR Simulator Training - January 11-17, 1980. Westinghouse Training Facility, Zion, Illinois.

Senior Reactor Operator Training - Phase 2 and 3 training April through August 1981. Westinghouse Training Facility, Zion, Illinois.

Power Plant Simulation - Short course at the University of California at Los Angeles, School of Engineering, July 1982.

PWR Simulator SRO Licensing Requalification Training - November 1982. Westinghouse Training Facility, Zion, Illinois.

JAMES A. SEXTON
Operations Manager

EMPLOYMENT HISTORY

PACIFIC GAS AND ELECTRIC COMPANY

1958 - 1960	Assigned to various conventional oil- and gas-fired power plants as operator.
1962 - 1970	Assigned to Humboldt Bay Power Plant as operator. Progressed from Auxiliary Operator to Control Operator. Received AEC Operator's License.
1970 - 1972	Assigned to Humboldt Bay Power Plant as Assistant Engineer.
1973 (9 mos.)	STONE AND WEBSTER ENGINEERING Advisory Engineer on construction of James A. Fitzpatrick BWR.
1973 - 1976	BURNS AND ROE Senior Plant Test and Operations Engineer and Operations Test Supervisor for startup activities at various conventional power plants and at Three Mile Island and PWR and Washington Public Power Supply BWR.

PACIFIC GAS AND ELECTRIC COMPANY

May 1977 - Nov 1977	Employed by PGandE General Construction Department. Assigned to Diablo Canyon as startup Group Supervisor.
Nov 1977 - Oct 1979	Transferred to Steam Department. Assigned to Diablo Canyon as Power Production Nuclear Engineer.
Oct 1979 July 1980	Senior Power Production Engineer (Operations) at Diablo Canyon.
July 1980	Promoted to Supervisor of Operations. Received NRC Senior Reactor Operator's License, June 1981.
July 1983	Promoted to Operations Manager DCPP Units 1 and 2.

NUCLEAR EXPERIENCE

Humboldt Bay - Served as a plant operator for eight years. Started as Auxiliary Operator responsible for operation of plant auxiliary systems; including high and low voltage power distribution systems, reactor refueling operations, steam and feedwater system operations, turbine and generator support systems operation. Progressed to Control Operator responsible for reactor and turbine generator operations. Performed all control room operations; including reactor and turbine generator power changes, reactor control rod timing, turbine generator trip tests and reactor refueling. Received AEC Reactor Operator's License.

James a Fitzpatrick Nuclear Power Plant - Directed initial nuclear vessel cold hydrostatic test; assigned startup responsibilities on reactor cleanup and radiation waste facilities.

Three Mile Island Nuclear Power Plant - Responsible for originating startup procedures on nuclear plant safety and instrument systems.

Washington Public Power Supply System, BWR, Unit 2 - Directed the preparation of nuclear systems descriptions and plant startup schedule. Directed the construction testing of nuclear plant systems providing technical guidance and inspection to the constructors for Burns and Roe Incorporated.

Diablo Canyon Nuclear Power Plant - Participated in the startup activities of nuclear systems as a group supervisor. Responsible for the origination of surveillance test procedures as Power Production Engineer (Nuclear). Transferred to Operations Department as Senior Power Production Engineer (Operations) to provide technical support to the Operations Department. Promoted to Supervisor of Operations in 1980. Responsible for all operator training and NRC operator licensing, Operating and Emergency Operating procedures, all plant equipment operations and functions as the Unit 1 and 2 startup coordinator. Received NRC Senior Reactor Operator's License (Unit 1) in June 1981. Received Diablo Canyon Power Plant NRC Senior Reactor Operator's License (Dual Unit 4) in May 1985.

Westinghouse PWR Owners Group Emergency Procedures Subcommittee member, 1979 to 1981.

EDUCATION

B.S. Engineering, California State University at Humboldt, 1972

FORMAL TRAINING COURSES

Participated in all formal training courses leading to AEC Operator's License at Humboldt Bay.

Seven-day training program on Westinghouse PWR Simulator at Zion, Illinois, October 1978.

Three-day training program on Westinghouse PWR Simulator at Zion, Illinois, July 1979.

Participated in training courses in preparation for NRC Senior Operators License at Diablo Canyon Power Plant.

Participated in requalification training on Westinghouse PWR simulator (three- and five-day training programs) at Zion, Illinois, 1980, 1981, 1982 and 1983.

BRUCE M. GROSSE
Senior Electrical Engineer

EDUCATION: B.S., Electrical Engineering, University of Colorado, 1976

PROFESSIONAL DATA:

Registered Professional Electrical Engineer in California
Member - Institute of Electrical and Electronics Engineers
Power Engineering Society
Industrial Applications Society

EXPERIENCE:

Mr. Grosse has been employed by PGandE from 1976 to the present. He is currently working in the Electrical Engineering Department and has been assigned to the Diablo Canyon Project since April 1981. Mr. Grosse is responsible for electrical engineering and design for various plant systems including the nuclear control and protection instrumentation, radiation monitoring, nuclear instrumentation, main annunciation and other areas involving the main control room. Since January 1983, he has been an active participant in the Control Room Design Review.

From February 1980 to April 1981, Mr. Grosse was assigned to the Geysers Project where he provided electrical engineering and design of pollution control systems for geothermal power plants.

From June 1976 to January 1980, Mr. Grosse worked in PGandE's General Construction Department and was assigned to the Geysers Project. At the Geysers Project, he inspected all phases of power plant construction. He also performed startup testing on power plant and pollution control equipment and systems.

JOSEPH J. LISBOA

ENGINEERING EXPERIENCE

1973 - Present BECHTEL POWER CORPORATION, San Francisco, California

- (1985 - Present). During the last two years, he has been the Diablo Canyon DCRDR Instrumentation and Controls Team Leader providing support in the area of control room instrumentation, systematic functional task analysis, Reg. 1.97 instrumentation and SPDS. He is also a team leader of the Project Engineering group responsible for the DCRDR design changes and their implementation.

- (April 1982 - May 1983). Instrumentation and Controls Consultant Engineer for the Atomic Energy Corporation (AEC) and the Electric Supply Commission (ESCOM), both of South Africa. Consultancy done in South Africa.

  Major responsibilities involved engineering assessment of 2 X 900 MW Framatome French PWRs (3 loop reactors).

  - Deterministic analysis of the reactor safety systems design, installation and operation, applying USA and French standards, criteria and regulatory guides.

  - Recommendations on Post-TMI Issues. R.G. 1.97 Requirements and System Interaction.

  - Reactor safety system components seismic/environmental analysis.

  - Plant safe-shutdown.

  - Testing and Commissioning.

  - Probabilistic Risk Analysis (PRA). Developed new methods for failure modes, fault trees and their statistical quantification.

- (January 1981 - March 1982). Instrumentation and Controls Group Supervisor for the Monticello (BWR) Nuclear Plant, Minnesota.

  - Directed engineering design, procurement, QA/QE, Qualification (Seismic/Environmental), installation and testing of instrumentation for the Scram Discharge Volume, Containment Isolation and the Plant Control Room Habitability System.

- (October 1973 - December 1980). Instrumentation and Controls Group Supervisor for the Pebble Springs Nuclear Reactor (PWR), Oregon State.

  - Developed a new solid state interposing logic system for plant equipment control.
  - Directed all engineering design of nuclear and non-nuclear systems.
  - Resolved licensing issues with the client and reactor vendor in response to NRC regulations.
  - Established safety criteria for instrumentation installation, protection and separation.

1967 - 1973 EBASCO SERVICES, INC., New York City, New York

Head Instrumentation and Controls/Electrical Engineer for nuclear plants. Major projects directed:

USA	Vermont Yankee (BWR) Shearon-Harris (W-PWR)(4 units) Robinson II (W-PWR)
Japan	Fukushima Units I, II, III and IV (BWR) SHIMANE (BWR)
Taiwan	Chin-Shan Units 1 & 2 (BWR)

Major responsibilities included:

- Advised local and foreign contractors on Instrumentation/ Electrical engineering, QA/QE and licensing matters.
- Interpreted IEEE, ASME, AEC regulations for local and foreign plants.

1959 - 1966 CONSOLIDATED EDISON COMPANY, New York City, New York
1956 - 1959 ELECTRICAL COMPANY OF CHILE, SANTIAGO

Senior Electrical Engineer for hydraulic and steam driven power plants.

Home Office - designed high and low voltage transmission and distribution systems including substations, network systems, power centers and controllers.

ENGINEERING INSTRUCTOR

1979 - Present (Evenings) CITY COLLEGE OF SAN FRANCISCO

GOLDEN GATE UNIVERSITY OF SAN FRANCISCO
Graduate level instruction.

EDUCATION

M. B. A. (Management) - Golden Gate University Graduate School of Business Administration, San Francisco, CA, 1979

Postmaster Degree (Computer Engineering) - New York University Graduate Engineering School, New York, 1973

M. E. E. (Automatic Control System) - Cooper Union Graduate School of Engineering, New York, 1969

B. E. E. (Electronics) - The City University of New York, New York, 1965

E. E. (Power) - Technical State University of Santiago, Chile, 1955

PROFESSIONAL LICENSES

Professional Engineer

- Electrical Engineering: California and Oregon
- Instrumentation and Control Systems: California
- Engineering Instructor in Computer Systems, Communications, Electronics: California

AFFILIATIONS

Senior Member, Institute of Electronics Electrical Engineers (IEEE)

Member, National Society of Professional Engineers

N. GARY SESHAGIRI, AIA

PROFESSIONAL EXPERIENCE

Jan. 1974 - Present

BECHTEL WESTERN POWER CORPORATION
San Francisco, CA

Current Position: Architectural Supervisor

Duties: Supervising architectural planning and design projects including:
- Preparing facility plans and budgets
- Establishing design criteria and identifying user needs
- Space planning
- Coordinating with design architects, engineers, construction personnel and consultants

Present Assignment: Since 1984, has been supporting PGandE's planning and design for facilities for personnel at Diablo Canyon. These include administration, training,

communications and controls, warehousing and maintenance buildings.

Previous Assignments
- Identifying user needs, conceptual design and site master planning for personnel facilities expansion at the Trojan site (for Portland General Electric)
- Architectural study of personnel facilities at Humboldt Bay Power Plant (for PGandE)
- Architectural work for personnel facilities expansion at Susquehanna Steam Electric Station (for Pennsylvania Power and Light)
- Architectural work for Pilgrim Station Unit 2 project (for Boston Edison). This included a human factors review of the Control Room design.
- Architectural work for Jim Bridges Power Plant (for Idaho Power)
- Architectural work for service buildings at Eastern Province International Airport in Saudi Arabia

EDUCATION

Master's Degree in Architecture, University of California at Berkeley (1971)
Master's Degree in Architectural Engineering, Oklahoma State University (1966)
Bechtel Operations Certificate (1981)

PROFESSIONAL

Licensed Architect in State of California
Member, American Institute of Architects

SIMON L. WONG
270 BAYVIEW CIRCLE
SAN FRANCISCO, CA 94124
(415) 821-1269

ACADEMIC CREDENTIAL:	Bachelor of Science in Electrical Engineering San Jose State University 1982 December
	Master of Science in Electrical Engineering (in progress) Santa Clara University Finished 28 units out of 45 required units.
WORK EXPERIENCE:	PACIFIC GAS AND ELECTRIC COMPANY SAN FRANCISCO, CALIFORNIA
April '83 - Jan. '86	Design Engineer (Geysers Geothermal Power Plant)
	THE GEYSERS POWER PLANT UNIT 16 & 20 -responsible for the plant turbine building embedded conduits, raceway, lighting and grounding design.
	THE GEYSERS POWER PLANT UNIT 21 -responsible for the control schematics design for the following auxiliary plant systems. o Compressed Air System o Fire Protection System o H2S Abatement System o 480V & 4160V motor control, protection and alarm scheme.
	-responsible for the plant auxiliary power system automatic transfer design.
Feb. '86 - May '87	Engineer, DIABLO CANYON NUCLEAR POWER PLANT -responsible for the electrical engineering and design of the 480V to 120/208V AC power for the plant secondary chemistry lab and secondary process control room.
	-responsible for the electrical aspect of the control room design review and the electrical engineering analysis and redesign of the plant main control board.
	-responsible for the investigation of the 500KV main step up transformer top tank stray flux heating phenomenon and the generator terminal box overheating problem.
June '87 - Present	Field Engineer, ONSITE PROJECT ENGINEERING GROUP (OPEG) at Diablo Canyon Power Plant -provide field walkdown verification for the Seismic Induced System Interaction Program (SISIP)

-responsible for the engineering and design for the 500KV main step up transformer spare bank testing.

-provide engineering support to plant maintenance and operation activities.

PROFESSIONAL AFFILIATION: IEEE Power Society Member

TRAINING:

-Computer Aid Design Drafting (CADD)
-Nuclear Plant Operation & Systems Training

JENNIFER R. (JENN) PARRIS

HUMAN FACTORS EXPERIENCE

Bechtel Western Power Corporation and Bechtel Eastern Power Division

10 months

I designed human factors enhancements for implementation on the Diablo Canyon Power Plant control room panels. I developed the overall approach to packaging the utility's enhancement concept into engineering drawings, and reviewed each panel's design execution for standardized application of the concept. I was directly responsible for half of the packages, as well as for monitoring preparation of the remainder and fielding human factors questions as they arise. In addition to panel surface enhancements, I also prepared design packages for instrument relocation and replacement to resolve other human engineering discrepancies on these panels.

15 months

I developed sections of the supplemental control room design review report of the Turkey Point Nuclear Units for utility submittal to regulatory agencies. This report completed documentation of the plant's compliance with control room human factors guidelines and regulations. I evaluated plant changes for human factors impact and for closure of known human engineering discrepancies, organized annunciator system modifications into design packages and identified implementation methods for human factors enhancements. To do this, I worked closely with both plant operations personnel and with engineering representatives of the client utility.

4 months

At the Limerick Nuclear Plant, I was the Bechtel engineer responsible for the highly time-critical implementation of the plant's control room human factors enhancement concept. This involved comprehensive paint, label and mimic application for which I had to develop special masking materials, application techniques, and a new engineering drawing for each panel. I also provided around-the-clock direction for crews of drafters and skilled tradesmen. As this project was on the critical path for plant fuel load, its completion on schedule and successful evaluation by regulatory agencies saved the client considerable money and rework.

1 1/4 years

I was the assistant team leader of a 22,000 man-hour task which redesigned the main control room panels of the Midland Nuclear Plant to incorporate human factors enhancements. For this, I developed new specifications and drawing standards. My responsibilities also involved functional supervision of the team's engineers and drafters, and coordination with computer-aided drafting, other engineering disciplines and the client utility's engineers. The task involved several hundred drawings and I received a performance award for being instrumental in keeping it on schedule. Later, I monitored its implementation at the plant site, resolving problems and materials questions until total plant construction cancellation by the client halted the work.

RELATED EXPERIENCE

9 months — Proposal and Qualification Document Preparation

As a staff engineer for Bechtel National, Inc., I was proposal engineer or proposal manager for the preparation of Bechtel's submittals to potential clients within the defense, space, nuclear waste processing, and power generation industries. I developed document requirements, composed technical and graphic input, and monitored both the information flow and physical preparation of the documents. Written or graphic input I formulated included project execution plans, personnel resumes, and descriptions of the company's capabilities and relevant project experience.

1 1/2 years — Reliability Engineering

Initally at Bechtel, I was a staff reliability engineer. I assisted in failure mode and effects (FMEA) reviews, probabilistic risk assessment (PRA) analysis and report writing; developed sampling plans per MIL standards; and compiled a computerized reliability data base. Less typical assignments included preparing and presenting a guide to business and technical letter writing, and developing and producing graphic aids for an engineering training videotape.

2 1/2 years — Industrial Engineering

Prior to my graduate studies, I was an industrial engineer for Cessna Aircraft Company. I coordinated styling and production improvements to aircraft parts and cabin interiors, and developed capital expenditure studies for new manufacturing equipment.Before joining Cessna, I was a systems and business planning engineer fo ALCOA (Aluminum Company of America), where I developed a master product code system for use with both production scheduling and marketing computer programs. I also acquired and organized business trend data for corporate presentation.

EDUCATION

BS, Industrial Engineering, University of Tennessee,1977
MS, Industrial and Operations Engineering, Occupational Safety and Health Engineering Option, University of Michigan,1981
— Graphic Design, University of Cincinnati

PROFESSIONAL DATA

Engineer-In-Training certificate, Tennessee
Member, American Institute of Industrial Engineers, Human Factors Society, and Alpha Pi Mu (Industrial Engineering honor society)
Graduate, Lifespring Basic and Advanced Training

PERSONAL

Born April 26, 1954
U.S. Citizen
Security Clearance - DOE "Q", December 1986; inactive
Private Pilot's License, ASEL

REFERENCES

Available upon request

JOSEPH L. SEMINARA
Human Factors Consultant

EDUCATION	B.A., General Psychology, New York University, 1950 M.A., Experimental Psychology, New York University, 1952

PROFESSIONAL DATA	Fellow:	Human Factors Society (HFS)
	Former Member:	HFS Executive Council HFS Publication Board
	Recipient:	Jack A. Kraft Award, 1983

EXPERIENCE

Mr. Seminara has served as a private consultant for the power industry since 1979. He has participated in DCRDR programs for the following plants: Cooper, Fermi, Perry, Peachbottom, Diablo Canyon, Hatch and Taiwanese control rooms (Kuosheng and Chinsan). He is also performing a research study for the Electric Power Research Institute (EPRI) to determine future HF research needs based on an analysis of 25 DCRDR reports submitted to the U.S.N.R.C.

Mr. Seminara worked as a human factors specialist with the Lockheed Corporation for 26 years. During that time he participated in a wide variety of complex man-machine system development programs ranging from Polaris System missile checkout equipment to lunar vehicles and bases. Since 1975, he was principal investigator on a series of EPRI projects documented in the following reports:

- Seminara, J.L., and Parsons, S.O., Human Factors Review of Power Plant Maintainability, Electric Power Research Institute, Palo Alto, CA, EPRI NP-1567, February 1981.

- Seminara, J.L., et al., Human Factors Methods for Nuclear Control Room Design, Electric Power Research Institute, Palo Alto, CA, EPRI NP-1118, Summary Volume, June 1979.

  Volume I: Human Factors Enhancement of Existing Nuclear Control Rooms, November 1979

  Volume II: Human Factors Survey of Control Room Design Practices, November 1979

  Volume III: Human Factors Methods for Conventional Control Board Design, February 1980

  Volume IV: Human Factors Considerations for Advanced Control Board Design, March 1980

- Seminara, J.L., Gonzalez, W.R., and Parsons, S.O., Human Factors Review of Nuclear Power Plant Control Room Design, Electric Power Research Institute, Palo Alto, CA, EPRI NP-309, November 1976.

In addition to his Lockheed experience, Mr. Seminara was with the United Technology Center during 1962-63 as head of the Human Factors department and at the Feldman Research and Engineering Laboratories in New Jersey where he was involved in the human factors design of the U.S. Army ordnance systems. While at the Rome Air Development Center, Rome, New York in the mid-1950s, he applied human factors design principles to the development of Air Force ground electronics systems and was deeply involved with the formation of human factors standards for the Air Force. During his military service (1952-54), he was a research assistant at the Human Research Unit, Fort Ord, California.

Since 1971, he has spent approximately two years in eastern Europe conducting research to define the scope and character of ergonomics, human factors and psychology in communist countries. Nine separate scientific exchange visits were sponsored by the National Academy of Sciences and the International Research Exchanges Board.

The International Atomic Energy Agency of the U.N. has designated Mr. Seminara a technical expert in power plant control room design and evaluation and sent him on two missions to the Republic of Korea in 1983 and 1984.

Mr. Seminara conducted workshops in Sweden (1982) and Israel (1981 and 1983) dealing with human factors evaluation methods in reviewing power plant control rooms and maintainability effectiveness.

GP

DONALD C. BURGY
Director, Human Performance Systems

EDUCATION

Ph.D. Candidate, Applied-Experimental Psychology, Catholic University of America

M.A., Applied-Experimental Psychology, Catholic University of America

B.A., Psychology, Swarthmore College

EXPERIENCE
1979 - Present

General Physics Corporation

Special qualifications include human factors engineering, man-machine systems design and evaluation, information processing, display technology, man-computer interfaces, performance evaluation, training system development, and speech/non-speech. Applied research background includes an emphasis in auditory and visual perception methods, multivariate statistical analysis, mini/micro computer applications and software psychology.

Managed a major 18-month Nuclear Regulatory Commission (NRC) research program on nuclear power plant control room crew task analysis. A data collection approach and methodology used to conduct a task analysis of nuclear power plant control room crews was developed in this program. The task analysis methodology used in this project was discussed and compared to traditional task analysis and job analysis methods in a Program Plan report. The data collection was conducted at eight power plant sites by teams comprised of human factors and operations personnel. Plants were sampled according to NSSS vendor, vintage, simulator availability, architect-engineer, and control room configuration. The results of the data collection effort were compiled in a computerized task data base.

Additional task analytic experience has been for the Navy SUBACS (Submarine Advanced Combat Systems) program. The human factors aspects of the SUBACS project involved the development of task analysis formats and collection methodology for the Fire Control and Acoustic Subsystems in the early Concept

Development Phase. Team performance improvement and training enhancement were primary goals of the systems development effort.

Research and development experience has included two Electric Power Research Institute studies entitled (1) Survey and Analysis of Communication Problems in Nuclear Power Plants, and (2) Operability Design Review of Prototype Large Breeder Reactors. Methodology for collection and analysis of real-time field data in power plant control rooms was developed as part of the communications study. Function/Task analyses and operational sequence diagrams were generated as part of the operational design review that involved the evaluation of six breeder reactor designs in their early design phase.

Industrial experience in nuclear power plant control room reviews has included on-site field evaluations at River Bend, Indian Point 3, Hatch, North Anna, Surry, Zion, LaSalle, Susquehanna (Advanced Control Room Design), Zimmer, Shoreham Salem, and Trojan Stations. Evaluations have included the application of current NRC Human Factors Engineering guidelines and existing military standards (MIL-STD-1472C) to control room designs as well as field and laboratory experimentation to validate criteria used in design trade-off analyses.

1978 - 1979 **Consultant**

Private consulting in statistical design and analysis, computer programming and applications, microcomputer systems and software psychology.

1976 - 1978 **Catholic University, Human Performance Laboratory**

Research Assistant

Applied and basic research experiments conducted on auditory signal classification of complex underwater sounds. Research sponsored by the Human Factors Engineering branch of the Office of Naval Research. Additional research and related areas included auditory and visual pattern recognition, performance measurement and evaluation, multidimensional scaling, and computer-based systems for acoustic and experimental data analysis. Computer experience involved programming experimental events and subsequent data analysis on Digital Equipment Corporation PDP-8/e, PDP-11/34 and DECSystem-10 Computers.

1975 - 1976	Eagleville Hospital and Rehabilitation Center Research Assistant and Interviewer Interviewed study participants and assisted in data processing for an Alcohol Abuse Research Grant and coordinated all programming and clerical needs for a sub-study on Life Stress Events. Skills in programming included JCL, SPSS, PL/1, and FORTRAN on IBM 370/168 system.
PROFESSIONAL ORGANIZATIONS	Acoustical Society of America American Psychology Association Human Factors Society National Conference on the Use of On-Line Computers in Psychology Psychometric Society Psychonomic Society Software Psychology Society Sigma XI
AWARDS	Grant-in-Aid of Research, National Sigma XI (1978) Grant-in-Aid of Research, The Catholic University of America Chapter of Sigma XI (1978)
PUBLICATIONS	Burgy, D., Lempges, C., Miller, A., Schroeder, L., Van Cott, H., Paramore, B. Task Analysis of Nuclear Power Plant Control Room Crews: Project Approach and Methodology (NUREG/CR-3371, Vol. 1). Washington, D.C.: U.S. Nuclear Regulatory Commission, September 1983. Burgy, D., Lempges, C., Miller, A., Schroeder, L., Van Cott, H., Paramore, B. Task Analysis of Nuclear Power Plant Control Room Crews: Data Results (NUREG/CR-3371 Vol. II). Washington, D.C.: U.S. Nuclear Regulatory Commission, September 1983. Burgy, D., Lempges, C., Miller, A., Schroeder, L., Van Cott, H., Paramore, B. Task Analysis of Nuclear Power Plan Control Room Crews: Task Data Forms (NUREG/CR-3371, Vol. 3). Washington, D.C.: U.S. Nuclear Regulatory Commission, December 1984.

Burgy, D., Lempges, C., Miller, A., Schroeder, L., Van Cott, H., Paramore, B. Task Analysis of Nuclear Power Plan Control Room Crews: Task Data Forms. (NUREG/CR-3371, Vol. 3). Washington, D.C.: U.S. Nuclear Regulatory Commission, December 1984.

Burgy, D., and Schroeder, L. Nuclear Power Plan Control Room Crew Task Analysis Database: SEEK System. (NUREG/CR-3606) Washington, D.C.: U.S. Nuclear Regulatory Commission, May 1984.

Topmiller, D. A., Burgy, D. C., Roth, D. R., Doyle, P. A., and Espey, J. J. Survey and Analysis of Communications Problems in Nuclear Power Plants (EPRI RP 501-5). Electric Power Research Institute; Palo Alto, CA, September 1981.

Burgy, D. C., Doyle, P. A., Barsam, H. F., and Liddle, R. J. Applied Human Factors in Power Plant Design and Operation. Columbia, MD; General Physics Corporation, 1980.

Howard, J. H., Jr., and Burgy, D. C. "Structure Preserving Transformations in the Comparison of Complex Steady-State Sounds" (Technical Report ONR-78-6). Washington, D.C., The Catholic University of America Human Performance Laboratory, December 1978.

Howard, J. H., Jr., Ballas, J. A., and Burgy, D. C. "Feature Extraction and Decision Processes in the Classification of Amplitude Modulated Noise Patterns" (Technical Report ONR-78-4). Washington, D.C., The Catholic University of American Human Performance Laboratory, July 1978.

Howard, J. H., Jr., Burgy, D. C., and Ballas, J. A. "A Deglitching Circuit for the AA50 D/A Converter." Behavior Research Methods and Instrumentation, 1978, 10, (6), 858-860.

Burgy, D. C. "Hemispheric Asymmetries in the Perception of Non-Speech Sound Characteristics." Unpublished master's thesis, The Catholic University of America, May 1978.

Howard, J. H., Jr., and Burgy, D. C. "Selective and Non-Selective Preparation Enhancement Effects of an Accessory Visual Stimulus on Auditory Reaction Time." Unpublished manuscript, The Catholic University of America, 1977.

"River Bend Station Detailed Control Room Design Review Summary Report: Methodology and Results" (Gulf States Utilities Company). Columbia, MD, General Physics Corporation, September 1984.

"Human Factors Maintenance Plan" (Gulf States Utilities Company). Columbia, MD, General Physics Corporation, November 1984.

"Human Factors Criteria" (Mississippi Power & Light Company). Columbia, MD, General Physics Corporation, March 1985.

"Task Analysis of Emergency Diesel Generator Loading" (Long Island Lighting Company). Columbia, MD, General Physics Corporation, April 1985.

REPORTS

"Preliminary Human Factors Engineering Recommendations for Near-Term Improvements of the Surry Nuclear Station Control Room" (Virginia Electric & Power Company, GP-R-705). Columbia, MD, General Physics Corporation, June 1980.

"Preliminary Human Factors Engineering Recommendations for Near-Term Improvements of the Zion Power Station Control Room" (Commonwealth Edison Company, GP-R-708). Columbia, MD, General Physics Corporation, June 1980.

"Human Factors Engineering Recommendations for Near-Term Improvements of the Zimmer Nuclear Power Station Control Room:" (Cincinnati Gas and Electric Company), GP-R-13002). Columbia, MD, General Physics Corporation, August 1980.

"Summary of the LaSalle County Nuclear Generating Station Noise Report" Commonwealth Edison Company, GP-R-13010). Columbia, MD, General Physics Corporation, August 1980.

"Summary of the LaSalle County Nuclear Generation Station Lighting Survey" (Commonwealth Edison Company, GP-R-13011). Columbia, MD, General Physics Corporation, August 1980.

Human Factors Engineering "Considerations for Implementing a 'Green Board' at Zion Nuclear Generating Station" (Commonwealth Edison Company, GP-R-13008). Columbia, MD, General Physics Corporation, August 1980.

"Human Factors Engineering Meter Banding Study" (Commonwealth Edison Company, GP-R-13016). Columbia, MD, General Physics Corporation, September 1980.

SECURITY CLEARANCE	SECRET

ROBERT DANNA
Director, Engineering Services

EDUCATION

M.S., Environmental Engineering, University of Central Florida
M.A., Physics, Hunter College of the City University of New York
B.A., Physics, Hunter College of the City University of New York

LICENSES AND CERTIFICATIONS

Registered Professional Engineer (Mechanical): Maryland, California

EXPERIENCE
1980-Present

General Physics Corporation
Mr. Danna serves as Director of the Engineering Services Department, responsible for approximately 40 engineers and specialists with an annual budget of over $3.8 million. He has been responsible for projects from $5,000 to $1.9 million, all of which were completed on schedule and within budget. Representative projects include:

- Pressure Vessels/Systems (PV/S) Certification, Residual Life Assessment Services
  Served as Project Director for over 50 man-years of engineering support for PV/S Certification/Residual Life Assessment Projects currently underway or completed at the U.S. Air Force Eastern Space and Missile Center, NASA Kennedy Space Center, Jet Propulsion Laboratory, Edwards Test Station, White Sands Test Facility, and Goddard Space Flight Center. Activities included the evaluation of approximately 1000 pressure vessels and over 400 systems containing high pressure gases (to 10,000 psig), cryogenics, rocket fuels, and hydraulics. Conducted field surveys; performed engineering analysis to ASME Section VIII, Divisions 1 and 2, B31 and other national standards; developed configuration management programs; and prescribed and evaluated nondestructive examination and test requirements. NDE included acoustic emissions, liquid penetrant, magnetic particle, ultrasonics, and radiography. Testing included hydrotests (to 10,000 psig) and relief valve and component certification. Two burst tests, along with destructive testing (tensile, impact), were conducted.

- Codes, Standards, and Regulatory Requirements Training
  Served as Project Manager and Lead Instructor for the development and delivery of GP's Codes and Standards Course. Instructed over 30 sessions and 400 utility

personnel at clients including the Institute for Nuclear Power Operations, Commonwealth Edison, Florida Power Corporation, Georgia Power, Gulf States Utilities, Iowa Electric Light and Power, Portland General Electric, Southern California Edison, and Texas Utilities.

- Technical Staff Training Program Development
  Served as Project Director for the development of engineer training programs for 6 utilities including nuclear licensing, material science, nuclear plant chemistry, repair and replacement, plant modifications, and environmental qualification.

- Technical Services to the Nuclear Utility Industry
  Managed or coordinated numerous technical support contracts including procedure reviews and upgrades, system design description development, configuration management program development, low level radioactive waste studies, and plant operability reviews. In addition provided coordination for staff augmentation in mechanical and electrical engineering desciplines to utilities including Pennsylvania Power and Light, Pacific Gas and Electric; Southern California Edison, Virginia Power and Tennessee Valley Authority.

1975 - 1980 United States Navy
Mr. Danna was the Director of the Physics Division at the Naval Nuclear Power School. He developed and taught the curriculum, revised the text, and trained new instructors. He also taught reactor dynamics, core characteristics, and reactor principles.

1973 - 1975 Hunter College of the City University of New York
Mr. Danna was a Lecturer and Research Assistant in the Physics Department. He taught a two-semester course in physics to science majors. In addition, he developed computer simulations for the study of chemical structures by resonance spectroscopy.

PROFESSIONAL AFFILIATIONS:
Member, American Society of Mechanical Engineers
Member, American Society for Quality Control

PUBLICATIONS AND PRESENTATIONS
Mr. Danna has co-authored over 10 technical publications in subject areas including pressure vessel/system failure prevention program development, configuration management, techncial staff training, and low level radioactive waste management for professional organizations including American Society of Mechanical Engineers, American Nuclear Society, Institute of Environmental Sciences, and Society for Computer Simulation.

(3/87)

MICHAEL W. DAWSON
Manager, Program Development

EDUCATION

M.S. Candidate, Nuclear Engineering/Health Physics, University of Cincinnati

B.B.A., Business Management, National University

U.S. Navy Nuclear Power Training Program

LICENSES AND CERTIFICATIONS

Certified PWR Senior Reactor Operator

Certified Level III Quality Assurance in accordance with ANSI N45.2.6 for Administration, Documentation and Training; Level II Quality Assurance for Operations Inspections

Electrical Operator: U.S. Navy Nuclear Power Program

Engineering Laboratory Technician: U.S. Navy Nuclear Power Program

EXPERIENCE
1981 - Present

General Physics Corporation
Mr. Dawson provides engineering, training, and management consulting services to industry and government clients. As Manager of Program Development for the Engineering Services Department, he is directly responsible for the coordination of projects in the western U.S. from GP's San Diego Regional office. Representative projects include:

- Station Operations Services
  Prepared system operating and administrative procedures, annunciator response, test, and surveillance test procedures. Walked-down systems and verified as-built conditions. Developed and prepared a surveillance test program to implement Environmental Technical Specifications at a BWR power plant. Participated in the validation of Emergency Operating Procedures for a PWR power plant.

- Training Program Development
  Prepared lesson plans for Licensed Operator systems training. Developed the Basic Radiation Protection training course, including lesson plans, training aids and demonstrations at the William H. Zimmer Nuclear Power Station. Developed course materials for: the legal basis for industry Codes and

Standards; nuclear regulatory issues of reportability and unreviewed safety questions; management organization and staffing; quality control inspection; procurement regulatory requirements and procedures.

- <u>Training Services</u>
  Administered and taught Radiation Protection course, the GP Nuclear Power Plant Fundamentals courses, and the academic fundamentals portion of Licensed Operator training onsite for a client. Has taught portions of the academic fundamentals to operator and STA candidates onsite, and portions of the GP Codes and Standards course for Technical Staff Engineers. Has taught courses in: Quality Assurance Fundamentals, Regulatory Requirements, and Standards; Procurement Regulatory Requirements and Procedures; Responsibilities of Management Review and Audit Committees.

- <u>Quality Assurance/Program Development Services</u>
  Prepared site organization and QA Administration procedures, and participated in the rewrite of the site QA Manual. Developed and wrote the program instructions for a computerized nonconformance reporting system. Developed the design control program for a utility assuming these responsibilities from an A/E. Participated in the review of administrative and implementing procedures, and the QA Manuals of contractors and vendors for QA Program compliance. Performed the Quality Engineering review and disposition of nonconformances and procurement documents. Performed inspections and surveillances of operations department activities, and participated in the development of the department Quality Control Manual at the Diablo Canyon Power Plant. Participated in audits and management reviews of programs and procedures in subjects including nonconformance reporting and dispositions, document control, training, clearance and jumper control, document and system turnover from construction to operations, and design modification control.

- <u>Human Factors Engineering Services</u>
  Participated in Detailed Control Room Design Review as the SRO Subject Matter Expert at both a PWR and a BWR. These projects included Emergency Operating Procedure validation, control room walk-throughs, and the independent assessment of control room I&C.

MURRAY EUGENE JENNEX
Project Manager

EDUCATION

Professional Certification, Micro-Computer Engineering, University of California at San Diego Extension

Master of Business Administration with emphasis in Computer Information Systems, National University

U. S. Navy Surface Warfare Officers School

U. S. Navy Nuclear Prototype

U. S. Navy Officer Candidate School

Bachelor of Arts in Chemistry and Physics, William Jewell College

EXPERIENCE
1981 - Present

GENERAL PHYSICS CORPORATION

Mr. Jennex served as a member of the Integrated Leak Rate Test (ILRT) Team in Station Technical Power Generation Group at the San Onofre Site. This involved serving as a computer operator during the Unit 1 ILRT, with performance of several local leak rate tests (LLRT) on Units 1, 2 and 3 Containment Isolation Vales and airlocks, as well as planning for the Unit 2 ILRT, as a computer operator for the Unit 2 ILRT, and assisting in development of the Unit 2 and 3 ILRT and LLRT Procedures and being the primary author of the Computer Program to be used in performing all future San Onofre Site ILRT's. Additional engineering duties included dispositioning Nonconformance Reports and Site Problem Reports for Units 1, 2 and 3, and designing an Airlock Interlock Failure Alarm for the Unit 1 control room. Mr. Jennex also has served as the General Physics On-site Project Manager during this time. His duties for this have included supervising five (5) on-site Engineers and serving as the on-site representative for General Physics.

Mr. Jennex served as the Technical Programatic Administrative Support Group Lead for Station Technical Plant Betterment Group at the San Onofre Site. His duties during this time included the supervision of the Proposed Facility Change/Design Change Package (PFC/DCP) Clerical Staff, PFC/DCP planning for the current outages, Unit 1 Return to Service and for all uupcoming outages including

the Unit 2 refueling outage, and review of all outage PFC/DCPs for potential Technical Specification Restraint Impact. Mr. Jennex also continued to improve and develop the PFC Tracking and Logging Program resulting in an improved system being implemented that tracks all PFCs and Turnovers for Units 1, 2 and 3.

Mr. Jennex served as a Plant Betterment Engineer for the Nuclear Steam Supply System (NSSS) support group at the San Onofre Nuclear Generating Station (SONGS) Units 1, 2 and 3. He was responsible for designing and implementing a proposed facility change tracking and logging program using the IBM PC and dBASE III relational database. The effort included program generation, troubleshooting, clerical staff training, and user's manual development. His other duties included reviewing and approving proposed facility changes, system turnovers, temporary modifications to the plant, test procedures and results, and procedure changes. His primary responsibility was ensuring the safety of the plant by doing the safety reviews for these items. Auxiliary duties included assisting in training and planning for the NSSS support group. During this time, Mr. Jennex was involved in several planned and unplanned plant outages, gaining experience in outage planning and scheduling and in ensuring work was performed and accepted on time. Mr. Jennex also gained expertise in developing proposed facility change and system turnover procedures and in the developing of a temporary modification program.

Mr. Jennex served as the Senior Technical Writer and on-site Editor for the San Onofre Nuclear Generating Station (SONGS) Units 2 and 3 System Description Project. His duties included writing specific system descriptions and editing of all descriptions for technical accuracy. Mr. Jennex also served as the project liaison between General Physics and the client. Mr. Jennex's auxiliary duties included researching data voids for the SONGS 2 and 3 simulator project. During this time, Mr. Jennex has achieved a high degree of technical expertise on the British built GEC Turbine-Generator and the main feedwater pump, incore and excore detector, control element drive mechanism, and reactor protection systems. Prior to this assignment, Mr. Jennex completed an Emergency Operating Facility (EOF) shield evaluation for the Saint Francisville Nuclear Power Station owned by Gulf States Utility. This evaluation included calculating shield design thickness for the various radiation hazards following a design base accident.

As a Staff Specialist for General Physics, Mr. Jennex served as a PWR Simulator Instructor, specializing in Chemistry and Radiation Protection. He has completed an eleven (11) week in-house Instructor Training Course including eight (8) weeks of classroom academics and three (3) weeks of training and classroom work on the Sequoyah Nuclear Power Plant Simulator. His auxiliary duties included technical writing for the Vogtle Nuclear Power Plant simulator training manual and the development of training materials for the various Simulator Training Centers managed by General Physics.

1978 - 1981 **U.S. NAVAL NUCLEAR POWER PROGRAM**

As an Engineering Officer of the Watch, Mr. Jennex has two (2) years experience in the Naval Nuclear Program. He served as a qualified watchstander at AIW Prototype in Idaho, and has experience in plant operations and major shutdowns for overhaul. As an officer onboard the USS BAINBRIDGE, Mr. Jennex gained further experience in plant operations, supply problems, training and personnel management.

1975 - 1978 **CHEMISTRY DEPARTMENT, WILLIAM JEWELL COLLEGE**

As a Laboratory Assistant, Mr. Jennex spent three (3) academic years operating and supervising the freshman laboratory. He was also responsible for instruction and safety in the Laboratory. He assumed the job of Lead Lab Assistant in his senior year, which also included the duties of sample and stock solution preparations and storeroom supervision and management.

LOTHAR R. SCHROEDER
Manager, Industrial Systems Technology

EDUCATION

Ph.D., Experimental/Applied Psychology, Lehigh University

M.S., Engineering Psychology, Lehigh University

B.S., General Engineering, University of Illinois

B. A., Psychology, University of Illinois

EXPERIENCE

1982 - Present

<u>General Physics Corporation</u>

Dr. Schroeder's areas of expertise include job and task analysis, procedures validation, equipment design studies, operations research, and organizational design and management. He has managed numerous projects which have provided human factors integration services for utilities in meeting their emergency response capability requirements.

Dr. Schroeder has supported an NRC research project, applying control crew task analysis data in areas of human engineering design and staffing. He has also managed a follow-on research project for the NRC which has used the existing task analysis database to identify training needs and to evaluate emergency procedures.

Dr. Schroeder has participated in the evaluation of training programs for the Technology Transfer Group and has supported the General Motors Model Maintenance Project. He is currently managing a staffing study for the Ameritech Publishing Company. In addition, Dr. Schroeder has developed and given numerous supervisory skills and diagnostic skills workshops for operations and technical staff.

1981 - 1982

<u>U.N.C. Nuclear Industries</u>

Dr. Schroeder worked as a human factors specialist, interfacing with engineers and other staff in identifying and solving problems relating to equipment design, the use of procedures, and training efforts at Hanford's N-Reactor. He also performed a human factors review of the 105-N control room in support of an on-going control room upgrade program.

1974 - 1980	**Department of Psychology, Moravian College** Dr. Schroeder's responsibilities as Assistant Professor and Department Chairperson included planning and coordinating a day and evening program in psychology involving over 100 majors, serving on several college committees, supervising individual field study, independent study, and honors projects, and serving as academic advisor to day and evening session students having an interest in applied psychology.
1973	**Wigdahl Electric Company** Dr. Schroeder worked as a consultant, identifying potential organization problems and conducting problem solving sessions.
1972	**Jewish Employment and Vocational Services** As an industrial psychologist, Dr. Schroeder consulted with several industries and governmental agencies in order to develop, validate and administer "job-related" personnel selection tests under a Department of Labor contract.
PROFESSIONAL AFFILIATIONS	Member, Human Factors Society Member, American Nuclear Society
PUBLICATIONS	"A Human Factors Guided Survey for Systems Development," American Nuclear Society Winter Meeting, December 1981, coauthor with D. R. Fowler. "Control Room Human Factors in Context," American Nuclear Society Winter Meeting, November, 1982, coauthor with D. R. Fowler & D. E. Friar. "Learning Style Data Applied to Nuclear Power Plant Training Programs." American Nuclear Society Annual Meeting, June 1983. "Task Analysis of Nuclear Power Plant Control Room Crews, Vol.", NUREG/CR-3371, U. S. Nuclear Regulatory Commission, June 1983. Authored with D. Burgy, C. Lempges, A. Miller, H. Van Cott, and B. Paramore. "Crew Task Analysis Database: SEEK System Users Manual NUREG/CR-3606, U. S. Nuclear Regulatory Commission, Authored with D. Burgy, March 1984.

JOHN J. VRANICAR
CRDR Team Leader, Phase I

EDUCATION	B.S., Mechanical Engineering, California State Polytechnic University, 1971
PROFESSIONAL DATA	Registered Professional Mechanical Engineer, State of California

EXPERIENCE

Pacific Gas and Electric Company
1981 - Present

Mr. Vranicar has been employed by PGandE since 1981. He is a Senior Nuclear Generation Engineer in Nuclear Operations Support. He was project manager for Phase I of the Control Room Design Review and was active in all phases of Phase I.

Participated in the review and critique of the Westinghouse generic emergency procedures task analysis.

Participated with INPO development of guides for performing a control room design review.

Performed reviews and assessments of nuclear industry events and their applicability to Diablo Canyon.

Performed safety evaluations of Diablo Canyon design changes.

Reviewed and assessed Diablo Canyon Licensee Event Reports (LERs) for accuracy, applicability and appropriateness of corrective action.

Assisted in the development of the yearly site emergency drill and acted as a controller in the control room during the drill. Provided plant parameters and realistic situational scenarios.

Tenessee Valley Authority
1978 - 1981

Member of an interdisciplinary task force which resolved system reliability problems involving the high pressure coolant injection (HPCI) system at Browns Ferry Nuclear Plant.

Reviewed and approved test instructions, test results and coordinated changes to systems for the Sequoyah Nuclear Plant.

EXPERIENCE (cont'd)

Performed safety evaluations of design changes for Sequoyah and Browns Ferry plants.

Investigated and developed a scenario as to the probable cause of a failure to successfully shutdown the reactor at Browns Ferry. Assisted in the redesign of the shutdown system.

Diagnosed problems of inadequate flow, cavitation and instrumentation during hot functional testing of Sequoyah Nuclear Power Plant. Interfaced with site and design personnel to expedite corrective action.

U. S. Navy
1971 - 1977

Machinist Mate. Served as staff instructor in the Navy nuclear power program at Idaho Falls, Idaho.

Served as staff instructor for operators aboard the nuclear carrier Dwight D. Eisenhower. Received letter of commendation for training efforts. Developed qualification standard, lesson plans and performed crew training. Also qualified as engine room supervisor.

ROBERT C. WASHINGTON
4176 Pickwick Drive
Concord, CA 94521
(415) 825-8244 (Home)
(415) 972-7023 (Work)

EDUCATION

California Polytechnic State University, San Luis Obispo
- B. S., Electrical Engineering (December 1979)

EMPLOYMENT HISTORY

Aug. 1977 - Dec. 1977	BECHTEL POWER CORPORATION, LA POWER DIVISION Engineering Assistant, Control Systems Dept., responsible for the design of flue gas scubber instrumentation associated with fossil fuel power plants.
Feb. 1978 - Mar. 1979	CALIFORNIA POLYTECHNIC STATE UNIV. - ENGINEERING SERVICES Boiler Control Technician, assisted with the installation and calibration of new electronic/puemantic control systems on the university steam boilers.
	PACIFIC GAS AND ELECTRIC COMPANY
Mar. 1980 - May 1983	Instrument and Control (I&C) Field Engineer, General Construction, Diablo Canyon Nuclear Power Plant (DCPP), responsible for the scheduling, material, installation design coordination and post installation testing of assigned instrument systems during the construction phases of DCPP.
May 1983 - Apr. 1984	Unit 1 Lead I&C Engineer, General Construction, DCPP, supervision of unit 1 I&C field engineers and support staff.
Apr. 1984 - Nov. 1984	Unit 1 Test Supervisor, General Construction, DCPP, Lead I&C Engineer responsibilities plus the supervison of the unit 1 I&C technicians.
Nov. 1984 - Present	Nuclear Generation Engineer, Nuclear Operations Support - Operations Engineering Group, responsible for providing technical support, staff engineering, and logistical support to the nuclear plant organizations (DCPP and Humbolt Bay PP), in the area of instrumentation and control.

JAMES BARRY NEALE

EDUCATION: B.S. Nuclear Engineering, Purdue University, 1979
SRO License Training, 1982

EMPLOYMENT HISTORY: Commonwealth Edison Company
September 1979 - November 1984

Worked in various positions as:

Project Coordinator - Night shift coordinator for work in response to NRC inspection and enforcement bulletin concerning seismic anchor bolted plates. Supervised location, diagrammjing and systematic logging of as-built anchor plates. Coordinated work between the architect engineer and the piping contractor.

System Engineer - For nuclear instrumentation system, incore flux mapping system and backup engineer for rod control system. Responsible for boron follow and reactivity anomalies.

Shift Foreman - Supervision of equipment operators. Review operator equipment readings, management verification of periodic equipment tests and system valve lineups. Shift fire chief; conducted periodic fire drills.

Shift Control Room Engineer - Control room personnel supervisor.

Pacific Gas and Electric Company
December 1984 - Present

Nuclear Generation Engineer - Evaluated nuclear industry operating experience for applicability to PGandE nuclear units. Scenario development of the Diablo Canyon emergency exercise.

SHEILA A. SCHAEN
1031A Mohr Lane
Concord, CA 94518

PROFESSIONAL OBJECTIVE	A mechanical engineering position dealing with nuclear power plant systems and operation including interfacing with plant personnel and outside organizations. Travel preferred.

EDUCATION

1984 - 1985	688 Class Nuclear Propulsion Plant Test Engineer Training Course, Electric Boat Division of GENERAL DYNAMICS, Groton, CT
1978 - 1982	Bachelor of Science in Mechanical Engineering, University of Connecticut, Storrs, CT

EXPERIENCE

PACIFIC GAS AND ELECTRIC COMPANY
77 Beale Street, San Francisco, CA 94106

1986 - Present Nuclear Generation Engineer.

Participated in the Safety Parameter Display System Human Factors Review and Update, the Detailed Control Room Design Review Project and the Plant Process Computer Replacement Project. Involved in Safety Classification issues relating to the plant components in the PIMS database. Assisted in the Surveillance Testing of pipe supports during two plant refueling outages.

Received a Pacific Gas and Electric Performance Recognition Award after nine months in this position.

GENERAL DYNAMICS, Electric Boat Division
Eastern Point Road, Groton, CT 06340

1984 - 1986 Nuclear Test Engineer.

Participated in Electric Boat Division's Nuclear Test Engineer Training Program. Included in the training were general shipyard operations, coordination of naval, trade and technical organizations involved with the testing program and test procedure implementation along with detecting and resolving technical problems.

1982 - 1984 Mechanical Design Engineer.

Worked in Propulsion Engineering as a Project Engineer on the development of a titanium heat exchanger.

Summer 1980 & 1981 Quality Assurance Engineering Assistant.

Worked in Quality Assurance (QA) reviewing drawings for non-destructive testing requirements during Electric Boat Division's Weld Review Program and responded to shipyard questions involving interpretation of drawing with regards to installation and QA requirements.

RESUME

SENIOR POWER PRODUCTION ENGINEER (OPERATIONS)

ROBERT L. FISHER

1. Birthdate: August 22, 1987
2. Citizenship: USA
3. Education: B. S. Nuclear Engineering, 1977
   Pennsylvania State University

4 Employment History: Joined PGandE in January 1979

   a. December, 1966 - February, 1977 -- Active Duty U. S. Navy. Served as Nuclear Reactor Operator in submarine service.

   b. February, 1974 - June, 1977 -- Employed by Pennsylvania State University as Senior Reactor Operator. Participated in University courses and research projects where research reactor facility was utilized.

   c. July, 1977 to March, 1978 -- Employed by General Electric Company, BWRTC Morris, Illinois. Licensed as Senior Reactor Operator on Dresden BWR Unit 2. Worked as Training Engineer (Dresden BWR Simulator Instructor) preparing Senior Operator candidates for NRC licensing examinations.

   d. May, 1987 to December 1978 -- Employed by Pullman Power Products as Quality Assurance Supervisor at Diablo Canyon Power Plant.

   e. January, 1979 -- Employed by PGandE. Assigned to Diablo Canyon as Quality Control Engineer.

   f. April, 1981 -- Assigned to Operations Department as a Power Production Engineer.

   g. December, 1982 -- Promoted to Senior Power Production Engineer. Assigned to Operations Department

5. Nuclear Experience:

   a. U. S. Navy

      1) March, 1969 to October 1969 -- Operator training on the Navy/SIC, prototype reactor (CE-PWR) at the Naval Reactor Facility, Windsor, Connecticut. Qualified as Nuclear Reactor Operator.

      2) February, 1970 to February, 1974 -- Assigned to the Engineering Department; Reactor Controls Division, nuclear submarine (W-PWR). Qualified Nuclear Reactor Operator, Engineering Watch Supervisor and Engineering Watch Officer. Experience in operating and maintenance of reactor control and monitoring equipment.

   b. Pennsylvania State University -- Served as Senior Reactor Operator for three years as shift supervisor for University research and experiments involving the reactor. Received NRC Senior Operator License, Dresden Unit 2, DPR-19, March 1978 (SOP 3211).

   c. Dresden Nuclear Power Station, Unit 2 - Licensed as Senior Reactor Operator as part of job requirements in order to conduct formal training courses on General Electric BWR simulator, Morris, Illinois for Senior Reactor Operator candidates. Received NRC Senior Operator License, Dresden Unit 2, DPR-19, March, 1978 (SOP-3211)

   d. Diablo Canyon Nuclear Power Plant

      1. As Quality Assurance Supervisor, participated in Pullman Power Products Quality Assurance Program for construction of ASME Code class piping in Units 1 and 2.

      2. Quality Control Engineer engaged in implementing the Quality Assurance Program in All safety related areas of Units 1 and 2.

      3. On temporary assignment to Operating Department Training group, participated in Cold License Training Program as an instructor of RO and SRO cold license candidates.

      4. As Power Production Engineer, participated in the preparation and review of Administrative, Operating and Emergency Procedures. Provides technical and administrative assistance to the Operations Manager, Senior Operations Supervisor, Shift Foremen, and the Senior Power Production Engineer (Operations). Provides relief support for the Shift Foremen and Shift Technical Advisors. Received an NRC Senior Operators License on Unit 1, June, 1981 (SOP-3961).

5. As Senior Power Production Engineer, originates, revises, and reviews normal, abnormal, and emergency operating procedures. Directly responsible for supervision and coordination of Shift Engineers (Shift Technical Advisors) and Power Production Engineers (Operations). Interviews and recommends for hire Operations Engineers and Auxiliary Operators. Provides Technical and Administrative assistance to the Operations Manager including assumption of duties during vacation, training, or sick days.

   The following is a list of specific activities either participated in, originated, coordinated, or directly supervised as a Senior Power Production Engineer:

   Shift Manager through commercial operation of Unit 1, Shift Manager through initial criticality of Unit 2, change over to the Responsible Budget System, chemical cleaning of Unit 1 secondary, rad-waste system task force, air inleakage reduction task force, acquisition of General Electric Transient Anaylsis Recording System (GETARS), Westinghouse Owners Group Representative and Control Room Design Review Representative.

   Received an ammendment to NRC Senior Operator License to include Unit 2, January 1985. (SOP-3961).

   Participated in both Unit 1 and 2 refueling outages (Sept. 86 - July, 1987. During Unit 1 outage was temporarily assigned to the outage organization as shift outage coordinator. This position acted as the senior outage management person on backshift. My duties were to provide overall management direction of outage activities, resolve all conflicts effecting the critical path, priorities and resources on my shift. Report job status and offer recommendations for improvement to the outage manager.

   During the Unit 2 first refueling outage was again temporarily assigned to the outage organization as functional department outage coordinator. The primary function of my position was to represent the operations department in support of the overall outage effort. This included overall responsibility for Unit 2 with respect to the operating departments activities. I reported to the operations manager and the outage manager.

e. California Polytechnic State University - Received an NRC Senior Operator License in December, 1981, (SOP-4123), on the University's research reactor AGN-201, limited to activities necessary to dismantle the reactor.

6. Formal Training Courses

   a. Graduate of Eighteen months U.S. Naval Electronics Technician "A" School Great Lakes, Illinois - January, 1967 - June, 1968.

   b. Graduate of six months U.S. Naval Nuclear Power School, Bainbridge, Maryland - Sept., 1968 to March, 1969.

   c. Graduate of six months U.S. Naval Operations Training Prototype, SIC, Windsor, Connecticut - April, 1969 to October, 1969.

   d. Simulator Training - General Electric BWR Training Center, Morris Illinois. (Twelve week course, 1977).

   e. BWR Nuclear Engineers Course - General Electric BWR Training Center Morris, Illinois (March, 1978)

   f. Quality Assurance Auditing School, Bechtel Power Corporation, April 1980.

   g. Participated in training courses for preparation of NRC Senior Operating License at Diablo Canyon Power Plant.

   h. Participated and completed Westinghouse training course for Shift Technical Advisors at Diablo Canyon Power Plant.

   i. Completed Thermodynamics, Heat Transfer and Fluid Flow pre-license review course given by Energy Consultants, Inc. on December 12, 1980.

   j. Completed mitigated core damage course given by Westinghouse Electric Corporation, July 24, 1981.

   k. Simulator Training - Westinghouse Nuclear Training Center, Zion, Illinois.

      1) Option III, (14 day course), 1980

      2) Retraining, (5 day course), 1981

      3) retraining, (5 day course), 1982

   l. Simulator Training - NRC License Requalification Program, Diablo Canyon Power Plant Simulator.

TERRANCE W. PELLISERO

EDUCATION — B.S. Nuclear Engineering, University of Virginia, 1978

LICENSES/ REGISTRATIONS — P.E. in Mechanical Engineering (California)
USNRC RO License OP-4562 (September 1977)

EXPERIENCE

Duke Power Company, Catawba Nuclear Station
June 1978 - July 1980

Developed pre-operational test procedures for Catawba and supported testing at McGuire Nuclear Station and outages at Oconee Nuclear Station. Also involved in nuclear fuel contract negotiations during assignment in the Charlotte General Office.

Impell Corporation, Walnut Creek, CA
August 1980 - February 1984

Developed emergency planning, administrative and surveillance test procedures. Developed eight emergency plan exercise scenarios and participated as the control room controller. Developed and conducted emergency plan training. Developed nuclear project cost reviews and participated in equipment qualification projects. Participated in cogeneration project work.

Pacific Gas and Electric Co., Diablo Canyon Power Plant
February 1984 - Present

Operations Department: Full scope responsibility for revision and maintenance of the emergency operating procedures; revised operating procedures; responsible for safe shutdown portion of major Appendix R audit; member of WOG Operations subcommittee; relief STA; completed 9-month STA certification portion of the SRO training program.

Engineering Department: Revise surveillance test procedures and perform testing; engineer and implement solutions to plant problems.

STEPHEN R. FRIDLEY
Senior Operatons Supervisor

EMPLOYMENT HISTORY

PACIFIC GAS AND ELECTRIC COMPANY

Aug. 1970	Assigned at Humboldt Bay Power Plant in the Operations Group.
June 1973	Promoted to Assistant Control Operator, Humboldt Bay Power Plant.
July 1974	Promoted to Control Operator (Reactor Operator) and assigned to Diablo Canyon Power Plant.
July 1978	Promoted to Assistant Training Coordinator at Diablo Canyon Power Plant.
April 1980	Appointed Operator Training Specialist.
Nov. 1980	Promoted to Shift Foreman.
Nov. 1981	Promoted to General Operating Foreman.
Jan. 1985	Appointed Senior Operations Supervisor.

NUCLEAR EXPERIENCE

Humboldt Bay - Advanced through all operating classifications at the plant to the position of Assistant Control Operator. Participated in startups, shutdowns, scram recoveries, power operation, refuelings and special tests. Received an AEC Reactor Operator's License in July 1973 (OP-3346).

Diablo Canyon - Participated in the initial training programs and startup testing of the plant.

EDUCATION

High School
College - 3 years

FORMAL TRAINING COURSES

Introduction to Nuclear Power, Humboldt Bay Power Plant, 1972.

Radiation Protection Training Course, Humboldt Bay Power Plant, 1972.

Humboldt Bay Equipment Description and Operations Course, Humboldt Bay Power Plant, 1972.

Diablo Canyon License Preparation - Consisted of reactor theory, radiation protection, equipment description and operation. Diablo Canyon Power Plant, 1974-1978.

Simulator Training - Westinghouse Nuclear Training Center, Zion, Illinois. Option II (one-week course), 1978.

Received NRC Senior Reactor Operators License for (SOP-3964) Diablo Canyon Power Plant on June 5, 1983.

DIABLO CANYON POWER PLANT
Operations Department

FROM: Ronald L. Ewing

SUBJECT: Resume

DATE: September 9, 1987

Gentlemen:

1. PERSONAL DATA

Full name: Ronald Lowell Ewing

Citizenship: United States of America

Age: 45 (born November 3,1941)

Present Employer: Pacific Gas and Electric Company
Diablo Canyon Power Plant
P.O. Box 56
Avila Beach, California 93424

Position Title: Shift Foreman
Diablo Canyon Power Plant

2. EDUCATION

High School

3. EMPLOYMENT HISTORY - Joined PGandE in November, 1966

a. October, 1961 to July, 1964 - U.S. Army.

b. November, 1966 to June, 1973 - Employed by PGandE at Humboldt Bay Power Plant. Promoted to Control Operator (Reactor Operator) in August, 1971.

c. June, 1973 - Promoted to Senior Control Operator and assigned to Diablo Canyon.

d. December, 1976 - promoted to Shift Foreman at Diablo Canyon.

4. NUCLEAR EXPERIENCE

a. Humboldt Bay - Advanced through all operating classifications at the plant to the position of Control Operator. Participated in startups, shutdowns, scram recoveries, power operation, refueling operation, and special tests.

b. Diablo Canyon - Participated in the initial training programs and startup testing of the plant.

5. PREVIOUS OPERATOR'S LICENSE

OP - Humboldt Bay Power Plant

6. OTHER INFORMATION

Mr. Ewing has completed the training for Diablo Canyon Power Plant Unit 1 and Unit 2 which is described below. He has acted as instructor for portions of this program.

Nuclear Power/Reactor Theory

The course, which was prepared and presented by the Plant Staff, consists of a review of mathematics used in nuclear calculations, atomic and nuclear physics, nuclear reactor theory, reactor safeguards principles, radioactive waste disposal, pressurized water reactor chemistry, physics, control of power distribution, and nuclear instrumentation theory. Approximately 232 hours.

Radiation Protection Training Course

This course, which was prepared and presented by the Plant Staff, consists of a review of atomic and nuclear physics, radiation protection theory, and the Radiation Control Standards and Procedures. Training in the use and limitations of radiation monitoring and survey instruments is also included. Approximately 109 hours.

Equipment Description and Operation Course

This course, which was prepared and presented by the Plant Staff, covers the description, operation, and operating procedures of each plant system and component. It includes a technical discussion of some general topics that have application in numerous systems. Approximately 953 hours.

Miscellaneous

This covers Heat Transfer and Fluid Mechanics, which was taught by an outside consultant. Other topics, prepared and presented by the Plant Staff, includes Mitigation of Core Damage, TMI review, and plant design changes. Approximately 155 hours.

Simulator Training

This course was presented by Westinghouse at the Nuclear Training Center, Zion, Illinois. A total of 27 days has been accumulated in a format designed to maximize the amount of "hands on" simulator time.

RONALD L. EWING

Resume of
CRAIG G. SMITH
1715 Diablo Drive
San Luis Obispo, Ca.
93401
phone (805) 541-0920

Present Status
Asst. Control Operator Diablo Canyon Power Plant

MILITARY EXPERIENCE

7/73 - 1/77	Reactor Plant Operator and Technician aboard U. S. Navy nuclear powered submarine.

EDUCATION

Military

1/73 - 6/73	Six months training at U. S. Naval Pressurized Water Reactor Prototype Plant, Schnectady, New York.
6/72 - 12/72	Six months academic training at U. S. Naval Nuclear Power Training School, Bainbridge, Maryland.
5/71 - 12/71	U. S. Naval Electronics Technician (Radar) Class A School, Treasure Island, San Francisco, California.
4/71 - 5/71	U. S. Naval Basic Electronics and Electricity course, San Diego, California.

Civilian

College:	Cuesta Community College, San Luis Obispo, California. Associate of Arts Degree, Dec. 1980 (Liberal Arts)
High School:	Sunnyside Senior High School, Sunnyside, Washington. Graduated in 1969.

CIVILIAN EMPLOYMENT (After military service)

2/77 - Present	Diablo Canyon Nuclear Power Plant Operations Dept. NRC Licensed Reactor Operator Present Position: Relief Asst. Control Operator

PERSONAL

Born: 8/14/51 in Sunnyside, Washington
Statistics: Height 5'10"; Weight 160 lbs.
Marital Status: Married
Military Awards: Good Conduct Medal; Qualified on Submarines
Military Security Clearance: Secret

LEE R. WATERS
421 Campana Pl.
Arroyo Grande, CA 83420

EDUCATION

Florida State University, B. S., Aeronautical Engineering, 1970

PROFESSIONAL ACHIEVEMENTS

Senior Operator License, Diablo Canyon Nuclear Power Plant, 1983

Nuclear Power Engineer, Westinghouse Electric Corporation, Naval Reactors Facility, 1980

PROFESSIONAL EXPERIENCE

Operations Shift Foreman, PGandE, March 1985 - Present

Supervise plant operations. Direct the activities of fuel loading, start up, low power and full power testing and commercial operations. Maximize plant operating efficiency and availability. Ensure compliance with technical specifications and environmental specifications. Assume the responsibilities of Interim Site Emergency Coordinator.

Shift Technical Advisor, PGandE, November 1982 - March 1985

Administer the surveillance testing program. Coordinate plant maintenance during routine operations, curtailments and outages. Initiate and review identified plant problems and disseminate this information on a computer based management information system. Technically advise the operating crew and the Operations Manager. Assume the responsibilities of Emergency Evaluation and Recovery Coordinator.

Nuclear Power Engineer, Westinghouse, June 1980 - November 1982

Direct the plant programs involving reactor testing and maintenance of Naval Reactors propulsion plants. Coordinate plant modifications, maintenance and overhaul. The demanding nature of this position required strength in areas such as reactor physics, heat transfer and fluid flow, systems operations and instrumentation controls.

ROGER L. JETT
Simulator Supervisor

EMPLOYMENT HISTORY

Feb. 1967 - May 1974	U.S. NAVY In the Navy nuclear power program. Final Rank E-6, qualified as an engineering watch supervisor.
May 1974 - May 1979	WESTINGHOUSE Training engineer at W.N.T.C. Zion, Illinois and in Pittsburgh, Pennsylvania. Promoted to senior engineer Grade B prior to resignation.
May 1979 - Nov. 1981	SELF-EMPLOYED Training consultant.
Dec. 1981 - Present	PACIFIC GAS AND ELECTRIC COMPANY Simulator Supervisor.

NUCLEAR EXPERIENCE

Seven years navy nuclear program, qualified at 53G prototype, qualified aboard USS Long Beach (CGN-9), qualified and instructed at AIW prototype, Idaho Falls, Idaho.

Instructor at the Westinghouse Nuclear Training Center, Zion, Illinois from 1974 to 1977. SRO licensed for Zion Unit 1 and 2 in May 1976.

Instructed at various plant sites around the world from 1977 to 1979 and involved with development of Ringhals Simulator for Sweden and Snupps Simulator in Zion, Illinois.

As a consultant, involved in instructing simulator training classes in Zion, Illinois for Westinghouse in July 1979 to December 1979. Involved with teaching license prep courses at Diablo Canyon Power Plant and providing simulator instruction for license candidates, January 1980 to November 1981.

EDUCATION

Glenville State College - 84 semester hours
Naval Nuclear Power School

FORMAL TRAINING COURSES

Electricians 'A' School USN, 1967
Nuclear Power School USN, 1968
Westinghouse Phase II and III Programs, 1974
Westinghouse License Training Program, 1976

RESUME' - J. D. LODGE

I. Personal History

Name: Jerry D. Lodge
Address: 599 Stanford Drive, San Luis Obispo, Ca. 93401
Telephone: Home (805)541-3157
Office (805)541-7145

Education: BS - Engineering Science (1959) U of Portland
MS - Nuclear Engineering (1966) U of Washington

Position: Power Production Engineer (Computer), Diablo Canyon Nuclear Power Plant
Pacific Gas and Electric Company

II. Summary of Qualifications

Nuclear Power Plant Operator Training Simulator experience in the areas of specifications, model development, computer programming, systems analysis and acceptance testing on the Fast Flux Test Facility Training Simulator, the Washington Public Power Supply System training simulator projects (1 and 3) and the Diablo Canyon Operator Training Simulator Project.

I&C test and startup work at the Fast Flux Test Facility on various plant systems including Flux Monitoring, Reactor Coolant and Plant Monitoring Systems.

Additional experience in Core Physics testing, Reactor Safeguards Analysis, Process Instrumentation and Reactor Operations with twenty plus years in the Nuclear Industry.

III. Professional Experience

Pacific Gas and Electric Company - Power Production Engineer (Computer) July 1982 to Present

Responsibilities include review of vendor Design Bases Documentation for plant system simulation models, providing the simulator data base and following all aspects of model integration and acceptance testing. Responsible for System Software/Hardware maintenance of the simulator upon delivery. (Westinghouse PWR Plant and Simulator/Gould-SEL Computer System)

Washington Public Power Supply System - Senior Engineer March 1980 to July 1982

Lead Engineer on the WNP-3 Operator Training Simulator Project (PWR-Combustion Engineering). Responsibilities include writing the bid specifications, supplying plant data, reviewing the vendors design concepts, acceptance testing of final systems and hardware/software maintenance of the simulator when delivered

III. Professional Experience (cont)

(1983). Also involved in software maintenance on the WNP-1 Operator Training Simulator (PWR-B&W) now in operation at the Supply System. Both systems use SEL computer equipment, 32/55 on WNP-1 and 32/77 on WNP-3.

Westinghouse Hanford Co. - Senior Engineer/Advanced Engineer
July 1970 to March 1980

Lead Engineer on the Fast Flux Test Facility (FFTF) Plant Monitoring System. Responsibilities included acceptance testing, system expansion and system maintenance (hardware/ software) of MODCOMP, DEC and HP computer systems.

Startup I&C work on Plant Control Systems, Flux Monitoring and Plant Monitoring Systems. Responsibilities included preparation of test procedures, craft scheduling and supervision, test coordination, and data report documentation.

Additional computer programming and simulation experience on the FFTF operator training simulator and various engineering simulation models. Also completed the Academic Operator Training program given to Operations Engineer at the FFTF.

Battelle Northwest - Development Engineer
October 1966 to July 1970 - Process Simulation & Analysis

Responsible Engineer for simulation and systems analysis studies on the Hanford N-Reactor (Dual Purpose PWR). Work included transient studies of the simulated overall N-Reactor plant and associated sub-system models.

Battelle Northwest & General Electric Co. - Engineer July 1959 to October 1966

Safeguards & Quality Assurance analysis of existing and proposed nuclear facilities on the Pacific Northwest Labs at Hanford. Nuclear physics experimentation for reactor dynamics studies. Development and application of instrumentation for measurement and control of variables in process plants and bench studies.

APPENDIX B

TASK ANALYSIS SCENARIO DESCRIPTIONS

1710S

PACIFIC GAS AND ELECTRIC COMPANY
DIABLO CANYON POWER PLANT

CONTROL ROOM DESIGN REVIEW

SCENARIOS

April 25, 1986

Prepared by:

General Physics Corporation
10650 Hickory Ridge Road
Columbia, MD 21044

Scenario 1   ATWS/Loss of Reactor Coolant

Procedures Used:	EP E-0	Reactor Trip or Safety Injection
	EP FR-S.1	Response to Nuclear Power Generation/ATWS
	EP E-0.1	Reactor Trip Response
	EP E-1	Loss of Reactor or Secondary Coolant
	EP E-1.2	Post LOCA Cooldown and Depressurization

Initial Conditions: End of life (EOL), Hot Full Power (HFP), Equilibrium Xenon

Scenario Sequence:

1. Initialize at HFP
2. Implement malfunctions to fail all automatic and manual reactor trips
3. Inform crew of plant conditions
4. Allow sufficient familiarization time
5. Implement turbine trip
6. Allow sufficient stabilization time after reactor is tripped. This will include recovery of 480V buses 13D and 13E (Unit 1) or 23D and 23E (Unit 2).
7. Implement malfunction small break LOCA (SBLOCA)

Expected Response: A reactor trip signal is generated from a spurious turbine trip. The reactor trip breakers do not open, resulting in an ATWS condition.

The operating crew attempts to trip the reactor via manual pushbuttons, which do not function properly. A transition is made to EP FR-S.1.

Once the crew determines safety injection is not required, a transition is made to EP E.0.1. EP E-0.1 is followed.

During reactor trip recovery, a SBLOCA develops. Safety injection occurs due to either pressurizer low pressure or manual actuation by the crew. This causes a return to Step 1 of EP E-0, and the automatic actions of SI are verified. As plant symptoms are diagnosed, a transition is made to EP E-1.

EP E-1 is followed. In Step 7, the criteria for SI termination are not met. At Step 13b the need for further cooldown and depressurization is established, resulting in a transition to procedure EP E-1.2.

The scenario should be continued until ECCS flow has been substantially decreased or is no longer required.

<u>Scenario Termination Criteria:</u> Discretion of CRDR Coordinator

Scenario 2 Large Break LOCA

Procedures Used:	EP E-0	Reactor Trip or Safety Injection
	EP E-1	Loss of Reactor or Secondary Coolant
	EP E-1.3	Transfer to Cold Leg Recirculation
	EP ECA-1.1	Loss of Emergency Coolant Recirculation
	EP FR-C.1	Response to Inadequate Core Cooling
	EP FR-C.1	Response to Degraded Core Cooling
	EP FR-Z.3	Response to High Containment Radiation Level
	EP E-1.4	Transfer to Hot Leg Recirculation

Initial Conditions: End of Life (EOL), Hot Full Power (HFP), Equilibrium Xenon

Scenario Sequence:

1. Initialize at HFP
2. Implement malfunction to fail RHR pumps
3. Inform crew of plant conditions
4. Allow sufficient familiarization time
5. Implement malfunction for large break LOCA (LBLOCA)
6. After SGs are depressurized to 160 psig, return an RHR pump to operable status

Expected Response: During normal, full power operations at EOL, a catastrophic rupture of an RCS hot leg occurs.

EP E-0 is immediately entered to verify automatic actuation of Safety Injection. In Step 22 of EP E-0 a transition to EP E-1 is made on abnormal containment conditions. The operating crew continues with EP E-1 until Step 14d is reached. At this point,

a transition to procedure EP E-1.3 is dependent on RWST level. When RWST level decreases to less than 33%, the transition is made.

During this time, an opportunity exists for the operating crew to implement EP FR-Z.3, "Response to High Containment Radiation Level."

During the SI transfer to cold-leg recirculation, recirculation capability is lost when the SI pumps trip. This situation causes a transition to EP ECA-1.1, "Loss of Emergency Coolant Recirculation." The RHR pumps cannot be manually started. During performance of EP ECA-1.1, core exit temperatures increase to a level sufficient to implement EP FR-C.2 and possibly EP FR-C.1. After SGs are depressurized to 160 psig, recirculation capability is reestablished with a start of an RHR pump. The transfer to cold-leg recirculation continues to completion.

After the plant is stable on cold-leg recirculation, EP E-1.4 is implemented at the direction of CRDR coordinator to demonstrate transfer to hot-leg recirculation.

<u>Scenario Termination Criteria</u>	Discretion of CRDR Coordinator

Scenario 3 SGTR W/Cooldown Using Backfill

Procedures Used:

EP E-0	Reactor Trip or Safety Injection
EP E-3	Steam Generator Tube Rupture

Initial Conditions: End of Life (EOL), Hot Full Power (HFP), Equilibrium Xenon

Scenario Sequence:

1. Initialize at HFP
2. Inform crew of plant conditions
3. Allow sufficient familiarization time
4. Implement malfunction for steam generator tube rupture

Expected Response:

A steam generator tube rupture occurs during normal, full power operation. An automatic SI occurs as a result of pressurizer pressure decrease, which causes the operating crew to implement EP E-0. In Step 21 of EP E-0, abnormal radiation indication from the steam jet air ejector causes a transition to EP E-3.

Once in EP E-3, the ruptured SG is identified and isolated. This action is followed by an RCS cooldown and depressurization to recover pressurizer level.

At Step 20 of EP E-3, SI termination criteria are met. Charging and letdown flows are used to control pressurizer level, and at Step 44 of EP E-3 the RCS is depressurized to allow backfill from ruptured SG for

ruptured SG cooldown. On low level the SG is refilled with auxiliary feedwater. This process is repeated until RCS tempratures are less than 200°F.

Scenario Termination Criteria: Discretion of CRDR Coordinator

Scenario 4 Secondary Break Inside Containment with Loss of Spray Capability

Procedures Used:

EP E-0	Reactor Trip or Safety Injection
EP E-1	Loss of Reactor or Secondary Coolant
EP E-2	Faulted Steam Generator Isolation
EP FR-P.2	Response to Anticipated Pressurized Thermal Shock Condition
EP FR-Z.1	Response to High Containment Pressure
EP E-1.1	SI Termination

Initial Conditions: End of Life (EOL), Hot Zero Power (HZP), Critical

Scenario Sequence:

1. Initialize at HZP
2. Implement malfunction to fail both containment spray pumps
3. Inform crew of plant conditions
4. Allow sufficient familiarization time
5. Implement malfunction for steam line break inside containment
6. Return containment spray pumps to operable status at the direction of the CRDR Coordinator

Expected Response:

At HZP and EOL a steam line ruptures inside reactor containment. A safety injection signal is generated from high containment pressure, which causes the operators to implement EP E-0.

Containment pressure continues to increase and exceeds the containment spray setpoint, but the containment spray pumps do not start. This forces a transition to EP FR-Z.1, "Response To High Containment Pressure." In EP FR-Z.1, all containment fan cooler units are

verified operable, and an investigation into the spray pump auto start failure is begun. After a brief time, the SRO receives a report that both spray pump motor breakers were open with the closing springs and charging motors de-energized. After re-energizing the closing springs the pump motors are manually restarted and the operating crew returns to EP E-0.

At Step 20 of EP E-0, a transition is made to EP E-2 to identify and isolate the faulted SG. After the faulted SG is isolated, the operating crew is directed to transfer to EP E-1.

During the faulted steam generator blowdown to containment, excessive cooldown rates will provide an opportunity for transfer to EP FR-P.2, "Response to Anticipated Pressurized Thermal Shock Condition."

In EP E-1, containment spray is terminated and safety injection termination criteria are met. At Step 8 of EP E-1, the operating crew is directed to EP E-1.1 for. SI termination criteria.

In EP E-1.1, safety injection is terminated, charging and letdown flows are established, and plant equipment is re-aligned for shutdown conditions.

<u>Scenario Termination Criteria</u>

Discretion of CRDR Coordinator

Scenario 5 Loss of Secondary Heat Sink

Procedures Used:

EP E-0	Reactor trip or Safety Injection
EP E-0.1	Reactor Trip Response
EP FR-H.1	Response to Loss of Secondary Heat Sink
EP E-1.1	SI Termination

Initial Conditions: End of Life (EOL), Hot Full Power (HFP), Equilibrium Xenon

Scenario Sequence:

1. Initialize to HFP
2. Implement malfunction to prevent all auxiliary feedwater pumps from starting
3. Inform crew of plant conditions
4. Allow sufficient familiarization time
5. Implement malfunction for reactor trip
6. When crew transitions to EP FR-H.1, implement malfunction for unit blackout
7. Return auxiliary feedwater pumps to operable status at the direction of the CRDR Coordinator

Expected Response: A reactor trip occurs at full power conditions. During the trip recovery, the reactor operator observes that no auxiliary feedwater pumps have started. This condition causes a transition to EP FR-H.1, "Response to Loss of Secondary Heat Sink." At the time the transition is made, the Station Auxiliary Transformer fails, blacking out the unit. All emergency diesels start as designed.

EP FR-H.1 directs the operating crew to attempt feedwater flow via condensate pumps, which are unavailable. The operating crew continues with EP FR-H.1 and establishes feed-and-bleed using safety injection. After feed-and-bleed has been established, auxiliary feedwater is reestablished. Feed-and-bleed is terminated using EP FR-H.1 and, after conditions are stabilized, the operating crew is directed to transfer to EP E-1.1 Step 11, "SI Termination."

Scenario Termination Criteria

At discretion of CRDR Coordinator after EP E-1.1 has been entered.

APPENDIX C

RHR PUMP CONTROL SWITCHES
RELOCATION AT BOARD VB1 MIMIC

DESIGN CHANGE PACKAGE (DCP) J-37348

PRIORITY ______ Required Condition, Mode, or Operability Status to Implement: Plant DCPP UNIT #1 System RHR Component CONTROL SWITCHES DCPP Reviewer Date	UNIT #1 No. J-37348-R 0 DESIGN CHANGE COVER SHEET SHEET 1 OF 2. CORRESPONDING DCNS FOR OTHER UNITS ARE: UNIT #2- DC2-EJ-38348 SIMULATOR- DCO-EJ-37349

DESCRIPTION OF CHANGE: REVISE THE ORDER OF THE SWITCHES FOR THE RHR PUMPS 1-1 AND 1-2 SUCH THAT THE SWITCHES FOR PUMPS 1-1 AND 1-2 ARE ARRANGED FROM LEFT TO RIGHT ON THE MIMIC. THUS THE LAYOUT OF THE SWITCHES AND AMMETERS WOULD BE IN THE CORRECT ORDER ON IVBI. THE DCP WORK TO BE CO-ORDINATED WITH PLANT OPERATIONS.

REASON FOR CHANGE: THE CONTROL SWITCHES FOR THE RHR PUMPS HAVE BEEN LAID OUT IN THE REVERSE ORDER FROM THE AMMETERS CAUSING CONFUSION TO OPERATORS.

THIS CHANGE ADDRESSES THE CONTROL ROOM DESIGN REVIEW (BL1060) WHICH IS A PG&E COMMITMENT TO NRC AS A CONDITION FOR MAINTAINING DCPP'S OPERATING LICENSE.

RELATIONSHIP TO OPEN DCNs:

[ ] Not Applicable

EFFECT ON PLANT OPERATION:

[X] Not Applicable REARRANGEMENT WILL ELIMINATE POSSIBLE OPERATOR CONFUSION BECAUSE THE AMMETERS AND CONTROL SWITCHES HAVE BEEN LAID OUT IN THE REVERSE ORDER ON VERTICAL BOARD IVBI. THIS CHANGE WILL ENHANCE CONTROL ROOM OPERATIONS.

RESPONSIBLE DISCIPLINE I & C

SIMULATOR

PRIORITY ______	
Required Condition, Mode, or Operability Status to Implement:	
Plant DCPP SIMULATOR	
System RHR	
Component CONTROL SWITCHES	
DCPP Reviewer	Date

No. J-37348-R 0

DESIGN CHANGE COVER SHEET SHEET 2 OF 2.

CORRESPONDING DCNS FOR OTHER UNITS ARE:
UNIT #1 DCN - DC1-EJ-37348
UNIT #2 DCN - DC2-EJ-38348

DESCRIPTION OF CHANGE: REVISE THE ORDER OF THE SWITCHES FOR THE RHR PUMPS 1-1 AND 1-2 SUCH THAT THE SWITCHES FOR PUMPS 1-1 AND 1-2 ARE ARRANGED FROM LEFT TO RIGHT ON THE MIMIC. THUS THE LAYOUT OF THE SWITCHES AND AMMETERS WOULD BE IN THE CORRECT ORDER ON VB1.

REASON FOR CHANGE: THE CONTROL SWITCHES FOR THE RHR PUMPS HAVE BEEN LAID OUT IN THE REVERSE ORDER FROM THE AMMETERS CAUSING CONFUSION TO OPERATORS.
THIS CHANGE ADDRESSES THE CONTROL ROOM DESIGN REVIEW (BLI 060) WHICH IS A PG&E COMMITMENT TO NRC AS A CONDITION FOR MAINTAINING DCPP'S OPERATING LICENSE.

RELATIONSHIP TO OPEN DCNs:

[ ] Not Applicable

EFFECT ON PLANT OPERATION:

[X] Not Applicable RE ARRANGEMENT WILL ELIMINATE POSSIBLE OPERATOR CONFUSION BECAUSE THE AMMETERS AND CONTROL SWITCHES HAVE BEEN LAID OUT IN THE REVERSE ORDER ON VERTICAL BOARD VB1.

RESPONSIBLE DISCIPLINE I&C

UNIT # 1 & SIMULATOR

DESIGN CHANGE PACKAGE FOLLOWER

DCP# J-37348 - R 0

ESTIMATE

COST & SCHEDULING/ENGINEERING

ESTIMATE

				By	Date
Engineering Start	8/25/87	Engineer		PKRangan	9/9/87
Construction Complete	END OF SRO	Approved (EGS)		[illegible]	9/17/87
Estimated Total Cost	$18,700.00	Accepted (PE)		KES[illegible]	9/22/87
Authorizing Job Estimate Number	BC1060 E; JE1944974				

Cost & Scheduling Spv. By [illegible] Date 9/21/87 [ ] Not Required (level of effort)

DESIGN

DESIGN ENGINEERING

	By	Date		By	Date
Engineer	PKRangan	9/9/87	Mechanical Discipline (Safety Evaluation)	[illegible]	9/21/87
Approved (GS)	[illegible] 9/13/87 [illegible]	9/17/87	Accepted (PE)	[illegible]	9/22/87

CONSTRUCTION

CONSTRUCTION PLANT STAFF REVIEW

	By	Date		By	Date
Received by EM			Imp to Safety or		
PSRC Approves (y/n) ___			Imp to Environ (y/n) ___		

Reason for Rejection ______________________

Plant Manager Approval ______________________ Date __________

Transmitted By ______________________ Date __________

CONSTRUCTION

	By	Date		By	Date
Received			Installation Complete		
Start Up Complete			As-Builts Attd (y/n) ___		

Released by Package Coordinator ______________________ Date __________

FC No. ______________ FCTs ______________________

ACCEPTANCE

ACCEPTANCE PLANT STAFF

	By	Date		By	Date
Received work complete			Staff review complete		

Plant Manager Final Approval ______________________ Date __________

Transmitted By ______________________ Date __________

ENGINEERING CLOSE OUT

All design documents issued for Operation: ______________ ______

Project Engineer Date

RMS Purged __________ __________ RMS Indexed __________ __________

By Date By Date

TABLE OF CONTENTS

DCP# J - 37348 - R 0

Page 1 of 1

DCN NUMBER	CURRENT REVISION	ATTACHED	LATER*	PREVIOUSLY ISSUED
DCI-EJ-37348	Ø	[X]	[ ]	[ ]
DCI-EE-37348	Ø	[X]	[ ]	[ ]
DCO-EJ-37349	Ø	[X]	[ ]	[ ]
		[ ]	[ ]	[ ]
		[ ]	[ ]	[ ]
		[ ]	[ ]	[ ]

SEE NEXT SHEET FOR ADDITIONAL DCNs [ ] Yes [X] No

* DCNs identified to be provided under approved package separation

FC NUMBER /FCT NUMBER	DCN NUMBER

SEE NEXT SHEET FOR ADDITIONAL FCs/FCTs [ ] Yes [X] No

PACIFIC GAS AND ELECTRIC COMPANY
DIABLO CANYON POWER PLANT UNIT NOS. 1 AND 2

DESIGN CHANGE SAFETY EVALUATION SUMMARY

SUBJECT: RELOCATE THE CONTROL SWITCHES FOR RHR PUMPS 1-1 AND 1-2 ON CONTROL BOARD VB1 MIMIC

1. DCP Number: J-37348, REV. Ø

2. Classification

	Yes	No
A. Does this change require a change to the Technical Specifications?	( )	(X)

If the above question is answered "Yes," a Licensing Amendment Report is required.

	Yes	No
B. Does this change require a change in the following documents:		
1. The SAR? (see definition, Section 2.2.12)	(X)	( )
2. Any Q or Class 1 items in the Q-List?	( )	(X)
3. The Environmental Qualification Report?	( )	(X)
C. Does this change affect:		
1. Security?	( )	(X)
2. Fire Protection?	( )	(X)
3. Emergency Planning?	( )	(X)
D. Is any of the affected equipment important to safety?	(X)	( )
E. Is radioactive material contained in the system?	( )	(X)
F. Is there a radioactive waste treatment change specified?	( )	(X)
G. Based on the Design and Safety Review, does a potential unreviewed safety question exist?	( )	(X)

If any of the previous questions have been answered "Yes," complete Questions 1 through 3 of the attached Safety Evaluation. If all questions have been answered "No," attach a justification detailing why no Safety Evaluation is required and answer Question J "No."

	Yes	No
H. Is any of the affected equipment important to environmental quality?	( )	(X)

DCP J-37348, REV. 0

		Yes	No
I.	Does the proposed change have the potential to impact the environment?	( )	(X)

If either Questions H or I have been answered "Yes'" complete the attached Environmental Evaluation. If Questions H and I have been answered "No," answer Question K "No."

		Yes	No
J.	Has this change been determined to constitute an unreviewed safety question?	( )	(X)

(Yes, if either Question 1, 2, or 3 of the attached Safety Evaluation is marked "Yes.")

K.	Has this change been determined to constitute an unreviewed environmental question?	( )	(X)

(Yes, if Question B on the attached Environmental Evaluation is marked "Yes.")

3. References:

SAR 7.7 FIG 7.7-20 TECH. SPECS SECT. 3.9.8.1,

Q-LIST PAGE IV-5 T.S SECT 3.9.8.2, 5.3 & 5.4

Performed by: George J. Wisbro P.K. RANGAN PKRangan Date: 9/9/1987

Reviewed by: Bill Ellis Date: 9/21/87

PSRC Review: ______ Date: ______

DCP J-37348, REV. 0

DESIGN CHANGE PACKAGE SAFETY EVALUATION

Prior to answering the following three questions from 10CFR50.59, present a description of the design change including the critical parameters, and how the functional requirements of the system, structure or component are satisfied:

UNIT 1 & SIMULATOR ON CONTROL BOARD VB1, THE CONTROL SWITCHES FOR RHR PUMPS 1-1 AND 1-2 READ FROM RIGHT TO LEFT. REVISE THE CONTROL SWITCHES TO READ FROM LEFT TO RIGHT ON THE MIMIC SO AS TO BE IN AGREEMENT WITH THE LAYOUT OF THE AMMETERS FOR THE RHR PUMPS. THE DCP WORK TO BE CO-ORDINATED WITH PLANT OPERATIONS. RHR OPERABILITY SHALL BE VERIFIED. REFER TO SECTION 6 OF THE TECH. SPECS.

1. Is the possibility of an accident or malfunction of a different type than any evaluated previously in the SAR created?

Yes	No
( )	(X)

THE RELOCATION OF THE SWITCHES FOR RHR PUMPS 1-1 AND 1-2 ON THE MIMIC ON VB1 WILL RESULT IN BETTER FUNCTIONAL GROUPING. THE RELOCATION OF THE UNIT 1 SWITCHES WILL NOT AFFECT SAFETY RELATED WIRING SEPARATION AND THE INTEGRITY OF CHANNELS IS MAINTAINED. THE MOUNTING OF THESE SWITCHES WILL ENSURE THAT THE SAFETY INTEGRITY OF THE UNIT 1 VERTICAL BOARD IS NOT AFFECTED. SEISMIC CALCULATION No. IS 46.2 DEMONSTRATES THAT THERE IS NO SEISMIC IMPACT CREATED BY THESE DEVICES ON 1VB1. THUS, THE POSSIBILITY OF ANY ACCIDENT BEYOND THAT ANALYSED IN THE SAR IS NOT ANTICIPATED. THIS CHANGE AFFECTS SAR DWG. #7.7-20 ICE LETTER TO LICENSING ICE-3890.
No new materials are introduced by this change.

DCP J-37348, REV. 0

DESIGN CHANGE PACKAGE SAFETY EVALUATION

2. Is the margin of safety as defined in the basis for any Technical Specification reduced?

	Yes	No
	( )	(X)

THE SWAPPING OF THE RHR SWITCHES IN THE CONTROL BOARD TO MATCH CORRESPONDING PUMP IS A FUNCTIONAL UPGRADE OF CONTROL ROOM OPERABILITY. OPERATORS WILL NOT BE CONFUSED WITH RHR PUMP CONTROL. THE TECH SPEC REQUIREMENTS* HAVE BEEN MET WITH THIS UPGRADE.

*TECH. SPEC. SECTIONS 3/4.9.8.1; 3/4.9.8.2, 3/4 5.2 & 3/4 5.3

3. Is the probability of occurrence or the consequences of an accident or malfunction of equipment important to safety previously evaluated in the SAR increased?

	Yes	No
	( )	(X)

THE RELOCATION OF THE SWITCHES FOR RHR PUMPS 1-1 AND 1-2 ON THE MIMIC ON VB1 WILL RESULT IN BETTER FUNCTIONAL GROUPING. THE RELOCATION OF THE UNIT 1 SWITCHES WILL NOT AFFECT SAFETY RELATED WIRING SEPARATION AND THE INTEGRITY OF CHANNELS IS MAINTAINED. THE MOUNTING OF THESE SWITCHES ~~SHOULD~~ WILL ENSURE THAT THE SAFETY INTEGRITY OF THE VERTICAL BOARD IS NOT AFFECTED. SEISMIC CALCULATION No. IS 46.2 DEMONSTRATES THAT THERE IS NO SEISMIC IMPACT ~~OF THESE DEVICES~~ BY THIS CHANGE ON 1VB1. THUS, THE POSSIBILITY OF ANY ACCIDENT PREVIOUSLY ANALYZED IN THE SAR IS NOT INCREASED. THIS CHANGE AFFECTS SAR DWG# 7.7-20 (I&C LETTER TO LICENSING ICE-3896).

PACIFIC GAS AND ELECTRIC COMPANY
DEPARTMENT OF NUCLEAR PLANT OPERATIONS
DIABLO CANYON POWER PLANT UNIT NOS. 1 AND 2

ATTACHMENT 1

PLANT MODIFICATION SYSTEMS INTERACTION EVALUATION

DCN NUMBER: DCI-EJ-37348 REV. Ø

TITLE/SUBJECT: RHR PUMPS 1-1 AND 1-2 CONTROL SWITCHES

EVALUATION METHOD — YES / NO

1) Does DCN meet Systems Interaction Exclusion Criteria? If yes, review is complete. Describe rationale: [X] [ ]

INSTRUMENT CHANGES WITHIN MAIN CONTROL BOARD VB1

2) Does DCN meet Source Screening Criteria? If yes, review is complete. Describe rationale: [ ] [ ]

3) Does DCN meet Target Screening Criteria? If yes, review is complete. Describe rationale: [ ] [ ]

4) Was review of DCN conducted in the field? [ ] [ ]

Comments on field inspection: ____

5) Were any interactions observed and postulated as a result of the field inspection? [ ] [ ]

a) If yes, was interaction previously documented? [ ] [ ]

IDS NO: ____

Does interaction resolution satisfactorily resolve the concerns for this DCN? [ ] [ ]

b) If interaction was not previously documented, initiate Action Request and IDS.

AR Number: ____

6) Are revisions to the SISIP Target List required? [ ] [X]

Performed By: [signature] [signature] 9/13/87 Date: 9/9/87

DC0150 7III

PACIFIC GAS AND ELECTRIC COMPANY
DEPARTMENT OF NUCLEAR PLANT OPERATIONS
DIABLO CANYON POWER PLANT UNIT NOS. 1 AND 2

ATTACHMENT 1

PLANT MODIFICATION SYSTEMS INTERACTION EVALUATION

DCN NUMBER: DCO-EJ-37349 REV. Ø

TITLE/SUBJECT: RHR PUMPS 1-1 AND 1-2 CONTROL SWITCHES

EVALUATION METHOD | YES | NO

1) Does DCN meet Systems Interaction Exclusion Criteria? If yes, review is complete. Describe rationale: [X] [ ]

INSTRUMENT CHANGES WITHIN SIMULATOR CONTROL BOARD VB1. SIMULATOR IS A NON-SAFETY RELATED PANEL AND DOES REQUIRE SEISMIC ANALYSIS.

2) Does DCN meet Source Screening Criteria? If yes, review is complete. Describe rationale: [ ] [ ]

3) Does DCN meet Target Screening Criteria? If yes, review is complete. Describe rationale: [ ] [ ]

4) Was review of DCN conducted in the field? [ ] [ ]

Comments on field inspection:

5) Were any interactions observed and postulated as a result of the field inspection? [ ] [ ]

a) If yes, was interaction previously documented? [ ] [ ]

IDS NO:

Does interaction resolution satisfactorily resolve the concerns for this DCN? [ ] [ ]

b) If interaction was not previously documented, initiate Action Request and IDS.

AR Number:

6) Are revisions to the SISIP Target List required? [ ] [X]

Performed By: [signature] Date: 9/9/87

DC0150 7III

PACIFIC GAS AND ELECTRIC COMPANY
DEPARTMENT OF NUCLEAR PLANT OPERATIONS
DIABLO CANYON POWER PLANT UNIT NOS. 1 AND 2

ATTACHMENT 1

PLANT MODIFICATION SYSTEMS INTERACTION EVALUATION

DCN NUMBER: DC1-EE-37348 REV. 0

TITLE/SUBJECT: REVISE RHR CONTROL SWITCH LOCATIONS

EVALUATION METHOD	YES	NO
1) Does DCN meet Systems Interaction Exclusion Criteria? If yes, review is complete. Describe rationale: RELOCATION OF EXISTING DEVICES WITHIN EXISTING PANEL	[X]	[ ]
2) Does DCN meet Source Screening Criteria? If yes, review is complete. Describe rationale:	[ ]	[ ]
3) Does DCN meet Target Screening Criteria? If yes, review is complete. Describe rationale:	[ ]	[ ]
4) Was review of DCN conducted in the field? Comments on field inspection:	[ ]	[ ]
5) Were any interactions observed and postulated as a result of the field inspection?	[ ]	[ ]
a) If yes, was interaction previously documented? IDS NO:	[ ]	[ ]
Does interaction resolution satisfactorily resolve the concerns for this DCN?	[ ]	[ ]
b) If interaction was not previously documented, initiate Action Request and IDS. AR Number:		
6) Are revisions to the SISIP Target List required?	[ ]	[X]

Performed By: Robert Fong Date: 9/14/87

DC0150' 7III

PG&E

DCPP UNIT #1

PACIFIC GAS AND ELECTRIC CO.
ENGINEERING DEPARTMENT
BLI NO. 060
PRIORITY NO. ____

NUCLEAR POWER PLANT
DESIGN CHANGE

DCPP UNIT #1
(PLANT)

DATE 9/9/87
DCN NO. DCI-EJ-37348
REV. NO. 0
SHEET 1 OF 20

REQUEST

To: KLEMME HERMAN From: PAT. K. RANGAN

Structure or System: CONTROL ROOM VERTICAL BOARD VB1 Sys No. 10

Component: RHR PUMPS 1-1 AND 1-2 CONTROL SWITCHES.

Description of Change: ON VERTICAL BOARD VB1, RELOCATE THE CONTROL SWITCHES OF RHR PUMPS 1-1 AND 1-2. REVISE THE MIMIC TO CONFORM WITH THE CHANGE, SO AS TO PROVIDE A LOGICAL LEFT TO RIGHT SEQUENCE OF THE CONTROLS THAT MATCHES THE ASSOCIATED AMMETERS.

Reason for Change: THE CONTROL SWITCHES FOR THE RHR PUMPS 1-1 AND 1-2 HAVE BEEN LAID OUT IN THE REVERSE ORDER FROM THE AMMETERS CAUSING CONFUSION TO OPERATORS.

Schedule/Justification: BEFORE THE END OF UNIT #1 2ND RFO.

Construction Status (for revised DCNs): N/A ☐ Not Started ☐ Partially Complete ☐ Completed

List of Attachments Not on DDL: SK-DCI-EJ-37348-2

Requested by: P.K. RANGAN

REVIEW AND APPROVAL

Requested Change is:

- ☐ Approved per delegation of authority
- ☒ Approved
- ☐ Noted, document change not required
- ☐ Rejected (explain) ____
- ☐ Per telecon with ____ on ____ (If required by delegation of authority)
- ☐ As-built documents required
- ☐ Approved, document change only

Safety-Related Work: ☒ Yes ☐ No

Important to Environmental Quality: ☐ Yes ☒ No

Not Safety-Related, requires Quality Assurance: ☐ Yes ☒ No

DCN required to close an NCR ☐ Yes ☒ No

NCR No. ____

Reviewed By: PKRangan 9/9/1987
Discipline Engineer Date

Checked By: [signature] 9/12/87
Date

Frank Mori 9/17/87
Group Supervisor Date

COMPLETION

	By	Date		By	Date
Installation Complete			Received Engineering		
Start-up Complete			Accepted Engineering		
Accepted NPG			Revisions Approved		

FCs Issued: ____ DIST: MNN, JJ, PB, KH

TECHNICAL REVIEW

DCN No. DCI-EJ-37348

Rev. No. 0

1a. DESIGN DOCUMENT REVIEW. The following documents are relevant to this change and have been originated, reviewed, or require revision as indicated:

Document, No., Rev.	Originated (Yes/No)	Reviewed (Yes/No)	Requires Revision (Yes/No)
• Q-List*		YES	
• Design Criteria Memorandums	NO	YES	NO
• Calculations 15 46.2	NO	YES	YES; REVISION COMPLETED
• Design Verification Reports	NO	NO	NO
• Design Change Notices	NO	NO	NO

* Revisions to the Q-List are to be transmitted to the Mechanical Engineering EGS.

1b. DESIGN SAFETY REVIEW. The following is a list (not all inclusive) of design and safety issues to be considered. Indicate by "Yes" or "No" whether or not each issue affects or is affected by the change. Unless the reason for a "No" answer is obvious, further explanation is required. If "Yes," explain why or how the issue is relevant and how it is resolved. If "No," explain why the issue is not relevant.

	Relevant Issue? (Yes/No)	Comments (Use additional sheets as necessary)
• Accident Analysis (FSAR Chapters 6 and 15)	YES	THE MIMIC WILL BE REVISED TO PROVIDE A LOGICAL LEFT TO RIGHT SEQUENCE OF SWITCHES RHR PUMPS 1-1 & 1-2. DOES NOT AFFECT ACCIDENT ANALYSIS
• ALARA	NO	
• Shielding/Radiation Zones	NO	
• Environmental Quality	NO	
• Fire Protection	NO	
• Unacceptable Components	NO	
• Codes and Standards	NO	COMPLIES WITH HUMAN FACTORS ENGINEERING GUIDELINES.

DCI-EJ-37348
Rev. 0
Sh. 3 of 20

	Relevant Issue? (Yes/No)	Comments
• Simulator	YES	DCO-EJ-37349
• System Interaction	NO	NO IMPACT- SWAPPING IDENTICAL DEVICES
• Regulatory Guides	NO	R. G. 1.75 HAS NOT BEEN VIOLATED
• Environmental Qualifications	NO	
• General Design Criteria	NO	
• Seismic Qualification	YES	SEE SHEET #7
• Water Hammer	NO	
• Inservice Inspection	NO	
• Heavy Loads	NO	
• Flooding	NO	
• Radioactive Piping	NO	
• High-Medium Energy Line Break	NO	
• Control Room Design Review (Including habitability)	YES	SEE SHT #7 OF THIS DCA
• Multi-Unit Impact	NO	NO INTERACTION WITH UNIT #1
• Aluminum Inside Containment	NO	
• Security	NO	
• HPRDS	NO	
• Personnel Safety	NO	
• Masonry Block Walls	NO	
• Core Drilling Impact	NO	
• Redundancy/Separation Requirements	YES	DESIGN CHANGE IN ACCORDANCE WITH R.G. 1.75
• Penetration Sealing	NO	
• Paint Inside Containment	NO	
• Refueling Operations	NO	
• Material Compatibility	NO	NO NEW MATERIAL ADDED.

	Relevant Issue? (Yes/No)	Comments
o Vital Bus Loading	NO	NO ADDED LOAD TO VITAL BUS LOADING.
o Maintainability/Accessibility	NO	
o Floor or Wall Loading	NO	
o Missiles	NO	
o Operability	YES	SEE SHT #7 OF THIS DC
o Electrical Design Considerations ~~(see attached sheet for discussion)~~	YES	SEE DCI-EE-37348
o Hydraulic Design Criteria ~~(see attached sheet for discussion)~~	NO	
o Chemistry Effects ~~(see attached sheet for discussion)~~	NO	
o I&C Design Considerations (see attached sheet #8 for discussion)	YES	SEE SHT # 8 OF THIS DCA
~~o HVAC Design Considerations (see attached sheet for discussion)~~	NO	
o For nonsafety-related modifications, discuss why the design resulting from the modifications will not affect (a) any safety-related structures, systems or components, and (b) items identified in Paragraph 4.4.4(e) of Procedure 3.6 ON.	NO	
o Other	NONE	
o FMEA Evaluation	NO IMPACT	

2. LICENSING. The following NRC Licensing submittals are relevant to this change and have been reviewed as indicated below. Where a revision is required, Licensing has been notified.

Document	Reviewed	Requires Revision
a. FSAR # FIG. 7.7-20	YES	YES
b. Technical Specification	YES	NO
c. Other	NA	NA

\# Revisions to the FSAR require an FSAR Change Notice to be transmitted to NRA.

3. COORDINATION.

Design Package Lead Discipline I&C

Coordination Required: [ ] No [X] Yes Coordinated With:

Department	Engineer (Signature)	Date	DCN Required	
ELECTRICAL	Robert Fong Robert Fong	9/12/87	* Yes (X)	No ( )
SEISMIC MECHANICAL	WINNIE CHU Winnie Chu	9/17/87	Yes ( )	No (X)
			Yes ( )	No ( )
			Yes ( )	No ( )
			Yes ( )	No ( )
			Yes ( )	No ( )
			Yes ( )	No ( )

* DCI-EE-37348

PG&E

DCI-EJ-37348
Rev. O
Sh. 6 of 20.

PACIFIC GAS AND ELECTRIC COMPANY
ENGINEERING DEPARTMENT
DESIGN DOCUMENTS LIST
DOCUMENTS AFFECTED BY DESIGN CHANGE NOTICE (DCN)

PLANT DCPP UNIT # 1 DCN NUMBER DCI-EJ-37348 REVISION 0 DATE 9/9/87 ENGINEER P.K RANGAN

DOCUMENT NUMBER	SHEET NO.	REVISION CURRENT	REVISION INTERIM	REVISION AS BUILT	DOCUMENT TITLE	DATE COMPLETED	BY	DATE APPROVED	BY
504481	1	*	*		MAIN CONTROL ROOM VERTICAL BOARD (VB) ARRANGEMENT				
					* TO BE DETERMINED BY DESIGN DRAFTING				

DCI-EJ-37348
REV. Ø
SHT. 7 OF 20

DESIGN CHANGE SAFETY EVALUATION

1.0 Introduction

A safety review of this DCN based on Procedure 3.6 ON, Attachment "E", indicates that four (4) issues require further analysis to determine the degree of safety impact to the control room boards. The issues to be analyzed are listed below:

Issue A: Seismic Qualification of Control Boards

Issue B: Control Room Design Review

Issue C: Operability

Issue D: Instrumentation and Controls Design Considerations

2.0 Safety Evaluation of Issues

2.1 Issue A: Seismic Qualification of Control Boards

WESTINGHOUSE REPORT WCAP-10358 (AUGUST 1983) - SEISMIC QUALIFICATION OF THE DIABLO CANYON MAIN CONTROL BOARD & CENTRAL CONSOLE IN COMBINATION WITH THE RESULTS OF DEVICE TESTING (WCAP-10359, AUG. 1983) HAVE VERIFIED THE SEISMIC INTEGRITY OF THE CONTROL BOARDS AND CONSOLES. --

THE CONTROL BOARD VB1 HAS BEEN REANALYZED BECAUSE OF THIS CHANGE PER SEISMIC ANALYSIS # IS 46.2.

THE SEISMIC INTEGRITY OF THE CONTROL BOARD HAS NOT BEEN ALTERED. NO EFFECT ON THE SAFETY-RELATED DEVICES. THESE ARE SWAPPED IN PLACE.

2.2 Issue B: Control Room Design Review

The present design change is a CRDR Team recommendation to improve the usability of the control room boards. This design change will reorder potentially misleading indication and controls to agree with control room design convention and good human factors practices. The proposed changes will enhance control room operation by reducing operator response time and potential for error.

2.3 Issue C: Operability

Implementation of the proposed changes will not negatively impact plant operability. The devices to be reordered will remain on the same control panel. No additions or deletions in plant parameter information is proposed.

2.4 Issue D: Instrumentation & Controls Design Considerations

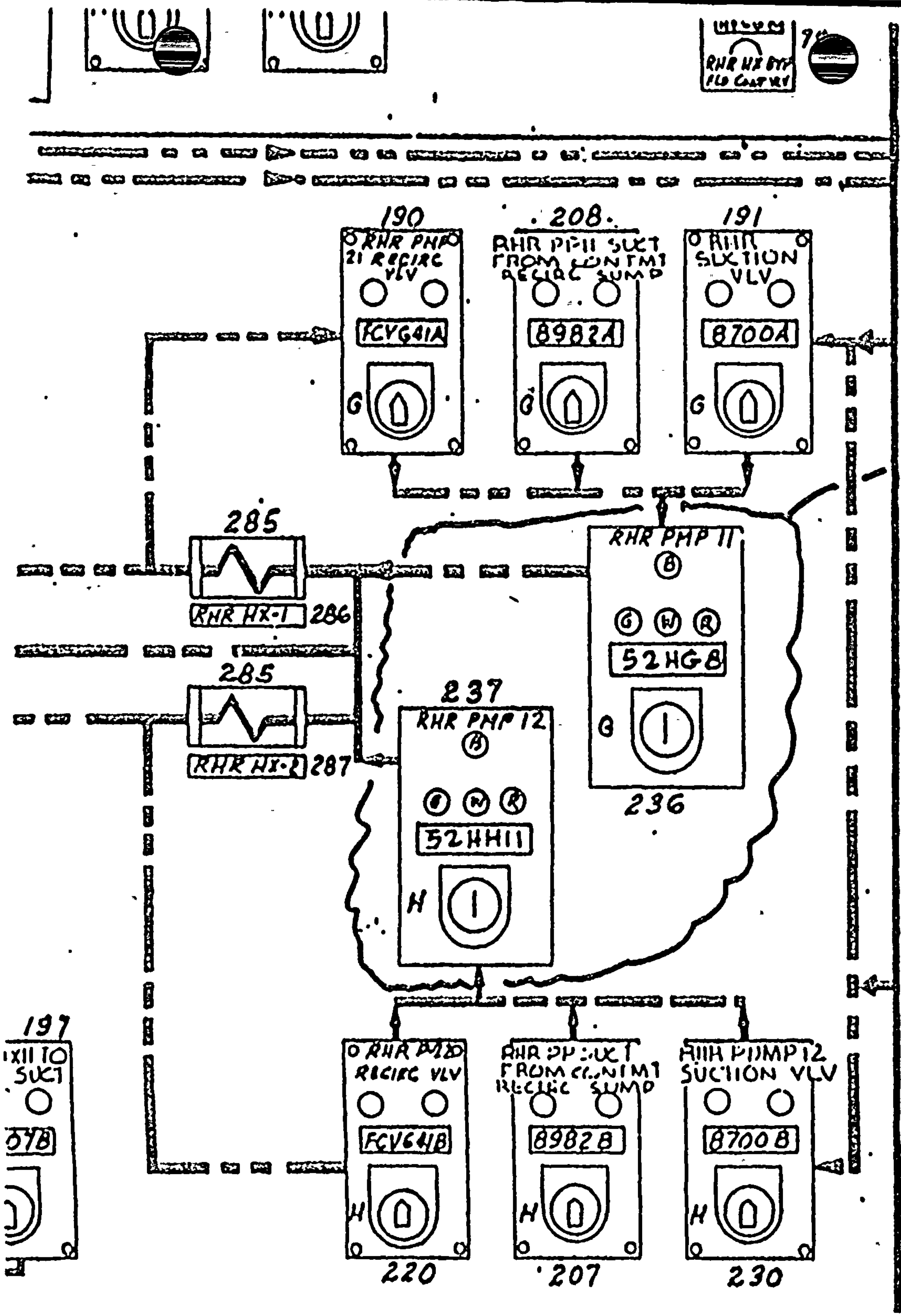
As the proposed changes do not include any change in instrument ranges, size, scales, inputs, selection devices or other features, all I & C design considerations can be considered to have been met. The CRDR Team evaluation has examined the human factors aspects of this design change and has found no guidelines or good practices to be violated.

Analysis by: K. RANGAN Date: 9/1/87

Signature: Jorge J. Lisboa 9/8/87

Approval by: F. MORI 9/17/87

Signature: Frank Mori



EXCERPT FROM DWG. #504481 (VBI ARRGMT.) REV. 9

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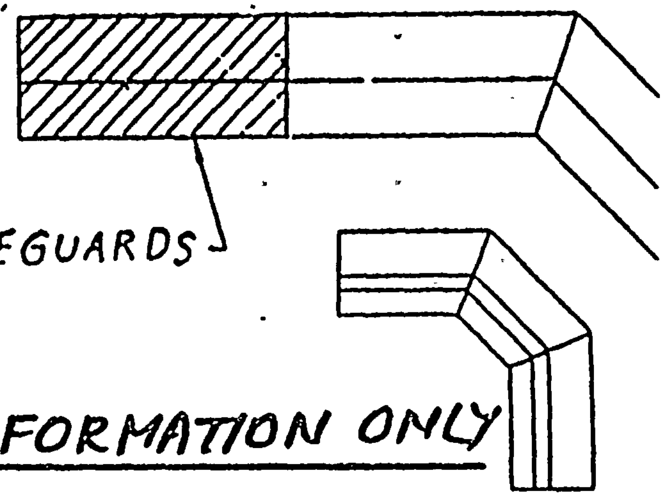
EXISTING LOCATION OF RHR PMP. 1 & 2 CONTROLS.
SEE SK-DCI-EJ-37348-1



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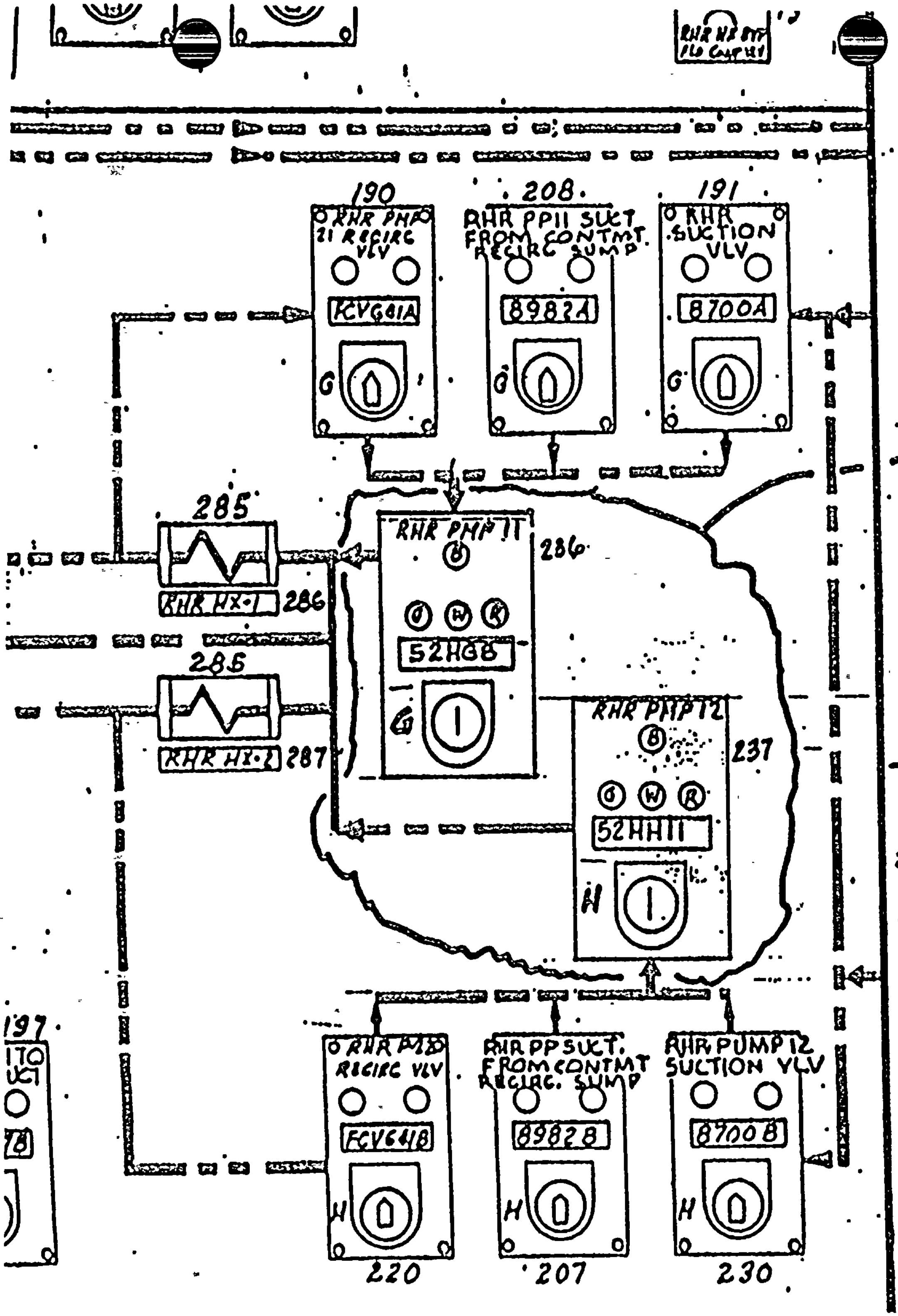
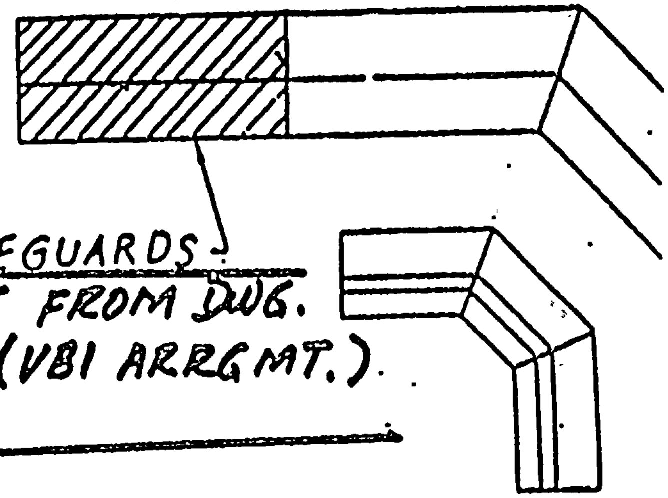


REVISED MIMIC - RHR PMP 1-1 & 1-2 CONTROLS RELOCATED.



SAFEGUARDS

EXCERPT FROM DWG. #504481 (VBI ARRGMT.) REV. 9.

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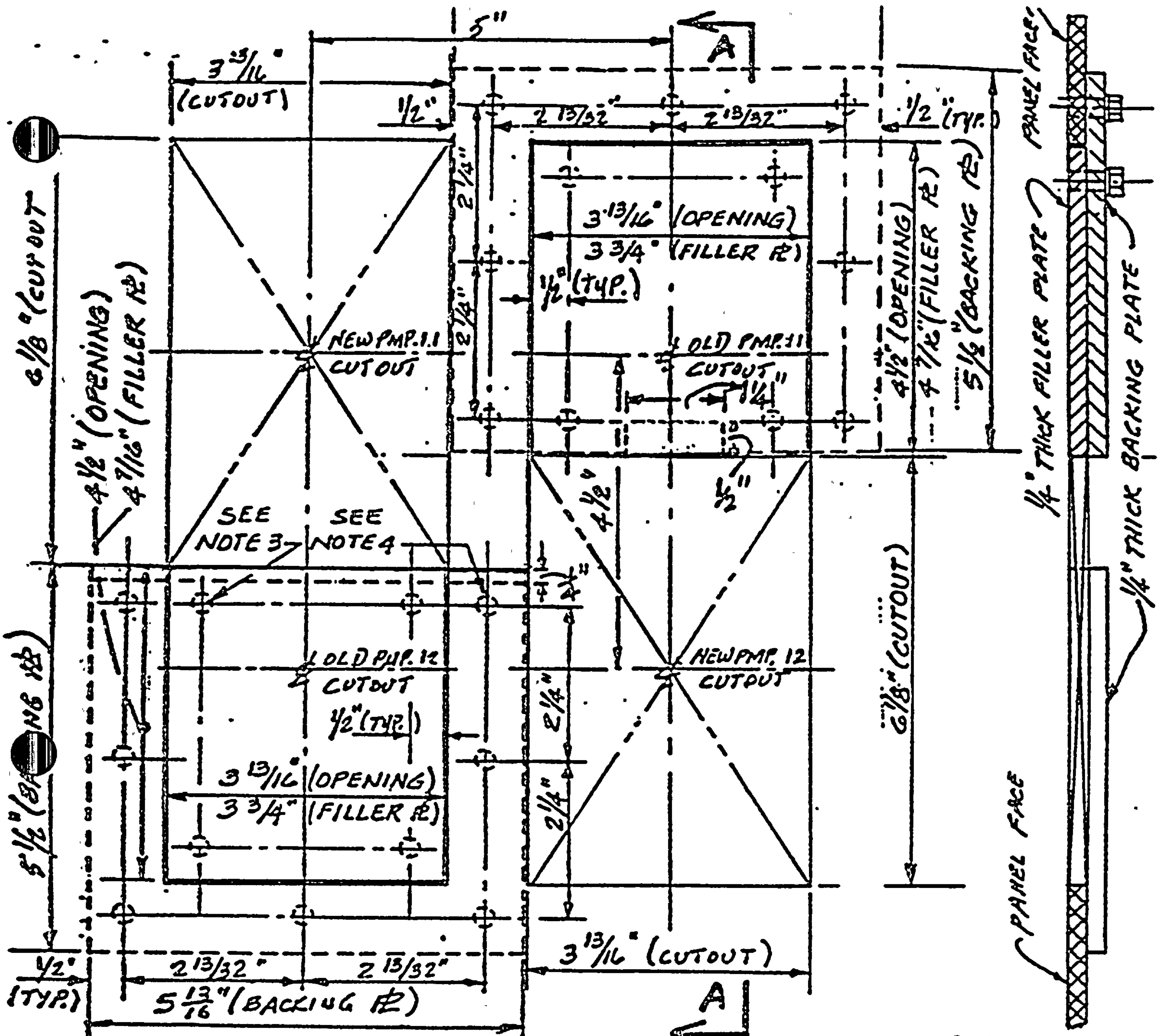
DRAWN: [signature] 8/16/87
[signature] 8/17/87
APPROVED: Frank Mori 8/17/87



SECT. A-A

CUTOUT / FILLER PL / BACKING PL DETAIL

NOTES:

1. BOLT TIGHTENING
BOLT SHOULD BE TIGHTENED TO A TORQUE VALUE OF 50 IN.-LBS (±2) BY A TORQUE WRENCH. (FOR #10-32 ASTM A574 BOLTS)

BOLTS ARE ALLOY STEEL ASTM A574.
LOCK WASHERS ARE MEDIUM C/STEEL ANSI B18.21.1.
FILLER & BACKING PLATES ARE ASTM A36 STEEL PLATE, 1/4" THICKNESS.

3. PROVIDE INDICATED HOLES FOR #10-32 HEXAGONAL SOCKET-HEAD CAP BOLTS WITH LOCK WASHERS. DRILL AND TAP MATCHING HOLES ON FILLER PLATE.

4. PROVIDE INDICATED HOLES FOR #10-32 HEXAGONAL SOCKET-HEAD CAP BOLTS WITH LOCK WASHERS. DRILL AND TAP MATCHING HOLES IN EXISTING PANEL

5. BACKING PLATE MAY BE CUT BACK 1/8" FROM THE EDGE OF THE FILLER PLATE AT THE DEVICE CUTOUT.

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SK-DCI-EJ-37348-2

DRAWN: [signature] 9/9/87
APPROVED: [signature] 9/17/87
[signature] Frank Mori 9/17/87

PROCEDURE FOR THE INSTALLATION OF
IN CONTROL BOARD VB1

1.0 Control Room Control Board VB1 is a Safety-Related, Seismic Category I structure containing Class 1E devices, so modifications effected in this board shall meet the requirements for a Q-equipment.

Modifications shown on this design change fall in above safety category, and, have been designed with back-up calculations, and analyses to ensure the integrity of the board and of adjacent mounted devices. Therefore, field work shall follow DCN drawings requirements and procedures as outlined below.

1.1 <u>Procedures</u>

1.1.1 <u>Materials</u>

Board fabrication materials shall be of the type and size specified by this design change notice. Standard materials to be used in the manufacturing of the fabricated parts are as follows:

a. Steel sheet in accordance with ASTM A36

b. Steel bar in accordance with ASTM A36

c. Steel angle in accordance with ASTM A36

1.1.2 <u>Dimensional Tolerances</u>

1. Overall dimensions (length, height and depth) of a bracket, structure are to be within ±0.125 inch tolerance.

2. Interjacent dimensions are to be within ±.125 inch tolerance.

3. Cutout locations are to be within ±.125 inch tolerance.

4. Cutout dimensions are to be within ±.062 inch tolerance.

5. Angular dimensions are to be within ±.50°.

6. Exterior surface flatness to be within ±1/8 inch tolerance for any two (2) foot square area.

1.1.3 Cutouts

Cutouts are to be made by the appropriate method; punching, nibbling, sawing and drilling, in accordance with chart tabulation.

1.1.4 Surface Preparation and Finishing

1. Clean panel with safety degreaser (fire retardant).
2. Grind all panels on outside and especially at any welded joints and corners.
3. Fill all bad scratches, etc. with filler, allow to dry and grind smooth.
4. Spray two (2) to three (3) coats of special gray or interior white primer on inside.
5. Spray three (3) to five (5) coats of primer surfacer on outside, allow to dry and sand with 320 and 400 grit sandpaper. Refill any remaining surface defects with glazing putty and resand as required.
6. Paint finish coat on outside of panel. Three (3) to six (6) coats for lacquer or two (2) coats for enamel. Finish coats to be of uniform thickness and free from sags, runs and/or smears.
7. External paint to be ~~#120 arbor green flat lacquer and~~ #121 arbor green light pastel epicote enamel (National Lead #45M27). External shall have a minimum total dry film thickness of 3 mills.
8. Interior to be R-CC interior white and have a minimum total dry film thickness of 2 mills.

1.2 Tests

The tests shall include the following:

a. Check of all panel mounted devices to assure that they are securely attached.

b. Check of all terminal blocks and interconnecting wiring to assure that they are firmly secure and have covers where applicable.

c. 100 percent point-to-point continuity tests and electrical insulation tests.

d. Tests performance should assure protection of instruments/devices from high-voltage input signals.

Attached checklists would help field-work completion, and can be used as checkpoints for the requirements of above procedures, and of DCN drawings.

The checklists are:

1. Checklist "A": Fabrication

2. Checklist "B": Grinding/Cutting

3. Checklist "C": Painting

4. Checklist "D": Tests

5. Checklist "E": Final Inspection

The above checklists are performed sequentially: A, B, C, D, and E sequence only.

CHECKLIST "A": FABRICATION

PANEL NO.: 1VB1

	OK	N/A	*
o Structure material is type and thickness specified.	[ ]	[ ]	[ ]
o Length, height and diagonal dimensions and flatness of structure (brackets, angles, bracing) surfaces are within specified tolerances.	[ ]	[ ]	[ ]
o Cutout and mounting holes are correctly located, seized and deburred.	[ ]	[ ]	[ ]
o Structural frame is correctly fabricated and dimensioned.	[ ]	[ ]	[ ]
o Welding is in accordance with applicable drawings and and specifications.	[ ]	[ ]	[ ]
o Switch coverplates, etc. are present, correct and identified.	[ ]	[ ]	[ ]
o Interior brackets, unistrut, stiffeners, wireways, etc. are correct.	[ ]	[ ]	[ ]
o Switch as mounted is correctly supported with uniform clearance and identified.	[ ]	[ ]	[ ]

* COMMENTS, CORRECTIONS, ETC.:

____

____

____

____

____

CHECKLIST "B": GRINDING/CUTTING
PANEL NO.: VB1

	OK	N/A	*
o Fabrication check completed.	[ ]	[ ]	[ ]
o Structure are smooth and free from indents, weld spatter, scratches, etc.	[ ]	[ ]	[ ]
o Surfaces to be painted have been steam cleaned with phosphatizing solution.	[ ]	[ ]	[ ]
o All surfaces to be painted have been prepared in accordance with applicable requirements.	[ ]	[ ]	[ ]

* COMMENTS, CORRECTIONS, ETC.:

CHECKLIST "C": PAINTING

PANEL NO.: VB1

	OK	N/A	*
o GRINDING/CUTTING CHECK COMPLETED.	[ ]	[ ]	[ ]
o Exterior and interior primer coats have been properly applied.	[ ]	[ ]	[ ]
o Finish coats are of the correct material, properly applied and treated.	[ ]	[ ]	[ ]
o All exterior surfaces are free from sags, runs or smears and finish is uniform.	[ ]	[ ]	[ ]
o Paint coating thickness is as specified.	[ ]	[ ]	[ ]
o Specified type of paint for panel interior/exterior was used.	[ ]	[ ]	[ ]

* COMMENTS, CORRECTIONS, ETC.:

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CHECKLIST "D": TESTS

PANEL NO.: VB1

	OK	N/A	*
o PAINTING CHECK COMPLETED	[ ]	[ ]	[ ]
o Calibrated Instruments used for tests..	[ ]	[ ]	[ ]
o Continuity test is in accordance with the applicable drawings.	[ ]	[ ]	[ ]
o High potential dielectric strength testing is done in accordance with the instrument manufacturer's instructions, and test results are acceptable.	[ ]	[ ]	[ ]
o Insulation resistance testing is done in accordance with the instrument manufacturer's instructions, and test results are acceptable.	[ ]	[ ]	[ ]
o Functional test results are acceptable.	[ ]	[ ]	[ ]
o Nameplates, labels, and designations properly identify the devices, and the proper materials are used in accordance with the applicable drawings.	[ ]	[ ]	[ ]
o Wiring methods and materials for the switch are in accordance with the applicable drawings and specifications.	[ ]	[ ]	[ ]
o Switch developments are correct.	[ ]	[ ]	[ ]

* COMMENTS, CORRECTIONS, ETC.:

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CHECKLIST "E": FINAL INSPECTION

PANEL NO.: VB1

	OK	N/A	*
o All test connections have been removed, terminal screws have been tightened, covers have been replaced, and all wiring disconnected for test has been reconnected.	[ ]	[ ]	[ ]
o TESTS CHECK COMPLETED	[ ]	[ ]	[ ]
o All material shortages have been documented.	[ ]	[ ]	[ ]
o All handrails, rubstrips, etc. are properly installed.	[ ]	[ ]	[ ]
o All paint blemishes have been touched up.	[ ]	[ ]	[ ]
o Panel has been properly cleaned.	[ ]	[ ]	[ ]
o Internal board bracing and brackets properly installed and cleaned.	[ ]	[ ]	[ ]
o Switch operates in accordance with Electrical DCN requirements	[ ]	[ ]	[ ]

* COMMENTS, CORRECTIONS, ETC.:

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HUMAN ENGINEERING DISCREPANCY RECORD
DIABLO CANYON NUCLEAR PLANT

DATE: 07/14/86

CHRON. NO: 356
TRACKING STATUS: ASSESSMENT

IDENTIFICATION:

DATA SOURCE: SURVEY,EOP VALIDATION

CATEGORY: C.5

LOCATION: VB1 UNIT 1: YES UNIT 2: YES

COMP NO: HED356-0 DESC: RHR PUMPS CONTROL SWITCHES AND AMMETERS

DESCRIPTION OF DISCREPANCY:

AMMETER ORIENTATION IS REVERSED WITH RESPECT TO THE CONTROL SWITCHES.

ASSESSMENT:

RECOMMENDATIONS:

A. NONE. B. REVISE THE MIMIC TO PROVIDE A LOGICAL LEFT-TO-RIGHT SEQUENCE OF RHR PUMP CONTROLS THAT MATCHES ASSOCIATED AMMETERS.

CORRECTIONS: Ref -59e

VERIFICATIONS:
VALIDATION:

DCR# (UNIT 1) : ISSUE DATE: / /
DCR# (UNIT 2) : ISSUE DATE: / /
OTHER : ISSUE DATE: / /

IMPLEMENTATION:

PRIORITY RATING: 2

IMPLEMENTATION SCHEDULE: **FOR INFORMATION**

COMMENTS:

PG&E

DCPP SIMULATOR

PACIFIC GAS AND ELECTRIC CO.
ENGINEERING DEPARTMENT
BLI NO. 060
PRIORITY NO.

NUCLEAR POWER PLANT DESIGN CHANGE

DCPP SIMULATOR
(PLANT)

DATE 9/9/87
DCN NO. DCO-EJ-37349
REV. NO. 0
SHEET 1 OF 22

To: KLEMME HERMAN   From: PAT. K. RANGAN

Structure or System: CONTROL ROOM VERTICAL BOARD VB1   Sys No. 10

Component: RHR PUMPS 1-1 AND 1-2 CONTROL SWITCHES.

REQUEST

Description of Change: ON VERTICAL BOARD VB1, RELOCATE THE CONTROL SWITCHES OF RHR PUMPS 1-1 AND 1-2. REVISE THE MIMIC TO CONFORM WITH THE CHANGE, SO AS TO PROVIDE A LOGICAL LEFT TO RIGHT SEQUENCE OF THE CONTROLS THAT MATCHES THE ASSOCIATED AMMETERS.

Reason for Change: THE CONTROL SWITCHES FOR THE RHR PUMPS 1-1 AND 1-2 HAVE BEEN LAID OUT IN THE REVERSE ORDER FROM THE AMMETERS CAUSING CONFUSION TO OPERATORS.

Schedule/Justification: BEFORE THE END OF UNIT #1 2ND RFO.

Construction Status (for revised DCNs): N/A ☐ Not Started ☐ Partially Complete ☐ Completed

List of Attachments Not on DDL: SK-DCO-EJ-37349-3

Requested by: P.K. RANGAN

REVIEW AND APPROVAL

Requested Change is:

- ☐ Approved per delegation of authority
- ☒ Approved
- ☐ Noted, document change not required
- ☐ Rejected (explain)
- ☐ Per telecon with ______ on ______ (If required by delegation of authority)
- ☐ As-built documents required
- ☐ Approved, document change only

Safety-Related Work: ☐ Yes ☒ No   DCN required to close an NCR ☐ Yes ☒ No

Important to Environmental Quality: ☐ Yes ☒ No   NCR No.

Not Safety-Related, requires Quality Assurance: ☐ Yes ☒ No

Reviewed By: PKRangan   9/9/1987
Discipline Engineer   Date

Checked By: Jorge J. Lisbon   9/10/87
Date

Frank Moi   9/17/87
Group Supervisor   Date

COMPLETION

	By	Date		By	Date
Installation Complete			Received Engineering		
Start-up Complete			Accepted Engineering		
Accepted NPG			Revisions Approved		

FCs Issued:

DIST: MNN, SS, PB, KN

TECHNICAL REVIEW

DCN No. DCO-EJ-37349
Rev. No. 0
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1a. DESIGN DOCUMENT REVIEW. The following documents are relevant to this change and have been originated, reviewed, or require revision as indicated:

Document, No., Rev.	Originated (Yes/No)	Reviewed (Yes/No)	Requires Revision (Yes/No)
• Q-List*		YES	NO
• Design Criteria Memorandums	NO	NO	NO
• Calculations	NO	NO	NO
• Design Verification Reports	NO	NO	NO
• Design Change Notices	NO	NO	NO

* Revisions to the Q-List are to be transmitted to the Mechanical Engineering EGS.

1b. DESIGN SAFETY REVIEW. The following is a list (not all inclusive) of design and safety issues to be considered. Indicate by "Yes" or "No" whether or not each issue affects or is affected by the change. Unless the reason for a "No" answer is obvious, further explanation is required. If "Yes," explain why or how the issue is relevant and how it is resolved. If "No," explain why the issue is not relevant.

	Relevant Issue? (Yes/No)	Comments (Use additional sheets as necessary)
• Accident Analysis (FSAR Chapters 6 and 15)	NO	CHANGES TO SIMULATOR DO NOT AFFECT PLANT OPERATION OR SAFETY
• ALARA	NO	
• Shielding/Radiation Zones	NO	
• Environmental Quality	NO	
• Fire Protection	NO	
• Unacceptable Components	NO	
• Codes and Standards	YES	RELEVANT CODES & STANDARDS HAVE BEEN ADHERED TO

	Relevant Issue? (Yes/No)	Comments
• Simulator	YES	THIS DCN
• System Interaction	NO	
• Regulatory Guides	NO	
• Environmental Qualifications	NO	
• General Design Criteria	NO	
• Seismic Qualification	NO	NOT REQUIRED FOR SIMULATOR
• Water Hammer	NO	
• Inservice Inspection	NO	
• Heavy Loads	NO	
• Flooding	NO	
• Radioactive Piping	NO	
• High-Medium Energy Line Break	NO	
• Control Room Design Review (Including habitability)	YES	SEE SHT. # 7 OF THIS DCN.
• Multi-Unit Impact	NO	
• Aluminum Inside Containment	NO	
• Security	NO	
• NPRDS	NO	
• Personnel Safety	NO	
• Masonry Block Walls	NO	
• Core Drilling Impact	NO	
• Redundancy/Separation Requirements	NO	
• Penetration Sealing	NO	
• Paint Inside Containment	NO	
• Refueling Operations	NO	
• Material Compatibility	NO	

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	Relevant Issue? (Yes/No)	Comments
o Vital Bus Loading	NO	
o Maintainability/Accessibility	NO	
o Floor or Wall Loading	NO	
o Missiles	NO	
o Operability	YES	SEE SHT#7 OF THIS DCN
o Electrical Design Considerations ~~(see attached sheet for discussion)~~	NO	
o Hydraulic Design Criteria ~~(see attached shee~~t ~~for discussio~~n)	NO	
o Chemistry Effects ~~(see attached shee~~t ~~for discussio~~n)	NO	
o I&C Design Considerations (see attached sheet #8 for discussion)	YES	SEE SHT#8 OF THIS DCN
o HVAC Design Considerations ~~(see attached s~~heet ~~for discussion)~~	NO	
o For nonsafety-related modifications, discuss why the design resulting from the modifications will not affect (a) any safety-related structures, systems or components, and (b) items identified in Paragraph 4.4.4(c) of Procedure 3.6 ON.	YES	NO (a) OR (b) ITEMS IN SIMULATOR AREA.
o Other	NONE	
o FMEA Evaluation	NO IMPACT	

2. LICENSING. The following NRC Licensing submittals are relevant to this change and have been reviewed as indicated below. Where a revision is required, Licensing has been notified.

Document	Reviewed	Requires Revision
a. FSAR #	YES	NO
b. Technical Specification	YES	NO
c. Other	NO	NO

Revisions to the FSAR require an FSAR Change Notice to be transmitted to NRA.

3. COORDINATION

Design Package Lead Discipline I&C

Coordination Required: [X] No [ ] Yes Coordinated With:

Department	Engineer (Signature)	Date	DCN Required	
			Yes ( )	No ( )
			Yes ( )	No ( )
			Yes ( )	No ( )
			Yes ( )	No ( )
			Yes ( )	No ( )
			Yes ( )	No ( )
			Yes ( )	No ( )

PG&E

PACIFIC GAS AND ELECTRIC COMPANY
ENGINEERING DEPARTMENT
DESIGN DOCUMENTS LIST
DOCUMENTS AFFECTED BY DESIGN CHANGE NOTICE (DCN)

PLANT DCPP SIMULATOR DCN NUMBER DCO-EJ-37349 REVISION 0 DATE 9/9/87 ENGINEER P.K. RANGAN

DOCUMENT NUMBER	SHEET NO.	REVISION			DOCUMENT TITLE	DATE COMPLETED	BY	DATE APPROVED	BY
		CURRENT	INTERIM	AS BUILT					
698798	10[illegible]	*	*		FRONT VIEW LAYOUT FOR VERT. BOARD VB1 D.C. SIMULATOR.				
698798	2	*	*		VB1 ARRANGEMENT				
					* TO BE DETERMINED BY DESIGN/DRAFTING.				

DESIGN CHANGE EVALUATION

1.0 A SAFETY EVALUATION OF THIS DCN BASED ON PROCEDURE 3.6ON, ATTACHMENT E INDICATES THAT THREE(3) ISSUES REQUIRE FURTHER ANALYSIS TO DETERMINE THE DEGREE OF SAFETY IMPACT TO THE SIMULATOR. THREE ISSUES WERE IDENTIFIED TO BE ANALYZED AS LISTED BELOW:

A. CONTROL ROOM DESIGN REVIEW (CRDR)

B. OPERABILITY

C. INSTRUMENTATION AND CONTROLS DESIGN CONSIDERATIONS.

2.0 ISSUE A - CONTROL ROOM DESIGN REVIEW

2.1 THE PRESENT DESIGN CHANGE INVOLVES IN REVISING THE ORDER OF RHR PUMP SWITCHES 1-1 AND 1-2 SUCH THAT THE SWITCHE FOR PUMPS 1-1 AND 1-2 ARE ARRANGED FROM LEFT TO RIGHT O. THE MIMIC ON VB1.

THE CRDR TEAM HAS REVIEWED THIS CHANGE AND CONCURS WITH THE REARRANGEMENT. THE PROPOSED CHANGES WILL ENHANCE CONTROL ROOM OPERATION BY REDUCING OPERATOR RESPONSE TIME

2.2. ISSUE B — OPERABILITY.

THE PROPOSED CHANGE TO REVISE THE ORDER OF RHR PUMP SWITCHES 1-1 AND 1-2 FROM LEFT TO RIGHT ON VB1 WILL NOT NEGATIVELY IMPACT PLANT OPERABILITY. NO ADDITIONS OR DELETIONS IN PLANT PARAMETER INFORMATION IS PROPOSED.

2.3 ISSUE C : INSTRUMENTATION & CONTROLS DESIGN CONSIDERATIONS

AS THE PROPOSED CHANGES DO NOT INCLUDE ANY CHANGE IN INSTRUMENT RANGES, SIZE, SCALES, INPUTS, SELECTION DEVICES OR OTHER FEATURES, ALL I&C DESIGN CONSIDERATIONS CAN BE CONSIDERED TO HAVE BEEN MET.

THE CRDR TEAM EVALUATION HAS EXAMINED THE HUMAN FACTORS ASP. OF THIS DESIGN CHANGE AND HAS FOUND NO GUIDELINES OR GOOD PRACTICES TO BE VIOLATED.

ANALYSIS BY: P. K. RANGAN DATE 9/9/1987

SIGNATURE P.K. Rangan
George J. Grisbon 9/16/87

APPROVAL BY: F. MORI DATE 9/17/1987

SIGNATURE Frank Mori

ITEM	TAG		DESCRIPTION	DEVICE	DWG.
186.	V1/231		CONTAINMENT FAN COOLER 11	M03KLCK	437600
187.	V1/232		CONTAINMENT FAN COOLER 12	M03KLCK	437600
188.	V1/233		CONTAINMENT FAN COOLER 13	M03KLCK	437600
189.	V1/234		CONTAINMENT FAN COOLER 14	M03KLCK	437600
190.	V1/235		CONTAINMENT FAN COOLER 15	M03KLCK	437600
191.	V1/236		RESIDUAL HEAT REMOVAL PUMP 11	M04PFU	437591
192.	V1/237		RESIDUAL HEAT REMOVAL PUMP 12	M04PFU	437591
193.	V1/238		CONTAINMENT SPRAY PUMP 11	M04PFU	437590
194.	V1/239		CONTAINMENT SPRAY PUMP 12	M04PFU	437590
195.	V1/240		SAFETY INJECTION PUMP 11	M04PFU	437589
196.	V1/241		SAFETY INJECTION PUMP 12	M04PFU	437589
197.	V1/242		AUXILIARY SALT WATER PUMP 11	M05SFU	437594
198.	V1/243		AUXILIARY SALT WATER PUMP 12	M05SFU	437594
199.	V1/244		COMPONET COOLING WATER PUMP 11	M05SFU	437593
200.	V1/245		COMPONET COOLING WATER PUMP 12	M05SFU	437593
201.	V1/246		COMPONET COOLING WATER PUMP 13	M05SFU	437593
202.	V1/247		SAFEGUARDS STATUS LIGHT BOX	LB4X20	663231
203.	V1/249		SAFEGUARDS STATUS LIGHT BOX	LB4X6	663231
204.	V1/251		SAFETY FEATURE VLVS. MONITOR LIGHT BOX A	LB4X11	437696
205.	V1/253		PHASE A/CONTMT VENT. ISOLATION MONITOR LIGHT BOX B	LB4X16	437697
206.	V1/255		S. I./FW ISO./STM GEN LEVEL MONITOR LIGHT BOX C	LB4X12	437698
207.	V1/257		PHASE B/MN. STM. ISOL. MONITOR LIGHT BOX D	LB4X8	437699
208.	V1/259	FI-970A	RHR FLOW TO COLD LEGS 1 & 2 FI-970A LOW	VX-252	663224-1
209.	V1/260	FI-970B	RHR FLOW TO COLD LEGS 1 & 2 FI-970B HIGH	VX-252	663224-1
210.	V1/262	LI-940	CONTMT RECIRC SUMP LEVEL LI-940	VX-252	171A033
211.	V1/264	LI-941	CONTMT RECIRC SUMP LEVEL LI-941	VX-252	173A033
212.	V1/266		CFCU CONDS. TRAY LEVEL LITES	LB3X5	437600
213.	V1/291	PK-1	5X5 ANNUNCIATOR LIGHT ARRAY	ANN5X5	501121/2
214.	V1/292	PK-2	5X5 ANNUNCIATOR LIGHT ARRAY	ANN5X5	501122/2
215.	V1/295	HC-8716A	RHR PUMP CROSS-TIE 8716A	M02DEBA	437605
216.	V1/296	HC-8716B	RHR PUMP CROSS-TIE 8716B	M02DEBA	437605
217.	V1/297	HC-8823	SIS C. L. 1 TO 4 ISOL. VALVE 8823	M02DCJX	437588
218.	V1/298	HC-8824	SIS H. L. 3 & 4 ISOL. VALVE 8824	M02DCJX	437588
219.	V1/299	HC-8825	SIS H. L. 1 & 2 ISOL. VALVE 8825	M02DCJ	437588
220.	V1/300		CONTAINMENT ISOLATION PHASE B TRAIN A RESET	M00UAH	663231-2
221.	V1/301		CONTAINMENT ISOLATION PHASE B TRAIN A RESET	M00UAH	663231-2
222.	V1/302	HC-8843	B. I. TK. OUT HDR BACK LEAK ISOL. VLV. 8843	M02DCJ	437588

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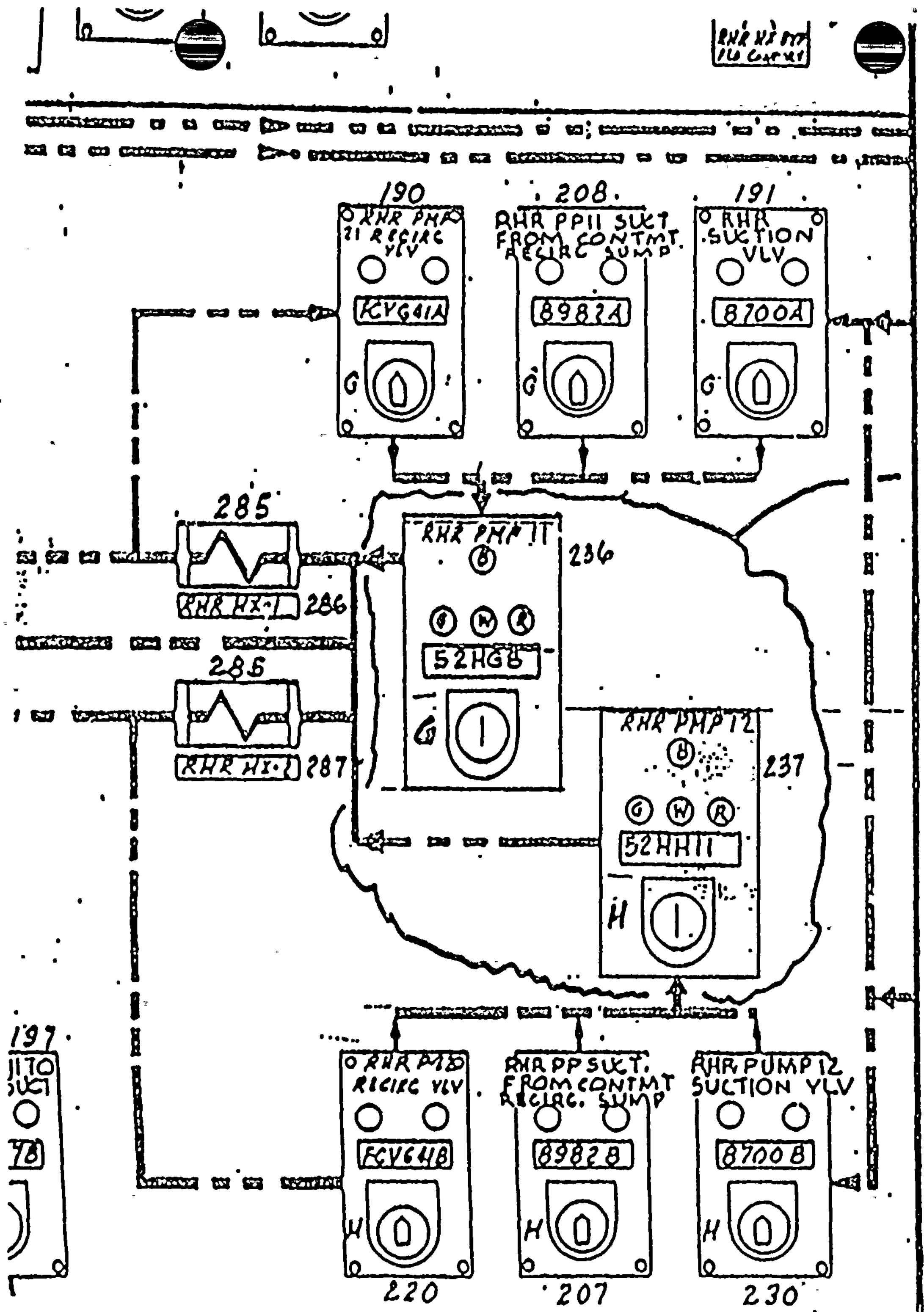
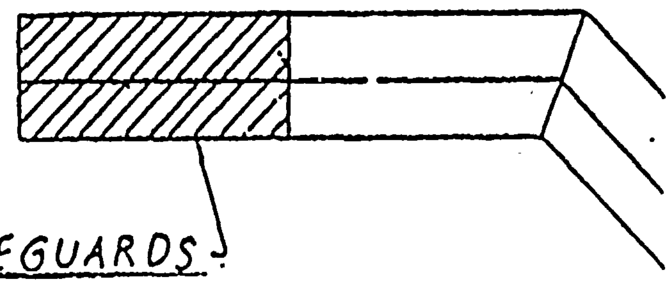
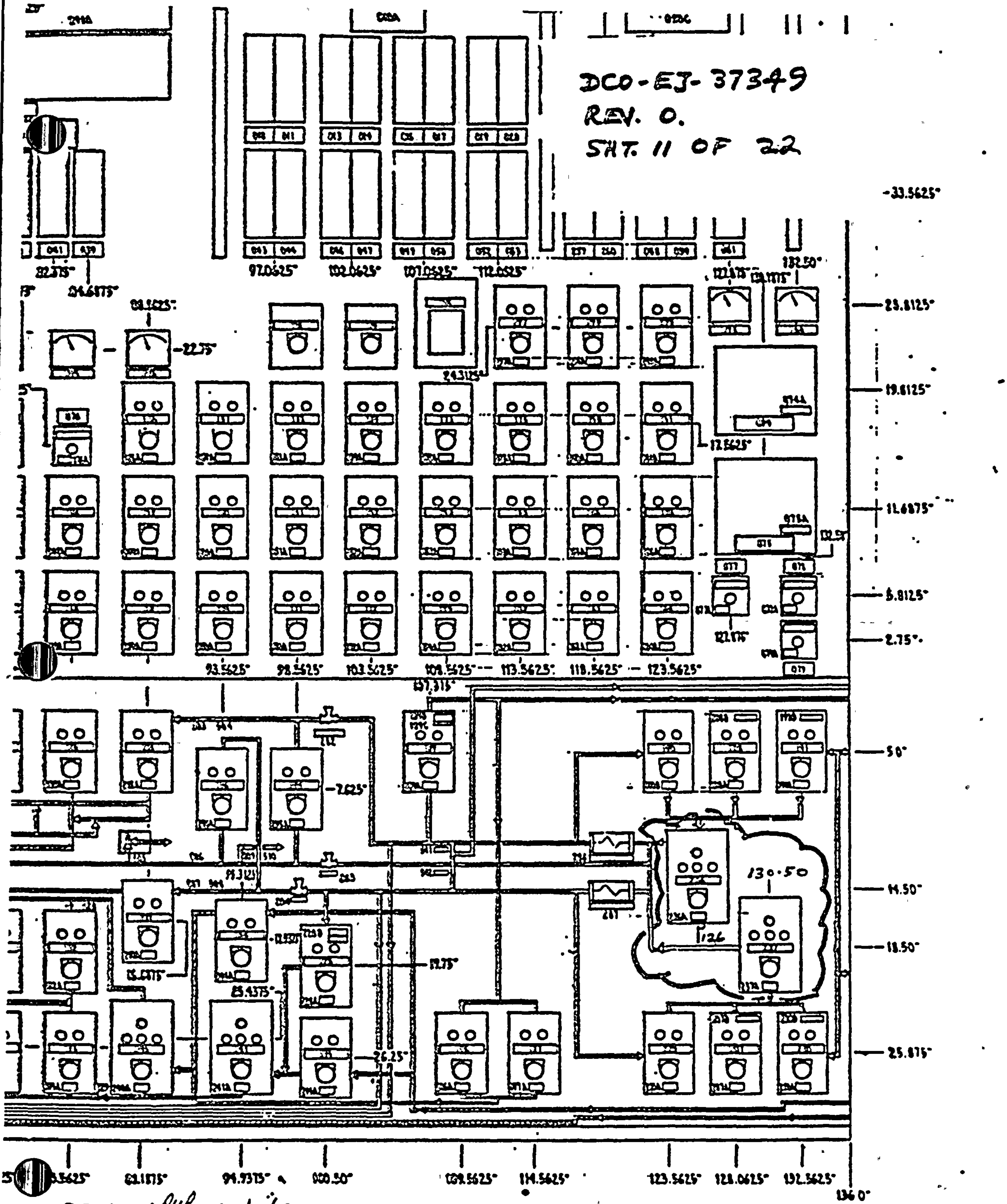


REVISED MIMIC - RHR PMP 1-1 & 1-2 CONTROLS RELOCATED.



DWG # 698798-2-1

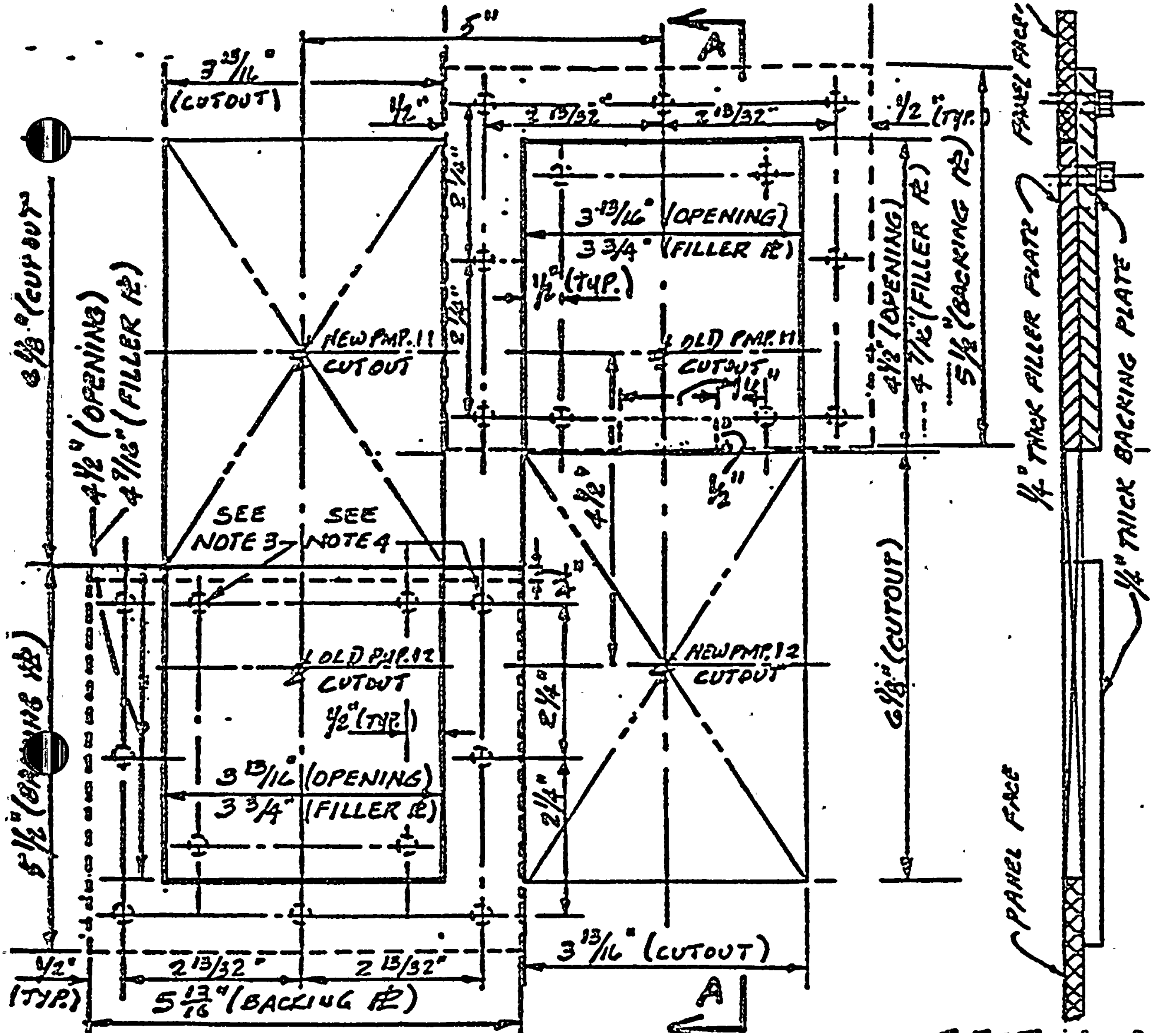
SK-DCO-EJ-37349-1 REV. 0	
DRAWN:	[signature] 8/16/87, [signature] 9/16/87
APPROVED:	Frank Moi 9/17/87



DRAWN: [signature] 7/9/87 DWG # 698798-2-1

APPROVED: Frank Moi 9/17/87

SKETCH. SK-DCO-EJ 37349-2



SECT. A-A

CUTOUT / FILLER PL / BACKING PL DETAIL

DCO-EJ-37349
REV. 0
SHT 12 OF 22

NOTES:

1. BOLT TIGHTENING
   BOLT SHOULD BE TIGHTENED TO A TORQUE VALUE OF 50 IN-LBS (±2) BY A TORQUE WRENCH. (FOR #10-32 ASTM A574 BOLTS)
2. BOLTS ARE ALLOY STEEL ASTM A574. LOCK WASHERS ARE MEDIUM C/STEEL ANSI B18.21.1.
   FILLER & BACKING PLATES ARE ASTM A36 STEEL PLATE, 1/4" THICKNESS.
3. PROVIDE INDICATED HOLES FOR #10-32 HEXAGONAL SOCKET-HEAD CAP BOLTS WITH LOCK WASHERS. DRILL AND TAP MATCHING HOLES IN FILLER PLATE.
4. PROVIDE INDICATED HOLES FOR #10-32 HEXAGONAL SOCKET-HEAD CAP BOLTS WITH LOCK WASHERS. DRILL AND TAP MATCHING HOLES IN EXISTING PANEL.
5. BACKING PLATE MAY BE CUT BACK 1/8" FROM THE EDGE OF THE FILLER PLATE AT THE DEVICE CUTOUT.

SK-DCO-EJ-37349-3

DRAWN: [signature] 9/9/87
APPROVED: [signature] 9/17/87
Frank Mori 9/17/87

PROCEDURE FOR THE INSTALLATION OF SWITCHES FOR RHR PUMPS 1-1 & 1-2 IN SIMULATOR CONTROL BOARD VB 1

1.0 The Simulator Control Board VB1 is a non-Seismic Category I structure, non-safety related board, and does not contain Class 1E devices. Therefore there are not Q requirements to be met in work required by this DCN. However, the procedures delineated below are engineering requirements to mount devices/instruments in this structure, and follow panel designer/manufacturer requirements. In addition, DCN brackets and support details as established in this design will ensure that recorder and modules as installed are adequately supported in the board.

1.1 Procedures

1.1.1 Materials

Board fabrication materials shall be of the type and size specified by this design change notice. Standard materials to be used in the manufacturing of the fabricated parts are as follows:

a. Steel sheet in accordance with ASTM A36

b. Steel bar in accordance with ASTM A36

c. Steel angle in accordance with ASTM A36

d. Welding wire in accordance with AWS A5.18-65, Class E70S-6

e. Welding rod in accordance with AWS A5.1-69, Class E-6013

- All completed welding shall exhibit a smooth, even contour with no irregularities in evidence, such as uneven starts and stops, poor weld contour, excessive rollover, lack of fusion, or an undercut limit of 1/64 inch.

- Welds shall have no cracks.

- Welds shall be subject to visual inspection with the aid of 5X magnification. Visual welding inspection shall be completed before the application of fillers or paint. Preferably, inspection should be done after finish grinding and sandblasting. The following correction requirements for nonconforming welds are applicable:

  a. Excessive overlap on the rear weld of square groove and double vee groove connections shall be corrected by removing enough weld metal to show that the connection is properly fused.

  b. Excessive concavity of weld, undersize welds, and undercutting shall be corrected by cleaning the weld and depositing additional weld metal.

  c. Cracks and pinholes shall be repaired by removing the defective portion of the weld and then rewelding the joint.

1.1.5 <u>Surface Preparation and Finishing</u>

1. Clean panel with safety degreaser (Fire retardant).

2. Grind all panels on outside and especially at any welded joints and corners.

1.1.2 Dimensional Tolerances

1. Overall dimensions (length, height and depth) of a bracket, structure are to be within ± 0.062 inch tolerance.

2. Interjacent dimensions are to be within ± .125 inch tolerance.

3. Cutout locations are to be within ± .062 inch tolerance

4. Cutout dimensions are to be within ± .062 inch tolerance

5. Angular dimensions are to be within ± .50°.

6. Exterior surface flatness to be within ± 1/8 inch tolerance for any two (2) foot square area.

1.1.3 Cutouts

Cutouts are to be made by the appropriate method; punching, nibbling, sawing and drilling, in accordance with chart tabulation.

1.1.4 Welding (Brackets, Angles, Bracing, etc.)

o All welding for board is to be completed using the GMAW process. Any deviations from this process shall be submitted for engineering approval.

o Fabrication welding shall follow YEP Standard Operating Procedure #155, Revision 0 as modified by Westinghouse Specification PJCV 8300, Revision 3, Appendix D.

3. Fill all bad scratches, etc. with filler, allow to dry and grind smooth.

4. Steam clean inside and outside with phosphatizing solution (Dura-Gard steam cleaner and phosphatizer).

5. Spray two (2) to three (3) coats of special gray or interior white primer on inside.

6. Spray three (3) to five (5) coats of primer surfacer on outside, allow to dry and sand with 320 and 400 grit sandpaper. Refill any remaining surface defects with glazing putty and resand as required.

7. Paint finish coat on outside of panel. Three (3) to six (6) coats for lacquer or two (2) coats for enamel. Finish coats to be of uniform thickness and free from sags, runs and/or smears.

8. The interior of the panel shall be painted Polane White #F63W13 as manufactured by Sherwin Williams.

9. The exterior of the panel shall be painted with PG&E #121 Arbor green light pastel ~~as manufactured by Sherwin Williams.~~ (National Lead # 45M27). Paint shall have a minimum total dry film thickness of 3 mils.

3.2 Tests

The tests shall include the following:

a. Check of all panel mounted devices to assure that they are securely attached.

b. Check of all terminal blocks and interconnecting wiring to assure that they are firmly secure and have covers where applicable.

c. 100 percent point-to-point continuity tests and electrical insulation tests.

d. Control circuits/indicating modules functional tests, as far as possible by actuating switches during the application of appropriate signals while observing the reponse of panel devices. Tests performance should assure protection of instruments/devices from high-voltage input signals.

Attached checklists would help field work completion, and may be used as check-points for the requirements of above procedures, and of the DCN drawings. The checklists are:

1. Checklist "A": Fabrication
2. Checklist "B": Grinding/Cutting
3. Checklist "C": Painting
4. Checklist "D": Tests
5. Checklist "E": Final Inspection

The above checklists are performed sequentially: A, B, C, D, and E sequence only.

CHECKLIST "A": FABRICATION
PANEL NO.: VB1

	OK	N/A	*
o Structure material is type and thickness specified.	[ ]	[ ]	[ ]
o Length, height and diagonal dimensions and flatness of structure (brackets, angles, bracing) surfaces are within specified tolerances.	[ ]	[ ]	[ ]
o Cutout and mounting holes are correctly located, seized and deburred.	[ ]	[ ]	[ ]
o Structural frame is correctly fabricated and dimensioned.	[ ]	[ ]	[ ]
o Welding is in accordance with applicable drawings and and specifications.	[ ]	[ ]	[ ]
o SWITCHES coverplates, etc. are present, correct and identified.	[ ]	[ ]	[ ]
o Interior brackets, unistrut, stiffeners, wireways, etc. are correct.	[ ]	[ ]	[ ]
o SWITCHES as mounted ~~are~~ are correctly supported with uniform clearance and identified.	[ ]	[ ]	[ ]

* COMMENTS, CORRECTIONS, ETC.:

CHECKLIST "B": GRINDING/CUTTING
PANEL NO.: VB/

	OK	N/A	*
o Fabrication check completed.	[ ]	[ ]	[ ]
o Structure are smooth and free from indents, weld spatter, scratches, etc.	[ ]	[ ]	[ ]
o Surfaces to be painted have been steam cleaned with phosphatizing solution.	[ ]	[ ]	[ ]
o All surfaces to be painted have been prepared in accordance with applicable requirements.	[ ]	[ ]	[ ]

* COMMENTS, CORRECTIONS, ETC.:

CHECKLIST "C": PAINTING
PANEL NO.: VB1

	OK	N/A	*
o GRINDING/CUTTING CHECK COMPLETED.	[ ]	[ ]	[ ]
o Exterior and interior primer coats have been properly applied.	[ ]	[ ]	[ ]
o Finish coats are of the correct material, properly applied and treated.	[ ]	[ ]	[ ]
o All exterior surfaces are free from sags, runs or smears and finish is uniform.	[ ]	[ ]	[ ]
o Paint coating thickness is as specified.	[ ]	[ ]	[ ]
o Specified type of paint for panel interior/exterior was used.	[ ]	[ ]	[ ]

* COMMENTS, CORRECTIONS, ETC.:

DCN-DCO-EJ-37349
REV 0
SHEET 21 OF 22

CHECKLIST "D": TESTS

PANEL NO.: VB1

	OK	N/A	*
o PAINTING CHECK COMPLETED.	[ ]	[ ]	[ ]
o Calibrated Instruments used for tests..	[ ]	[ ]	[ ]
o Continuity test is in accordance with the applicable drawings.	[ ]	[ ]	[ ]
o High potential dielectric strength testing is done in accordance with the instrument manufacturer's instructions, and test results are acceptable.	[ ]	[ ]	[ ]
o Insulation resistance testing is done in accordance with the instrument manufacturer's instructions, and test results are acceptable.	[ ]	[ ]	[ ]
o Functional test results are acceptable.	[ ]	[ ]	[ ]
o Nameplates, labels, and designations properly identify the devices, and the proper materials are used in accordance with the applicable drawings.	[ ]	[ ]	[ ]
o Wiring methods and materials are in accordance with the applicable drawings and specifications.	[ ]	[ ]	[ ]
	[ ]	[ ]	[ ]

* COMMENTS, CORRECTIONS, ETC.:

____

____

____

____

DCN-DCO-EJ-37349
REV 0
SHEET 22 OF 22

CHECKLIST "E": FINAL INSPECTION
PANEL NO.: VB1

	OK	N/A	*
o All test connections have been removed, terminal screws have been tightened, covers have been replaced, and all wiring disconnected for test has been reconnected.	[ ]	[ ]	[ ]
o TESTS CHECK COMPLETED.	[ ]	[ ]	[ ]
o All material shortages have been documented.	[ ]	[ ]	[ ]
o All handrails, rubstrips, etc. are properly installed.	[ ]	[ ]	[ ]
o All paint blemishes have been touched up.	[ ]	[ ]	[ ]
o Panel has been properly cleaned.	[ ]	[ ]	[ ]
o Internal recorder bracing and brackets properly installed, cleaned, etc.	[ ]	[ ]	[ ]
o Switch works properly in accordance with Data requirements.	[ ]	[ ]	[ ]
o Switch operates properly in accordance with electrical DCN requirements.	[ ]	[ ]	[ ]

* COMMENTS, CORRECTIONS, ETC.:

____
____
____
____
____
____

0242T/0004T-23

PG&E

Procedure 3.6 on
Attachment C
Page 1 of 5

NUCLEAR POWER PLANT DESIGN CHANGE

PACIFIC GAS AND ELECTRIC CO.
ENGINEERING DEPARTMENT
BLI NO. 060
PRIORITY NO. 482/880

DCPP UNIT 1
(PLANT)

DATE 8/26/87
DCN NO. DC1-EE-37348
REV. NO. 0
SHEET 1 OF 8

To: S. AUER   From: S. L. WONG

REQUEST

Structure or System: RHR SYSTEM   Sys No. 10

Component: CONTROL SWITCHES ON VB1

Description of Change: REVISE THE LOCATION OF RHR PUMP 11 & 12 CONTROL SWITCHES ON VB1 TO CONFORM TO A LOGICAL LEFT TO RIGHT SEQUENCE WHICH MATCHES THE ASSOCIATED AMMETERS.

~~THIS WORK MUST BE ACCOMPLISHED WHEN FUEL IS OFFLOAD~~

Reason for Change: TO AVOID OPERATOR CONFUSION. THE EXISTING LAYOUT OF THE CONTROL SWITCHES IS NOT IN THE CONVENTIONAL LEFT TO RIGHT SEQUENCE AND DOES NOT MATCH WITH THE LAYOUT OF THE AMMETERS LOCATED ON THE VERTICAL BOARD.

Schedule/Justification: WORK TO BE DONE DURING THE NEXT PLANT OUTAGE.

Construction Status (for revised DCNs): ☒ Not Started   ☐ Partially Complete   ☐ Completed

List of Attachments Not on DDL: NONE

Requested by: SIMON L. WONG

REVIEW AND APPROVAL

Requested Change is:

- ☐ Approved per delegation of authority
- ☒ Approved
- ☐ Noted, document change not required
- ☐ Rejected (explain)
- ☐ Per telecon with ______ on ______ (If required by delegation of authority)
- ☐ As-built documents required
- ☐ Approved, document change only

Safety-Related Work: ☒ Yes ☐ No
DCN required to close an NCR ☐ Yes ☒ No
Important to Environmental Quality: ☐ Yes ☒ No
NCR No.
Not Safety-Related, requires Quality Assurance: ☐ Yes ☒ No

Reviewed By: Simon L. Wong   8/24/87
Discipline Engineer   Date

Checked By: [signature]   8/24/87
Date

[signature]   8/24/87
Group Supervisor   Date

COMPLETION

	By	Date		By	Date
Installation Complete			Received Engineering		
Start-up Complete			Accepted Engineering		
Accepted NPG			Revisions Approved		
FCs Issued:					

DIST: MNN, JS, PB, SA, SF

TECHNICAL REVIEW

DCN No. DCI-EE-37348
Rev. No. 0
Page 2 of 8

1a. DESIGN DOCUMENT REVIEW. The following documents are relevant to this change and have been originated, reviewed, or require revision as indicated:

Document, No., Rev.	Originated (Yes/No)	Reviewed (Yes/No)	Requires Revision (Yes/No)
o Q-List*		YES	NO
o Design Criteria Memorandums			
NA			
o Calculations			
NA			
o Design Verification Reports			
NA			
o Design Change Notices			
DCI-EJ-37348	NO	YES	NO

* Revisions to the Q-List are to be transmitted to the Mechanical Engineering EGS.

1b. DESIGN SAFETY REVIEW. The following is a list (not all inclusive) of design and safety issues to be considered. Indicate by "Yes" or "No" whether or not each issue affects or is affected by the change. Unless the reason for a "No" answer is obvious, further explanation is required. If "Yes," explain why or how the issue is relevant and how it is resolved. If "No," explain why the issue is not relevant.

	Relevant Issue? (Yes/No)	Comments (Use additional sheets as necessary)
o Accident Analysis (FSAR Chapters 6 and 15)	NO	
o ALARA	NO	
o Shielding/Radiation Zones	NO	
o Environmental Quality	NO	
o Fire Protection	NO	
o Unacceptable Components	NO	
o Codes and Standards	YES	COMPLIES WITH HUMAN FACTORS ENGINEERING GUIDELINES

	Relevant Issue? (Yes/No)	Comments
Simulator	YES	SEE DCO-EJ-37349
System Interaction	YES	SEE SISIP FORM
Regulatory Guides	NO	
Environmental Qualifications	NO	
General Design Criteria	NO	
Seismic Qualification	YES	APPROVED BY SEISMIC REVIEW GROUP
Water Hammer	NO	
Inservice Inspection	NO	
Heavy Loads	NO	
Flooding	NO	
Radioactive Piping	NO	
High-Medium Energy Line Break	NO	
Control Room Design Review (Including habitability)	YES	GENERATED BY CRDR TEAM
Multi-Unit Impact	NO	
Aluminum Inside Containment	NO	
Security	NO	
NPRDS	NO	
Personnel Safety	NO	
Masonry Block Walls	NO	
Core Drilling Impact	NO	
Redundancy/Separation Requirements	YES	MAINTAIN EXISTING SEPARATION
Penetration Sealing	NO	
Paint Inside Containment	NO	
Refueling Operations	NO	
Material Compatibility	NO	

	Relevant Issue? (Yes/No)	Comments
o Vital Bus Loading	NO	
o Maintainability/Accessibility	NO	
o Floor or Wall Loading	NO	
o Missiles	NO	
o Operability	YES	OPERATOR TRAINING REQD
o Electrical Design Considerations ~~(see attached sheet for discussion)~~	YES	MAINTAIN EXISTING SEPARATION
o Hydraulic Design Criteria (~~see attached sheet for discussion)~~	NO	
o Chemistry Effects (~~see attached sheet for discussion)~~	NO	
o I&C Design Considerations ~~(see attached sheet for discussion)~~	NO	
o HVAC Design Considerations (~~see attached sheet for discussion)~~	NO	
o For nonsafety-related modifications, discuss why the design resulting from the modifications will not affect (a) any safety-related structures, systems or components, and (b) items identified in Paragraph 4.4.4(e) of Procedure 3.6 ON.	NO	
o Other	NO	N/A

o FMEA Evaluation — CHANGING SWITCH POSITIONS HAS NO IMPACT ON FAILURE MODE

DCI-EE-37348
SHEET 5 OF 8

2. LICENSING. The following NRC Licensing submittals are relevant to this change and have been reviewed as indicated below. Where a revision is required, Licensing has been notified.

Document	Reviewed	Requires Revision
a. FSAR #	YES	NO
b. Technical Specification	YES	NO
c. Other NONE		

Revisions to the FSAR require an FSAR Change Notice to be transmitted to NRA.

3. COORDINATION
Design Package Lead Discipline I&C

Coordination Required: [ ] No [X] Yes Coordinated With:

Department	Engineer (Signature)	Date	DCN Required	
I &C	J. LISBOA J Lisbon	9/14/87	Yes (X)	No ( )
			Yes ( )	No ( )
			Yes ( )	No ( )
			Yes ( )	No ( )
			Yes ( )	No ( )
			Yes ( )	No ( )
			Yes ( )	No ( )

PG&E

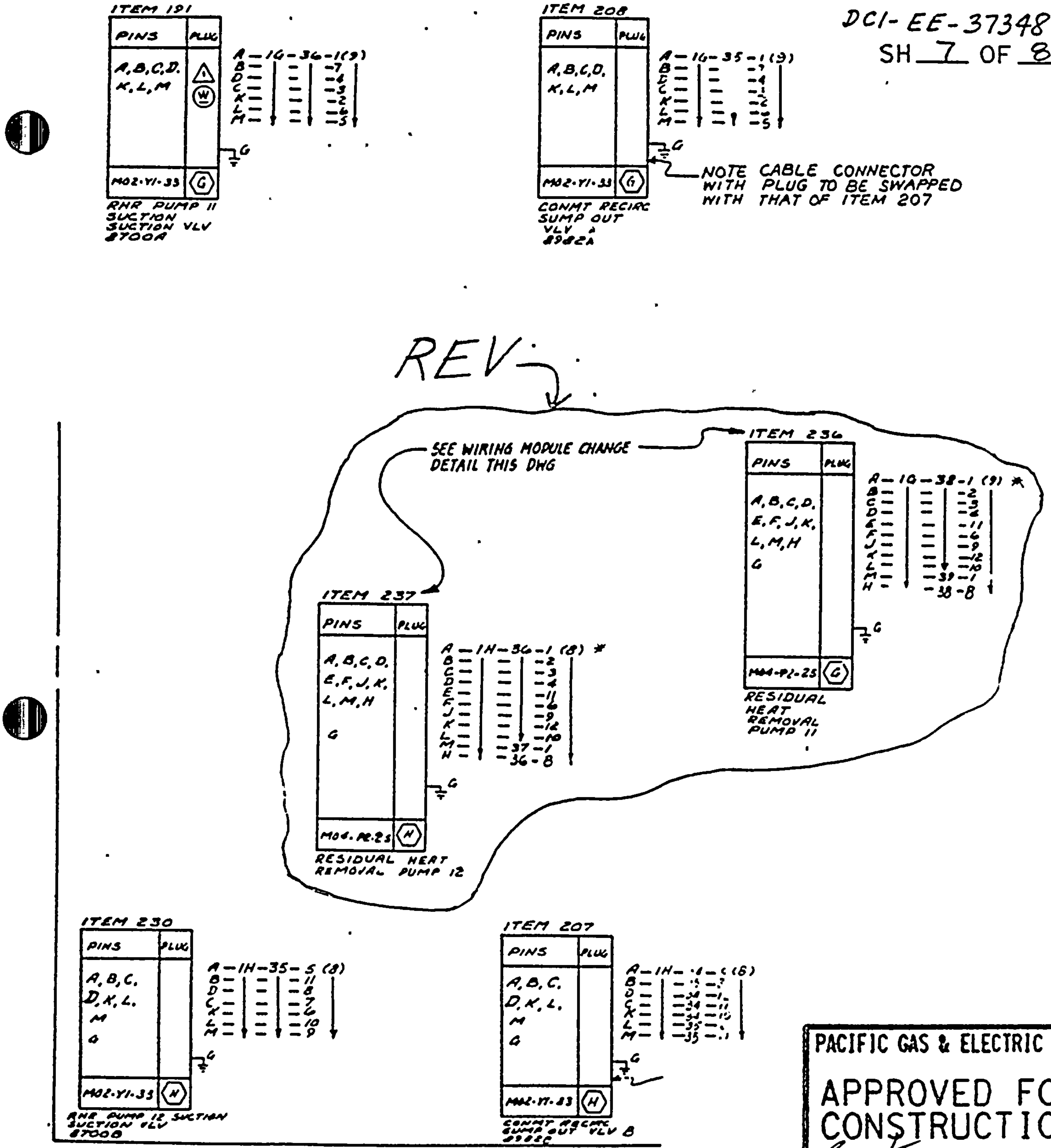
PACIFIC GAS AND ELECTRIC COMPANY
ENGINEERING DEPARTMENT
DESIGN DOCUMENTS LIST
DOCUMENTS AFFECTED BY DESIGN CHANGE NOTICE (DCN)

Procedure 3.6 ON
Attachment D
Page 1 of 1

PLANT DCPP UNIT 1 DCN NUMBER DC1-EE-37348 REVISION 0 DATE 3/6/87 ENGINEER S.L. GIANG

DOCUMENT NUMBER	SHEET NO.	REVISION			DOCUMENT TITLE	DATE COMPLETED	BY	DATE APPROVED	BY
		CURRENT	INTERIM	AS BUILT					
452379	-	7	7A		D/C MAIN CONTROL BD, 1VB1				

DCI-EE-37348
SH 7 OF 8



PACIFIC GAS & ELECTRIC CO.

APPROVED FOR CONSTRUCTION

ENGINEERING DEPARTMENT

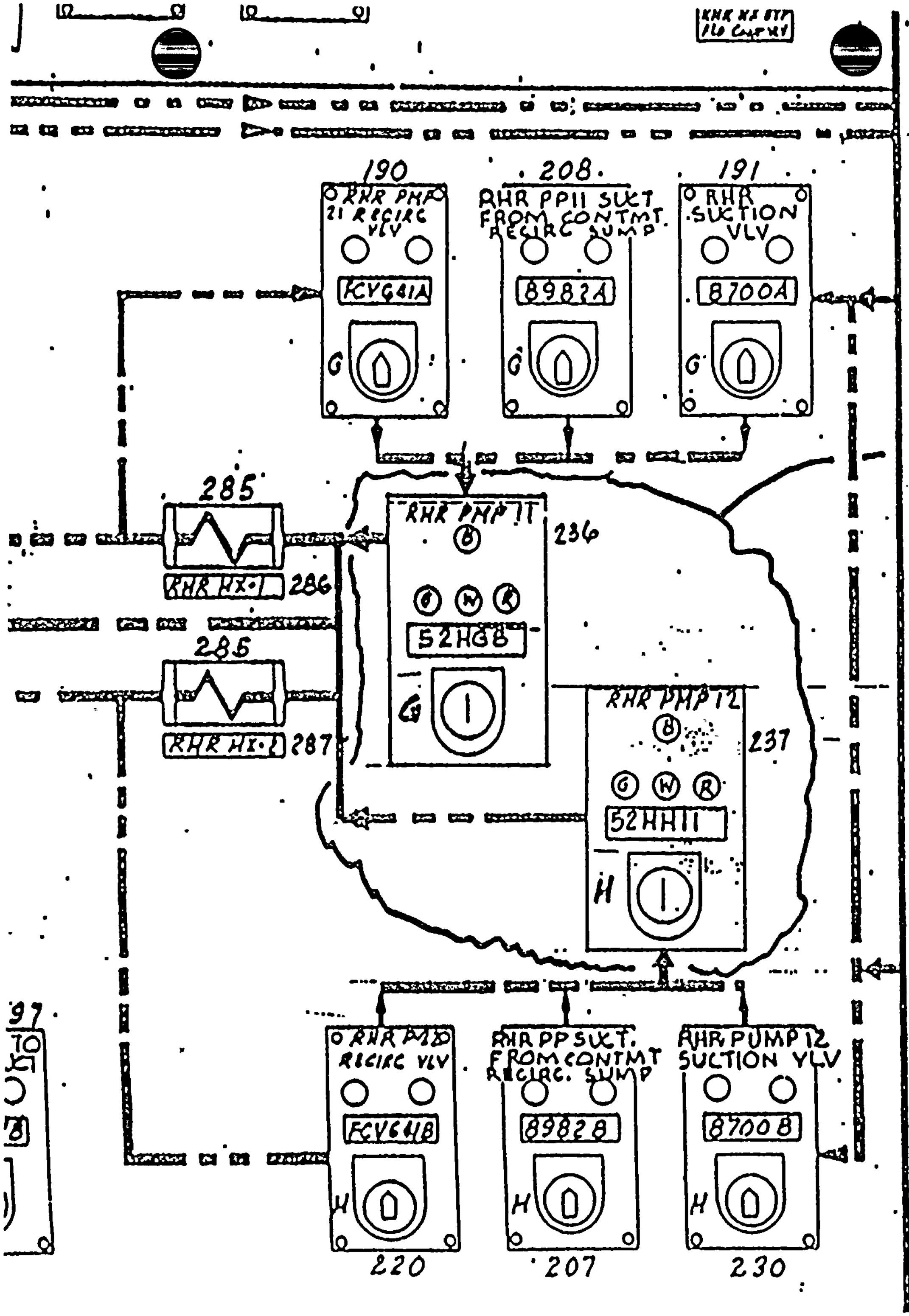
BY [signature] DATE 3/24/87

TRACE	BY DWL	CHECK	SUPV	DATE 3/23/87	DIABLO CANYON UNIT NO. 1	
ELECTRICAL REVISION SHEET	RELOCATE ITEMS 236 & 237				SHEET 1 OF 1 SHEETS	
					DRAWING NUMBER SK452379	REV 7A

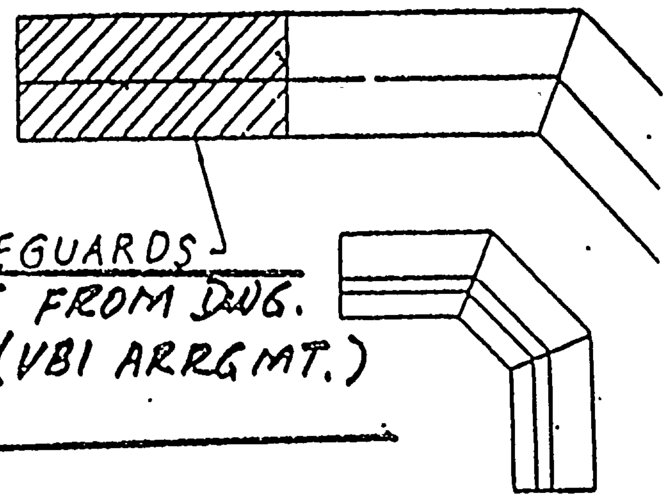
[160,130]840746.02C 05-23-84 MJC

DCI-EE-37348
REV. 0
SHT. 8 OF 8

3/23/87



REVISED MIMIC - RHR PMP 1-1 & 1-2 CONTROLS RELOCATED.



EXCERPT FROM DWG.
#504481 (VBI ARRGMT.)
REV.

FOR INFORMATION ONLY

3/24/87

SYS 38 DCN ESTIMATE INPUT FORM

DATA ENTRY DATE ____________

DCN COSTS ESTIMATE DNUESTM1

PAGE 1 OF 1

UNT	DPT	DIS	DCN SEQ	Rev	CU	MANUAL DIRECT MHRS	DIRECT DISTRB MHRS	DDX	NON-MANUAL MHRS	NM %	ENGRG MHRS	MATRL COSTS	S/C COSTS	SC ∯	ESTIM LOG NO.	DCP/REV # (FOR REF. ONLY)
1	E	J	37348			Ø	Ø		Ø		32	Ø.	Ø.		SF	J37348 Rev 0
1	E	E	37348			50		N		N	48	0.1	Ø.		SF	↓
0	E	J	37349			35		N		N	24	0.1	Ø.		SF	↓
																BLE 60

NOTES ⊛ "CHANGE" ENTRY MUST BE COMPLETE DATA REPLACEMENT FOR DCN/REV.

⊛ TYPE OF ENTRY:
- ADD = NO DCN/REV ESTIMATE RECORD YET EXISTS
- CHANGE = DCN/REV ESTIMATE RECORD EXISTS
- DELETE = REMOVE ENTIRE DCN/REV ESTIM. RECORD

∯ = ALPHA CHARACTER Ø = ZERO

⊛ REV:
- Rev. 0 = Blank
- Rev. 1 = RØ1
- Rev. 2 = RØ2

⊛ DIRECT DISTRIB MHR ONLY ENTERED WHEN DIFFERENT FROM STD. %. AN "N" MUST BE ENTERED IN DD%.

⊛ NON-MANUAL MHR ENTRY PROCEDURE SAME AS FOR DIRECT DISTRIB & NM %.

⊛ LOG #:
- SF = SFHO ESTIMATE
- xxxF = FIELD ESTIM BY LOG#

⊛ MAT'L & S/C COSTS ARE IN 1000'S TO NEAREST 100 IE $1200 = 1.2

4/1/87

UNIT: 1 BLI NO.: 60

COST STUDY NO. ______ REV. ____

DCP NO. J37348 REV. 0 Disc. J

DCN NO. 1EJ37348 REV. 0 Priority ____ Disc. J Facility ____

PACIFIC GAS AND ELECTRIC COMPANY

DIABLO CANYON POWER PLANT

TAKEOFF MA APPROVED ______

PRICED MA DATE 9/19/87

CHECKED Cz SHEET 1 OF 3

ITEM & DESCRIPTION	QUANTITY	UNIT	UNIT COST MAT'L	UNIT COST S/C	MANHOURS UNIT	MANHOURS TOTAL	COST MATERIAL	COST SUBCONTRACT
WORK TO BE DONE UNDER ELECT						8		
DCN 1EE37348								
ENGR. ONLY 32 HR								
THIS DCN COST = 32 x $79 = $2,500								
ELECT. DCN 1EE37348 SH 2 $10,000								
SIM DCN 0EJ37349 SH 3 6,200								
TOTAL DCP PROJ COST → $18,700								

UNIT: 1 BLI NO.: 60

PACIFIC GAS AND ELECTRIC COMPANY

DIABLO CANYON POWER PLANT

COST STUDY NO. ______ REV. ______

DCP NO. J37348 REV. 0 Disc. J

DCN NO. 1EE37348 REV. 0 Priority ______ Disc. E Facility ______

TAKEOFF [initials] APPROVED ______

PRICED [initials] DATE 9/19/87

CHECKED [initials] SHEET 2 OF 3

ITEM & DESCRIPTION	QUANTITY	UNIT	UNIT COST MAT'L	UNIT COST S/C	MANHOURS UNIT	MANHOURS TOTAL	COST MATERIAL	COST SUBCONTRACT
SWAP LOCATION OF RHR PUMP SWITCHES	2	EA			4	8		
CUT-OUT PANEL FOR THE SWITCH	2	EA			4	8		
FAB & INSTALL FILLER PLATES	2	EA	50		8	16	100	
REWORK MIMIC LINES	1	LT			3	3		
TERM & DETER	44	EA			.25	11		
TEST & CALIB	1	LT			4	4		
FIELD COST = (50 x $121) + $100 = $6,200								
ENGR COST = 48 x $79 = $3,800								
TOTAL DCN 1EE37348 $10,000								
						50	100	

UNIT: 0 BLI NO.: 60

PACIFIC GAS AND ELECTRIC COMPANY

DIABLO CANYON POWER PLANT

COST STUDY NO. ______ REV. ____

DCP NO. J37348 REV. 0 Disc. J

DCN NO. 0EJ37349 REV. 0 Priority ____ Disc. J Facility ____

TAKEOFF [illegible] APPROVED ____

PRICED [illegible] DATE 9/19/87

CHECKED [illegible] SHEET 3 OF 3

ITEM & DESCRIPTION	QUANTITY	UNIT	UNIT COST MAT'L	UNIT COST S/C	MANHOURS UNIT	MANHOURS TOTAL	COST MATERIAL	COST SUBCONTRACT
SWAP LOCATION OF RHR PUMP SWITCHES	2	EA			3	6		
CUTOUT PANEL FOR THE SWITCHES	2	EA			4	8		
FAB & INSTALL FILLER PLATE	2	EA	50		6	12	100	
TERM & DETERM	44				.2	9		
FIELD COST = ($35 x 121) + 100 = $4,300								
ENGR COST = 24 x $79 = $1,900								
THIS DCN 0EJ37349 → $6,200								
						35		

APPENDIX D

PAM PANEL DEMARCATION AND HIERARCHICAL LABELING

DESIGN CHANGE PACKAGE (DCP) J-38114

TPL

PRIORITY ______
Required Condition, Mode, or Operability Status to Implement:
Plant DCPP 2
System Various
Component control panel face (PAM 1)
DCPP Reviewer   Date

No. J-38114-R 1

DESIGN CHANGE COVER SHEET 1 of 1

Corresponding Unit 1 DCN is DC1-EJ-37114

Corresponding Simulator DCN is DC0-EJ-37115

DESCRIPTION OF CHANGE: Apply panel demarcation and new hierarchical labeling to enhance functional grouping of instruments on panel PAM-1. Label placement and engraving should be done per the "Supplement to the Bill of Materials" in this DCP, & SK-DC2-EJ-38114-1, -2, -3 & -4.

REASON FOR CHANGE: Panel demarcation and hierarchical labeling will enhance functional grouping and identification of instruments on panel, correlating related devices. This change addresses control room design review, BL1060, which is a PG&E commitment to the NRC for maintaining DCPP operating license. This change specifically addresses HED's. 263, 459 & 61.

RELATIONSHIP TO OPEN DCNs:

[ ] Not Applicable

"FOR INFORMATION ONLY"

Pre-requisites — DCNs providing functional grouping:

DC2-EJ-38096
DC2-EJ-38098
DC2-EJ-38090

Rev. 1 of this DCP adds painting instructions & checklist

EFFECT ON PLANT OPERATION: Demarcation and hierarchical labeling will enhance device functional grouping and assist operator identification of devices and device relationships. New label color & style will standardize labeling to be highly legible and consistently worded. Changes will eliminate potential sources of operator confusion and enhance plant operation.

[ ] Not Applicable

RESPONSIBLE DISCIPLINE I&C

UNIT 2

DESIGN CHANGE PACKAGE FOLLOWER

DCP# J-38114 - R 1

COST & SCHEDULING/ENGINEERING

ESTIMATE

			By	Date
Engineering Start	10-6-86	Engineer	HParris	12-19-86
Construction Complete	by end of U2 outage	Approved (EGS)	IMoi	4/10/87
Estimated Total Cost	$0	Accepted (PE)	[illegible]	4/21/87
Authorizing Job Estimate Number	BU060Z, JE1944374			

By Date

Cost & Scheduling Spv. [illegible] 4/21/87 ☐ Not Required (level of effort)

DESIGN — ENGINEERING

	By	Date		By	Date
Engineer	[illegible]	4/09/87	Mechanical Discipline (Safety Evaluation)	[illegible]	4/21/87
Approved (grp ldr)	IMoi	4/10/87	Accepted (PE)	[illegible]	4/4/87

CONSTRUCTION — PLANT STAFF REVIEW

	By	Date		By	Date
Received by EM			Imp to Safety or		
PSRC Approves (y/n) ___			Imp to Environ (y/n) ___		

Reason for Rejection ______

Plant Manager Approval ______ Date ______

Transmitted By ______ Date ______

CONSTRUCTION

	By	Date		By	Date
Received			Installation Complete		
Start Up Complete			As-Builts Attd (y/n) ___		

Released by Package Coordinator ______ Date ______

FC No. ______ FCTs ______

ACCEPTANCE — PLANT STAFF

	By'	Date		By	Date
Received work complete			Staff review complete		

Plant Manager Final Approval ______ Date ______

Transmitted By ______ Date ______

ENGINEERING CLOSE OUT

All design documents issued for Operation: ______ ______

Project Engineer Date

RMS Purged ______ ______ RMS Indexed ______ ______

By Date By Date

NOTE: DCP revised to add painting instructions & checklist and to delete unnecessary label 955.

UNIT 2

DESIGN CHANGE PACKAGE FOLLOWER

DCP# J-38114 - R 0

ESTIMATE

COST & SCHEDULING/ENGINEERING

ESTIMATE

			By	Date
Engineering Start	10-6-86	Engineer	[illegible]	12-19-86
Construction Complete	by end of U2 outage	Approved (ECS)	[illegible]	12-20-86
Estimated Total Cost	$15,200.00	Accepted (PE)	[illegible]	12-26-86

Authorizing Job Estimate Number BL0602; JE194974

	By	Date		
Cost & Scheduling Spv.	[illegible]	12/24/86	☐	Not Required (level of effort)

DESIGN

DESIGN ENGINEERING

	By	Date		By	Date
Engineer	[illegible]	12/26/86	Mechanical Discipline (Safety Evaluation)	[illegible]	12/24/86
Approved (grp ldr)	[illegible]	12/26/86	Accepted (PE)	[illegible]	12/26/86

CONSTRUCTION

CONSTRUCTION PLANT STAFF REVIEW

		By	Date			By	Date
Received by EM				Imp to Safety or			
PSRC Approves (y/n)				Imp to Environ (y/n)			

Reason for Rejection ______

Plant Manager Approval ______ Date ______

Transmitted By ______ Date ______

CONSTRUCTION

		By	Date			By	Date
Received				Installation Complete			
Start Up Complete				As-Builts Attd (y/n)			

Released by Package Coordinator ______ Date ______

PC No. ______ FCTs ______

ACCEPTANCE

ACCEPTANCE PLANT STAFF

	By	Date		By	Date
Received work complete			Staff review complete		

Plant Manager Final Approval ______ Date ______

Transmitted By ______ Date ______

FOR INFORMATION ONLY

ENGINEERING CLOSE OUT

All design documents issued for Operation: ______ ______

Project Engineer Date

RMS Purged ______ ______ RMS Indexed ______ ______

By Date By Date

FOR INFORMATION ONLY.

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DCN NUMBER	CURRENT REVISION	ATTACHED	LATER*	PREVIOUSLY ISSUED
DCZ-EJ-38114	1	/X/	/_/	/_/
		/_/	/_/	/_/
		/_/	/_/	/_/
		/_/	/_/	/_/
		/_/	/_/	/_/
		/_/	/_/	/_/

SEE NEXT SHEET FOR ADDITIONAL DCNs /_/ Yes /X/ No

* DCNs identified to be provided under approved package separation

FC NUMBER /FCT NUMBER	DCN NUMBER

FOR INFORMATION ONLY

SEE NEXT SHEET FOR ADDITIONAL FCs/FCTs /_/ Yes /X/ No

PACIFIC GAS AND ELECTRIC COMPANY
DIABLO CANYON POWER PLANT UNIT NOS. 1 AND 2

DESIGN CHANGE SAFETY EVALUATION SUMMARY

SUBJECT: Apply panel demarcation and new hierarchical labeling to panel PAM-1.

1. DCP Number: J-38114 (UNIT 2)

2. Classification

	Yes	No
A. Does this change require a change to the Technical Specifications?	( )	(X)

If the above question is answered "Yes," a Licensing Amendment Report is required.

	Yes	No
B. Does this change require a change in the following documents:		
1. The SAR? (see definition, Section 2.2.12)	( )	(X)
2. Any Q or Class 1 items in the Q-List?	( )	(X)
3. The Environmental Qualification Report?	( )	(X)
C. Does this change affect:		
1. Security?	( )	(X)
2. Fire Protection?	( )	(X)
3. Emergency Planning?	( )	(X)
D. Is any of the affected equipment important to safety?	(X)	( )
E. Is radioactive material contained in the system?	( )	(X)
F. Is there a radioactive waste treatment change specified?		(X)
G. Based on the Design and Safety Review, does a potential unreviewed safety question exist?	( )	(X)

"FOR INFORMATION ONLY"

If any of the previous questions have been answered "Yes," complete Questions 1 through 3 of the attached Safety Evaluation. If all questions have been answered "No," attach a justification detailing why no Safety Evaluation is required and answer Question J "No."

	Yes	No
H. Is any of the affected equipment important to environmental quality?	( )	(X)

J-38114

	Yes	No
I. Does the proposed change have the potential to impact the environment?	( )	(X)

If either Questions H or I have been answered "Yes'" complete the attached Environmental Evaluation. If Questions H and I have been answered "No," answer Question K "No."

	Yes	No
J. Has this change been determined to constitute an unreviewed safety question?	( )	(X)

(Yes, if either Question 1, 2, or 3 of the attached Safety Evaluation is marked "Yes.")

K. Has this change been determined to constitute an unreviewed environmental question?	( )	(X)

(Yes, if Question B on the attached Environmental Evaluation is marked "Yes.")

3. References:

SAR 7.5

Performed by: SRParis per Date: 3-25-87

Reviewed by: Simon Li Date: 4/20/87

PSRC Review: Date:

"FOR INFORMATION ONLY"

J-38114

DESIGN CHANGE PACKAGE SAFETY EVALUATION

Prior to answering the following three questions from 10CFR50.59, present a description of the design change including the critical parameters, and how the functional requirements of the system, structure or component are satisfied:

This change provides panel demarcation of functional groups within the panel, and new hierarchical labeling of groups and devices on the panel. Labeling is of new color and style for consistency of readability and wording. No device relocations or modifications are included in this change, and also no wiring changes. All Unit 2 Class 1E wiring and separation requirements, and seismic analyses are unaffected by these changes. Label and demarcation materials are within fire protection requirements for panel use.

1. Is the possibility of an accident or malfunction of a different type than any evaluated previously in the SAR created?

Yes	No
( )	X

This change adds only panel demarcation and labeling. New labels will provide all information currently available to operators, but will now be in an easier-to-use hierarchical format. All device functions remain unchanged.

J-38114.

DESIGN CHANGE PACKAGE SAFETY EVALUATION

2. Is the margin of safety as defined in the basis for any Technical Specification reduced? Yes ( ) No (X)

No device changes such as control options, setpoints, alarms, ranges, status or availability are being changed. Work will be panel surface enhancements only. This change will enhance operator interface with the control board, as determined by the CRDR team. (See Sec. 2.3 on sh. 7 of DCN DC2-EA-38114.) 1

3. Is the probability of occurrence or the consequences of an accident or malfunction of equipment important to safety previously evaluated in the SAR increased? Yes ( ) No (X)

Panel demarcation and improved labeling will potentially enhance speed and accuracy of operator response to a plant transient or emergency, by enhancing device and system relationships of panel instrumentation. No changes to device or system function are included.

"FOR INFORMATION ONLY"

PG&E

PACIFIC GAS AND ELECTRIC CO.
ENGINEERING DEPARTMENT
3LI NO. 060
PRIORITY NO. 800

NUCLEAR POWER PLANT DESIGN CHANGE

DCPP
(PLANT)

DATE 3-25-87
DCN NO. DC2-EJ-38114
REV. NO. 1
SHEET 1 OF 26

To: K. Herman
From: JRParris

Structure or System: Main Control Room panel PAM 1 Sys No. Various
Component: control panel face
Description of Change: Apply panel demarcation and new hierarchical labeling to panel PAM-1.

REQUEST

Reason for Change: Demarcation and hierarchical labeling will enhance device functional grouping and identification of devices on the panel.
Schedule/Justification: by end of U2 outage/CRDR commitment

Construction Status (for revised DCNs): n/a ☐ Not Started ☐ Partially Complete ☐ Completed
List of Attachments Not on DDL: SK-DC2-EJ-38114-1; SK-DC2-EJ-38114-2; SK-DC2-EJ-38114-3, sheets 1 to 5; SK-DC2-EJ-38114-4, sh. 1 to 3.
Requested by: JRParris

REVIEW AND APPROVAL

Requested Change is:

- ☐ Approved per delegation of authority
- ☒ Approved
- ☐ Noted, document change not required
- ☐ Rejected (explain)
- ☐ Per telecon with ______ on ______ (If required by delegation of authority)
- ☒ As-built documents required
- ☐ Approved, document change only

INFORMATION ONLY

[PANELS ARE SAFETY RELATED]

Safety-Related Work: ☒ Yes ☐ No
DCN required to close an NCR ☐ Yes ☒ No
Important to Environmental Quality: ☐ Yes ☒ No
NCR No. —
Not Safety-Related, requires Quality Assurance: ☐ Yes ☒ No

Reviewed By: JRParris 3-25-87
Discipline Engineer Date

Checked By: DRangum 4/09/87 Date
Frank Mori 4/10/87
Group Supervisor Date

COMPLETION

	By	Date		By	Date
Installation Complete			Received Engineering		
Start-up Complete			Accepted Engineering		
Accepted NPG			Revisions Approved		
FCs Issued:					

DISTR. MNN, JS, PB, KH

7 - 12/15/86

TECHNICAL REVIEW

DCN No. DCI-EJ-38114
Rev. No. 1
Page 2 of 26

1a. DESIGN DOCUMENT REVIEW. The following documents are relevant to this change and have been originated, reviewed, or require revision as indicated:

Document, No., Rev.	Originated (Yes/No)	Reviewed (Yes/No)	Requires Revision (Yes/No)
o Q-List*		YES	NO
o Design Criteria Memorandums	NO	YES	NO
o Calculations	NO	YES	NO
o Design Verification Reports	NO	YES	NO
o Design Change Notices	NO	YES	NO

* Revisions to the Q-List are to be transmitted to the Mechanical Engineering EGS.

1b. DESIGN SAFETY REVIEW. The following is a list (not all inclusive) of design and safety issues to be considered. Indicate by "Yes" or "No" whether or not each issue affects or is affected by the change. Unless the reason for a "No" answer is obvious, further explanation is required. If "Yes," explain why or how the issue is relevant and how it is resolved. If "No," explain why the issue is not relevant.

FOR INFORMATION ONLY

	Relevant Issue? (Yes/No)	Comments (Use additional sheets as necessary)
o Accident Analysis (FSAR Chapters 6 and 15)	YES	changes are to panel surface only, not to device or system function (within control room)
o ALARA	NO	(↓ ↓)
o Shielding/Radiation Zones	NO	(↓ ↓)
o Environmental Quality	NO	(↓ ↓)
o Fire Protection	YES	See SHT. 8 this DCN ~~Materials will meet fire protection requirements for the control room~~
o Unacceptable Components	NO	(no component changes)
o Codes and Standards	NO	(↓ ↓)

	Relevant Issue? (Yes/No)	Comments
• Simulator	YES	See DCO-EJ-39115
• System Interaction	NO	(no component changes)
• Regulatory Guides	YES	design meets reqts. of RG 1.97
• Environmental Qualifications	NO	(within control room)
• General Design Criteria	YES	meets requirements of 10 CFR 50 App. A & FSAR 3.1
• Seismic Qualification	YES ~~NO~~	SEE SHEET 8 of this DCN ~~(panel surface only)~~
• Water Hammer	NO	(within control room)
• Inservice Inspection	NO	( )
• Heavy Loads	NO	( )
• Flooding	NO	( )
• Radioactive Piping	NO	( )
• High-Medium Energy Line Break	NO	( )
• Control Room Design Review (Including habitability)	YES	See p. 7 of this DCN
• Multi-Unit Impact	NO	(No component changes)
• Aluminum Inside Containment	NO	(within control room)
• Security	NO	(no component changes)
• NPRDS	NO	( )
• Personnel Safety	YES	materials used consider fire safety in control room
• Masonry Block Walls	NO	(within control room)
• Core Drilling Impact	NO	( )
• Redundancy/Separation Requirements	YES	panel surface only, channel labels provided where needed
• Penetration Sealing	YES ~~NO~~	SEE SHT. 9 painting note ~~(within control room)~~
• Paint Inside Containment	NO	(within control room)
• Refueling Operations	NO	(does not require outage)
• Material Compatibility	NO	(no component changes)

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	Relevant Issue? (Yes/No)	Comments
o Vital Bus Loading	NO	(no component changes)
o Maintainability/Accessibility	NO	( )
o Floor or Wall Loading	NO	( )
o Missiles	NO	(within control room)
o Operability	YES	See p. 7 of this DCN
o Electrical Design Considerations (see attached sheet for discussion)	NO	(no component changes, panel surface only)
o Hydraulic Design Criteria (see attached sheet for discussion)	NO	(no component changes)
o Chemistry Effects (see attached sheet for discussion)	YES ~~NO~~	ADDITION OF NEW MAT'L HAS NO ADDITIONAL EFFECT OTHER THAN THAT ALREADY ANALYZED. ~~(no component changes, control room only)~~
o I&C Design Considerations (see attached sheet for discussion)	YES	See p. 7 of this DCN
o HVAC Design Considerations (see attached sheet for discussion)	NO	(no component changes, panel surface only)
o For nonsafety-related modifications, discuss why the design resulting from the modifications will not affect (a) any safety-related structures, systems or components, and (b) items identified in Paragraph 4.4.4(e) of Procedure 3.6 ON.	NO	(panels are safety-related)
o Other	NONE	—
o FMEA Evaluation	NO	(no component changes or system changes)

2. LICENSING. The following NRC Licensing submittals are relevant to this change and have been reviewed as indicated below. Where a revision is required, Licensing has been notified.

Document	Reviewed	Requires Revision
a. FSAR *	YES	NO
b. Technical Specification	YES	NO
c. Other	—	—

* Revisions to the FSAR require an FSAR Change Notice to be transmitted to NRA.

3. COORDINATION
Design Package Lead Discipline I&C

Coordination Required: [X] No [ ] Yes Coordinated With:

Department	Engineer (Signature)	Date	DCN Required	
(none)			Yes ( )	No ( )
			Yes ( )	No ( )
			Yes ( )	No ( )
			Yes ( )	No ( )
			Yes ( )	No ( )
			Yes ( )	No ( )
			Yes ( )	No ( )

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PG&E

PACIFIC GAS AND ELECTRIC COMPANY
ENGINEERING DEPARTMENT
DESIGN DOCUMENTS LIST
DOCUMENTS AFFECTED BY DESIGN CHANGE NOTICE (DCN)

PLANT DCPP 2 DCN NUMBER DC2-EJ-38114 REVISION 1 DATE 12-19-86 ENGINEER JRParris

DOCUMENT NUMBER	SHEET NO.	REVISION			DOCUMENT TITLE	DATE COMPLETED	BY	DATE APPROVED	BY
		CURRENT	INTERIM	AS BUILT					
697191	9	*			POST ACCIDENT MONITORING PANEL 1 (PAM-1)				
					* Design/Drafting to determine				

INFORMATION ONLY

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DESIGN CHANGE SAFETY EVALUATION

1.0 Introduction

A safety review of this DCN based on Procedure 3.6 ON, Attachment "E", indicates that ~~three (3)~~ FIVE (5) issues require further analysis to determine the degree of safety impact to the control room panels. The issues to be analyzed are listed below:

Issue A: Control Room Design Review

Issue B: Operability

Issue C: Instrumentation and Controls Design Considerations
ISSUE D: SEISMIC QUALIFICATION
ISSUE E: FIRE PROTECTION

2.0 Safety Evaluation of Issues

2.1 Issue A: Control Room Design Review

The present design change is a CRDR Team recommendation to improve the usability of the PAMS instrumentation. This design change will provide panel demarcation lines and new hierarchical labeling for the panel face to enhance the previous functional grouping of the panels. The proposed changes will enhance plant operation by reducing operator response time and potential for error.

2.2 Issue B: Operability

Implementation of the proposed changes will not negatively impact plant operability. The intended demarcation and labeling affect the panel surface only, and do not modify any devices, themselves. No additions or deletions in plant parameter information is proposed, nor is any deletion in available label information of use to the operators. No special or additional training is required for operators or maintenance personnel, and compliance with Reg. Guide 1.97 has not been affected.

2.3 Issue C: Instrumentation & Controls Design Considerations

As the proposed changes do not include any change in instrument ranges, size, scales, inputs, selection devices or other features, all I & C design considerations can be considered to have been met. The CRDR Team evaluation has examined the human factors aspects of this design change and has found no guidelines or good practices to be violated.

Analysis by: J R PARRIS Date: 12-19-86

Signature: J R Parris

Approval by: FRANK MORI Date: 4/10/87

Signature: Frank Mori

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2.4 ISSUE D: SEISMIC QUALIFICATION

THIS DESIGN CHANGE INVOLVES THE REPLACEMENT OF EXISTING PANEL LABELS WITH NEW LABELS (GRAVOPLY 1/16" MATERIAL) AND THE ADDITION OF TAPED DEMARKATION LINES MADE FROM COLORED TAPE. THE ADDITION OF THESE MATERIALS TO THE PANEL SURFACE IS CONSIDERED TO HAVE NO IMPACT TO THE MASS OF THE PANEL, AS MOST OF THE LABELS ARE A REPLACEMENT OF PREVIOUS MATERIAL. THIS CHANGE IS CONSIDERED TO NOT IMPACT THE PANELS PREVIOUS SEISMIC INTEGRITY, AS ANALYZED BY WESTINGHOUSE.

2.5 ISSUE E: FIRE PROTECTION

THE DESIGN CHANGE INVOLVES THE REPLACEMENT OF EXISTING PANEL LABELS WITH NEW LABELS (GRAVOPLY 1/16" MATERIAL) AND THE ADDITION OF DEMARKATION LINES MADE FROM COLORED TAPE. THE LABEL MATERIAL IS OF SIMILAR COMPOSITION TO THAT ORIGINALLY FURNISHED BY WESTINGHOUSE. THE NOMINAL AMOUNT OF COMBUSTIBLE MATERIAL PROVIDED BY THE STRIPING TAPE IS CONSIDERED INSIGNIFICANT TO IMPACT THE EXISTING FIRE PROTECTION REQUIREMENTS FOR THE MAIN CONTROL ROOM.

PAINT REQUIREMENTS FOR CONTROL ROOM BOARDS/CONSOLES
UNITS 1 + 2 AND SIMULATOR

As a part of the Control Room Design Review (CRDR) the control boards are undergoing a total relabeling of devices. The new labels will be hierarchical, size-graduated and will generally identify devices from above rather than below as is presently the case. As a result, in Units 1 + 2, many screw holes from the present labels will be left exposed (i.e. not covered by the new label). Screws were not used to adhere labels on the simulator panels but removal may leave flaws in the panel surface. Demarcation lines will also be added to functionally group related devices. The use of the lines of demarcation requires that the existing black borders around control switches and other modules be muted.

Field work to fill the exposed screw holes, finish panel surfaces and paint control switch borders shall proceed as follows:

(NOTE: Unit 1 + 2 control room control panel PAM-1 is a Safety Related Seismic Category I structure containing Class IE devices, so modifications effected in this board shall meet the requirements for 'Q' equipment.)

1

Surface Preparation and Finishing

1. Clean panel with safety degreaser (fire retardant).
2. Fill screw holes as necessary, from removed labels, etc. with filler, allow to dry and sand smooth. Ensure that all equipment in the panel is protected from any filler material.
3. Spray one coat of the required primer on external surface of panel and/or control switch border. Allow to dry and sand with 320 and 400 grit sandpaper. Refill any remaining surface defects with glazing putty and resand as required.
4. Spray finish coat(s) on panel surface and control switch borders to match existing surface of panel. Finish coats to be of uniform thickness and free from sags, runs and smears.
5. Paint to be # 121 arbor green light pastel epicote enamel for the panels (National Lead # 45M27). Paint shall have a minimum total dry film thickness of 3 mils.
6. Complete checklist 'A'.

CHECKLIST 'A': PAINTING

PANEL NO.: PAM-1

	OK	N/A	*
o Exterior primer coat(s) have been properly applied.	( )	( )	( )
o Finish coat(s) are of the correct material, properly applied and treated.	( )	( )	( )
o All exterior surfaces are free from sags, runs or smears and finish is uniform.	( )	( )	( )
o Paint coating thickness is as specified.	( )	( )	( )

1

* COMMENTS, CORRECTIONS, ETC.:

FOR CHANGES TO CLOUDED AREA, SEE SKETCH SK-DC2-EJ-38114-1 & -2.

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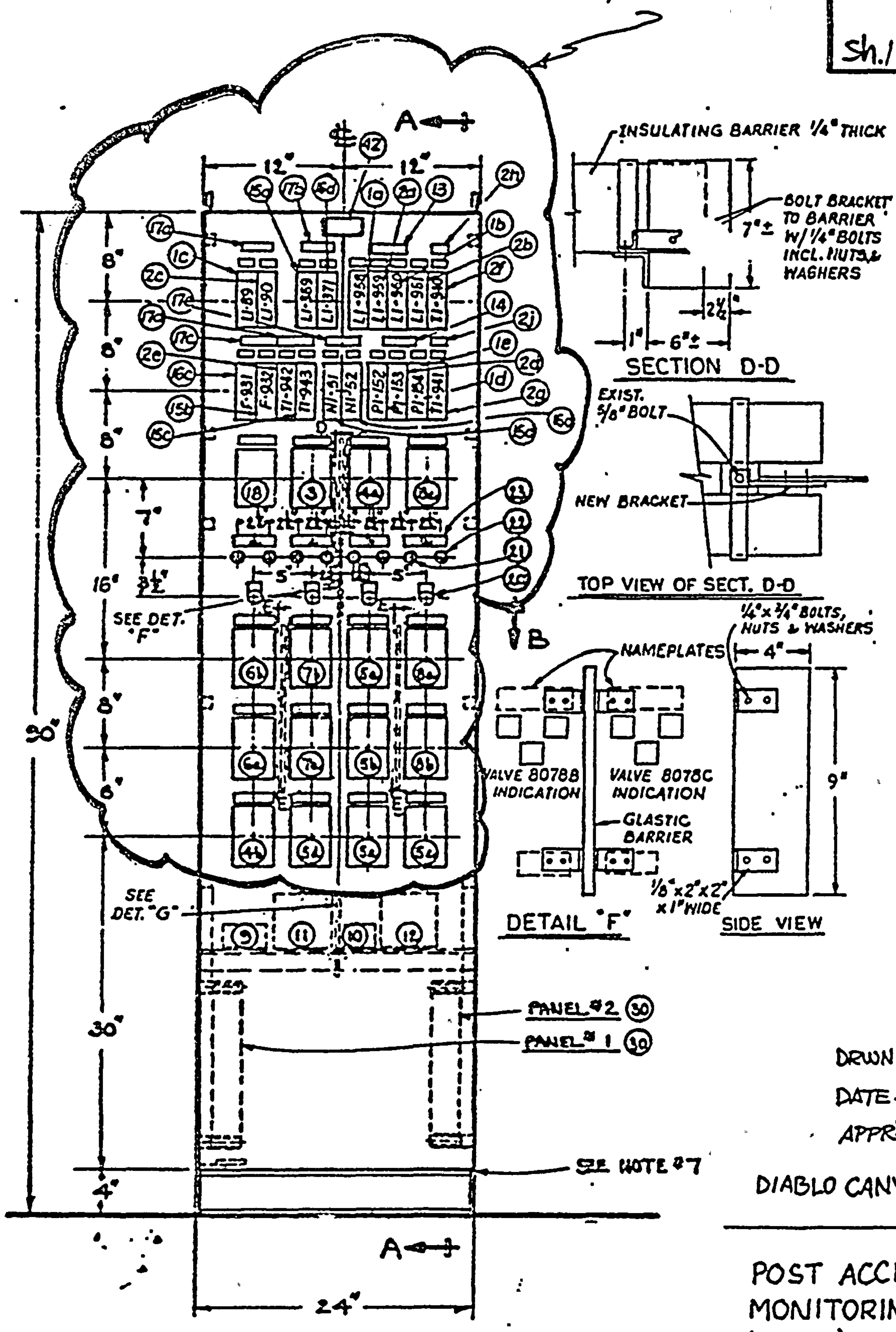

REAR VIEW

DRWN JRParrish
DATE 12-19-86
APPR Frank Mori 4/10/87

DIABLO CANYON

POST ACCIDENT MONITORING PANEL 1 (PAM-1)

DC 697191-9

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FOR CHANGES IN NAMEPLATE INFORMATION IN ITEMS 1 TO 23, SEE SKETCH SK-DC2-EJ-38114-3

BILL OF MATERIALS

ITEM	QTY	DESCRIPTION
		MATERIALS PROVIDED BY P. G. & E. EXCEPT NAMEPLATES, WIRED BY TRAYER
1	4	WESTINGHOUSE MODEL VX252 INDICATOR, POWERED FROM CHANNEL 1 AND INSTRUMENT LOOPS - ACOPIAN POWER SUPPLY UG775, WITH ENGRAVED WHITE LETTERS ON BLACK NAMEPLATE 1-1/2" X 3/4" FORMICA NO. 840 LETTER SIZE 7, READS: a. "2-1" LI-938 b. "2-3" LI-940 d. "2-1" PI-932 e. "2-3" PI-934
	1	WESTINGHOUSE MODEL VX252 INDICATOR, POWERED FROM CHANNEL 1 AND INSTRUMENT LOOPS - ACOPIAN POWER SUPPLY UG775, WITH ENGRAVED WHITE LETTERS ON BLACK NAMEPLATE 1-1/2" X 3/4" FORMICA NO. 840 LETTER SIZE 7 READS: e. "RHR PUMP RM. SUMP LEVEL" LI-89
2	5	WESTINGHOUSE MODEL VX252 INDICATOR, POWERED FROM CHANNEL 2 AND INSTRUMENT LOOPS - ACOPIAN POWER SUPPLY UG776, WITH ENGRAVED WHITE LETTERS ON BLACK NAMEPLATE 1-1/2" X 3/4" FORMICA NO. 840 LETTER SIZE 7, READS: a. "2-2" LI-939 b. "2-4" LI-941 d. "2-2" PI-933 h. CONT TEMP j. CONT TEMP
	4	WESTINGHOUSE MODEL VX252 INDICATOR POWERED FROM CHANNEL 2 AND INSTRUMENT LOOPS - ACOPIAN POWER SUPPLY UG776, WITH ENGRAVED WHITE LETTERS ON BLACK NAMEPLATE 1-1/2" X 3/4" FORMICA NO. 840 LETTERS SIZE 7, READS: d. "RHR PUMP RM. SUMP LEVEL" LI-90 e. "CONTAINMENT WTR. TEMPERATURE" TI-942 f. "CONTAINMENT AIR TEMPERATURE" TI-940 g. "CONTAINMENT AIR TEMPERATURE" TI-941
3	1	LEEDS & NORTHRUP MODEL 430 2 PEN RECORDER, POWERED FROM CHANNEL 1 WITH ENGRAVED WHITE LETTERS ON BLACK NAMEPLATE 3-1/2" X 3/4" FORMICA NO. 840 LETTER SIZE 7, READS: "REACTOR VESSEL AND PLENUM LEVEL" LR-199, LR-200, LR-201
4	2	LEEDS & NORTHRUP MODEL 430 2 PEN RECORDER, POWERED FROM CHANNEL 2 WITH ENGRAVED WHITE LETTERS ON BLACK NAMEPLATE 3-1/2" X 3/4" FORMICA NO. 840 LETTER SIZE 7, READS: a. "REACTOR VESSEL AND PLENUM LEVEL" LR-202, LR-203 b. ALARA RLD RR-34 ALUE RR-35
5	3	LEEDS & NORTHRUP MODEL 430 1 PEN RECORDER, POWERED FROM CHANNEL 1 WITH ENGRAVED WHITE LETTERS ON BLACK NAMEPLATE 3-1/2" X 3/4 FORMICA NO. 840 LETTER SIZE 7, READS: a. "CONTAINMENT H_2" ANR-82 b. "CONTAINMENT GROSS ACTIVITY" RR-30 c. "PLANT VENT GROSS ACTIVITY" RR-29 d. "PLANT VENT IODINE" RR-32 e. "PLANT VENT NOBLE GAS" RR-33
6	2	LEEDS & NORTHRUP MODEL 430 1 PEN RECORDER, POWERED FROM CHANNEL 1 AND INSTRUMENT LOOPS - ACOPIAN POWER SUPPLY UG775 WITH ENGRAVED WHITE LETTERS ON BLACK NAMEPLATE 3-1/2" X 3/4" FORMICA NO. 840 LETTER SIZE 7, READS: a. "CONTAINMENT W/R LEVEL" LR-942 b. "CONTAINMENT W/R PRESS." PR-939
7	2	LEEDS & NORTHRUP MODEL 430 1 PEN RECORDER, POWERED FROM CHANNEL 2 AND INSTRUMENT LOOPS - ACOPIAN POWER SUPPLY UG776 WITH ENGRAVED WHITE LETTERS ON BLACK NAMEPLATE 3-1/2" X 3/4" FORMICA NO. 840 LETTER SIZE 7, READS: a. "CONTAINMENT W/R LEVEL" LR-943 b. "CONTAINMENT W/R PRESS." PR-935
8	3	LEEDS & NORTHRUP MODEL 430 1 PEN RECORDER, POWERED FROM CHANNEL 2 WITH ENGRAVED WHITE LETTERS ON BLACK NAMEPLATE 3-1/2" X 3/4" FORMICA NO. 840 LETTER SIZE 7, READS: a. "CONTAINMENT H_2" ANR-83 b. "CONTAINMENT GROSS ACTIVITY" RR-31 c. "DEGREE OF SUBCOOLING" YR-31
9	1	ACOPIAN MODEL YA40HT75 POWER SUPPLY, INSTRUMENT LOOP POWER UG775 (CHANNEL 1)
10	1	ACOPIAN MODEL YA40HT75 POWER SUPPLY, INSTRUMENT LOOP POWER UG776 (CHANNEL 2)
11	1	LEEDS & NORTHRUP MODEL 099260 POWER SUPPLY, RECORDER POWER CHANNEL 1
12	1	LEEDS & NORTHRUP MODEL 099260 POWER SUPPLY, RECORDER POWER CHANNEL 2
13	1	ENGRAVED WHITE LETTERS ON BLACK NAMEPLATE 3-1/2" X 3/4" FORMICA NO. 840 LETTER SIZE 7, READS: "ACCUMULATOR TK. W/R LEVEL"
14	1	ENGRAVED WHITE LETTERS ON BLACK NAMEPLATE 3-1/2" X 3/4" FORMICA NO. 840 LETTER SIZE 7, READS: "GAS DECAY TANK PRESSURE"
		MATERIALS PROVIDED & WIRED BY PG&E
15	4	WESTINGHOUSE MODEL VX-252 INDICATOR, POWERED FROM CHANNEL 1 UG775, WITH ENGRAVED WHITE LETTERS ON BLACK NAMEPLATE 1-1/2" X 3/4" FORMICA NO. 840 LETTER SIZE 7, READS: a. LI-369 b. FI-932 c. TI-943 d. RI-31
16	4	WESTINGHOUSE MODEL VX252 INDICATOR, POWERED FROM CHANNEL 2 UG776 WITH ENGRAVED WHITE LETTERS ON BLACK NAMEPLATE 1-1/2" X 3/4" FORMICA NO. 840 LETTER SIZE 7, READS: a. LI-371 b. c. FI-931 d. RI-32
17	5	ENGRAVED WITH WHITE LETTERS ON BLACK NAMEPLATE 3-1/2" X 3/4" FORMICA NO. 840 LETTERS SIZE 7, READS: a. RHR SUMP LEVEL b. LIQUID HOLD-UP TANK LEVEL c. CNT. SPRAY PUMP FLOW d. CONTAINMENT REACTOR CAVITY SUMP TEMP e. WIDE RANGE NEUTRON FLUX
18	1	LEEDS & NORTHRUP MODEL 430 3 PEN RECORDER, WITH ENGRAVED WHITE LETTERS ON BLACK NAMEPLATE 3-1/2" X 3/4" FORMICA NO. 840, LETTER SIZE 7, READS: CNT/REACTOR SUMP 1LM60/1LM61/1LM62
19		
		MATERIALS PROVIDED & WIRED BY TRAYER
20	4	CUTLER HAMMER CAT. #10250T1311-1N36 SELECTOR SWITCH, TYPE T, OIL TIGHT, 2 POSITION MAINTAINED WITH 1 N.O. AND 1 N.C. CONTACT BLOCK #10250T-1 COMPLETE WITH LEGEND PLATE MARKED: "CLOSE-OPEN"
21	4	CUTLER HAMMER CAT. #10250T201C8, LAMP INDICATING, TYPE T, OIL TIGHT, 120V, COMPLETE WITH RESISTOR AND GREEN LENS
22	4	CUTLER HAMMER CAT. #10250T201C7, LAMP INDICATING, TYPE T, OIL TIGHT, 120V, COMPLETE WITH RESISTOR AND RED LENS
23	4	ENGRAVED WITH WHITE LETTERS ON BLACK NAMEPLATE 3-1/2" X 3/4" FORMICA NO. 840 LETTER SIZE 7, READS: a. "REACTOR VESSEL VENT VALVE 8078A" b. "REACTOR VESSEL VENT VALVE 8078B" c. "REACTOR VESSEL VENT VALVE 8078C" d. "REACTOR VESSEL VENT VALVE 8078D"
24	60	BUSS CAT. NO. HLD. FUSE HOLDER FOR 1/4" X 1-1/4" INDICATING FUSES, BAYONET TYPE TRANSPARENT KNOB 15A., 250 VOLT
25	11	BUSS CAT. NO. GLD. FUSE, NON-TIME DELAY, 5A., 125 VOLT
26	21	BUSS CAT. NO. GLD. FUSE, NON-TIME DELAY, 3/4A., 125 VOLT
27	27	FOR FUSE PANEL AT ENGRAVED WITH WHITE LETTERS ON BLACK NAMEPLATE 3-1/2" X 3/4" FORMICA NO. 840 LETTERS SIZE 7, READS: a. "LR-203" LR-201 b. "PR-939" c. "ANR-82" d. "LR-942" e. "RR-30" f. "RR-29" g. "RR-32"

APPR Frank Mai 4/10/87

DATE 12-19-86

DRWN JRParris

POST ACCIDENT MONITORING PANEL 1 (PAM 1)
BILL OF MATERIALS 697191-10 p.1 of 2

FOR CHANGES IN NAMEPLATE INFORMATION IN ITEM 42, SEE SKETCH SK-DC2-EJ-38114-3

DC2-EJ-38114
R1
Sh 13 of 26

p. 2

BILL OF MATERIALS		
ITEM	QTY	DESCRIPTION
		h. "RR-33"
		j.
		k.
		l.
		m.
		n. "LR-942"
		p. "PR-939"
		q. "PI-152"
		r. "PI-154"
		s. "LI-958"
		t. "LI-960"
		u. "LI-89"
		v.
		w.
		x.
		y.
		z.
		aa) "REACTOR VESSEL VENT VALVE 8078C"
		bb) "REACTOR VESSEL VENT VALVE 8078D"
		cc)
28		
29	27	FOR FUSE PANEL #2, WITH ENGRAVED WHITE LETTERS ON BLACK NAMEPLATE 3-1/2" X 3/4" FORMICA NO. 840 LETTER SIZE 7, READS:
		a. "LR-202 LR-203"
		b. "PR-950"
		c. "AMR-83"
		d. "LR-943"
		e. "RR-31"
		f. "RR-34 RR-35"
		g.
		h.
		j. "YR-31"
		k.
		l.
		m.
		n. "LR-945"
		p. "PR-958"
		q. "PI-153"
		r.
		s. "LI-959"
		t. "LI-961"
		u. "LI-90"
		v.
		w.
		x.
		y.
		z.
		aa) "REACTOR VESSEL VENT VALVE 8078A"

BILL OF MATERIALS		
ITEM	QTY	DESCRIPTION
		bb) "REACTOR VESSEL VENT VALVE 8078B"
		cc)
30	2	23-1/8" W. X 15" H. X 2" 11 GA. MILD STEEL PAN FOR FUSE PANEL #1 AND FUSE PANEL #2 WITH 4 SPACERS
31	1	11 GA. MILD STEEL ENCLOSURE WITH 1/4" THICK 23-3/4" W. X 86 H. MILD STEEL REAR STEEL PLATE
32	1	5.4# -- 4" CHANNEL BASE
33	1	PADLOCKABLE 3-POINT LATCHING DOOR HANDLE
34	1	T.E.C. FOOT OPERATED DOOR STOP
35	1	BRAZE BONDED GROUND LUG
36	2	1/2" REMOVABLE EYEBOLT
37	4	1/2" X 5" X 5" FLAT BAR WELDED TO BASE FOR ANCHORING
38	YES	NEOPRENE AIR CUSHION DOOR GASKET ALL AROUND DOOR
39	1	1/4" X 2" X 78" FLAT BAR WELDED ON DOOR FOR REINFORCEMENT
40	3	TRAYER HINGE WITH 1/4" STAINLESS STEEL PIN
41	1	T.E.C. STAINLESS STEEL NAMEPLATE
42	1	ENGRAVED LAMICOID NAMEPLATE BLACK WITH WHITE LETTERS, READS: "PAM 1"
43	20	12-POINT TERMINAL BLOCK GENERAL ELECTRIC CAT. #CR151B2
44	14	1" X 1" X 35-3/8" MILD STEEL SQUARE BAR
45	2	1/4" X 2" X 2" X 23-3/4" ANGLE
46	1	1/4" X 2" X 3" X 23-3/4" ANGLE
47	2	11 GA. MILD STEEL 1" D. X 14" W. X 14-1/2" H. PAN WITH 4 SPACERS
48	2	11 GA. MILD STEEL 1" D. X 7" W. X 43" H. PAN WITH 6 SPACERS
49	2	11 GA. MILD STEEL 1" D. X 7" W. X 14-1/2" H. PAN WITH 4 SPACERS
50	2	1/2" X 2" MILD STEEL BAR 85-5/8" LONG WELDED TO CABINET 2" IN EVERY 6"
51	6	1/2" X 2" X 4" MILD STEEL BAR CONTINUOUSLY WELDED TO ITEM #50) FOR SUPPORT OR REMOVABLE CHANNEL
52	3	5.4# - 4" REMOVABLE CHANNEL 17-1/2" LONG MOUNTED WITH (4) 1/2" - 13NC GRADE 5 BOLT
53	1	1/4" X 1" X 23-3/4" LONG FLAT BAR
54	2	4 - 1/2" X 1/2" X 2" SPACER BAR WELDED TO 1/4" X 2" X 2" X 23-3/4" ANGLE FOR SUPPORT OF INDICATORS
55	1	1/4" X 1-1/4" X 23-3/4" FLAT BAR
56	3	1/4" X 4-3/4" X 23-3/4" FLAT BAR
57	4	1" X 1" X 8-1/2" MILD STEEL SQUARE BAR WELDED TO TOP AND BOTTOM SQUARE BAR AND TO BACK OF ANGLE
		MATERIALS PROVIDED & WIRED BY P.G.&E.
58		
59		LEEDS & NORTHRUP MODEL AA SYNCHRONIZING TRANSFORMER
60		
61		

BILL OF MATERIALS		
ITEM	QTY	DESCRIPTION
62		
63		
64		
65		
66		
67		
68		
69		
70		12-POINT TERMINAL BLOCK "T.B.-KA"
71		CWF 1515 MINUTE ALARM LAMP
72		3/4" R.I. NIPPLE

NOTE: THIS DRAWING TO ACCOMPANY PAM 1-UNIT 2 ARRANGEMENT DRAWING # 697191 SH. 9

BILL OF MATERIAL FOR
PAM 1 - POST ACCIDENT MONITORING
PANEL 1 - P.G.&E.
DIABLO CANYON - UNIT 2
(SH 2 OF 2)

2

[illegible]

2	[illegible]	[illegible]					
NO.	DATE	DESCRIPTION	BY				
P.G.&E. REVISION							
[illegible]	1	697191	10	2			

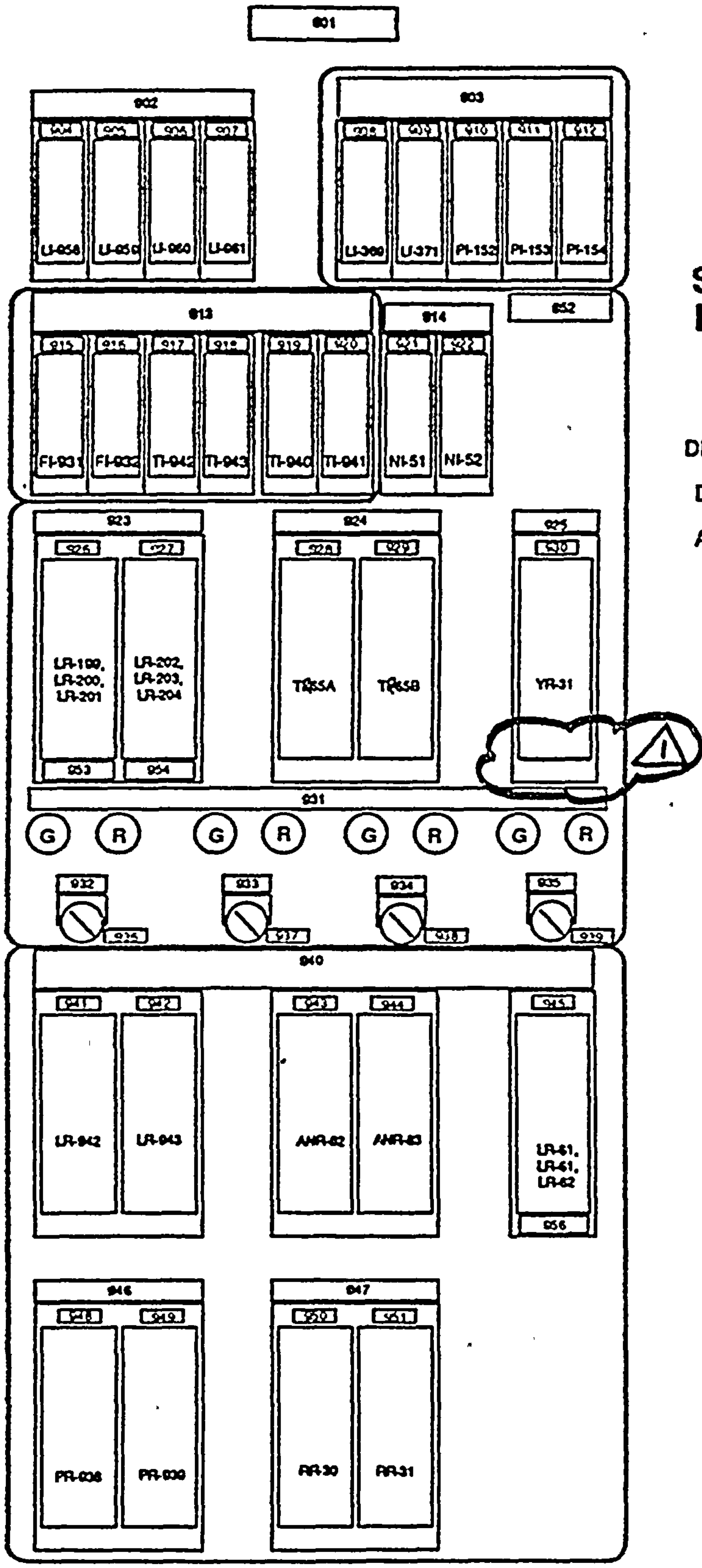
POST ACCIDENT MONITORING PANEL, PAM 1
BILL OF MATERIALS
697191-10 p. 2/2

DRWN J.R. Farris
DATE 12-19-86
APPR [illegible] 4/10/87

DC2-EJ-38114

R 1

SH. 14 OF 26 .



SK-DC2-EJ-38114-2 SH. 1 OF 1
LABELING LAYOUT
PAM-1

DRWN J R Parris

DATE 12-19-86

APPR F. Mori 4/10/87

DC2-EJ-38114

R 1

SH 15 OF 26.

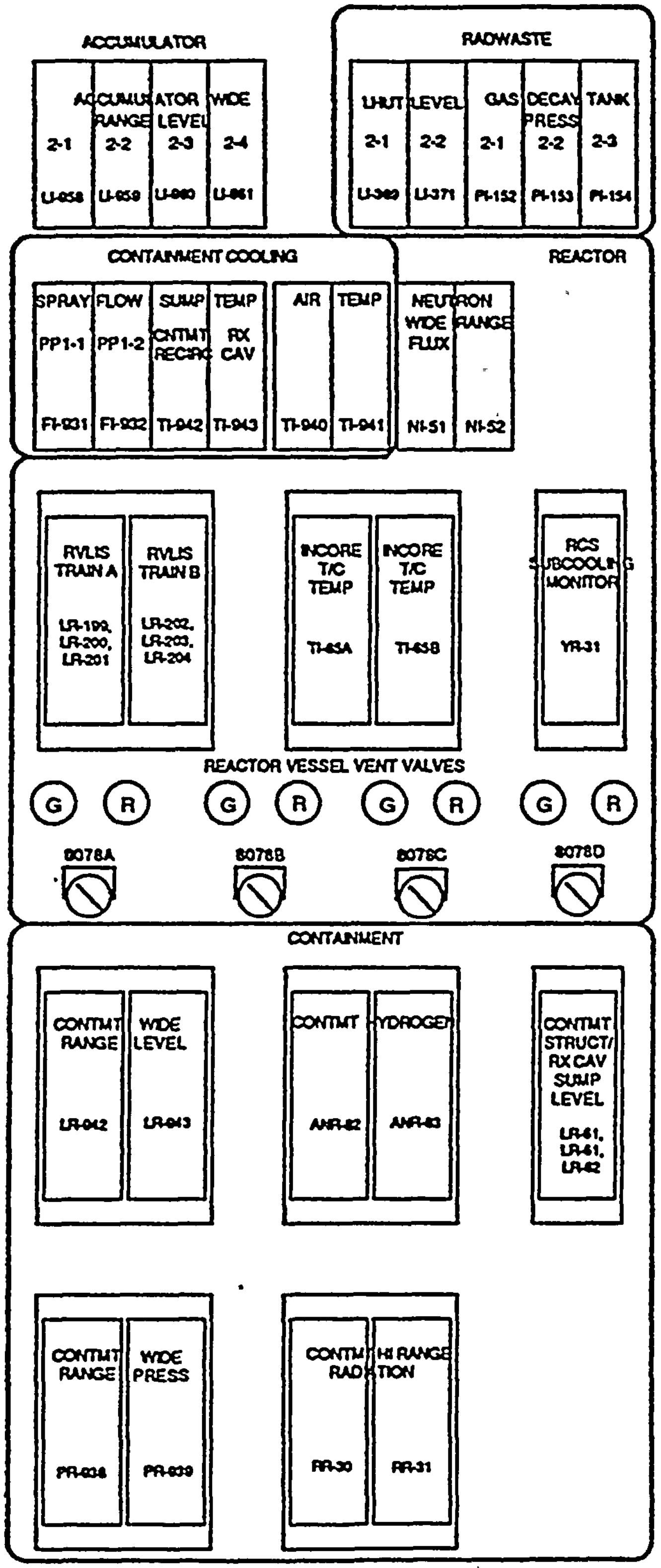
SK-DC2-EJ-38114-1 SH. 1 OF 1
DEMARCATION LAYOUT, PAM-1

DRWN J.R. Parris

DATE 12-19-86

APPR F. Moi 4/10/87

PAM 1



SK-DC2-EJ-38114-3, Sh. 1 of 5

BILL OF MATERIALS FOR HUMAN FACTORS LABELING CHANGES

PANEL PAM 1

ITEM NO.	QTY.	LINE	DESCRIPTION	LABEL SIZE	PRINT SIZE
901	1	1	UNIT 2 - PAM 1	A	1
902	1	1	ACCUM WR LVL	D	3
		2	2-1 2-2 2-3 2-4 (a)	-	5
903	1	1	RADWASTE (center over all 5 indicators) (b)	D	3
		2	(center over first 2) LHUT LVL (center over last 3) GAS DECAY TK PRESS (b)		4
		3	2-1 2-2 2-1 2-2 2-3 (b)		5
904	1	1	LI-958	H	8
905	1	1	LI-959	H	8
906	1	1	LI-960	H	8
907	1	1	LI-961	H	8
908	1	1	LI-369	H	8
909	1	1	LI-371	H	8
910	1	1	PI-152	H	8
911	1	1	PI-153	H	8
912	1	1	PI-154	H	8
913	1	1	CONTMT COOLING (b)	D	3
		2	SPRAY FLOW SUMP TEMP AIR TEMP (b)	-	4
		3	PP 2-1 PP 2-2 CONTMT RECIRC RX CAV (b)	-	5
914	1	1	NEUTRON	F	5
		2	WR FLUX	-	5

DRWN BY JRParris DATE 12-19-86 APPR BY J Moi 4/10/87 DCN NO. DC2-EJ-38114 Rev.1 SH.16 OF 26

(a) space to center over individual indicators or recorders in set
(b) see attached sketch for layout details

SK-DC2-EJ-38114-3, Sh. 2 of 5

BILL OF MATERIALS FOR HUMAN FACTORS LABELING CHANGES PANEL PAM 1

ITEM NO.	QTY.	LINE	DESCRIPTION	LABEL SIZE	PRINT SIZE
915	1	1	FI-931	H	8
916	1	1	FI-932	H	8
917	1	1	TI-942	H	8
918	1	1	TI-943	H	8
919	1	1	TI-940	H	8
920	1	1	TI-941	H	8
921	1	1	NI-51	H	8
922	1	1	NI-52	H	8
923	1	1	RVLIS	F	4
		2	TRAIN A TRAIN B (a)	–	5
924	1	1	INCORE T/C	F	4
		2	TRAIN A TRAIN B (a)	–	5
925	1	1	RCS SUBCOOL	F	4
		2	MONITOR	–	4
926	1	1	LR-199, 200, 201	L	8
927	1	1	LR-202, 203, 204	L	8
928	1	1	TR-65A	H	8
929	1	1	TR-65B	H	8
930	1	1	YR-31	H	8
931	1	1	RX VESSEL VENT VALVES	E	4
932	1	1	A	E	6
933	1	1	B	E	6

DRWN BY DATE APPR BY DCN NO. DC2-EJ-38114 Rev. 1 SH. 17 OF 26

(a) Space to center over individual indicators or recorders in set

SK-DC2-EI-38114-3, Sh. 3 of 5

BILL OF MATERIALS FOR HUMAN FACTORS LABELING CHANGES PANEL PAM 1

ITEM NO.	QTY.	LINE	DESCRIPTION	LABEL SIZE	PRINT SIZE
934	1	1	C	E	6
935	1	1	D	E	6
936	1	1	8078A	H	8
937	1	1	8078B	H	8
938	1	1	8078C	H	8
939	1	1	8078D	H	8
940	1	1	CONTAINMENT	D	3
		2	WR LVL HYDROGEN STRUCT/RX CAV	–	4
		3	SUMP LVL	–	5
		:		–	5
941	1	1	LR-942	H	8
942	1	1	LR-943	H	8
943	1	1	ANR-82	H	8
944	1	1	ANR-83	H	8
945	1	1	LR-60,61,62	L	8
946	1	1	WR PRESS	E	4
947	1	1	HI RANGE RAD	E	4
948	1	1	PR-938	H	8
949	1	1	PR-939	H	8
950	1	1	RR-30	H	8
951	1	1	RR-31	H	8
952	1	1	REACTOR	C	3

DRWN BY DATE APPR BY DCN NO. DC2-EI-38114 SH. 18 OF 26

Rev. 2

SK-DCZ-EJ-38114-3, sh 4 of 5

BILL OF MATERIALS FOR HUMAN FACTORS LABELING CHANGES PANEL PAM 1

ITEM NO.	QTY.	LINE	DESCRIPTION	LABEL SIZE	PRINT SIZE
953	1	1	RED — UPPER RANGE (LR-199)	E	9
		2	BLUE — FULL RANGE (LR-200)	—	9
		3	GREEN — DYNAMIC HEAD (LR-201)	—	9
954	1	1	RED — UPPER RANGE (LR-202	E	9
		2	BLUE — FULL RANGE (LR-203)	—	9
		3	GREEN — DYNAMIC HEAD (LR-204)	—	9
~~955~~	~~1~~	~~1~~	~~RED — TEMP (°F)~~	~~E~~	~~9~~
		~~2~~	~~BLUE — PRESSURE (PSI)~~ DELETE	~~—~~	~~9~~
956	1	1	RED — CONTMT SUMP LVL 2-1 (LR-60)	E	9
		2	BLUE — CONTMT SUMP LVL 2-2 (LR-61)	—	9
		3	GREEN — RX CAVITY SUMP LVL (LR-62)	—	9

DRWN BY DATE APPR BY DCN NO. DCZ-EJ-38114 SH. 19 OF 26

Rev. 1

RADWASTE
LHUT LVL GAS DECAY TK PRESS
2-1 2-2 2-1 2-2 2-3

DEVICE EDGES

CONTMT COOLING
SPRAY FLOW SUMP TEMP AIR TEMP
PP 2-1 PP 2-2 CONTMT RECIRC RX CAV

CONDENSE THIS DEVICE'S LABEL AS NEEDED TO FIT

LABEL EXAMPLES, HALF SIZE SK-DC2-EJ-38114-3, SH 5 OF 5

SK-DC2-EJ-38114-4
Sh. 1 of 3
LABEL CODE INFORMATION

LABEL SIZE (PLAQUE) IN ALPHA

PRINT SIZE (CHARACTER) IN NUMBERS

LABEL FOR	VERTICAL BOARD	CONTROL CONSOLE
A – Console ID	1.25 in. by (as req'd)	
B – Subpanel Segment	1.00 in. by (as req'd)	
C – Subgroup (freestanding)	.75 in. by (as req'd)	.50 in. by (as req'd)
D – Subgroup w/Subhead I	1.625 in. by (as req'd)	.625 in. by (as req'd)
w/Subhead I & II	1.625 in. by (as req'd)	N/A
w/2 lines Subhead II	1.625 in. by (as req'd)	N/A
E – Subhead I (freestanding)	.625 in. by (as req'd)	.375 in. by (as req'd)
F – Subhead I w/Subhead II	1.00 in. by (as req'd)	N/A
– Control Switch Description	.700 in. by 3.00 in.	.700 in. by 3.00 in.
H – Instrument Tag Number	.50 in. by 1.375 in.	.50 in. by 1.375 in.
K – Breaker ID	.375 in. by ________	.375 in. by ________
L – Subhead II (freestanding)	.50 in. by (as req'd)	
M –	.375 in. by (as req'd)	

LABEL/PRINT SIZE COMBINATIONS

A – #3 & #5 = 1.25 in. by (as req'd)
A – 2 of #4 = 1.25 in. by (as req'd)
D – 2 of #4 or #3 = 1.625 in. by (as req'd)
D – 3 of #4 = 1.625 in. by (as req'd)
D – 2 of #4 & 1 – #5 = 1.625 in. by (as req'd)
D – #3 & #4 = 1.625 in. by (as req'd)
D – #4 & 2 of #5 = 1.625 in. by (as req'd)
D – #3, #4, #5 = 1.625 in. by (as req'd)
F – #4 & #5 = 1.00 in. by (as req'd)

LABEL STOCK IS LIGHT GREEN WITH BLACK CORE, PRECUT SIZES.
(Mfg. Code no. 092586, .062 ga. 17438
Hermes code 610/310-221
mint green/black)

DC2-EJ-38114
R 1
Sh. 21 of 26

Table 7-3: Recommended label plaque dimensions.

DRWN JRParris pul DATE 12-19-86 APPR [illegible] 4/10/87

SK-DC2-EJ-38114-4

sh. 2 of 3

DC2-EJ-38114
R1
Sh. 22 of 26

LABEL CHARACTER DIMENSIONS

Labels should be constructed exclusively with capital letters.

Letter style shall be Helvetica (this is written in HELVETICA)

Characters and line spacing should observe the following dimensions:

Letter width-to-height ratio should be 3:5. Exceptions are for the numeral "4" which should be one stroke width wider, the numeral "1" and letter "I" which should be one stroke width, and the letters "W" and "M" which should be 3 stroke widths wider.

Stroke width-to-character height should be between 1:6 and 1:7 for light characters on a dark background, and between 1:7 and 1:8 for dark characters on a light background. Dark on light is preferred contrast scheme. See Table below for actual recommended stroke widths.

The minimum space between characters should be one stroke width.

The minimum space between words should be one character width.

The minimum space between lines of labeling should be one-third the character height. When lines of different character height are used on a single label, the minimum space between lines should be one-third the height of the tallest character size used.

Recommended stroke widths for letter heights

LETTER HEIGHT (IN.)	STROKE (CUTTER)	RATIO
1.00	.150	1/6.66
.75	.125	1/6.00
.47	.075	1/6.27
.38	.060	1/6.33
.30	.050	1/6.00
.25	.040	1/6.25
.19	.030	1/6.33
.13	.020	1/6.50

PRINTING FOR:	CHARACTER HEIGHT VERTICAL BOARD	 CONTROL CONSOLE
1. Console	1.00 in.	.47 in.
2. Subpanel Segment	.75 in.	.38 in.
3. Subgroup	.47 in.	.25 in.
4. Subgroup subhead I	.38 in.	.19 in.
5. Subgroup subhead II	.30 in.	---
6. Single line letter or number designation	.30 in.	.19 in.
7. Single or dual line word descriptors	.19 in.	.19 in.
8. Instrument tag numbers	.19 in.	.19 in.
9. Control options	.13 in.	.13 in.
10. Miscellaneous secondary information (breaker number, interlock, etc.)	.13 in.	.13 in.

Table 7-1: Minimum letter heights for vertical board and control console labels.

Incorporated B2, B3

HUMAN ENGINEERING DISCREPANCY

CHRON B1

LOC

UNIT 1 - 2 (CIRCLE WHICH)

ANALYST JLS/JJL

ITEM / LOCATION (S):

PAM 1 & 2

CHECKLIST CODE (S):

Functional grouping (Q.S, 8, 14) 6.8.1.1.b

DISCREPANCY (MEASUREMENTS AS APPROPRIATE):

PAM displays are not functionally grouped on panels. The containment air temp indicators are not grouped, nor side by side for easy comparison of readings.

FOR INFORMATION ONLY

PRIORITIZATION

PRELIMINARY SIGNIFICANCE ESTIMATE: 3

DC2-EJ-38114
R1
Sh. 24 of 26

PRELIMINARY BACKFIT FEASIBILITY ESTIMATE:

PRELIMINARY RECOMMENDATION:

Use demarcation, labeling and rearrangement to highlight correct functional grouping by safety functions.

HUMAN ENGINEERING DISCREPANCY CHRON 459 LOC B-P

UNIT 1 - 2 (CIRCLE WHICH) ANALYST

ITEM/LOCATION(S) PAM-1

CHECKLIST CODE(S) LMD-13

DISCREPANCY (MEASUREMENTS AS APPROPRIATE) No Demarcation used on panels.

FOR INFORMATION ONLY

DCZ-EJ-38114 R1 Sh. 25 of 26

PRIORITIZATION

PRELIMINARY SIGNIFICANCE ESTIMATE: MOD

PRELIMINARY BACKFIT FEASIBILITY ESTIMATE: MOD

PRELIMINARY RECOMMENDATION

See Chron # 14 & 29

C.1 D

DCZ-EJ-38114
R1
Sh.26 of 26

CHRON # 263

CATEGORY E.9a

HUMAN ENGINEERING DISCREPANCY ASSESSMENT

TITLE/SUBJECT Identification of Pens, Scales, Channels
(Complete description is on attached HED)

ASSESSMENT: 8-30-85

Reviewers

SAFETY RELATED:	YES	NO	
HIGH ERROR POTENTIAL:	YES	NO	
SIGNIFICANT SAFETY CONSEQUENCE:	YES	NO	N/A
OPERABILITY CONCERN:	YES	NO	N/A

PRIORITY: I

RATIONALE: None required

FOR INFORMATION ONLY

DISSENT: None

RECOMMENDED CORRECTION PLAN:

A. Immediate Compensatory Actions/Justification for Continued Operation:
~~Install temporary labels until final enhancements are implemented.~~ New labels were installed after the HED was recorded. The new label adequately identify the pens.
6/3/85

B. ~~Suggested~~ Correction:
Enhance labels to clearly associate the pens with their function as part of the overall panel enhancement.

CONCURRENCE: ____________________ ________
Management Team Representative Date

REJECTED BY: ____________________ Reason:

UNIT: 2 BLI NO.: 060

COST STUDY NO. — REV.

DCP NO. J38114 REV. 1 DISC. J

DCN NO. 2EJ38114 REV. 1 Priority Disc. J Facility

PACIFIC GAS AND ELECTRIC COMPANY

DIABLO CANYON POWER PLANT

TAKEOFF MA APPROVED

PRICED MA DATE 4/21/87

CHECKED SHEET 1 OF 1

ITEM & DESCRIPTION	QUANTITY	UNIT	UNIT COST MAT'L	UNIT COST S/C	MANHOURS UNIT	MANHOURS TOTAL	COST MATERIAL	COST SUBCONTRACT
PAINTING IS NOT INCLUDED.						0		
WILL BE COVERED UNDER								
PLANT BUDGET								