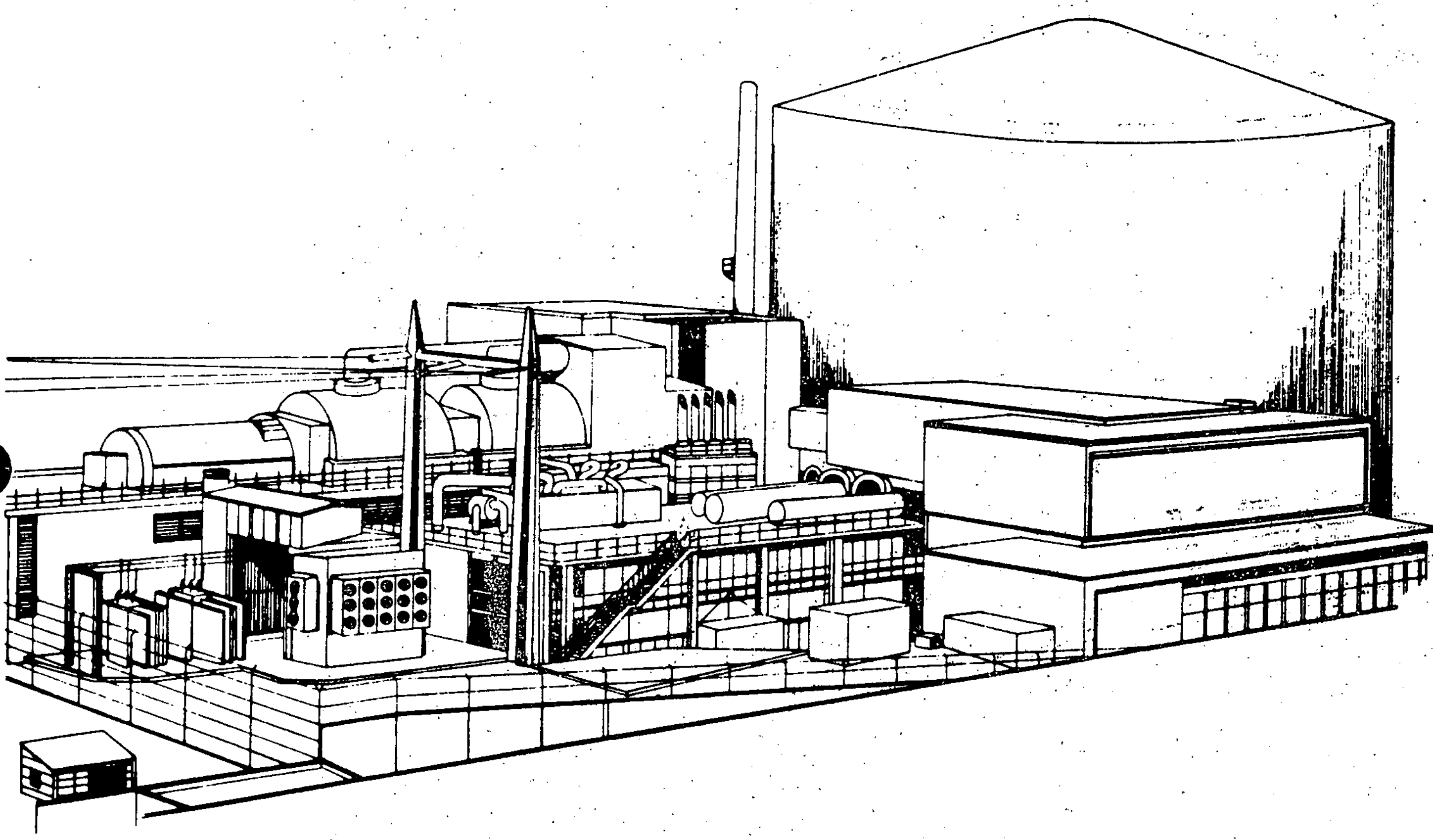
SAN ONOFRE UNIT 1

CONTROL ROOM DESIGN REVIEW REPORT
DECEMBER 1987



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VOLUME 1

Southern California Edison

SONGS 1

CONTROL ROOM DESIGN REVIEW

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CONTROL ROOM DESIGN REVIEW

LIST OF ACRONYMS AND ABBREVIATIONS

AC - Alternating Current
AFW - Auxiliary Feedwater
AFWAS - Auxiliary Feedwater Actuation Signal
AFWP - Auxiliary Feedwater Pump
AFWS - Auxiliary Feedwater System
AIMS - Action Items Management System
AMS - Accident Monitoring System
ANNUN - Annunciator
AOI - Abnormal Operating Instruction
AUX - Auxiliary

BAMU - Boric Acid Makeup
BAS - Boric Acid System
BAST - Boric Acid Storage Tank
BOP - Balance-of-Plant

CAR - Corrective Action Report
CAS - Compressed Air System
CCW - Component Cooling Water
CCWS - Component Cooling Water System
CDM - Corporate Document Management
CEA - Control Element Assembly
CFMS - Critical Function Monitoring System
CIAS - Containment Isolation Actuation Signal
CIS - Containment Isolation System
CO - Control Operator
CR - Control Room
CRDM - Control Rod Drive Mechanism
CRDR - Control Room Design Review
CRIS - Control Room Isolation Signal
CRS - Control Room Supervisor
CRSD - Control Room Standards Document
CRT - Cathode Ray Tube
CS - Containment Spray
CSAS - Containment Spray Actuation Signal
CSS - Containment Spray System
CST - Condensate Storage Tank
CWO - Construction Work Order
CVCS - Chemical and Volume Control System

dB - Decibel
dBA - Decibel (ambient)
DBA - Design Basis Accident
DBMS - Data Base Management System
DC - Direct Current
DCN - Drawing Change Notice
DCP - Design Change Package
DG - Diesel Generator
DNB - Departure from Nucleate Boiling
DNBR - Departure from Nucleate Boiling Ratio
DSD - Dedicated Safe Shut Down

LIST OF ACRONYMS AND ABBREVIATIONS (cont.)

ECCS	- Emergency Core Cooling System
EFAS	- Emergency Feedwater Actuation Signal
EHC	- Electro-Hydraulic Control
EOF	- Emergency Operations Facility
EOI	- Emergency Operating Instructions
EOL	- End Of Life
EOP	- Emergency Operating Procedures
EPG	- Emergency Procedure Guidelines
EPTG	- Emergency Procedure Technical Guidelines
EPRI	- Electric Power Research Institute
ERF	- Emergency Response Facility
ERG	- Emergency Response Guidelines
ESF	- Engineered Safety Features
ESFAS	- Engineered Safety Features Actuation
FCN	- Field Change Notice
FCR	- Field Change Request
FEMA	- Federal Emergency Management Association
FRG	- Functional Recovery Guideline
FSA	- Final Safety Analysis
FW	- Feedwater
FWS	- Feedwater System
FWCS	- Feedwater Control System
GE	- General Electric
GEN	- Generator
GP	- General Physics
GPM	- Gallons per Minute (gal/min preferred)
HED	- Human Engineering Discrepancy
HELBA	- High Energy Line Break Analysis
HF	- Human Factors
HFP	- Hot Full Power
HP	- Health Physics
hp	- High Pressure
HPCS	- Health Physics Computer System
HVAC	- Heating, Ventilating, and Air Conditioning
I&C	- Instrument and Control
ICC	- Inadequate Core Cooling
ID	- Identification (Number)
IDCN	- Interim Drawing Change Notice
IEEE	- Institute of Electrical and Electronic Engineers
IIS	- Integrated Implementation Schedule
INPO	- Institute of Nuclear Power Operations
ISEG	- Independent Safety Evaluation Group
kV	- Kilovolt
LER	- Licensing Event Report
LBLOCA	- Large Break Loss-of-Coolant Accident
LOCA	- Loss-of-Coolant Accident

LIST OF ACRONYMS AND ABBREVIATIONS (cont.)

MCC - Motor Control Center
MFWP - Main Feedwater Pump
MFWS - Main Feedwater System
Mhz - Megahertz
MO - Maintenance Order
MOV - Motor Operated Valve
MSLB - Main Steam Line Break
MSS - Main Steam System
MW - Megawatts

NC - Normally Closed
NCR - Nonconformance Report
NO - Normally Open
NOA - Nuclear Operations Assistant
NOI - Normal Operating Instruction
NO. - Number
NRC - Nuclear Regulatory Commission
NRR - Nuclear Reactor Regulation
NSSS - Nuclear Steam Supply System
NTD - Nuclear Training Division

OPG - Operations Procedures Group
ORP - Operator Requalification Program
OSC - Operations Support Center

P&ID - Piping and Instrumentation Diagram
PASS - Post-Accident Sampling System
PEO - Plant Equipment Operator
PFC - Proposed Facility Change
PGP - Procedures Generation Package
PI - Pressure Indicator
PMRC - Plant Modification Review Committee
PMS - Plant Monitoring System
PORV - Power-Operated Relief Valves
psi - Pounds per Square Inch
psig - Pounds per Square Inch Gauge
PWR - Pressurized Water REactor
Pzr - Pressurizer

QA - Quality Assurance

RCDT - Reactor Coolant Drain Tank
RCP - Reactor Coolant Pump
RCS - Reactor Coolant System
RCSS - Reactor Cycle Sampling System
RHR - Residual Heat Removal
RO - Reactor Operator
RPS - Reactor Protection System
RRS - Rod Control System
RSP - Reactor Shutdown Panel
RTD - Resistance Temperature Detector

LIST OF ACRONYMS AND ABBREVIATIONS (cont.)

SBLOCA - Small Break Loss-of-Coolant Accident
SCE - Southern California Edison
SCRAM - Safety Control Rod Ax Man
SEB - Sphere Enclosure Building
SER - Safety Evaluation Report
Also: Significant Event Report (INPO)
SFP - Spent Fuel Pool
SFTA - System Function Task Analysis
SG - Steam Generator
SGBS - Steam Generator Blowdown System
SGLC - Steam Generator Level Control
SGTR - Steam Generator Tube Rupture
SI - Safety Injection
SIS - Safety Injection System
SIAS - Safety Injection Actuation Signal
SIR - Station Incident Report
SLSS - Safeguard Load Sequencing System
SMM - Subcooling Margin Monitor
SOCR - San Onofre Commitment Register
SOER - Significant Operating Experience Report (INPO)
SOMMS - San Onofre Maintenance Management System
SONGS - San Onofre Nuclear Generating Station
SPDS - Safety Parameter Display System
SPR - Site Problem Report
SRO - Senior Reactor Operator
SS - Shift Supervisor
STA - Shift Technical Advisor
SWCS - Saltwater Cooling Water
SWO - Station Work Order

TCN - Temporary Change Notice
TG - Turbine Generator
TMI - Three Mile Island
TPCWS - Turbine Plant Cooling Water System
TRIMS - Training Records Information Management System
TSC - Technical Support Center

UPS - Uninteruptible Power Supply
USMC - United States Marine Corps

VCT - Volume Control Tank

WO - Work Order
WOG - Westinghouse Owners Group
WR - Work Request
WRN - Work Request Number

XFCR - Field Change Request (Initiated by Construction)
XFR - Transfer

EXECUTIVE SUMMARY

The activities of the CRDR began in June 1985 with the evaluation and selection of a human factors engineering consultant. General Physics Corporation was chosen based on their prior experience in the control room design reviews of other nuclear power reactors of similar design and vintage to SONGS 1, and the extensive personnel qualifications in human factors engineering. General Physics Corporation was responsible for performance of several of the CRDR tasks with coordination, support and review as necessary from SCE personnel.

The methodology used to perform the CRDR was based on the NUREG-0700 guidelines. The CRDR was divided into component phases similar to those recommended in NUREG-0700. These phases were developed and executed in a thorough and comprehensive manner to ensure the CRDR resulted in identification, evaluation, and correction of control room man-machine interface deficiencies.

As part of the CRDR the following was performed:

- Development of a multidisciplined CRDR Team responsible for coordination, technical support, working participation, and management oversight of CRDR activities performed by SCE and General Physics.

- Review of plant operating experience including a historical documentation review and survey and interviews of control room operating personnel.

- Control room human engineering survey to evaluate compliance of existing control room instrumentation and controls with NUREG-0700 guidelines.

- Identification and evaluation of plant specific system requirements and associated operator tasks.

- Verification of operator task performance capabilities including availability of required instrumentation and controls and suitability of existing equipment.

- Validation of control room functions via operator walkthroughs of accident scenarios simulated in the full-scale photographic control room mockup.

- Evaluation, prioritization and identification of corrective actions for all human engineering discrepancies (HEDs) identified.

- Coordination and integration of related NUREG-0737, Supplement 1 initiatives.

As a result of the CRDR, the following is a summary of the enhancements and modifications approved for implementation:

<u>Enhancements</u>

A. Provide functional system color coding and demarcation by repainting the panels and all instrument bezels.

B. Implement control room labeling scheme to provide a five-level labeling hierarchy with clear, concise and consistent information; relocate the labels to the top of the instruments; and replace pushbutton labeling wherever required.

C. Provide scale coding and re-scaling for indicators and recorders to show key operating information, legibility, engineering units and proper ranging.

D. Prioritize annunciator system by the use of colored windows and replace the legends to improve size, consistency and clarity of characters.

E. Address glare problems by utilizing non-glare paint and non-glare lenses where indicated.

Modifications

A. Redesign of the Emergency Diesel Generator Control Panels C41 and C42 to improve control/display integration.

B. Significant design modifications to the Nuclear Control Auxiliary Panel C09 including over forty (40) I&C component relocations.

C. Deletion of 16 abandoned controls and indicators on various panels.

D. Component relocations on the Remote Shutdown Panel C38, Auxiliary Equipment Control Panel C13, and Recorder Panel C05.

E. Replacement of several indicators, recorders, and controls judged unsuitable.

F. Annunciator system upgrades including elimination of boric acid heat trace nuisance alarms via addition of new alarm points, color code prioritization, tile replacement to improve legibility, consistency and accuracy, and elimination of abandoned points.

G. Miscellaneous actions to complete in the areas of Communications, Environment, and Procedures.

H. Installation of protective hinge covers and barriers.

I. Installation of additional indicators and controls.

Although not all inclusive of the modifications to be implemented by the CRDR, the above lists provide an overview of the significant changes that will be made to the control room.

1.0 CONTROL ROOM DESIGN REVIEW INTRODUCTION

1.1 FORWARD

This report presents the methodology and results of Southern California Edison (SCE) Company's Control Room Design Review (CRDR) of the San Onofre Nuclear Generating Station (SONGS) Unit 1 control room. The activities of the CRDR were performed in accordance with the CRDR Program Plan for the purpose of assessing the degree to which the control room conformed to applicable human factors criteria and principles. The primary effort was directed to those aspects established by NRC precedent to be most relevant and contributory in reducing the risk of operator error. Based on the CRDR, control room modifications intended to improve the man-machine interface and conditions that could contribute to operator error, confusion and fatigue were identified.

1.2 OBJECTIVES

This control room design review report is being submitted to the NRC in compliance with NUREG-0737, Supplement 1, Item 5.2.b. Furthermore, this report meets the intent of the appropriate draft evaluation criteria of NUREG-0801. Upon implementing the recommendations of this report, the control room for SONGS Unit 1 will:

1. Conform to the established criteria provided in NUREG-0700 to the extent practicable.

2. Conform to good human engineering practices currently employed in the industry.

3. Meet the requirements of NUREG-0737, Supplement 1.

The activities described in this report derived their bases from the Control Room Design Review Program Plan written by SCE to implement the CRDR and submitted to the NRC by SCE letter dated December 16, 1985. The plan encompassed the guidelines provided in NUREG-0700, "Guidelines for Control Room Design Reviews".

The content of this report reflects the evolution of CRDR activities conducted on SONGS 1 by SCE. Specific details of each area of the CRDR process are addressed generically so that the reader can comprehend the scope of each item without excessively burdening the report.

This report is submitted to provide documentation of the SCE commitment to control room enhancement. Activities described in this report meet the functional intent of the NRC NUREG-0737, Supplement 1, Item 5.1.b. Moreover, recognizing the dynamics of continual plant design evolution and of continued operating experience, the criteria developed as part of the CRDR will be factored into any future control room modification. To this end, the information contained in this report is considered a "snap-shot" of a continuing process as well as a statement of the SCE application of human engineering to the control room man-machine interface.

1.3 GENERAL PLANT DESCRIPTION

1.3.1 Site Description

The San Onofre site is located on the coast of Southern California in San Diego County, approximately 62 miles southeast of Los Angeles and 51 miles northwest of San Diego. The site is located entirely within the boundaries of the United States Marine Corps Base, Camp Pendleton, California, near the northwest end of the 18-mile shoreline. The site is approximately 4,500 feet long and 800 feet wide, comprising 84 acres. The site consists of three pressurized water reactors. Unit 1 is a Westinghouse design reactor and Units 2 and 3 are Combustion Engineering design reactors. Approximately 16 acres are occupied by Unit 1. Units 2 and 3 cover 52.8 acres of which the power block and site switchyard occupy 27.7 acres and parking and access area another 25.1 acres. The remaining 15.2 acres are occupied by the administration building or are available for auxiliary usage. Units 2 and 3 are located southeast of and immediately adjacent to Unit 1.

1.3.2 Plant Characteristics

The San Onofre Unit 1 plant characteristics are as follows:

Reactor Type	Pressurized Water Reactor (PWR)
Reactor Designer	Westinghouse
Generator Manufacturer	Westinghouse
Capacity	450 MWe (gross)
Engineer	Bechtel
Commercial Operation	January 1968

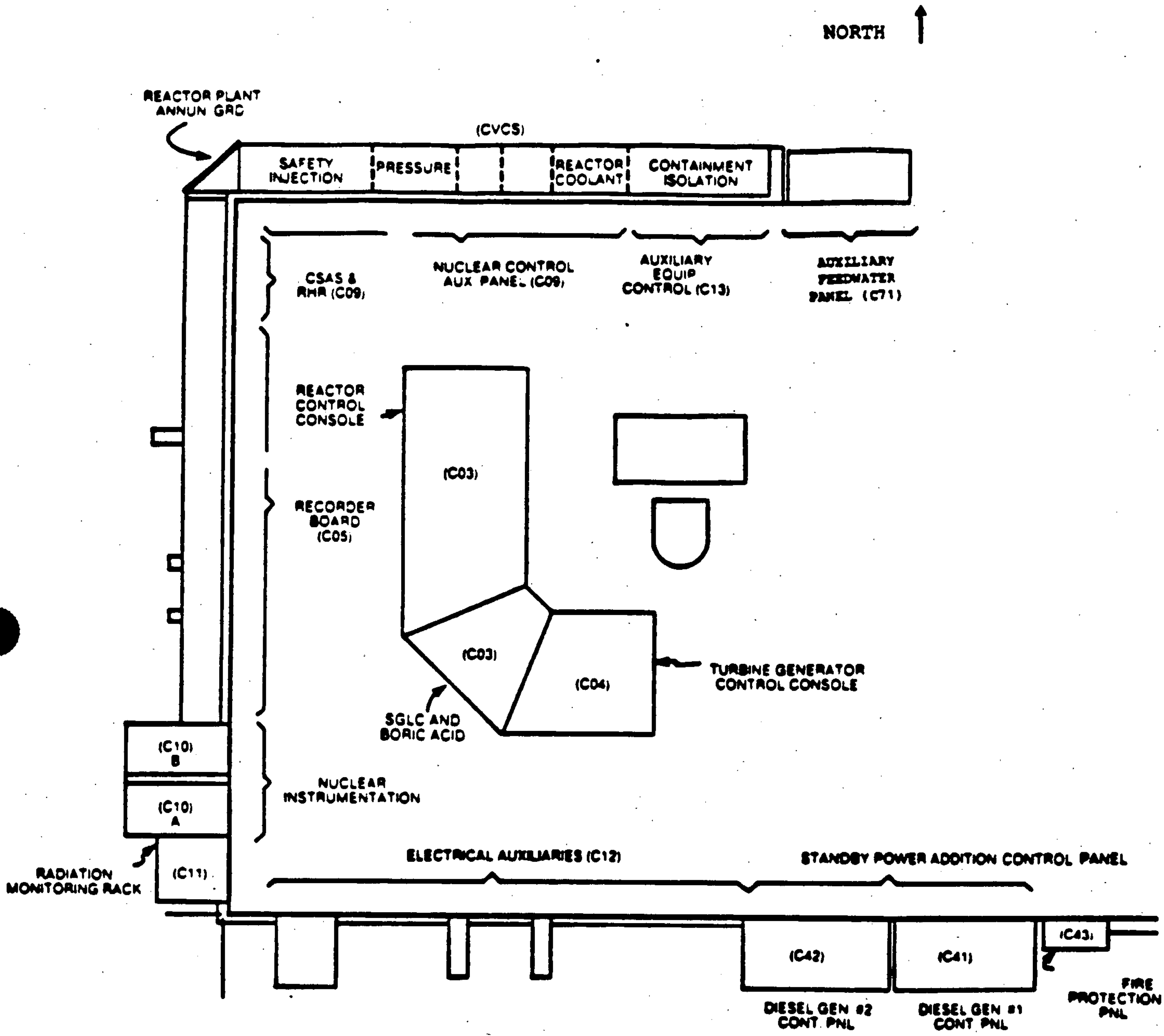
The ultimate heat sink for all safety related cooling water systems is saltwater from the Pacific Ocean, supplied to the component cooling water heat exchangers by saltwater cooling pumps located within an intake structure separate from Units 2 and 3. Seawater pumped from the intake structure by the circulating water pumps serves as the heat sink for heat rejected by the main condensers and the turbine plant cooling water system.

1.3.3 Control Room Configuration

The SONGS 1 control room is illustrated in Figure 1-1. Vertical control panels form three sides of the control room perimeter, surrounding a J-shaped console which utilizes a combined bench/vertical operating surface contour. The functional location of instruments and controls is also outlined. Behind the main control panels are additional panels accommodating equipment such as meteorological data monitoring, miscellaneous recorders, etc.

Figure 1-1

CONTROL ROOM LAYOUT



Abbreviation:

ANNUN	- Annunciator
AUX	- Auxiliary
CONT	- Control
CVCS	- Chemical and Volume Control System
GEN	- Generator
PNL	- Panel
RHR	- Residual Heat Removal
SGLC	- Steam Generator Level Control

2.0 MANAGEMENT AND ORGANIZATION

2.1 INTRODUCTION

The management approach used to perform the CRDR and prepare this report was to establish a CRDR team to coordinate and participate in CRDR activities. The CRDR team consisted of a multidisciplined group of individuals knowledgeable in the areas necessary to perform a control room review. This group was given the direction that cost, schedule, and original design considerations should not be taken into account when assessing the findings or making recommendations.

2.2 OBJECTIVES

The major objective of the CRDR was to identify specific instrumentation and control components, environmental factors, and other man-machine interface aspects that were less than optimal, and could cause confusion, difficulty, or undue fatigue for the plant operators in the performance of their duties.

The problems identified through the CRDR are referred to as Human Engineering Discrepancies (HEDs). All instruments in the control room inventory were reviewed and HEDs were recorded and reported. The HEDs were classified as to seriousness and priority of need for correction or improvement in accordance with NUREG-0801, and recommendations for means of improvement were submitted to SCE management for approval.

The recommendations that were accepted will be implemented either by issuing design change packages (DCPs), field change notices (FCNs), or maintenance orders (MOs). These documents define the design change and provide directions for the implementation, testing, and placing the improvement in operation in the plant.

2.3 METHODOLOGY

The methodology developed to perform the CRDR was as follows: (1) establish a CRDR team, (2) review the control room panels for compliance with NUREG-0700, (3) document the findings and assessments, (4) make recommendations to resolve the findings and assessments, and (5) implement the recommendations. Each of these CRDR processes is further described in subsequent sections.

2.3.1 CRDR Team Organization

The SCE approach to management of the control room design review is outlined in Figure 2-1. The primary elements include the Plant Modification Review Committee, the CRDR HED Assessment and Evaluation Teams, General Physics human factors consultants, and the CRDR team members.

SCE management personnel have the ultimate responsibility for the control room design review. The day-to-day conduct of the review, however, was the responsibility of the CRDR Team Leader and team members. The CRDR Team provided the management oversight to ensure the integration of the project objectives and to meet the regulatory intent of the review.

2.3.1.1 Responsibilities

A. CRDR Team

The CRDR Team was responsible for planning, scheduling and conducting the detailed, integrated control room review and work activities performed by SCE and General Physics. The CRDR Team Leader provided management oversight for the CRDR team activities and a focal point for communication between SCE and General Physics.

The CRDR Team activities included developing the methodologies for the review, establishing the detailed plan and schedule for the control room design review, participation in CRDR tasks, coordination and implementation of tasks, technical support, and review and approval of activities.

B. Human Factors Engineering Consultants

The human factors engineering consultants, General Physics personnel, provided existing recognized human factors engineering criteria for the CRDR program. General Physics provided a CRDR Project Manager responsible for coordinating, implementing and documenting the historical document review, operating personnel survey, control room survey, system function task analysis, control room inventory, verification and validation of control room instruments and functions, and compilation of HEDs. General Physics also provided working participation in the assessment of HEDs and preparation of CRDR documentation. In addition, General Physics provided indoctrination on human factors engineering considerations for the CRDR team members.

C. CRDR Human Engineering Discrepancy Assessment Team

Following preparation of a HED by General Physics, the HED was forwarded to the CRDR HED Assessment Team. The Assessment Team was responsible for review, determination if the HED should actually be a HED, establishment of the correction priority, formulation of alternative corrective actions, and recommendation for correction actions. The Assessment Team was composed of the CRDR Team Leader and six other members with the following expertise; nuclear licensing, instrumentation and controls engineering, human factors engineering, senior reactor operator, nuclear engineering, and balance-of-plant (BOP) engineering. After all the HEDs had been assessed, the HEDs were forwarded to the CRDR HED Evaluation Team.

D. CRDR Human Engineering Discrepancy Evaluation Team

Upon receipt of the HED from the HED Assessment Team, the HED Evaluation Team was responsible for reviewing assessment findings and confirming or rejecting assessment recommendations.

The Evaluation Team was composed of the CRDR Team Leader and eight members with the following expertise; nuclear engineering, station technical, station operations, station training, station management, instrumentation and controls engineering, nuclear licensing, and human factors engineering. Following completion of the HED evaluation, the HED was forwarded to the SCE Plant Modification Review Committee for approval.

E. Plant Modification Review Committee

The primary responsibility of the Plant Modification Review Committee (PMRC) was to provide management review to ensure meaningful control room improvements will be provided by the CRDR proposed modifications. The PMRC evaluated proposed modifications utilizing criteria relating to safety, compliance, technical specifications, cost benefit analysis, operation and ALARA. The PMRC had responsibility and authority to review and accept or reject the scope, priority, and budget category of proposed CRDR modifications.

F. SCE Executive Approval

The San Onofre Vice President and Site Manager, Vice President of Engineering and Construction, and Vice President of Nuclear Engineering, Safety and Licensing meet on a regular basis and review and approve all plant design changes. These individuals have authority to accept or reject proposed CRDR modifications.

2.3.1.2 Reporting Relationships

A combination of the SCE CRDR Team Leader and the General Physics CRDR Project Manager provided the focal point for coordinating and implementing CRDR activities (see Figure 2-1). As part of this responsibility, these individuals provided the functional communication link between the primary elements of the CRDR. The CRDR Team Leader maintained ultimate responsibility for resolution of differences and other concerns identified by the CRDR Team, HED Evaluation and Assessment Teams, and the PMRC. Any items not resolved were referred to higher levels of management.

2.3.2 Control Room Design Review Process

2.3.2.1 Objective

The major goal of the CRDR was to identify HEDs that exist in the control room, and which may create unnecessary difficulty, fatigue, or confusion for the operator in the performance of their duties, or in recognizing and understanding existing and developing plant conditions.

As required by NUREG-0700, the control room design review concentrated in the areas listed below:

- o Control panel reviews
- o Control room design and layout
- o Control room instrumentation, controls, and equipment
- o Control room environment review
- o System function and task analysis identification, and control room function verification
- o Validation of panel design and emergency operating instructions
- o Development/documentation of HEDs
- o Integration of other NUREG-0737, Supplement 1 items

2.3.2.2 Criteria Development

A. Control Panel Review Criteria

The control panel review criteria including instrumentation, controls, equipment, control room design and layout, and environment were based on NUREG-0700. Based on the review criteria and methodology, General Physics performed the physical review of the controls and instruments.

B. System Function and Task Analysis Criteria (SFTA)

The criteria used as the basis for the SFTA were based on the generic Westinghouse Owner's Group (WOG) System Review and Task Analysis, the WOG Emergency Response Guidelines, plant specific documentation including the SONGS 1 Emergency Operating Instructions (EOIs), SONGS 1 System Descriptions, EOI Bases Documents, and the SONGS 1 Q-List.

Based on the above criteria, General Physics performed the CRDR System Function and Task Analysis.

2.3.2.3 CRDR Process Methodology

The CRDR process consisted of five phases as functionally illustrated in Figure 2-2. These phases are (1) planning, (2) review, (3) assessment, (4) correction and implementation, and (5) documentation and reporting. This process was developed to provide an optimal implementation plan for completion of the control room design review. Within each phase are several individual tasks established to comply with NUREG-0700 and properly perform and document the CRDR process.

A. CRDR Preparation and Planning

The SONGS 1 Control Room Design Review Program Plan was submitted to the NRC in December 1985. This document described the basic process by which SCE intended to perform the CRDR. In accordance with this document, SCE began the preparation process for organizing, scheduling, implementing and documenting the various phases of the CRDR. Manpower was allocated to establish the CRDR team consisting of a multidisciplined group with the necessary expertise to support the various facets of the CRDR. Individual and functional responsibilities were established, including program overview orientation and reporting relationships.

B. CRDR Review - Identification of HEDs

The review phase consisted of six major processes by which human engineering discrepancies were identified. These processes were based on and conducted in accordance with NUREG-0700. The various processes were performed by General Physics with SCE participation, support and review as necessary. These processes were:

- Operating Experience Review - identify factors or conditions that could cause, and/or previously caused human performance problems via review of documented occurrences.

- Control Room Inventory - develop current listing of all control room instruments, controls, and equipment with which the operators interface.

- Control Room Human Engineering Survey - identify characteristics of instrumentation and controls, equipment, physical layout, and environmental conditions that do not conform to precepts of good human engineering practice.

- System Function and Task Analysis - determine the information and control requirements of the control room crew for emergency operation and ensure that required systems can be efficiently and reliably operated under the conditions of emergency operation by available personnel.

- Verification of Task Performance Capabilities - verify instrumentation and controls identified in the Task Analysis are present in the control room and effectively designed to support correct task performance.

- Validation of Control Room Functions - ensure functions allocated to the control room operating crew can be accomplished effectively within the structure of the SONGS 1 EOIs and the control room design as it exists.

Discrepant items identified by the above processes were identified as HEDs and processed in accordance with the guidelines established by the HED Assessment phase.

C. HED Assessment

The objective of the CRDR assessment phase was to evaluate and assess the significance and relative importance of the HEDs discovered in the review phase. Recommendations were made concerning corrective actions to be taken, and justification was given in those instances when it was determined that no action was necessary.

The transition to the assessment phase of CRDR proceeded as a natural extension of the review phase. Criteria were developed to assess and categorize the HEDs as to seriousness of potential consequences, particularly with regard to safety. Additional evaluations were made concerning possibilities and alternate options for improvement and for difficulty of implementation.

The HEDs that were minor in nature, had no safety consequences or other potential significant consequences that could have resulted in loss of plant availability or equipment damage, were given a lower priority for correction in accordance with NUREG-0801. All HEDs were given consideration although it was recognized that control panel design embodies numerous compromises among requirements competing for priority. Therefore, situations may exist where the most direct means of improvement for one feature or aspect would have a detrimental effect on some other feature or on overall design. Where that occurred, an attempt was made to find the best overall solution.

D. HED Correction and Implementation

The objective of the Implementation Phase was to remedy significant HEDs identified in the Assessment Phase. An effort was made during the Assessment Phase to give the most important items priority for corrective action. Priorities were assigned in accordance with NUREG-0801 and based on the potential for error, degree of safety importance, and potential for unsafe condition or Technical Specification violation.

Control room and control panel modifications recommended by the CRDR Evaluation Team and accepted by the PMRC will be implemented by an established, closely controlled, and scheduled procedure in accordance with SCE's design control program.

Implementation of modifications will be scheduled in accordance with the Integrated Implementation Schedule (IIS). The IIS establishes a hierarchical ranking system whereby pending plant modifications are evaluated based on safety and non-safety criteria in relationship to all other pending modifications.

E. Documentation and Reporting

In order to provide a systematic and consistent means of conducting the CRDR, working documents were generated as required for the work activities of the CRDR, and retained for long-term storage, either in conventional files and/or microfilm along with significant documents that support CRDR determinations, decisions, and conclusions.

To this end, documentation was generated and retained to ensure:

o A record was provided of all documents used by the CRDR Team as references during the various phases.

o A record was provided of all documents produced by the CRDR Team as project output.

o An audit path was generated through the project documentation.

o Project files were developed in a manner that allows future access to help determine the effects of control room changes proposed in the future.

The following documents served as reference material used during the CRDR process:

o Licensee Event Reports (LERs)
o Station Incident Reports (SIRs)
o System Descriptions
o Piping and Instrumentation Drawings
o I&C Index
o Control Room Floor Plan
o Panel Layout Drawings
o Panel Photographs
o SONGS 1 EOIs
o SONGS 1 EOI Bases Documents
o Abnormal Operating Instructions

- o Westinghouse Generic Systems Review and Task Analysis (SRTA)
- o Westinghouse Generic Emergency Response Guidelines (ERGs)
- o SONGS 1 Q-List
- o SONGS 1 Regulatory Guide 1.97 Review
- o SONGS 1 SPDS Conceptual Study
- o Nonconformance Reports (NCRs)
- o Site Problem Reports (SPRs)
- o Functional Specifications of the Technical Data Display and Transmit System (FOX 3)
- o Regulatory Guide 1.97, Rev. 2
- o Environmental Qualification Master List
- o Final Safety Analysis
- o Technical Specifications

Throughout the CRDR, documents were processed in order to facilitate the systematic assessment and comparison of actual control room features against desired standards, record the results of the design review, identify HEDs, and provide recommendations for achievable, cost-effective design improvements. The documents generated during the CRDR include:

- o Program Plan
- o Project Schedule
- o Operator Questionnaire
- o LER Review Results Forms
- o Control Room Inventory Worksheets
- o Panel Checklists (from the Control Room Survey)
- o Task Analysis Worksheets
- o Videotapes of Validation
- o HED Assessment Team Meeting Minutes
- o HED Evaluation Team Meeting Minutes
- o PMRC Meeting Minutes
- o CRDR Team Meeting Minutes
- o All HED Records
- o Final Summary Report

The primary means for storage of the large amount of data and information produced during the CRDR was a computerized Data Base Management System (DBMS). The focus of the DBMS was an IBM XT computer. The DBMS software was based on the dBASE III program by Ashton-Tate, as modified by General Physics for the CRDR project. The DBMS allowed for selective sorts and lists of data collected through the CRDR. The following data was entered into the DBMS files:

- All HED Records
- Task Analysis Data
- Equipment Characteristics Data
- System Function Description List

Each of the input data files allowed for rapid, convenient management and tracking of review findings and results. The HED file provided a look-alike output form that was used in the Final Summary Report and other documentation.

2.3.3 <u>CRDR Team Selection, Qualifications and Task Assignments</u>

2.3.3.1 Plant Modification Review Committee

The Plant Modification Review Committee (PMRC) is an existing organization within the SONGS operating structure. The PMRC was not an organization formed for purposes of performing CRDR related tasks. The qualifications of the individuals forming the PMRC established an appropriate medium to incorporate engineering-management considerations for approval or disapproval of proposed modifications. The individuals forming the PMRC include:

- H. E. Morgan - Station Manager
- R. W. Krieger - Operations Manager
- J. T. Reilly - Station Technical Manager
- J. J. Wambold - Project Manager
- D. E. Shull - Maintenance Manager

2.3.3.2 CRDR Team Composition and Qualifications

The CRDR Team consisted of a core group of specialists in human factors engineering, plant operations, and nuclear and electrical/instrumentation and controls (I&C) engineering. This core group included personnel also knowledgeable in licensing, training, program management, and other NUREG-0737, Supplement 1 programs, such as SPDS, upgrade of Emergency Operating Instructions (EOIs), and Regulatory Guide 1.97.

The team members were carefully selected to obtain an optimum blend of past experience to ensure the best possible CRDR, analysis, and recommendations. All CRDR team members were chosen to be part of the group because of their expertise with nuclear power plant fundamentals, design, and operation. General academic background as well as nuclear academics and extensive control systems design experience were considered as part of the qualification.

The qualifications of the CRDR Team members meet the NUREG-0800 criteria. The team members' resumes are provided in Appendix 2A.

2.3.3.3 CRDR Team Orientation and Training

General Physics provided an orientation and training course in human factors engineering for the CRDR team members at the beginning of the CRDR effort.

Training was to familiarize personnel with the principles of human factors engineering and their application to the CRDR. The importance of proper preparation and training for all CRDR activities was recognized. During the course of the CRDR as specific areas of training were identified, appropriate training or orientation was provided to meet these needs, including practical application of human engineering criteria to these areas of the CRDR.

The specific areas of human factors training that were covered during the two-day orientation course included:

- o Systems Analysis Techniques
- o Human Factors Convention
- o Anthropometry
- o Panel Layout Principles
- o Labeling
- o Man-Machine Interface
- o Color Psychology/Color Use

The General Physics training course was entitled, "Applied Human Factors in Power Plant Design and Operation." The text for the training course was written by D. C. Burgy, P. A. Doyle, H. F. Barsam and R. J. Liddle. The course training instructors were Dr. L. R. Schroeder and Dr. H. N. Tobey.

2.3.3.4 Task Assignments

The CRDR team task assignments were made to best utilize the group members' area of expertise in the preparation of the CRDR report. The level of participation of the various CRDR support groups is illustrated in Table 2-1.

The primary responsibility for coordination and implementation for each CRDR activity was held by the CRDR Team Leader and/or the General Physics CRDR Project Manager. The remaining team members provided working participation in task performance, review and comment, or technical support in appropriate areas of expertise.

Figure 2-1
FUNCTIONAL CRDR TEAM ORGANIZATION

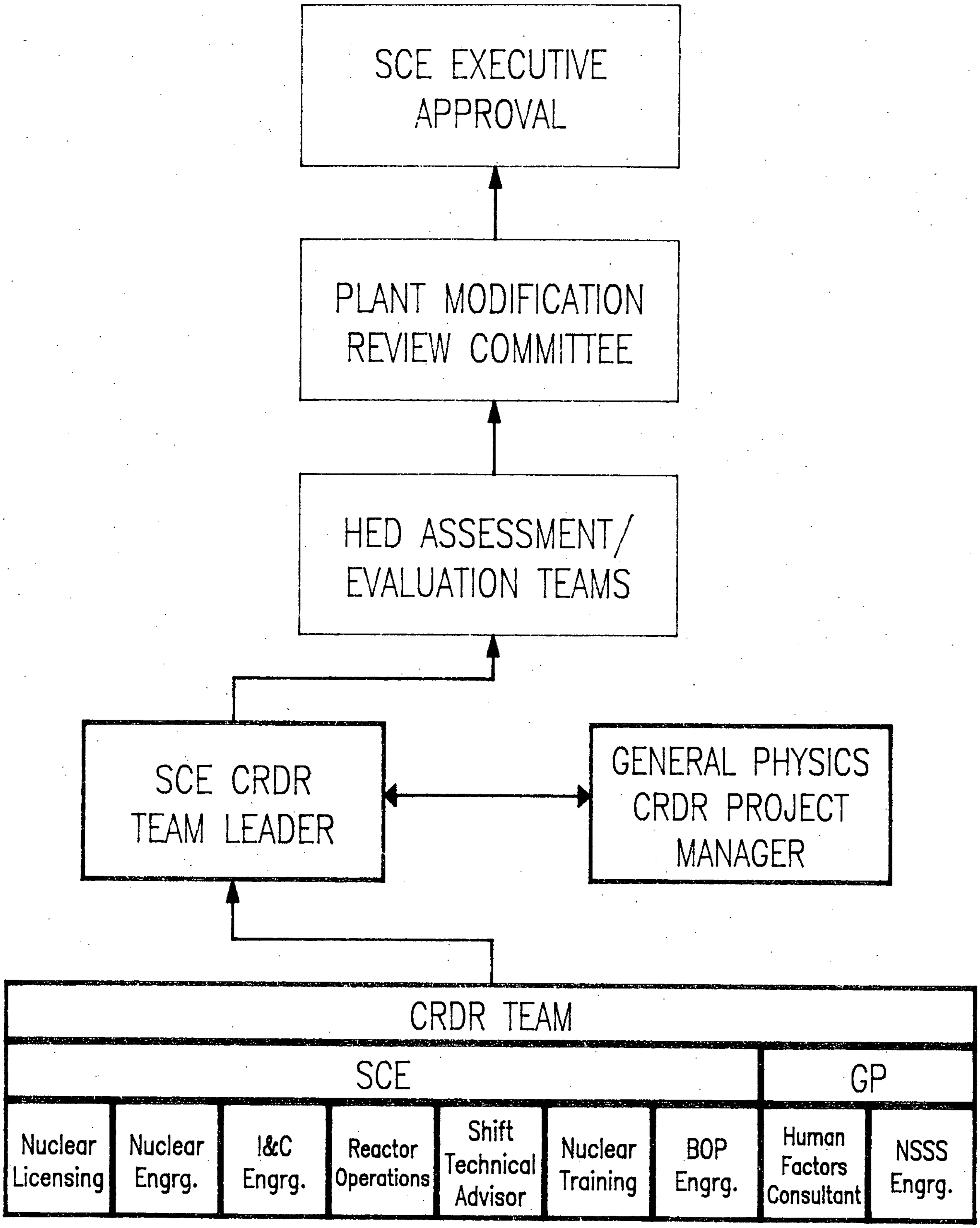


Figure 2-2

CRDR PROCESS FLOWCHART

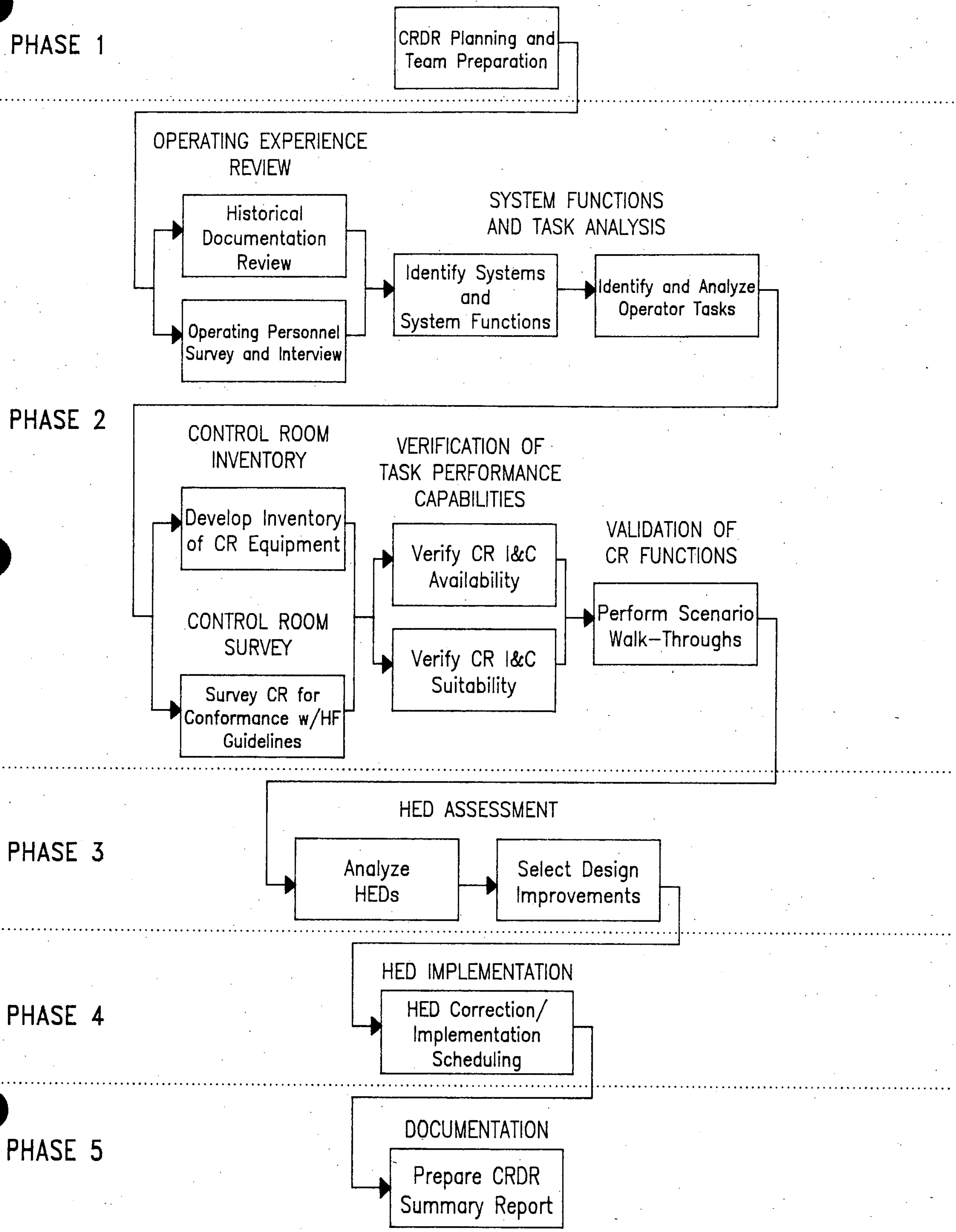


Table 2-1 SONGS-1 CRDR Level of Participation Summary

PHASE/ACTIVITY	CRDR TEAM LDR	HF ENGR (GP)	I&C ENG	STA OPS	NUC SYS ENG	NUC TRNG	NUC LIC	STA TECH	STA MGMT	CRDR TEAM
	INDIVIDUAL POSITION OR DEPARTMENT									
1.0 Planning Phase										
1.1 Select HFC	C/RA									
1.2 Procure CR Mockup	C/RA	W								
1.3 Conduct HF Orientation	C	C/W	W	W	W		W			W
2.0 Review Phase										
2.1 Plant Operating Experience Review	C/RA	W	RC	RC	RC		RC			RC
2.1.1 Historical Document Review	C/RA	W	RC	RC	RC		RC			RC
2.1.2 Operating Personnel Survey	C/RA	W	RC	W/RC	RC		RC			RC
2.2 CR Inventory	C/RA	W	RC	RC	RC		RC			RC
2.3 CR Survey	C/RA	W	RC	T/RC	RC		RC			RC
2.4 SFTA	C/RA	W	RC	RC	RC		RC			RC
2.5 Verify Instruments	C/RA	W	RC	RC	RC		RC			RC
2.6 Validate CR Functions	C/RA	W	RC	RC	RC		RC			RC
2.7 Compile HEDs	C/RA	W	RC	RC	RC		RC			RC
3.0 Assessment Phase										
3.1 HED Assessment	C/W/RA	W	W	W	W		W			T
3.2 HED Evaluation	C/W	RC	W	W	W	W	W	W	W/RA	T
3.3 PMRC	C/T	T	W	W	T		T	W	W/RA	T
4.0 Implementation Phase										
4.1 Scheduling HED Correction	T			W	W		C/W/RA			RC
5.0 Documentation Phase										
5.1 Task Reports	C/RA	W	RC	RC	RC		RC			RC
5.2 Summary Report	C/RA	W	RC	RC	RC	RC	W/RC	RC	RC	RC
5.3 General Documentation	C/W/RA	W					RC			RC

C – Responsibility for coordination and implementation
W – Working participation in CRDR task
RC – Review and comment role
RA – Review and approval authority
T – Technical support and/or input

APPENDIX 2A

REVIEW TEAM RESUMES

The Resumes for the CRDR Team, HED Assessment Team and HED Evaluation Team members are contained in this appendix in the following order:

Southern California Edison

- A. A. Hernandez
- J. G. Ibarra
- M. J. Kirby
- W. McGhee, Jr.
- M. B. McKinley
- J. L. Prickett
- J. P. Reynoso
- E. Siemion
- D. D. Snuggs
- J. R. Tate
- M. J. Thomas

General Physics Corporation

- D. B. Barks
- D. C. Burgy
- R. Danna
- M. W. Dawson
- M. E. Jennex
- W. A. Martin
- M. K. Pumphrey
- L. R. Schroeder
- H. N. Tobey
- M. D. Venters III

ARMANDO A. HERNANDEZ

EDUCATION:

BS Electrical and Electronics Engineering, California Polytechnic State University, Pomona, 1974
Human Factors Engineering Workshop/Certification
ISA Controls Professional Engineer Exam Review Course
Electrical Power Distribution - Industrial and Commercial Plants, UCLA Extension
Applied Protective Relaying, Cal Poly, Pomona

SUMMARY:

Present: San Onofre Nuclear Generating Station Unit One Project Lead Controls Engineer -- Direct in-house controls engineering and manage engineering services contractors for preparation of NSSS and Balance of Plant systems modifications. Member of CRDR Project Team providing I&C cognizance.

2 Years: I&C engineer San Onofre Nuclear Generating Station Units 2 and 3 Project-Controls engineering support for the following projects: Makeup Demineralizer, Full Flow Condensate Polishing Demineralizer, Condensate Tank Level Control, MSIV/FWIV/FWBV Hydraulic Actuators, miscellaneous balance of plant modifications.

3 Years: I&C Engineer San Onofre Unit One Project -- Performed Controls engineering for the following design packages: Wide range radiation monitoring of effluent and containment, post-accident containment monitoring of pressure, water level and hydrogen, automated alert system, inadequate core cooling preliminary engineering, health physics computer installation and input signal processing.

6 Years: Electrical Engineer -- Performed electrical engineering and design for generation related plant improvement projects including: Mohave auxiliary power study, Mohave primary air duct burner control, Mohave side stream softener preliminary engineering, Mohave bottom ash T.V.'s, Mohave coal ponds 5-8, Big Creek 3, Unit 5, fuel oil pipeline renovation-cathodic protection, Mohave fuel oil startup system, Mohave coal slurry, storage and reclaim facility, Long Beach combined cycle cathodic protection system design, catalina island flushing system, Brea fuel oil facility and Dominguez Hills pipeline preliminary engineering, alamitos 5 & 6 crude oil conversion construction support, Digital dispatch security monitoring system.

EXPERIENCE:

Mr. Hernandez has, for the last seven years, been involved in nuclear power plant instrumentation and controls systems modifications. In the last two and one half years this has been in the capacity of the lead controls discipline engineer for SONGS Unit 1.

Previously, his work has been as an electrical discipline engineer on generation plant related improvement projects. Assignments were both systems and design oriented with continual emphasis on auxiliary power and electrical implementation of controls and instrumentation.

Usual duties in the Controls and instrumentation discipline included: in-house design, management of engineering contractor preparing design packages, engineering support during construction/start up, and engineering support for Nuclear Licensing and Engineering.

For the electrical discipline, Mr. Hernandez' duties included equipment sizing, including voltage drop, short circuit and capacity calculations for medium and low voltage switchgear, motors, transformers and cable, electrical apparatus bid packages from development through evaluation, award and vendor coordination, development of one line and elementary diagrams, direction of designers in final design of physicals, wiring, bills of material and schedules, and implementation of electrical requirements for controls and instrumentation through interface with Controls and Mechanical Engineers and documents, e.g., logic diagrams, P&IDs.

PROFESSIONAL AFFILIATION:

Registered Professional Engineer in the State of California, Certificate Number E9129.

JOSE G. IBARRA

EDUCATION:

BS Electrical Engineering, New Mexico State University, 1975
18 Semester Credits toward MBA, University of Nevada/Las Vegas
Nuclear Reactor Safety, Summer Course, Massachusetts Institute of Technology, 1979
Man-Machine Interface, Summer Course, Massachusetts Institute of Technology, 1980

SUMMARY:

Present: Nuclear Systems Engineer - SONGS 1 CRDR Team Co-Leader. Providing Human Factors support to Utility Simulator Facility Group on developing standard for non-plant reference simulator.

3 Years: I&C Engineer in the Nuclear Engineering Organization providing, conceptual design, Nuclear Engineering support to plant modifications.

1 Year: Startup Engineer at SONGS Unit 1, starting up several TMI Radiation Monitoring System Retrofits.

6 Years: Electronic Engineer doing system design for nuclear weapons testing at the Nevada Test Site.

3 Years: Consultant to the NRC on several Electrical and I&C Systems. Heavy involvement with the TMI Short-Term Lessons Learned and TMI Implementation Plan.

3 Years: Satellite Tracking Operator and Instructor. Responsible for operations and maintenance of remote satellite tracking systems for worldwide coverage. Responsible for teaching station procedures and station maintenance.

EXPERIENCE:

Mr. Ibarra has over 14 years of experience in the design and application of instrumentation in the application of nuclear weapons research and the nuclear power industry. The last 8 years have been in TMI related issues first working with the NRC Staff and the last 5 years working as a Nuclear Engineer for Southern California Edison.

His present job of Nuclear Engineer with Southern California Edison involves doing conceptual designs and providing Nuclear Engineering input to plant modifications. He has provided the technical lead for Southern California Edison on SPDS, Regulatory Guide 1.97 and the other NUREG-0737 systems.

EXPERIENCE: (continued)

Before the nuclear power industry, he was involved in nuclear weapons testing. As an Electronic Engineer for Lawrence Livermore National Laboratory he designed state-of-the-art systems for nano-second data resolution. Responsibilities included fielding the systems, and data analysis.

His nuclear power involvement was begun as a consultant to the NRC for the Electrical and I&C Systems branches. Tasks involved doing technical evaluation and the writing of the TER's. He was on loan to the NRC for a period of one year in Bethesda. During this time, he was the electrical representative on site inspection teams for the TMI Short-Term Lessons Learned Implementation. While on loan, he also worked with the I&C Systems branch in reviews of near-term licensees on the TMI Implementation Plan.

He did system startups at San Onofre Nuclear Generating Station Unit 1. He put into operations several of the Post-TMI systems. Responsibilities including writing the prerequisite and preoperational test procedures and in charge of the technicians while performing the testing.

He was involved 3 years in worldwide satellite tracking operations both as an operator and as an instructor. As an operator he was responsible for station operations and maintenance of the tracking systems. As an instructor he taught operation and maintenance procedures.

His satellite tracking operator experience has encouraged him to pursue the human factors interface interests. He has attended the MIT Human Factors Engineering summer course. He has attended human factors lecturers at the NRC Headquarters and in his pursuit of his MBA has concentrated on Organization Behavior courses.

Mr. Ibarra was a member of the SONGS Units 2 & 3 CRDR Team. Since mid year, he has been a member of the Utility Simulator Facility Group. This group is developing a non-plant reference simulator standard to comply with the new rule 10 CFR 50.45.

PUBLICATIONS: J. G. Ibarra, "SPDS, Once Again, The San Onofre Experience", presented at EPRI SPDS Implementation and ERF Seminar, Boston, MA, May 1986.

USFG member, "Guidance for Development of a Simulation Facility to Meet the Requirements of 10 CFR 55.45", Draft October, 1987.

MICHAEL J. KIRBY

EDUCATION:

Orange Coast College - Costa Mesa, California
Associate of Arts Degree

U. S. Navy Nuclear Power School

SUMMARY:

1975 - Present Southern California Edison Company

Present — Nuclear Training Administrator - In charge of all aspects of operator training for SONGS Unit 1 (Non-Licensed, Licensed, STA and Requalification.)

1980-1982 — Nuclear Training Instructor - Responsible for conducting classroom training for SONGS 1 operators (RO, SRO, and Requalification.)

1975-1980 — Operator SONGS 1 - Progressed from non-licensed operator to RO (11-76) to SRO (1-79) to Operating Foreman.

1965-1973 — U. S. Navy - Navy Nuclear Program, Mechanical Operator, Qualified EWS, and Prototype Instructor.

EXPERIENCE:

As Nuclear Training Administrator has responsibility for development, presentation and evaluation of all aspects of operator training. Both Initial for RO's and SRO's. Training of STA's and requalification training of all licensed and non-licensed operators as well as STA's. Helped achieve successful INPO accreditation in 1985. Worked with Westinghouse Electric Corporation in the development and modification of non-plant reference simulator for the conduct of simulator training. Also on the Westinghouse Owners group procedure subcommittee for the development of Emergency response guidelines.

As Nuclear Training Instructor was in charge of conducting training for all Unit 1 operator positions, both in the classroom and the simulator. Only SRO licensed training instructor. Conducted periodic evaluations both on and off shift of the operations staff. Additionally, conducted requalification training for all operations positions.

Served as Unit 1 operator in the following positions: NPEO 1975 to 1976 responsible for operation of secondary plant components, 1976 - 1978 ACO (RO licensed) responsible for all aspects of nuclear reactor plant operations. 1979 - 1980 CO and Operating Foreman (SRO Licensed) responsible for the direct supervision of plant operations and direction to other licensed and non-licensed operators. Spent periodic assignments in conducting training for classes of non-licensed operators.

While in U. S. Navy Nuclear Program, Qualified Engineering Watch Supervisor aboard dual reactor power plant. Leading Mechanical Operator responsible for all aspects of power plant operation and maintenance aboard naval vessel. Prototype instructor West Milton New York, responsible for the training of Navy nuclear mechanical operators at a land based training reactor and facility.

WILLARD McGHEE JR.

EDUCATION:

U.S. Navy Nuclear Power Training Program, 1966
Reactor Operator License Training Program, San Onofre Nuclear Generating Station - Unit 1, 1974
Senior Reactor Operator Training Program, San Onofre Nuclear Generating Station - Unit 1, 1977

SUMMARY:

Present:	Coordination Supervisor - Member of the Unit 1 CRDR Project Team providing operational input to the process.
3 Years:	Primary Operations reviewer for various documents associated with NUREG-0737 and other regulatory issues.
2 Years:	Participated as an author, reviewer, and Project Manager for the San Onofre Unit 1 and Units 2/3 Emergency Operating Instruction upgrade.
1 Year:	Supervised the development and implementation of the Unit 1 Operations Procedures Group.
2 Years:	Nuclear Training Administrator responsible for all aspects of operator training.
7 Years:	Machinist Mate and Engineering Laboratory Technician in the U.S. Navy Nuclear Power Program.

EXPERIENCE:

Mr. McGhee has over 22 years experience in the operation of nuclear reactor propulsion and stationary power generating plants. The last 15 years have been at San Onofre Nuclear Generating Station performing a variety of tasks within the Operations Department.

His present position as Coordination Supervisor involves various manpower related activities as well as the review of documents related to SPDS, Regulatory Guide 1.97, Emergency Operating Procedures, and 10CFR50 Appendix R.

He participated in the Westinghouse and Combustion Engineering Owners Group Operations Subcommittees responsible for development of generic Emergency Procedure Guidelines (EPG's). As part of the Emergency Procedure (EP) upgrade project, he developed the Procedures Generation Package (PGP), drafted plant specific EP's, reviewed EP's developed by others, and periodically performed various duties as Project Manager.

EDUCATION: (Continued)

In formation of a separate Procedures Group within the Operations Department, he initiated actions to formalize procedural revision/development, feedback incorporation, validation, revision priority, retrieval, and commitment compliance for the Unit 1 Operating Instruction set.

As part of the initial manning group for Units 2/3 he interviewed, hired, and was responsible for the training of the initial operator manning group. As the Training Administrator, he was responsible for the development of training program revisions in response to evolving TMI training needs. In formation of a separate Nuclear Training Department within the Southern California Edison Company, he initiated a variety of actions in support of operator training and NRC licensing for all three units, training program development for Shift Technical Advisors, and design and construction of a new training facility.

During his tour of duty in the Navy, he performed a variety of tasks related to machinery preventative and repair maintenance, testing, and operation. In the capacity of an Engineering Laboratory Technician, he performed duties related to the radiological controls associated with the repair of contaminated machinery, and the transport, storage, and disposal of radioactive materials evolving from the operation of Navy nuclear powered craft.

MARK B. MCKINLEY

EDUCATION:
- BS Naval Architecture, U. S. Naval Academy, 1976
- U. S. Naval Nuclear Propulsion Training Program, 1977
- SONGS Unit 1 Shift Technical Advisor Training Program, 1982
- SONGS Unit 1 Senior Reactor Operator License Training, 1983

SUMMARY:
- Six years Nuclear Generating Station Engineering (four years as licensed SRO)
- One and one-half years Naval Nuclear Propulsion Program
- Two years Naval Surface Warfare Operations

EXPERIENCE:

- Present: Supervising Engineer, Unit 1 Shift Technical Advisors. Functional as first-line supervisor for nine man team of engineers (STAs), and as cognizant engineer for Appendix R/Dedicated Shutdown Plant Modifications during Cycle IX Refueling (and as contributing author for complex network of operating instructions implementing the design). Additionally functioned as a Statin Technical representative for Unit 1 CRDR matters.

- 2 Years: Unit 1 STA Group Leader. Coordinated work assignments and shift rotation for five engineers/STAs. During this period, functioned collaterally as Project Administrator for the SONGS Unit 1 Emergency Operating Instruction Upgrade of 1984, including simulator validation of the EOIs. Additionally functioned as instructor for the Operations Department on new/upgraded EOIs.

- 5 Years: Shift Technical Advisor, SONGS Unit 1. Provided technical and operational support to the operating shift in matters pertaining to technical specification compliance, and abnormal events/plant transients. Concurrently held Senior Reactor Operator qualification commencing December 1983.

- 1 Year: Station Engineer. Performed plant engineering duties primarily in the area of incident investigation. Prior to entering STA training for last nine months of this period, functioned as an Independent Safey Engineering Group member, reviewing NRC and industry correspondence for applicability to SONGS Unit 1.

EXPERIENCE (Continued):

- 5 Years: Naval Service. One and one-half years as propulsion plant watch officer aboard USS Enterprise (CVN-65), preceded by three years of service in the nucler propulsion training program, and aboard USS John Paul Jones (DDG-32) as combat Information Center Officer and Fire Control Officer.

PROFESSIONAL:
- SRO License SONGS Unit 1 effective December 1983
- Professional Engineer (Mechnical), California, 1987

JERRY L. PRICKETT

EDUCATION

BS, Electrical Engineering, Tri-State University, 1956

Graduate courses taken:
Principles of Management, Organization/Human Behavior, Psychology, Computer Design, Data Processing

Other courses/training:
SCE - Supervisory sessions, Assertive Management, Conflict Management, Motivation and Leadership, Telemetry/Supervisory Control Systems, Combustion Engineering - Nuclear Reactor and Steam Supply System Course, Human Factors Engineering and Design Reliability

DOD/U.S. Navy - Joint Chiefs of Staff/Armed Forces Staff College, Sr. Program Managers School for Systems Procurement and Logistics Control, Aircraft Instruments and Controls, Celestial Navigation, Avionics Weapons, Surveillance Satellites and Data Acquisition Systems, Leadership and Command

SUMMARY

Present — CRDR Team Leader for SCE San Onofre Nuclear Generating Station, Unit 1.

7 years — Special Projects Coordinator/Team Leader for numerous task groups, I&C systems at SONGS 2 and 3, including DCRDR Coordinator, SONGS 2/3.

14 years — Extensive supervisory and lead engineer positions in control systems and console design, installation and testing for several agencies such as NASA, AEC, USAF, public utility and commercial projects, including the LM/FBR program at Hanford, Washington, and the George C. Marshall Space Flight Center at Huntsville, Alabama (NASA).

4 years — Supervisory positions in Operations/Test for complex, heavily instrumented and computerized facilities (Aerojet-General and TRW Systems)

5 years — Captain/USNR - Program Manager of Avionics Weapons Systems Task Force efforts for Naval Air Systems Command Headquarters, Washington, D.C.

EXPERIENCE

Mr. Prickett is currently the Team Leader for the SONGS One Control Room Design Review and was the Group Coordinator for the SONGS 2 and 3 Detailed Control Room Design (DCRDR) completed at Bechtel Power Corporation/Norwalk in January 1986.

Prior to this, he has been the Special Projects Group Leader for numerous task groups formed and directed by him for installation and testing at SONGS 2 and 3, including radiation monitoring, all computer systems (as plant computer, critical functions monitor, core protection computer, plant security, health physics), EMI/RFI Noise Suppression, Toxic Gas, 1980 CRDR Task Force, etc. Earlier he was the I&C Engineering Site Representative for all I&C systems at SONGS 2 and 3. Previously, he was the Project Group Leader for the Kaiparowits Project which included in-house design.

Before joining SCE, Mr. Prickett worked for Aerojet-General Corporation (AGC) and TRW Systems for 13 years as follows:

1971-1974 <u>Supervisor</u>, Electrical, Controls and Instrumentation Design Department for the FFTF (Fast Breeder) Project at Hanford, Washington. He was responsible for the formation and direction of the Electrical Design Department in the development of special power instruments, controls, video, and communication systems design for numerous large test complexes for the AEC, USAF, and NASA. He was also responsible for the development of design criteria, projected work plans, procurement specifications, cost estimates, establishment of all drawing format and standards, control consoles, design and cabling distribution systems for remote handling of core components, fuel pin assemblies and analysis equipment in a high-radiation environment. His related duties included customer and management presentations, design review meetings, mechanical design interface, and updating of detailed work plans, schedules, and costs.

1970-1971 <u>Supervisor/Test Conductor</u> for USAF Satellite Testing Facility (AGC, Azusa). He supervised all testing and maintenance for a multiple systems' infrared satellite test complex consisting of a digital data acquisition and control system, a large hi-vacuum test chamber and associated pumping systems, internal dual phase heating and cooling shrouds, and large optical alignment fixtures. His duties required an intimate knowledge of analog and digital data compression and transmission technique, computer software and peripheral euqipment, hi-vacuum pumping systems and controls, infrared and optical systems and cryogenics.

1968-1969 <u>Lead Instrumentation Engineer</u>, TRW Capistrano Test Site, LEMDE Program. He was responsible for supervision and technical direction of the instrumentation and data acquisition systems for the LEMDE Static Fire Test Area. His coordinate duties included design of special instrumentation and control system requirements, transducer applications, proposal efforts, volatile gas-flow measurement studies, and specification and bid evaluation.

1967-1968 <u>Project Controls Engineer</u> for the design of a USAF satellite test facility. He supervised the design of all vacuum chamber I&C systems and control panelboards for testing complex satellite systems in a hostile environment.

1966-1967 <u>Lead Control Systems Engineer/Consultant</u> to General Electric and Lear-Siegler at the NASA Mississippi Test Facility, Bay St. Louis, Mississippi. He served as <u>Technical Consultant and Group Leader</u> for redesign, installation, and checkout of hydrogen gas and fire-detection systems, oxidizer and propellant loading, transfer, and storage systems for Saturn SI-C and SII Test Complex.

1963-1965 <u>Resident Field Engineer</u>, NASA/Saturn V Test Complex; Redstone Arsenal, Huntsville, Alabama. He completed this job as <u>Resident Manager</u>. He was responsible for supervision of the installation and checkout of I&C systems and instruction of NASA personnel.

1961-1963 Group Leader and Systems Design Engineer, NASA/Saturn V projects. He designed numerous automatic control and power distribution systems for the NASA Saturn V Test Complex at Marshall Space Flight Center. Complete design drawing packages included schematics, console and equipment fabrication drawings, control console and panel layout, and conduit and cable tray installations in conformance with NEC Code, NEMA and JIC Standards, and Human Factors Criteria (MIL Spec).

Prior positions held were Project Engineer and Plant Electrical Engineer, Plant Engineering Department, Wolverine Tube Division, Calumet and Hecla, Detroit, Michigan. Mr. Prickett was responsible for all electrical projects in the plant during a 5-million dollar modernization program. He redesigned the primary power distribution system (including Substations), automated several new processes, designed over thirty control consoles, developed material handling systems, established nondestructive testing programs, supervised subcontractor installation work and checkout, and assisted in Procurement and Contracts Administration.

MILITARY U.S. Naval Air Reserve/Active. He is presently assigned as Captain/Program Manager to Pacific Missile Test Center, Pt. Mugu, California, responsible for P3 Avionics Weapon System Projects. Capt. Prickett directs efforts of three Project Managers and eight Project Officers on high visibility projects for the NAVAIRSYSCOM HQ, Washington, D.C. (Regular Navy)

He has held prior positions as Project Officer, Operations Officer, Flight Officer, Training Officer, Personnel Officer, Electronic Division Officer, and Navigation/Tactics Officer. He was formerly a Chief Petty Officer, Maintenance Division Chief, and Instructor of Electrical/Electronic Theory and Aircraft Electrical and Instrument Systems.

PROFESSIONAL AFFILIATIONS

Registered Control Systems Engineer, Calif. No. 808, June 1976
Member, Instrument Society of America (ISA)
Member, Pacific Coast Electrical Association (PCEA)
Member, Institute of Electrical and Electronic Engineers (IEEE)
Sponsor, Instrumentation and Controls Group, PCEA (1978-79)

JOHN P. REYNOSO

EDUCATION:	-BS Agricultural Engineering/Mechanical, University of Arizona, Tucson. 1977 -U.S.Naval Nuclear Power Training.,1978 -U.S.Navy Nuclear Engineering Training.,1981 -Shift Technical Advisor Training. SONGS1,1982 -Senior Reactor Operator Training. SONGS1,1983 -Power System Protection, SONGS1,1986 -Nuclear Reactor Safety, Summer Course, Massachusetts Institute of Technology, 1987
SUMMARY:	Present: Shift Technical Advisor/Senior Reactor Operator for SONGS Unit 1. Member of the CRDR team for SCE providing interface between SCE and Project as a collateral duty. Involved with the updating of the SONGS Unit 1 EOI's. 4 Years: Shift Technical Advisor/Senior Reactor Operator. Conducted various Independent Safety Engineering investigations as part of ISEG. Assisted in the Unit 1 return to service working with Plannig and control. Authored various SONGD 1 Apppendix R Alternative Shutdown Procedures. 5 Years: Engineering Officer of the Watch U.S.Nuclear Submarine forces. Oualified Reactor Controls Assistant,Damage Control Assistant and Electrical Officer.
EXPERIENCE:	Mr. Reynoso has over 9 years of experience in the operation and control of nuclear power systems. His nuclear experience started in the military service on board nuclear powered fast attack submarine involving one year of land based training prior to serving 4 years at sea. He has proven to a competent in the operation and control of nuclear power plants by his accomplishments in the area of qualification as a STA and SRO on a commerical nuclear power plant.
PROFESSIONAL AFFLIATION	Professional Reactor Operator Society, Member

EDWARD SIEMION

EDUCATION:

BS Chemistry, Detroit Institute of Technology
Certificate in Advertising and Sales Management, Detroit Institute of Technology
Professional Certificate in Power Plant Engineering (Nuclear and Fossil), University of California, Los Angeles
Certificate in Plant Engineering, University of California, Los Angeles
DOD/U.S. Navy, Industrial College of the Armed Forces (In a Naval Reserve capacity attached to Office of Naval Research, Pasadena, California)

SUMMARY:

Present: Control Systems Engineer for SCE San Onofre Nuclear Generating Station, Unit 1 supporting the human factors task force preparing the Control Room Design Review.

6 Years: Control Systems Engineer for SCE San Onofre Nuclear Generating Station, Units 1, 2 and 3.

10 Years: Control Systems Engineer for SCE's fossil fired generating stations.

4 Years: Control Systems Engineer doing engineering design for Petroleum Refinery and Petrochemical projects at C. F. Braun Company (Engineers/Constructors).

18 Years: Sales and Application of Instruments, Controls and Safety Shutdown Control Panels for Robertshaw Controls Company.

EXPERIENCE:

Mr. Siemion has 38 years experience in the design and application of instruments and controls for nuclear/fossil generating stations, petroleum refineries and petrochemical/chemical plants. In acquiring this experience he participated in the design of many control panels and safety shutdown panels.

He was a Control Systems Design Engineer on SONGS 1 rework, resulting from TMI upgrading requirements. When this work was completed, he became assigned to SONGS 2 and 3 during these units final construction phase to commercial operation and post operational design additions/modifications. His work consisted of design change packages, plant facility change documents, evaluating suppliers equipment and performance, investigating NRC advice on equipment failure at other nuclear plants and studying/documenting possible effects at San Onofre Nuclear Generating Station.

EXPERIENCE: (continued)

In addition to the aforementioned nuclear experience, he participated in the design/construction/start-up of the following SCE fossil fired plants as a Control Systems Engineer:

Cool Water Generating Plant - a combined cycle oil fired gas turbine and steam turbine plant.

Long Beach Combined Cycle Plant - similar concept to above but different equipment.

Mohave Generating Station - the first coal slurry fired generating station in the U.S.

Between the above fossil power plant assignments, he wrote and provided 95% input into the SCE Controls Engineering Guide, which serves as the standard for design and work practices for SCE's fossil power plants and their modifications.

He is a Naval Aviator and Lt. Commander, USNR, whose experience ranges from anti-submarine patrol work to Naval Aircraft Overhaul Facilities. His final years in the Naval Reserve were in a Naval Reserve Research Company attached to the Office of Naval Research at Pasadena, California. This experience encompassed the whole gamut of Naval Operations regarding basic research.

PROFESSIONAL AFFILIATIONS:

Registered Control System Engineer, California
Senior Member, Instrument Society of America
Member, Southern California Meter Association

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DOREMUS D. SNUGGS

EDUCATION: BS Electrical Engineering at California State University, Long Beach (CSULB), 1974
15 semester units toward MSEE, CSULB

TRAINING:
a) Hewlett Packard HP-1000 Computer Courses: (1982)
   - RTE 6/vm System Manager
   - RTE 6/vm Sessions Monitor
   - Introduction to HP-1000
b) Integrated Computer Systems: Database Management Systems (1984)
c) Design Process Workshop - SCE (1985)
d) INTEL, IAPX Assembly Programming Language (1985)
e) Westinghouse WDPF System Course (1986)

SUMMARY:

Present: Responsible Controls Engineer:

4 years: SONGS 1 Security Computer System Upgrade - Provided engineering and design to extend Honeywell security computer to Unit 1.

SONGS 1 Turbine Generator Vibration Expansion Monitoring System - Engineering and design for replacing turbine generator instrumentation system with programmable recorder and engineering workstation in control room.

SONGS 1 CRDR Team Member - Provide Human Factors I&C engineering support.

2 years: Big Creek Automation Project - Responsible Engineer for the engineering and design of Big Creek Powerhouse automation using Hewlett Packard computer system for automatic generation and control.

2 years: Project Engineer, Century Data Systems - Responsible for hardware design of Winchester hard disk drives.

1 year: Project Engineer, TRW Communication Systems - Engineering and design of computer hardware for point-of-sale terminals.

3 years: Senior Engineer, Litton G&C Systems - Engineer and design of computer hardware and memory systems for military guidance and control systems.

3 years: Engineer II, Rockwell International - Software math-modeling and development for space shuttle program.

EXPERIENCE: Mr. Snuggs has over 16 years engineering experience in the engineering and design of computer-related control circuits and instrumentation. The last 6 years' experience involve nuclear process computer engineering and controls instrumentation for Southern California Edison.

His nuclear controls experience includes Responsible Controls Engineer for modifications of SONGS Site Security Computer System, CRDR, Turbine Generator Supervisory System, COLLS Backup Computer System, and SONGS Health Physics Computer System design and installation.

Before Southern California Edison, he was a Project Engineer for computer hardware design manufacturers at Century Data Systems. As a Project Engineer, he was responsible for the design of Winchester hard disk drives and computer interface controllers.

Mr. Snuggs was also a Project Engineer at TRW CS&S. He was responsible for design of logic and control circuitry for TRW computer systems. This included circuit design using state-of-the-art circuit components for computer hardware subsystems.

His controls experience also included a Senior Engineer position at Litton Guidance and Controls Systems where he was responsible for the engineering and design of military hardware for Cruise missile, F-18, and F-15 programs. He designed logic control circuits for a computer memory system which are used for guidance and control subsystems for the above projects.

Mr. Snuggs was also the I&C Responsible Engineer for the Big Creek Automation project. He was responsible for engineering and design for automating hydro electric plants in the Big Creek area using Hewlett Packard computers. Also, he was responsible for the software design for remote operation of each plant from a centralized computer system at Big Creek Powerhouse No. 3.

JOE. R. TATE

EDUCATION: Completed High School, Redland, California, 1953

San Bernardino Valley College, 1957 - 1962

Completed various courses in mathematics and electrical theory

SUMMARY: Mr. Tate has been employed by the Southern California Edison Company since 1955. He has been engaged in power plant operations throughout this 32-year period. Seventeen years of Mr. Tate's time has been in a supervisory capacity. Twenty-two years in nuclear power, and ten years experience with gas and oil fueled power plants.

Present: Assistant Manager of Operations - San Onofre Nuclear Generating Station, Units 1, 2, and 3

3 years: Superintendent - San Onofre Nuclear Generating Station, Units 2 and 3

3 years: Supervisor of Plant Operations - San Onofre Nuclear Generating Station, Unit 1

7 years: Watch Engineer (Shift Supervisor) - San Onofre Nuclear Generating Station, Unit 1

2 years: Operating Foreman - San Onofre Nuclear Generating Station, Unit 1

4 years: Control Operator - San Onofre Nuclear Generating Station, Unit 1

10 years: Held position of Auxiliary Operator, Assistant Control Operator, and Control Operator in fossil fueled generating station.

EXPERIENCE Licensed Reactor Operator, San Onofre Nuclear Generating Station, Unit 1, 1967 - 1968

Licensed Senior Reactor Operator, San Onofre Nuclear Generating Station, Unit 1, 1968 - 1981

Licensed Senior Reactor Operator, San Onofre Nuclear Generating Station, Units 2 and 3, 1982 - 1987

Past member of the Westinghouse Owners Group Operation Subcommittee

Current member of the Combustion Engineering Owners Group Operations Subcommittee.

Participated, as an alternate, on the Operation Nuclear Society Subcommittee for review and revision of several ANS-3 Standards following the Three Mile Island incident.

MICHAEL J. THOMAS

EDUCATION: Idaho State University
Pocatello, Idaho
Bachelor of Science, Engineering, 1982

SUMMARY: Present: Nuclear Licensing -- Member of the Unit 1 CRDR Team providing working support to the CRDR process.

5 Years: Southern California Edison Company working in the Nuclear Licensing Department.

EXPERIENCE: Mr. Thomas is currently working in the SONGS 1 Nuclear Licensing Group. Current responsibilities include participating in the SONGS 1 CRDR effort as the responsible engineer for this discipline. Additional responsibilities include Regulatory Guide 1.97, SPDS, Generic Letter 83-28, ATWS Mitigating System Actuation Circuitry, and Technical Specifications including station batteries, snubber surveillance and testing, Appendix J, containment isolation and reactor protection system.

Mr. Thomas has been involved in steam generator tube inspections, environmental qualification efforts, the Systematic Evaluation Program, and acted as the Assistant Refueling Engineer during the San Onofre Unit 3 Cycle 2 refueling outage. As a licensing engineer, his duties include technical specification interpretations; preparation and submittal of technical specification revisions, responses to NRC generic letters, and NRC informational inquiries; coordination of multi-disciplined tasks; and consultant interfacing.

DAVID B. BARKS
Senior Analyst

EDUCATION

Post Baccalaureate Computer Science, University of Tennessee, Chattanooga
B.S., Psychology, University of Tennessee, Chattanooga

EXPERIENCE

1980 - Present

General Physics Corporation
Responsibilities include project management, the development of database and office systems software, writing proposals, and software manuals, training on software packages, and development of a human factors user-computer interface for display systems.

- Database Development
  Developed database management systems for Wisconsin Electric Power Company, Mississippi Power and Light Company, Long Island Lighting Company, Nuclear Regulatory Commission, Omaha Public Power District, Gulf States Utilities, Southern California Edison, Pacific Gas and Electric Company, Niagara Mohawk Company, and the United States Army. In database development, Mr. Barks has used dBASE III, dBASE II, C, PROLOG, and BASIC. Other applications used LOTUS 1-2-3, SYMPHONY, Ashton-Tate's FRAMEWORK, and Apples MacIntosh.

  - The database management systems for Nuclear Utilites were used to keep track of NRC auditable documents, and their relationships to schedules, procedures, inventories, and generic plant information.

  - The database management system for the US Army was a pilot system for laboratory training of samples in the BZ demilitarization project.

  - The database system for Niagara Mohawk Company was used to track responsibility for engineering modifications and scheduling of the modifications.

- Application Development
  Designed and developed applications for Vertical Markets. Duties included writing software specification, marketing strategy, marketing plan, marketing material. Overseeing software development and product packaging.

  - Developed a photo retrieval application for nuclear power plants. Using video disk and CD/ROM technology linked to existing IBM AT database technology a method was developed to link equipment information with the image of the component.

  - Exam Bank Utility Developed Software specification, marketing plant, demonstration disk, and oversaw the development of the completed Exam Bank Utility (EBU). The EBU provided a database of standard questions and answers to allow instructors to build exams by objective.

- Training
  Mr. Barks has trained people the use of the hardware and software listed below:

- Apple MacIntosh	WORD PERFECT
- IBM PC	Displaywrite 3
- PC DOS	SUPERKEY
- dBASE III	SIDEKICK
- dBASE II	PROKEY (4.0)
- dBASE IIIE	MS WORD
- LOTUS 1-2-3	REFLEX
- LOTUS SYMPHONY	Page Maken
- FRAMEWORK	EXCEL
- WORDSTAR	CLIPPER

- Shift Technical Advisor (STA) Training, Georgia Power Plant, Plant Hatch
  Taught a 6-week program on behavioral science and management to Shift Technical Advisors.

- Human Factors Design Review
  Participated in human factors control room design reviews at several nuclear plants including Wisconsin Electric Power Company, Point Beach Nuclear Plant, and Mississippi Power & Light, Grand Gulf Nuclear Station, Omaha Public Power District's Fort Calhoun Station, Gulf States Utility's River

Bend Station and Georgia Powers' Plant Vogtle Preliminary Design review. Participated in human factors design reviews of letter sorting machines for the United States Postal Workers Union.

- Syncrude Canada Ltd.
  Wrote system's training manuals for a cogeneration power plant.

- PWR Task Analysis Pilot Study, Oak Ridge National Laboratory
  Investigated and authorized this pilot study (NUREG-CR-2498) to demonstrate what information task analysis can provide for various applications.

- BWR Task Analysis Pilot Study, Oak Ridge National Laboratory
  As project manager for development and testing mathematically predictable model of operator performance.

- BWR 1983-84 Research Study, Oak Ridge National Laboratory
  As Chief Task Analyst, the analysis of actions to develop performance criteria was the objective of this project.

- Safety Related Operators Actions Wrap Up, Oak Ridge National Laboratory
  The development and implementation of a computer actuated model of nuclear power plant operator performance to augment the ANSI N660 standard was the result of this project.

1978 - 1980

Henry J. Kaiser Company
Mr. Barks' duties included the performance and evaluation of construction testing for the Cincinnati Gas and Electric Company. As Principal Generation Construction Turnover Engineer he was responsible for seeing that all items turned over to the client were accurate and that appropriate documentation was on file. As part of the documentation aspect of his work, Mr. Barks worked on and assisted development of a computerized system index test matrix to keep track of all testing and documentation for system turnover. His other duties included work for the client in a quality assurance function. As Principal Quality Assurance Turnover Group Engineer, he reviewed all turnover

documentation against all applicable documentation prior to turnover. Mr. Barks also assisted operations in the performance of preoperational startup tests.

PUBLICATIONS

"Safety Related Operator Actions Wrap Up; Criteria of Operator Performance NUREG-CR-XXX". Coauthor with E. J. Kozinsky, A. M. Beare, F. Gomer, and L. H. Gray

"Nuclear Power Plant Control Room Task Analysis: Pilot Study for Pressurized Water Reactors," NUREG-CR-2598, May 1982, Coauthor with E. J. Kozinsky, and S. Echols.
"Criteria for Safety-Related Nuclear Power Plant Operator Actions: Initial Boiling Water Reactor (BWR) Simulator Exercises," (DRAFT) NUREG/CR-2534 (ORNL/NUREG/TM-8195), September 1981, Coauthor with E. J. Kozinsky, A. N. Beare, P. M. Haas.

"Nuclear Power Plant Control Room Task Analysis: Pilot Study for Boiling Water Reactor Study". Coauthor with F. Gomer, G. Moody.

"Task Analysis Methodologies for Safety Related Operator Actions", American Nuclear Society Winter Meeting 1981.

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DONALD C. BURGY
Director, Human Factors Engineering

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 <u>Consultant</u>

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

1976 - 1978 <u>Catholic University, Human Performance Laboratory</u>

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	<u>Eagleville Hospital and Rehabilitation Center</u> 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. <u>Task Analysis of Nuclear Power Plant Control Room Crews</u>: <u>Project Approach and Methodology</u> (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. <u>Task Analysis of Nuclear Power Plant Control Room Crews</u>: <u>Data Results</u> (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. <u>Task Analysis of Nuclear Power Plan Control Room Crews</u>: <u>Task Data Forms</u> (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 Mechanical Engineer: Maryland

EXPERIENCE

1980 - Present

General Physics Corporation
Mr. Danna directs engineering consulting services for government and utility clients. He has been responsible for projects from $5,000 to $1.8 million. All projects were completed on schedule and within budget. Representative projects include:

- An Evaluation of the Benefits, Risks, and Costs of Establishing Regulatory Cut-off (De Minimis) Levels for Radioactivity in Certain Waste Streams from Nuclear Power Plants, Atomic Industrial Forum, Inc.
  Serves as project engineer for the National Environmental Studies project to examine the sources, concentrations, and quantities of very low level radioactive wastes (VLLW) generated at light water nuclear power plants. The final report will be usable as technical support for an industry petition for rulemaking to NRC, under 10CFR20 and 10CFR61, to establish a regulatory cut-off for VLLW.

- Pressure Vessels/Systems Certification Support, NASA Kennedy Space Center (KSC)
  Serves as Project Manager for a program designed to certify all pressure systems at KSC; responsibilities include the coordination of on- and off-site engineering staff, the evaluation of current automated data processing of certification documentation, and the monitoring of the Lockheed, EG&G, and McDonnell Douglas certification effort.

- Safety Review Committee Program Development, Gulf States Utility Company, Sacramento Municipal Utility District, Toledo Edison Company.
  Directed the development of Plant Review Committee and Management Safety Review Committee training programs. The programs provide for the training and qualifications of key plant and corporate management personnel responsible for assessing the safety of plant operations or configuration changes.

- Technical Staff Training Program Development, New York Power Authority, Texas Utilities, Baltimore Gas & Electric, Commonwealth Edison, Niagara Mohawk, and Houston Lighting & Power
  Serves as project director for the development of engineer training programs which include Codes and Standards, Nuclear Licensing, Equipment Qualification, Material Science, and Plant Chemistry.

- Startup Test Procedure Review, Pennsylvania Power and Light Company
  As Project Director, supervised the detailed technical review of all the Susquehanna Steam Electric Station Unit 2 startup test procedures. The test procedures were reviewed against the design criteria established in the FSAR and GE design documents to insure that they demonstrated compliance with these criteria.

- Multilayered Vessel Recertification Analysis, National Aeronantics and Space Administration
  As Project Director, provided technical direction to the analysis of pressure vessels at White Sands Test Facility to meet ASME Section VIII Divisions 1 and 2.

- San Onofre Nuclear Generating Station Unit 2 Simulator Trainer Configuration Management Program, Southern California Edison Company
  Supervised, as Project Manager, the review and evaluation of all SONGS Unit 2 Design Change Packages (DCPs) and Proposed Facility Changes (PFCs) to determine their impact on the simulator trainer baseline configuration. Developed project procedures and overall program guidelines for use in utility management of CM program.

- Energy in Municipal Wastewater Treatment: An Energy Audit Procedure and Supporting Data Base, U.S. Environmental Protection Agency
  Supervised, as Project Manager, the development of a data base which compiles all reported literature on energy use in municipal wastewater treatment plants; developed a generic methodology to assess the total energy required for construction and operation of wastewater treatment plants.

- **Shift Technical Advisor and Senior Reactor Operator Training Programs**
  Managed or instructed courses in Reactor Physics, Thermal-Hydraulic Analysis, Accident Assessment, and Nuclear Plant Materials to utility engineers seeking qualification as Shift Technical Advisor and Senior Reactor Operator at twelve power plants.

1976 - 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 - 1976 **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 Metals

PUBLICATIONS AND PRESENTATIONS:

J. P. Davis, R. Danna, "De Minimis Concentrations of Radionuclides in Various Waste Media, Status Report," Transactions of the American Nuclear Society, 47, p 101 (1984).

D. E. Sharp, R. Danna, J. E. Stoneking, T. G. Carley, "Failure Prevention Program Implementation: A Case Study of High Pressure Gas Storage Vessels," American Society of Mechanical Engineers, 84-PVP-66, pp 1-6 (1984).

E. G. Landauer, R. Danna, "The Need for Technical Staff Training," Transactions of the American Nuclear Society, 46, pp 44-46 (1984).

K. J. Rebeck, R. Danna, G. S. Miller, R. T. Hollingsworth, "Recertification Analysis and Inspection Planning for Environmental Test Facilities," Proceedings of the Institute of Environmental Sciences, pp 328-335 (1984), also published in the Journal of Environmental Sciences, 27, pp 33-39 (1984).

C. S. Trent, R. Danna, "Development of a Configuration Management Program for Nuclear Power Plant Simulators," All About Simulators, 1984, Society for Computer Simulation, 14, pp 18-24 (1984).

R. Danna, C. S. Trent, "Implementation of a Configuration Management Program for Nuclear Plant Simulators," Transactions of the American Nuclear Society, 45, pp 558-559 (1983).

R. Danna, "Overview of Configuration Management Program Development and Implementation for Ground Based Pressure Vessels and Systems," NASA Pressu Systems Seminar, White Sands Test Facility, September (1983).

R. Danna, K. J. Rebeck, "Failure Prevention Program Development: An Application of Pressure Vessel and System Recertification and Inspection Planning," Failure Prevention and Reliability - 1983, American Society of Mechanical Engineers, pp 109-117 (1983).

R. Danna, "Critical Exposure Pathways: An Analysis of the Environmental Impact of Gaseous Effluents from Light-Water-Cooled Reactors," Research Paper, University of Central Florida (1979).

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/Facility Services
  Prepared system operating procedures, annunciator response procedures, test and surveillance test procedures. Developed and prepared a surveillance test program to implement Environmental Technical Specifications. Participated in the procedures validation of the Emergency Operating Procedures for a PWR power plant.

- Quality Assurance/Program Development Services
  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>Training Program Development</u>
  Prepared lesson plans for Licensed Operator systems training. Developed the Basic Radiation Protection training course, including lesson plans, and all training aids and demonstrations at the William H. Zimmer Nuclear Power Station.

- <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 on-site for a client. Has taught portions of the academic fundamentals to operator and STA candidates on-site, and portions of the GP Codes and Standards course for Technical Staff Engineers.

- <u>Human Factors Engineering</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.

1979 - 1981 <u>General Atomic Company</u>
Mr. Dawson served as the Health Physics Representative on a total of seven projects with General Atomic. He was responsible for independently carrying out the Health Physics Programs on these projects, which included HTGR fuel fabrication, TRIGA facilities, hot cell facilities, and radwaste.

1978 - 1979	**Franzen & Associates** Mr. Dawson investigated and marketed personal savings and investment programs. He researched and designed business plans for small businesses including structuring and maintaining accounting systems. He prepared tax returns and tax planning programs.
1969 - 1978	**U.S. Navy** Engineering Laboratory Technician Mr. Dawson served in progressive assignments as Electrical Operator and Engineering Laboratory Technician. He was responsible for operation and maintenance of electrical distribution systems and radiac and sampling equipment. He prepared and delivered shipboard training programs in radiation protection. He served as Prototype Instructor for plant systems, radiation protection, and chemistry.
PROFESSIONAL AFFILIATIONS	Plenary Member, Health Physics Society Member, American Society for Quality Control Secretary of the Modifications (Design) Subcommittee of the Committee for QA of Operating Power Plants - Standards Committee of ASQC

(11/85)

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.

WADE A. MARTIN
Staff Specialist

EDUCATION

B. S. Candidate, Nuclear Technology, University of the State of New York
A. S. General Studies, University of the State of New York
U. S. Navy Nuclear Power Training Program

EXPERIENCE
1983 - Present

<u>GENERAL PHYSICS CORPORATION</u>

Mr. Martin provides technical and engineering support for nuclear and fossil fuel commercial power plants and Navy nuclear and fossil fuel ships. He prepares operation, maintenance and monitoring procedures; evaluates fluid and mechanical systems and develops system and component performance criteria; performs engineering feasibility studies and design reviews; and develops training programs and materials. Representative projects include:

- <u>Systems Review and Task Analysis for Nuclear Power Plant Control Room Design Review and Implementation Services</u> - Mr. Martin assists the human factors engineering department in the development and preparation of task analysis methodologies and system descriptions to be utilized in the collection of dynamic human performance data. These tasks include procedure analyses, scenario development and verification, and validation processes. He is currently providing his task analysis expertise to both PWR and BWR type commercial nuclear power plants for emergency procedure walkthroughs and equipment availability projects. Additionally, Mr. Martin develops Human Factors Standards for both BWR and PWR commercial power plants and aids those utilities in implementing design changes in accordance with those standards. Representative projects include:

  - Hatch Nuclear Power Plant Control Room Design Review and EOP verification and Validation effort
  - Vermont Yankee Nuclear Power Plant Human Engineering Discrepancy implementation program and EOP flowchart validation project
  - Diablo Canyon Nuclear Power Plant Control Room Design Review
  - San Onofre Unit 1 Nuclear Power Plant Control Room Design Review
  - Fort Calhoun Nuclear Power Plant Human Factors Maintenance Plan Development
  - Salem Nuclear Power Plant Human Factors Maintenance Plan Development and EOP Verification and Validation project.

- Point Beach Nuclear Plant implementation program

- <u>Reliability Centered Maintenance Program Development</u>
  Mr. Martin provided engineering support to Consolidated Edison Company of New York in the development of a pilot Reliability Centered Maintenance (RCM) program. This pilot RCM program was developed for two high maintenance systems.

- <u>Managing Predictive Maintenance, Dayton Power and Light</u>
  Developed training materials for a course entitled "Managing Predictive Maintenance." Wrote "state-of-the-art" training material sections on Lubrication, Acoustic Emissions Monitoring, and Fiber Optic Inspection.

- <u>Equipment Performance Monitoring and Submarine System Review, Naval Sea Systems Command, Submarine Systems Monitoring, Maintenance and Support (SMMS) Office.</u>
  Developed submarine mechanical equipment performance criteria and monitoring procedures for submarine propulsion plant systems; and conducted several mechanical system design reviews.

- <u>Waste Heat Boiler Deaerating Feedwater Tank System</u>
  Developed training manual for the waste heat boiler deaerating feedwater tank system located onboard DD 963 Class ships. Additionally, Mr. Martin conducted the training courses onboard these vessels instructing Navy personnel in the proper operation and maintenance of the system.

1977 - 1983 **<u>UNITED STATES NAVY</u>**

Mr. Martin served as Propulsion Plant Watch Supervisor on the USS EISENHOWER (CVN-69) where he was responsible for overall supervision and control of all aspects of nuclear reactor operations, steam plant operations, and auxiliary functions. He was the main machinery room work center supervisor accountable for the training and administrative duties related to a forty-man work center. Additionally, Mr. Martin was responsible for scheduling and performing corrective and preventive maintenance on distilling units, turbine generators, main engines, and associated support equipment.

As Quality Assurance Supervisor, he was responsible for initiating and ensuring quality assurance procedures were carried out on all mechanical systems.

PROFESSIONAL AFFILIATIONS American Society of Mechanical Engineers

Michael K. Pumphrey
Staff Specialist

EDUCATION B.A., University of Maryland, College Park, Maryland

EXPERIENCE

Present

General Physics Corporation
Mr. Pumphrey provides production assistance for control room design review implementation services and for department marketing materials. He is in a supporting role for video services, such as computer-based training and videodisc training projects.

1983 - 1987

Advanced Resource Development Corporation
As Graphics Director, Mr. Pumphrey was instrumental in the operation of a complete computer graphics section. He was proficient in all aspects of desktop publishing (which decreased the use of outside vendors and dramatically enhanced the production quality of graphics, reports and promotional materials. His expertise included creating initial sketches, final drawings, illustrations, floor plans, isometrics, forms, charts, vugraphs, slides, promotional material, prints and industrial models. He was also actively involved in a number of human factors, research and corporate development projects for industry and government clients.

1980 - 1983

Wordscape, Incorporated
As Manager, Mr. Pumphrey was responsible for supervising a staff of 13 paste-up artists, proofreaders, and typesetters. In this capacity, he was actively involved in specing type, designing layouts, drafting, pricing, illustrating, making photostats for magazines, books, brochures, maps, advertisements, logos, charts, displays, vugraphs, slides, mechanicals, isometrics and cartoons. He was also involved with workload and cost projections, client relations, and personnel management.

1974 - 1980

Anne Arundel County Bureau of Engineers
Mr. Pumphrey researched and wrote traffic engineering reports for the Northern Anne Arundel County area. As a Traffic Analyst, his photography and drafting experience were instrumental in the accomplishment of projects.

(3/87)

LOTHAR R. SCHROEDER
Principal Scientist

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

General Physics Corporation

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.N.C. Nuclear Industries

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.

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HENRY N. TOBEY
Human Factors Psychologist

EDUCATION

Ph.D., M.A., Human Factors/Environmental Psychology, The Catholic University of America

M.A., Psychology, Indiana State University

B.A., Psychology, Philosophy, Biology, Washington University (St. Louis, Mo.)

LICENSES AND CERTIFICATIONS

Licensed Psychologist, State of Maryland

EXPERIENCE

1985

General Physics Corporation

Dr. Tobey is a member of the Human Factors Engineering Department where he is supporting human factors evaluations of control room designs and emergency operating procedures upgrade projects for utility clients. Projects include:

- Detailed Control Room Design Review
  Contributing to Detailed Control Room Design Review project for a PWR facility. Participation includes conducting operating experience reviews, operator interviews, control room survey and task analysis based on Emergency Operating Procedures.

- Emergency Procedures Preparation
  Conducting reviews of symptom-based Emergency Operating Procedures for PWR facility of verification efforts. Also contributing to system review and task analysis efforts as part of procedures upgrade program using WOG ERGs.

Dr. Tobey's professional competencies include research design, environmental programming and design, survey and sampling design (including questionnaire and interview techniques), product design, urban planning research, transportation analysis, and product safety analysis.

1984 and 1978 - 1981

Biotechnology, Inc., Falls Church, VA

Served on a Detailed Control Room Design Review performed for Taiwan Power Corporation and was responsible for redesigning DCRDR tasks as necessary for their execution in a foreign environment. Prior work for Biotechnology included visitor behavior research for the National Air and Space Museum, transportation research and urban streetscapes analyses for the Federal Highway Administration and consumer product safety analyses for the Consumer Product Safety Commission.

1983 **Idaho National Engineering Laboratory**

In the Reactor Evaluation Program Division, Dr. Tobey conducted analyses and research on Nuclear Power Plants. Primary responsibilities included assessment of human performance measurement models and methods of human error estimation. As part of this project, future technology projections for nuclear reactor control rooms were evaluated. Other professional responsibilities included conducting indepth task analyses at the experimental Loss-Of-Fluid-Test Reactor, designing "friendly" person-computer software interfaces, and evaluating reactor operators' responses to seismic stress events.

1981 - 1983 **Center for Applied Research, Inc., Falls Church, VA**

As Research Director, Dr. Tobey was responsible for conducting several large scale safety related research projects as well as for marketing the company research program. Projects under his direct supervision included a natural study for the Federal Highway Administration on Pedestrian Risk Exposure Measures, a procedure review on task performance of guards at the National Air and Space Museum, and various equipment and environmental design projects utilizing anthropometric analyses and alternative material construction.

1975 - 1977 **Naval Ship Engineering Center, Washington, D.C.**

Dr. Tobey researched shipboard habitability for the Naval Ship Engineering Center (now subsumed within the Naval Sea Systems Command). This work was primarily design oriented, directed at the habitability of messing and berthing spaces of surface ships (destroyers, fast frigates, and carriers). The design research involved applications of a systems engineering approach utilizing such concepts from environmental psychology as privacy, territoriality, personal space, "neighborhood" design, and the negative effects of crowding and noise.

1976 - 1977 **Catholic University of America, Washington, D.C.**

Both in this country and in France, Dr. Tobey taught a graduate architecture course entitled, "Behavioral Criteria For Environmental Design". The purpose of the course was to instruct architecture and engineering students in ways to improve the designed "fit" between environments and the needs of their users. The applied skills taught included facility programming, design communication facilitation, design review techniques, post-occupancy evaluation, and ways to establish design criteria based on the behavioral task goals of various environments. As a graduate student at Catholic University, he also taught psychology laboratories in clinical research, perception, cognition and memory.

SELECTED PUBLICATIONS

Tobey, H. N. and Eichner, R. E. "Built Form, Land Use and Movement Patterns: How They Affect Streetscapes," Streets as Public Property, A. Vernez-Moudon (Ed.), Van Nostrand Reinhold; in print.

Tobey, H. N. A process-directed value system. Manuscript in progress.

Blackman, H. S. and Tobey, H. N. Computers in Process Control: The Human Element. The Design of Experiments to Test Man/Machine Systems. American Society of Mechanical Engineers. 1983.

Tobey, H. N., Shunaman, E., and Knoblauch, R. L. Pedestrian risk exposure measures. Prepared for the Federal Highway Administration under contract DTFH-61-81-C-00020. 1983.

Tobey, H. N. An analysis of museum guard performance. Prepared for the National Air and Space Museum of the Smithsonian Institution under contract SF-2072630000. 1982.

Eichner, R. B. and Tobey, H. N. Urban streetscape analysis. Prepared for the Federal Highway Administration under contract DTFH-61-81-C-0030. 1982.

Tobey, H. N. Connotative messages of single-family homes: A multidimensional scaling analysis. Proceedings of the Environmental Design Research Association Conference. 1982.

Tobey, H. N., et al. A study of visitor movement through the National Air and Space Museum's galleries and exhibits. Prepared for the Smithsonian Institution under contract FN-906016. 1979.

Tobey, H. N. Research survey and evaluation of four exhibit galleries in the National Air and Space Museum. Prepared for the Smithsonian Institution under contract FN-900576. 1979.

Tobey, H. N., and Logan, E. An emerging hazards analysis of household appliances. Prepared for the U.S. Consumer Product Safety Commission under contract CPSC-C-79-1204. 1980.

Tobey, H. N., et al. Hazard and human factors analysis of injuries associated with flame-fired appliances. Prepared for the U.S. Consumer Product Safety Commission under contract CPSC-C-79-1204. 1980.

Knoblauch, R. L., and Tobey, H. N. Safety aspects of using vehicle hazard warning lights. Prepared for the Federal Highway Administration under contract DOT-FH-II-9385. 1979.

MARK DELAMAR VENTERS III
Associate Scientist
Human Factors Specialist

EDUCATION	B.S., Geology, University of North Carolina at Wilmington
	Nuclear Auxiliary Operator Candidacy School, Shearon Harris Energy and Environmental Center, Carolina Power and Light Company, Raleigh, NC
	BWR Systems Training, Brunswick Nuclear Project, Carolina Power and Light Company, Southport, NC

EXPERIENCE
1986 - Present

General Physics Corporation

Mr. Venters is a member of the Human Factors Engineering Department where he supports human factors evaluations of Detailed Control Room Design Reviews (DCRDR), Safety Parameter Display System (SPDS) upgrades, control room staff upgrades, Emergency Operating Procedures (EOP) upgrade. Mr. Venters also provides systems experience for development, assessment, and implementation of Human Engineering Discrepancies (HEDs). Representative projects include:

- Emergency Operating Procedure Validation, SPDS Upgrade, and Control Room Staff Upgrade.

  Assisted in Emergency Operating Procedure Validation program at Wisconsin Electric Power Company, Point Beach Units 1 & 2. Performed SPDS upgrade study and Control Room Staff Upgrade study based on detailed analyses and interviews.

- DCRDR Instrumentation Verification and HED Development/Assessment

  Provided BWR systems experience for verification of instrumentation and development of HEDs in DCRDR task analysis program and assisted in HED assessment program at Georgia Power Company, Edwin I. Hatch Units 1 and 2.

- System Function Review and Task Analysis.

  Assisted in development of Task Analysis for DCRDR implementation and provided PWR systems experience in WOG safety system function comparison reviews and emergency scenario development for Southern California Edison Company, San Onofre Nuclear Generating Station Unit 1.

- SPDS Evaluation

  Provided BWR systems experience for scenario assessment in SPDS evaluation at Louisiana Power and Light Company, River Bend Unit 1.

- DCRDR and EOP Independent Verification

  Performed detailed independent verification of control room instrumentation in Task Analysis program for Mississippi Power & Light, Grand Gulf Nuclear Station, Unit 1.

- DCRDR Verification and HED Documentation

  Performed DCRDR Verification, managed Systems Function Review and Task Analysis program, and documented HEDs for Pacific Gas and Electric Company, Diablo Canyon Units 1 and 2.

1983 - 1986 **Carolina Power & Light Company, Brunswick Nuclear Project.**

As a Nuclear Auxiliary Operator, Mr. Venters' primary duties included: daily surveillance and performance testing of various types of operating instrumentation and equipment in strict accordance with plant operating manual and related technical specifications; providing auxiliary supervision of operating power producing equipment as necessary to meet load demands; troubleshooting as well as clearing out malfunctioning equipment for maintenance and repair; active member of operational procedure review committee and fire protection group, and active participation in a continuous related technical and on-the-job training program.

1974 - 1983 **Mark Venters Photography**

As a Photographer, Mr. Venters assisted in commercial, industrial, aerial and wedding photography. He is knowledgeable in ektachrome processing, analytical and sound motion picture production, plus has considerable experience in camera and projector repairs. He also assisted in the sales and rentals of photographic equipment and supplies.

PROFESSIONAL AFFILIATIONS

Member, American Nuclear Society
Member, American Association of Petroleum Geologists

3.0 PLANT OPERATING EXPERIENCE REVIEW

3.1 OBJECTIVE

The objective of the Plant Operating Experience Review was to identify factors or conditions that could cause and/or have previously caused human performance problems that could be alleviated by improved human engineering. The process of performing this review consisted of evaluation of station operating experience via historical plant documentation and confidential interaction with control room operating personnel. These two processes provided a basis for evaluating documented occurrences of human error or potential deficiencies in the control room man-machine interface and the suitability of the control room.

3.2 HISTORICAL DOCUMENTATION REVIEW

The objective of the Historical Documentation Review was to identify potential HEDs through evaluation of SONGS 1 operating experience using the station's documented reports. The historical documentation used in this review was selected considering the broad spectrum of areas from which problems potentially relating to control room man-machine interface deficiencies could be identified. The historical documents reviewed included:

- Licensing Event Reports (LERs) - The LER is the formal method used for reporting a serious event to the NRC. The Station Compliance organization is responsible for documentation of LERs.

- Station Incident Reports (SIRs) - SIRs are used to document incidents locally, and may disclose discrepancies in the design of a system. Reportable occurrences are handled by the Station Compliance organization using the LER.

- Nonconformance Reports (NCRs) - The NCR is used to identify nonconforming material, parts, or components for those structures, systems and components listed as safety-related. Quality Assurance is responsible for NCRs.

- Site Problem Reports (SPRs) - The SPR is used to identify safety-related or nonsafety-related equipment problems or discrepancies. Station Technical is responsible for SPRs.

Documentation generated in the last 5 years by these reporting mechanisms was reviewed for identification of potential problem areas. Existing programs are in place at SONGS to review this documentation and provide corrective action as necessary. However, the current programs do not necessarily account for and correct associated potential human engineering deficiencies in the control room. Consequently, this review was intended to identify potential HEDs not corrected in the resolution of the problems encountered.

3.2.1 Historical Documentation Review-Criteria

To qualify as a direct concern of the CRDR team, the source of an operating difficulty was directly related to the control room in one of the following ways:

A. The operating procedure was performed using controls and/or displays in the control room or remote shutdown panel.

B. The operating procedure was directed or coordinated from the control room or remote shutdown panel.

C. The operating difficulty appeared to be intensified due to lack of attention or needed support from the control room.

The event or incident was then analyzed to determine if it resulted or could have been amplified by any type of HED.

If the event had no discernable link to the control room, it was considered to be outside of the scope of the CRDR.

3.2.2 Historical Documentation Review - Methodology

The station operating experience documented in the reports noted in section 3.2 was reviewed and evaluated for potential HEDs that are pertinent to this CRDR effort. The plant-specific document review encompassed those documents that were generated in the last 5 years. The documentation was reviewed for problems that could impinge on control room operations or that reflect control room design deficiencies. The pertinent problems were documented on the Historical Document HED Review Summary form as illustrated in Figure 3-1. This form provided for easy dissemination of pertinent information relating to the event including:

- o Identification of the incident, problem or error
- o Explanation/presentation of the incident, problem, or error including summarization of events preceding the occurrence
- o Identification of probable cause
- o Corrective action/recommendations

A subsequent review of the historical documentation produced during the CRDR effort was also performed. This subsequent review was initiated to identify potential human engineering problems that have surfaced while the CRDR was being performed. This subsequent review produced no new problems not previously addressed by the CRDR process.

3.2.3 Historical Documentation Review - Findings and Assessments

Evaluation of the plant operating experience reports that satisfy the evaluation criteria of section 3.2.1 resulted in the preparation of a HED. The Historical Document HED Review Summary Form documents the problem, provides an assessment, a recommended corrective action, and a reference to the corresponding HED, if appropriate. The HED report generally duplicates the historical document review form with the addition of data as suggested by NUREG-0801.

The findings and assessments that resulted from evaluation of the historical document review are identified on HEDs 500 through 526.

3.3 CONTROL ROOM OPERATING PERSONNEL SURVEY AND INTERVIEW

The objective of the personnel survey and interviews was to obtain direct operator input to aid in identifying potential or actual deficiencies in the control room layout or design, or in operating procedures that resulted in confusion (mental activities), difficulty (manual activities), or distraction (the work space and environment). A questionnaire was prepared for the control room operators. The intent of the survey was to determine the aspects pertinent to the suitability of the control room, in the opinion of the operator. Where problems existed the specific component, system, or situation was identified. In addition, face-to-face interviews were conducted by our human factor specialists.

The operator survey was completely independent of Southern California Edison (SCE) Company involvement in order to protect the confidentiality of the operators and to have no bias on the survey.

3.3.1 Personnel Survey and Interview - Criteria

The survey and interviews were structured to meet the requirements and intent of NUREG-0700, Section 3.3.2. The 43 item questionnaire solicited operator feedback regarding:

- Workspace Layout and Environment
- Controls
- Displays
- Annunciators
- Communication Systems
- Computer System
- CRT Displays
- Corrective and Preventive Maintenance

- o Procedures
- o Staffing and Job Design
- o Training
- o Miscellaneous

3.3.2 Personnel Survey and Interview - Methodology

A two-pronged approach was used to gather information from operating personnel. First, a self-administered questionnaire was distributed to a sample of operating room personnel to be completed and mailed directly to the human factors consultants who were not employees of SCE. The survey was structured to elicit preliminary indications of potential human factors problems. Second, a representative sample of the personnel who were given questionnaires were interviewed. The interviews focused on the responses to the questionnaire. The interviews explored the specific nature of the problem, its prevalence, effects on control room operations, and suggestions for improvement. In addition, open-ended questions exploring the positive aspects of the control room were asked. The interviews were conducted by the human factors consultants with no SCE management present. The operating personnel interviewed were senior reactor operators (SROs) and reactor operators (ROs).

3.3.2.1 Questionnaire Content

The 43-item questionnaire distributed to the sample of operating personnel is shown in Appendix 3A. The format requested the respondents to identify problems relating to the area of the questions. They were specifically requested not to consider such issues as seriousness, pervasiveness, or consequence of the condition, nor whether it was feasible to correct the problem. The intent was to gather all possible problems that would be explored further in the interviews and other phases of the overall control room design review.

Table 3-1 lists the item numbers of the questions classified by the content areas specified in NUREG-0700, Paragraph 3.3.2.2. The last item of the questionnaire (No. 43) was an open-ended question asking the respondent to list any other problems observed during control room operations. As can be seen from table 3-1, the items address the issues raised in NUREG-0700 and thus serve as a valid screening device for identification of potential human factors problems.

3.3.2.2 Interview Format

Two human factors consultants from General Physics conducted the interviews together. The pair of interviewers interviewed each person. SCE management personnel were not present during the interviews. The interview was structured around the survey responses of the interviewees. Each problem identified was explored to determine: (1) the specific nature of the problem including the specific controls, displays, procedures, etc., where the problem exists, (2) how often it has been a problem, (3) what the consequences of the problem, in terms of safety and productivity, has been or could be, and (4) suggestions for improving the situation. In addition, general open-ended questions were asked to solicit positive features of the control room and to uncover the existence of other problems not included on the questionnaire. The interviews took from 30 to 60 minutes each and took place on company time.

3.3.3 Personnel Survey and Interview - Results

The results of the personnel survey and interviews are listed in Appendix 3A. Summaries of the responses to each of the 43 items on the questionnaire are provided with a response frequency tabulation to indicate the number of respondees identifying particular problems.

The results were reviewed by the CRDR team and the General Physics CRDR Project Manager to identify those survey responses that provide the basis for HEDs. A total of 79 HEDs were generated from the personnel survey and interviews. These discrepancies were documented as HEDs 400 through 478.

Overall, the survey and interviews served as a valuable source of information for identifying possible human engineering deficiencies and directed the team members to salient problems in the control room, many of which were already being addressed in design change packages initiated by SCE or by other CRDR processes.

Fig. 3-1

Historical Document HED Review Summary Form

LER 84-12 **Other (Specify)** SIR

Report Number: S01-84-13 **Report Date:** 11-9-84 **Occurrence Date:** 10-10-84

Error Categorization: Operator Error **Work Station:**

Instruments Involved: CCW Temp. Alarm **Procedures Involved:** S01-7-11
S01-14-5
S01-4-19
S01-4-9

Major System Involved: Saltwater Cooling
Component Cooling Water

Identification of Occurrence: High and Rising CCW Temperature Alarm

Summarize Events Preceding Occurrence: UNIT SHUTDOWN IN Mode 5

480V Busses were restored to a normal lineup following termination of the Temporary Emergency Diesel Preoperability Test
SWC flow was isolated to the lower CCW HX E-20B

Summarize Events During Occurrence:

Operators mistakenly lined up CCW flow to the lower CCW Hx which had no SWC flow, thus rendering both trains of RHR inoperable.

Identification of Probable Cause:

Lack of awareness of plant status. Inadequate investigation/evaluation of abnormal indications. Lack of caution or informational tags, too much work in progress which required operations support.

Corrective Action Taken/Proposed:

Establish preshift briefings
Tailboard before evolutions
Stress compliance over accomplishing work
Revise procedures to notify operators if in abnormal valve lineup.

Additional Recommendations:

See HED 507

Table 3-1

CONTROL ROOM OPERATING PERSONNEL QUESTIONNAIRE
ITEMS CLASSIFIED INTO CATEGORIES LISTED IN NUREG-0700,
PARAGRAPH 3.3.2.2.
(See Appendix 3A for a copy of the questionnaire)

NUREG-0700 Category	Questionnaire Item Number
Work Space Layout and Environment	1, 2, 3, 4, 5, 6, 7, 8, 9, 10, 15
Panel Design	2, 7, 9, 10, 11, 12, 13, 14, 15, 16, 17, 18, 19
Annunciator Warning System	10, 19, 20, 21, 22, 23, 24, 38
Communications	8, 25, 26
Process Computers	14, 27, 28, 29
Corrective & Preventive Maintenance	30, 31
Procedures	6, 32, 33, 37, 38, 41
Staffing & Job Design	34, 35, 36, 37, 38
Training	39, 40, 41, 42

APPENDIX 3A

CONTROL ROOM OPERATOR QUESTIONNAIRE

AND

RESPONSE SUMMARY

WORKSPACE LAYOUT AND ENVIRONMENT

1. What aspects, if any, of the control room layout, furniture and equipment make your job hard to do?

   (2) o None

   [Inadequate Communications Equipment]

   (3) (Specific responses have been integrated with item #25)

   [Furniture]

   (1) o The CRS Desk is too small and does not have enough drawers.

   (1) o Additional bookcases are needed above TSC window to support a technical library.

   [Miscellaneous Comments]

   (1) o There are not enough electrical outlets.

   (1) o Systems controls are spread out on different control boards - this is especially the case where: "add-on" work has been done.

   (1) o Some controls are not labeled clearly (see item #2).

   (1) o Meters should have linear scales (see item #16).

2. What problems are there in the control room with color coding or labeling?

   (4) o None

   [Color Coding Problems]

   (4) o There is little or no color coding at present or lines of demarcation around equipment areas or systems.

   (2) o There is no coding to alert the operator of the significance of some controls and SW's - i.e., CV334 emergency boration, Chg. - pps., PORVs, Refueling water pps., test pp., 220 & 4 KV bkrs.

   [Labeling Problems]

   (1) o The following labels are inaccurate:

   CB 4072 & 6072 still labeled Villa Park - should be Serrano

   CB 13A&B label says "Red-CIS Override" while it calls out a red coding, there is no red coding, it is gray.

   (1) o The following instruments are not labeled.

   Charging pump SI Lockout Reset, and FCV-1115D, E, F - Power light.

(1) o The names of many level, pressure, or temperature gauges are very hard to read. This is especially true on the North Vertical Board where meters on top row have clearly marked I&C numbers above but - the names are written vertically on the meter face.

(2) o Some controls and displays are not labeled clearly. An example is that some pump control switches are labeled "G-#" and do not have noun names which are usable such as "East - Pump". Also, all meters in Control Room should be labeled - not by numbers, but with words or abbreviations.

(1) o There are too many temporary labels.

(1) o Critical information may not be present on controls and indicators (i.e., power supply, failure mode on Loss of Power, environmentally qualified instrumentation, etc.).

3. Please identify any areas of the control room that have inadequate lighting?

(5) o None

(3) o Relay rack area behind the north vertical board where daily surveillance is done.

(1) o Behind vital busses.

(1) o Lighting directly on each desk could be improved.

4. Have you ever seen the use of the emergency lighting? If Yes, were any aspects of it inadequate?

(8) o In general, the emergency lighting is adequate.

5. Is heating/ventilation adequate?

(9) o Yes

(1) o There is very little air flow in the NOA office.

(1) o Equipment is old and often breaks down; increased surveillance should be done by A.C. personnel.

6. What would you change in your work environment to reduce stress, fatigue or boredom?

(4) o There is a need for a radio to provide background music.

(1) o Develop procedures and work control methods that are "user-friendly". Administrative procedures and paperwork for repairing plant equipment or operating the Plant is lengthy and complicated.

(1) o Change control room lighting from "cool white" to "full spectrum" bulbs. The lighting in the control room is not state-of-the-art and small inexpensive improvements could realize good positive effects. There is also a need for individual incandescent desk lamps.

(1) o Institute a relief policy for the NCO halfway through the shift. It would be of benefit if operators could take a short break outside of control room.

(1) o Outline areas/groupings of instrument/controls (see also item #2).

(1) o Less management interference during normal operations, such as startup; and especially during emergency conditions (too many different managers call in the control room).

(1) o Ensure procedures do not fall apart (binders). Possibly too many procedures in 1 binder. Also, need a larger print rack.

(1) o Caution tags are used for almost everything now and yellow is predominant color on control boards, i.e., hard to distinguish between Cautions tags and EDMR's and LCOARs.

7. Based on your operational experience, have any errors or incidents occurred which could have been averted through improved control room design?

(2) o No

The following controls need greater separation/covers to prevent inadvertent activation:

(5) o South charging pump switch on J-console) too close to annunciator controls (was accidentally activated).

(4) o Emergency Boration Control CV-334 (this has been advertently activated) too close to CV-333.

(2) o The feedwater pump SW is next to the governor load limit switch (was inadvertently stopped during stop valve test).

(2) o The Diesel Generator output BKR SW is too close to other switches (it was closed instead of operating speed controls).

(1) o A present problem is that on a unit trip the Steam Generator Blowdown is allowed to continue in its pre-trip condition.

(1) o Exciter and Exciter Field Bkr. controls on J-console.

8. Which (if any) noise levels are particularly high? Is communication between operators made difficult as a result of high noise levels?

   (7) o None or No

   (1) o The control room is subjected to noise daily that precludes effective communication. This is especially true when the janitor vacuums the carpet.

   (4) o High noise areas in the plant have been noted: D.G. area, Reactor Area, Feed pumps area, Air compressor areas.

9. Is there a particular panel which you consider more difficult or confusing to operate than the others?

   (3) o No or none

   (2) o North Vertical Board - spread out and has many systems (partially or wholly) represented, e.g., CVCS just not organized logically.

   (2) o J-Console Turbine Control Eng - many controls and indications with no emphasis given to importance of the control. (i.e., Feed Pump Switch is same size and area as Feed Pump Oil Pump Switch). Also Turbine Controls are laid out without a particular pattern. FC-1107B is confusing because it is a dual action controller depending on whether plant is on-line.

   (3) o Diesel - Handles on operating devices too close together, similar and may operate in wrong direction. Auto Voltage Adjustor, Governors Speed Control and Breaker Control. Control display not in order. Especially, the Governor Speed Control - the "raise" and "lower" switches are in an order that violates stereotypes. Volt and speed control switches should have color and size changed to help distinguish these from each other.

   (1) o AFW Control Board - Too many controls and meters to support two little pumps.

   (1) o On West Vertical Board, controllers are reverse Foxboro (100% means closed as opposed to open). Also, Aux. Feed Controls (FCV-2300 and FCV-2301) meters are too high.

   (1) o Controllers used in the control room could be identified as Open, Close, Increase or Decrease.

   (1) o Wide Range Gas Monitor (R-1254) - Needs additional training (see item #39).

CONTROLS

10. Which controls do you think are difficult to find or access?

- (3) o None

- (1) o Anything on the AFW Panel (C-71). An operator may vent the RX head or start an AFW pump. These are unrelated controls on the same board? SI system valves - random locations on control boards (Feed pumps on J-console; rest of SI on North Wall); all confusing to the operators.

- (1) o All annunciator panels outside of the Control Room should have slaves in the CR to better inform the CR of plant conditions. Radwaste, heating and ventilation panel, boric acid temperature, and examples of alarms without specific indication.

- (1) o CVCS valves. Excess letdown inlets/outlets. TCV-1105. Letdown LCV-1112.

- (1) o CCW valves on West Vertical board (Saltwater pump controls on North Vertical board).

- (1) o Test pump on North Vertical Pump switch should be with the charging pump switches on J-console.

11. Which controls, if any, are poorly designed or built for handling or operating, and why?

- (2) o The S.I. Test push buttons for pressure instruments on North Vertical Board, if actuated would initiate an S.I. signal. This is an old design, before TMI, and has not been used for testing, it should be removed.

- (1) o Some controls are reverse acting (open to left rather than the right). (See item #9.)

- (3) o Reheater temperature controls RMC-3 uses pressure for process control and should use temp. Does not work in automatic.

- (1) o Feedwater Controllers and Console - it is difficult to "Null" and transfer from Auto to Manual and back.

- (1) o Switches being next to each other has caused incorrect activation. Governor Speed Changer, and East Feedwater Pump control switch, B. A. System (CVS 333 and 334). (See item #9.)

- (1) o Turbine Switches: Auto Stop Latch Reset on J-console - does not work. Governor on turbine - does not work.

CONTROLS (Cont.)

(1) o Refueling Water Level Detector is unreliable and has no redundancy.

12. Which controls are most likely to be operated in error, and why (for example, due to location or label, etc.)?

(1) o D.G. panels - (see item #9).

(1) o CV-334 - (see item #7).

(1) o CV-334/CV-333 - located next to each other, similar labels.

(1) o Feedwater pump valving - controls are separate from pump controls and valves and are labeled poorly.

(1) o The CSAS Pumps and Valves initiation push buttons (Train A/B) poorly labeled as to function, (i.e., initiation or reset). Some operators think that these switches are reset push buttons, they are not. They should be labeled "Pump Initiation".

(1) o Sync By-pass, #1 & #2 D/G control switch.

13. Are there any controls not currently in the control room, that are needed to respond to normal or emergency situations?

(1) o Sphere sump pump controls.

(1) o RCS Drain Tank Pump controls.

(1) o Yard Sumps, Intake Sumps, and Reheater Sump controls.

(1) o It would be an improvement to motorize the main steam stops (maintenance block valves).

(1) o 24" main steam isolation valves.

(1) o S/G blowdown valve indication/control.

(1) o Ventilation controls should be located in the control room area. We should not have to send an operator out to Start or Stop containment or Sphere Enclosure Building (SEB) ventilation.

(1) o Some needed displays suggested (see item #19).

14. Are there any controls on back panels that should be on front panels or vice-versa?

(3) o No

CONTROLS (Cont.)

(1) o Condenser back pressure recorder should be on Front Panel.

(5) o Delta T recorder should be on front panel to help identify natural circulation starting pumps, etc.

(1) o Instrument Failures VCT defeat switch.

(1) o Thermocouple monitoring system.

(1) o RCP vibration recorder.

(1) o VCT blend controller (primary water).

(1) o Sealing filter.

(1) o P meters.

(3) o Boric Acid Heat Trace Recorder.

(1) o Heating and Vent Recorder (R-9).

(1) o Fox-3 Slave in the Control Room for plant trend conditions (see item #28).

(1) o Rad Monitors 2100 and 2101.

DISPLAYS

15. Which displays are hard to locate or access. Please explain.

(3) o None

(1) o Most displays are readable, but labeling is lacking name or is missing on some (RCP Seal Flow, Wide Range, or Post Accident Monitors). (See item #2.)

(1) o Recorders need better point identification to tie equipment and area being monitored together. Identifying EQ equipment would be helpful.

(1) o Condenser pressure on recorder in S. Aux. Rails - operator has to go behind Vertical Boards to access this recorder.

(1) o SI parameters should be displayed on one panel. (See item #10.)

(1) o The amps red lines are present marked by a yellow, not a red zone. Plus, where the red zone begins does not accurately reflect the amp capacity of all pumps.

DISPLAYS (Cont.)

16. Which displays are difficult to read, and why?

(1) o None

(3) o Diesel generator meters are too high on panel and have curved faces making them hard to read.

(1) o FI-606 CCW Hx Outlet Flow and FI-602 RHR Loop Flow - type of scale makes accurate readings difficult at low end of scale.

(1) o Recorders like Delta T and flow recorders print over themselves.

(1) o Vital bus potential lights are small and hard to read. The actions required are different for each light.

(1) o The new bypass CV Flow indicators have non-linear scales as do the RHR Flow indicators.

(1) o Most multi-point recorders, especially the RAD Monitor multi-point recorder. The numbers from the many points regularly are printed on top of each other.

(1) o The new SI Flow meters are too small. The flat face is too reflective.

17. Which important indicators are difficult to see during normal or emergency operation, and why?

(4) o None

(2) o The SI "blue light" positions. Intermediate position is white and blue which looks about the same as blue.

(1) o #1, #2 D/G Meter: board is set about 12 inches too high for most personnel. (See item #16.)

(1) o Pen recorders YR-456, 457, 458, TR 401, 405 - Fail to Ink Properly.

18. Are there any displays in the control room that you feel are unnecessary?

(7) o No

(2) o Feedwater Flow Integrators - West Vertical Board and STM Dump Elapsed Timer - West Vertical Board.

(1) o SI test push buttons on North Vertical Board. Remove, if not to be used again (see item #11).

(1) o T SAT Recorder on J-Console not used .

DISPLAYS (Cont.)

19. What displays, not now in the control room are needed to respond to normal or emergency situations?

    (3) o None

    (1) o Loop Delta T Ind. (chart).

    (1) o Feed Temp Ind. (chart).

    (1) o Cond. Vacuum Chart.

    (2) o Atmospheric Steam Dump Valve Positions - open/closed.

    (2) o S/G Blowdown indication (for valves CV-100, 100A, 100B)

    (1) o Condenser and atmospheric valves.

    (1) o Refueling water pump (used for Containment Sprays) amps.

    (1) o D/G bus #2 ground detection meters (bus #1 is behind boards).

    (1) o Alarms from the Heating and Vent Board and the Radwaste Control Board.

    (1) o 220 KV trouble alarm for positions #1 and 3 specifically initiating on Low Air/Gas (low or High) Press.

    (1) o The generator hydrogen panel system.

ANNUNCIATORS

20. How can the annunciators be improved (e.g., content of legend, hardware, etc.)?

    (2) o No improvements needed.

    (2) o Each annunciator should have some method to tell the operator when the alarm condition has cleared.

    (2) o All alarms with multiple inputs should have reflash capability. Example of alarms with over 30 inputs, but once one alarm is received, the system is blind to other inputs.

    (1) o A slightly different tone for each annunciator panel would enable the operator to more rapidly locate an alarm.

    (2) o Fire panel alarms do not all annunciate.

    (1) o Reactor Plant Permissive Panel should have audible alarm.

    (1) o Common equipment can be grouped together better. Such as, CCW Sup, RCPs, Cont. Spray, Secondary system.

(1) o Retired annunciator windows (Evaporation system on North Vertical Board) need to be replaced with blanks.

(1) o A method to locate grounds on the system would help (individual ground indicators for each annunciator with an alarm on each panel).

(1) o Standardize the print, increase the clarity of annunciator windows, thereby improving the contrast of print on each window.

21. On what panel(s) does the placement of individual annunciators not follow a logical pattern?

(1) o None

(1) o RX Plant #2 Annunciators - Yard Sump, SMA-3 End of Tape.

(2) o Auxiliary Annunciator Panel - the primary systems should be removed to the associated systems, i.e., Hydrazine Spray Pump loss of control power should be in the area of the Hydrazine controls on RP 1 or 2. RP#2 - CSIAS Test and CSIAS Latch Alarms should be closer. Also, Cont. Spray.

(1) o Fire Systems Panel - All 480V should be grouped together. All 4KV Rolm should be grouped together, etc.

(1) o This is a generic problem.

22. Identify annunciators that are difficult to interpret or do not help diagnose a problem.

(2) o None

(1) o Hydrazine Spray Pump Loss of Power Alarm - does not identify which current is deactivated.

(2) o There are many which have several inputs (Radwaste, generator hydrogen, heat trace), and therefore do not directly tell what the problem area is.

(1) o RP 1st out window 35 (Pressure Transient in progress) - Unclear, means you have an overpressurization beyond 500 lbs.

(1) o Provide an alarm and test circuit for "Permissive Display Panel".

23. Identify annunciators that alarm too late to allow operator action.

(2) o None

(1) o Alarms exist for "18KV System Isolated", or "Auto Transformer End of Sequence"; only have lights.

(1) o Condenser Low Vacuum.

ANNUNCIATORS (Cont.)

24. Identify nuisance alarms.

(1) o None

(7) o Boric Acid Heat Trace on RP #1, window 80.

(1) o R-1254, 1255, 1256 A & B, 1257, 1258 A & B, 1259 should be tied to annunciators or at least given audible alarm.

(1) o Loss of MW, Safety Injection - these come in due to Voltage Spikes.

(1) o Fire Alarms.

(1) o Feedwater Heater level alarms during startup/shutdown.

(1) o CSAS Hydrazine low flow.

(1) o Rad. Area Entry Alarm.

COMMUNICATIONS SYSTEM

25. Is more or better communication equipment needed in the control room?

(1) o No

(2) o Operators need a dedicated radio channel for their use.

(1) o The CRS desk and the SS desk should have paging capabilities.

(8) o Both the telephone and radio systems are old and unreliable. Also, radio coverage is inadequate and communication is difficult or impossible to some areas of the plant (especially Containment).

26. Are verbal messages in the control room ever unclear?

(5) o No

(2) o Yes, but training/standardization (feedback method) has helped in the past to reduce problem.

(1) o Yes, operators are not trained well on verbal communications during emergency events. There does not appear to be a standard.

COMPUTER SYSTEM

27. Is the computer useful in providing you accurate, timely, and easily usable data regarding important system parameters under normal, abnormal, and emergency conditions? If no, please explain.

    (1) o Yes

    (5) o No, the FOX-3 does not currently serve a useful function - it is not located in the control room.

28. Is the computer difficult to use in retrieving important system data?

    (2) o The SOMMS (the Plant Maintenance Computer) is normally out of service during the time of day when the operators have the most time (midnight shift) available to input to it, it is also not user-friendly.

    (1) o Yes, on power failures, the FOX-3 computer seems to fail and important data for the FOX-3 is lost.

    (4) o Training is required for retrieving data from the computer (FOX-3).

    (3) o FOX-3 information is not in control room and is poor design - we cannot access. (See item #27.)

CRT DISPLAYS

29. Are there any problems with any of the following characteristics of the CRT displays?

    a. visibility (glare or location)
    b. imaged quality
    c. coding (for example, color, symbol)
    d. organization of call-up displays
    e. format of displays
    f. response time
    g. keyboard (or other entry techniques)

    (3) o None

    (1) o The few CRT-computer systems we have are not user-friendly (look at SOMM Maintenance order program). We have very limited computer/CRT items available. (See item #28.)

    (1) o We have no training on it. (See item #28.)

CORRECTIVE AND PREVENTIVE MAINTENANCE

30. Is the control room preventive maintenance program effective? Are the maintenance procedures effective?

    (3) o Yes

    (1) o Too much administrative work required for fix simple things (clamp, missing screws and covers, etc.).

    (2) o Need more maintenance on recorders.

    (1) o Priority system is not adhered to. Repairs take too long.

    (2) o There is no specific Control Room Preventive Maintenance Program.

31. Would you recommend any changes to current maintenance of surveillance testing efforts in the control room?

    (6) o No or none.

    (1) o Where possible, assign a group (relief operators) to start and complete surveillance, clearance for everyday work.

PROCEDURES

32. Can you find the procedure binder you need when you need it? Can you easily find the specific procedure or procedural step you need?

    (5) o Yes

    (5) o Each binder should have an up-to-date index, better coding and cross-referencing and be kept in the same location.

    (1) o No. Some procedures (SO1-5-13 for example) are too detailed and long, making it hard to use.

33. Are there any specific procedures that are so unclear that portions of them should be rewritten, and why?

    (3) o No

    (1) o SO1-5-13. 75 pages of how to operate a Radiation Monitor. Too detailed, omit T.S. section since any Tech Spec question must be answered by looking at the Technical Specifications.

    (3) o Many procedures are too detailed, hard to follow, incorrect and very hard to use and interpret.

STAFFING AND JOB DESIGN

34. What problems of control room shift staffing interfere with smooth, continuous system operation?

    (5) o None

    (1) o There is not enough lead time on schedule changes. STA's are often not close participants in shift operations and do not rotate with the same shift.

    (1) o There are similar supervisory responsibilities between SS & CRS. The "Chain of Command" is often not used.

    (1) o The reactor operator responsible for operating the plant should not have most of his time taken up with administrative duties, i.e., work authorizing surveillance, routines. Up to 70% of operator's time spent on paperwork.

35. Under what circumstances are individual responsibilities and chain-of-command not clearly understood, and how could this be improved?

    (4) o None

    (1) o Teamwork of the shift crew needs to be emphasized in training programs and enforced on shift.

    (1) o To improve the everyday chain-of-command each shift superintendent, control room supervisor, control operator, assistant control operator should have their own desk name tag and lapel tag.

    (1) o CRS introduction disrupted and continues to disrupt old chain-of-command. No longer possible to eliminate CRS, therefore, no improvement possible. (See item #34.)

    (1) o The SS & CRS & CO are not allowed the authority to meet their responsibilities. Management often gets too involved with decisions that should be handled at a lower level. Suggest management, including SS & CRS, let people do the job they are paid to do.

36. Are there any duties you are required to perform that you consider unreasonable or distracting in your primary responsibility as SRO or RO?

    (6) o No or None.

    (1) o SRO is required to implement EPIP's as Emergency Coordinator which will consume much of his time, during an emergency event. Yet he is responsible for the plant at the same time.

(1) o Too many non-operators personnel or personnel who do not need to contact the control room for plant information still call the control room for information, i.e., engineers, designers.

(1) o Yes - Area Management Program. Much of this program could be done by Maintenance. Engineering should manage the program.

37. What administrative procedures do you think could be implemented more efficiently (e.g., shift change, control room access)?

(4) o Basic problem is that there are too many administrative procedures and too many changes to them.

(1) o More awareness of Equipment Control Surveillance and plant operations sometime needs to be implemented.

(1) o Maintenance Management, EPEP's.

38. In off-normal situations, describe any workload problems you have encountered?

(1) o None

(2) o In off-normal situations workload problems are expected.

(1) o Nuisance alarms and sirens during emergency condition. (See item #24.)

(1) o Too many non-emergency personnel calling the control room to ask what is going on.

(1) o Interface with higher management required too much of the Shift Superintendents time during off-normal situations.

(1) o Open line with NRC during off-normal events.

(1) o Operator confined to desk all shift processing paper. TFM's design changes, TCN approvals, procedures in progress review, surveillance reviews, WARs-WAMs, logs, special orders, tag audits. Looking at the plant and control boards come last, if at all. (See item #37.)

TRAINING

39. On what system or panels would more practice or training be useful and why?

(3) o None

(1) o Turbine - Generator Controls. Infrequent operation.

(2) o Wide Range Gas Monitor - R-1254.

(1) o All aspects of Electrical training including: theory, relay protection, major system disturbances and system operating bulletins.

(1) o Fire Protection System.

(1) o FOX-3. (See item #28.)

(1) o D.S.D. Shutdown.

(1) o SOMM's.

(1) o AFW.

40. Do you have any recommendations for making training more effective?

(3) o No

(4) o Develop a plan for instructors to get in-plant operating experience. Right now they only teach the lesson plan - not operating philosophies. Annual requalification exams do not reflect the material presented in requalification training - they are more like an NRC exam.

(1) o Need Plant specific simulator.

(1) o Less formal systems (teach to objectives and challenge the student and instructor).

(1) o More control room/panel training and testing would help the operator be familiar with the boards sooner.

41. In what areas would refresher training be helpful for more effective operation?

(2) o None

(1) o Update older operators on Nuclear Theory and teach them Heat Transfer and Fluid Flow. Many items were either taught differently in the past or not at all. Operators with a significant time lapse since original training (7-10 years) have difficulty in retaining the knowledge, especially if they are now Supervisors with no hands-on operations.

(1) o Need to make sure everything is eventually reviewed.

TRAINING (Cont.)

(1) o Administrative Procedures -

"What To Do With Records" (Attachments/Forms/Logs, etc.)
"Use of Procedures"
"Control of Systems Alignments"

(1) o Electrical Systems.

42. In what technical or skill areas would additional training be helpful?

(1) o Supervisory skills

(1) o Teamwork

(1) o Company Policies

(1) o Nuclear Theory

(1) o Heat Transfer/Fluid Flow

(1) o Electrical

(1) o Emergency Response Operations

(1) o Water hammers, and operation of check valves.

GENERAL

43. If you have any additional comments that have not been covered elsewhere, please note them in the space below.

(1) o None

(1) o Implement a Supervisor Training Program for CRS and SS. Create a "Lessons Learned Program". (See item #42.)

(1) o Determine how other plants operate with 10 percent of the personnel on-site than we have. This could be due to fewer managers and/or more experienced personnel.

4.0 CONTROL ROOM INVENTORY

4.1 OBJECTIVE

The objective of compiling a control room equipment inventory was to obtain an accurate and current list of all instruments, controls, and equipment in the control room that the operators interface with during the course of their assigned activities and responsibilities. The inventory includes relevant instrument characteristics including panel location, I&C description, type of instrument, scale range and coding information.

The control room inventory provided a necessary tool in verification of availability and suitability of control room instrumentation and controls needed to support SONGS 1 emergency operations. The utilization of the inventory is discussed in Section 7.

4.2 METHODOLOGY

The control room inventory was compiled using the SONGS 1 photographic mock-up. In order to ensure independence between the control room inventory and the system function and task analysis, a different group of individuals performed each function. This independence ensured that the identification of required instrumentation and controls produced by the system function and task analysis would not be biased by previous knowledge of existing instrumentation and controls gained through the inventory check.

The control room inventory consisted of data, in the form of equipment characteristics, that was entered on an Equipment Characteristics Form as illustrated in Figure 4-1. This data was also entered in a computer database file for use during the verification phase. The following data was generated for each control room instrument and control:

- o Reviewer and Date - the name of the person filling out the equipment characteristics form and the date it was performed.

- I&C Description - description of the instrument or control and parameter measured if applicable.

- I&C Number - the alphanumeric identification given to the instrument or control.

- Panel ID - panel identification that the instrument is located on.

- Instrument Type - identification for the type of instrument or control including operational characteristics if appropriate, i.e., switch-rotating, slide, etc., recorder-continuous or discrete, controller-on/off or proportional, meter, potentiometer, pushbutton, indicating light, etc.

- Instrument Range - as appropriate, identification of the meter range from minimum to maximum on the scale.

- Units - the standard of measurement of the parameter, i.e., GPM, AMPS, PERCENT, PSIG, etc.

- Divisions and Scale - identification of number of major and minor graduation divisions, and indication of linear or logarithmic scale.

- Controls and Lights - for control switches, the number of positions were listed including function (e.g., 2 positions - Pump 1A/1B), for indicating lights the color and meaning when illuminated were listed.

Figure 4-1

EQUIPMENT CHARACTERISTICS FORM

Reviewer

Date

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
Diesel Fuel Storage Tank Level	LI-14A	DG1	Vert. Meter	0-100	%	20/2 Linear	None
Diesel Fuel Pump Sel SW	HS-6A	DG1	ROT SW	NA	NA	NA	2 POS: Pump 1A, 1B
Silence	HS-128A	DG1	PB SW	NA	NA	NA	Amber

5.0 CONTROL ROOM HUMAN ENGINEERING SURVEY

5.1 OBJECTIVE

The objective of the control room survey was to identify characteristics of control room instrumentation and controls, equipment, physical layout, and environmental conditions that do not conform to precepts of good human engineering practice. This survey was accomplished by conducting a systematic comparison of existing control room design features with human engineering guidelines as established in Section 6 of NUREG-0700.

For SONGS 1 this aspect of the control room design review was considered a critical process for identification of discrepancies relating to control display integration and functional grouping of components. In light of the age of SONGS 1 and the significant number of control room changes implemented since the original design, the potential for significant improvements in these areas exists. Consequently, the results of the control room survey were considered in conjunction with aspects of operational dynamics including operator task sequence requirements. This process provided evaluation of the static and dynamic aspects of functional grouping and control display integration.

5.2 CRITERIA

The control room survey was conducted using detailed guidelines defining human engineering suitability. To facilitate systematic application of the guidelines, a checklist format was used to evaluate compliance with each guideline. The checklist provided in Section 6 of NUREG-0700 established the comparative basis for evaluation of the control room features. The checklist was divided into nine sections as defined below.

5.2.1 Control Room Workspace

This section establishes guidance for general control room layout including accessibility of instrumentation, furniture and equipment layout, documentation organization and storage, and control room access. Other guideline areas include equipment and console dimensions, availability of emergency equipment, and control room environmental conditions (i.e., ventilation, illumination, emergency lighting, etc.).

5.2.2 Communications

This section establishes guidelines for voice communication systems including telephone systems, walkie-talkie radios, intercom systems, etc. Guidelines are also provided for auditory signal systems including use of auditory signals, coding techniques, propagation of signals, frequency, intensity, and reliability.

5.2.3 Annunciator Warning Systems

This section establishes guidelines for annunciator system design, prioritization, first-out annunciation, annunciator tile legends, and annunciator response controls.

5.2.4 Controls

This section establishes guidelines for the various types of controls in the control room including control design principles, selection of controls, as well as specifications for design and operation of rotary, pushbutton and other types of controls.

5.2.5 Visual Displays

This section establishes guidelines for the principles of visual displays including information displayed, usability and readability of displayed values, and scale coding. These guidelines are provided for meters, light indicators, graphic recorders, and miscellaneous type displays.

5.2.6 Labels and Location Aids

This section establishes guidelines for labeling principles including need for labeling and hierarchical labeling schemes. Guidelines are also provided for label location, content, style and readability of lettering, use and control of temporary labels, and use of location aids including color, demarcation and mimics.

5.2.7 Process Computers

This section establishes guidelines for computer access relating to prompting and structuring, data entry, function controls, response time, etc. Also provided are guidelines for CRT display characteristics and printer characteristics.

5.2.8 Panel Layout

This section establishes guidelines for control panel design including contents, effectiveness of layout, component recognition, logistics, functional relationships, separation of controls, component strings, and mirror imaging.

5.2.9 Control-Display Integration

This section establishes guidelines for basic control-display integration, control grouping and dynamic control display relationships. Guideline areas include single control and display pairs, multiple controls or displays, location and arrangement of control display groups, and general movement relationships.

5.3 METHODOLOGY

The control room human engineering survey encompassed all instrumentation and controls located within the control room, and additionally included the remote shutdown panel and the FOX-3 computer located outside of the control room. A human factors psychologist and an operations specialist from General Physics, trained in the specifics of control room surveys, performed the survey.

The checklist used to perform the survey was based on the criteria defined in Section 6 of NUREG-0700 and briefly explained in Section 5.2 of this report. The checklist was designed to include principles or explanatory statements followed by specific categorical or numerical statements that require a "yes" or "no" response. The procedure was to observe, or measure, as required, and check compliance with each categorical or numerical statement.

Some checklist items were addressed on a control room-wide basis, such as items that fall into the categories of communications, control room layout, and environmental factors. Other items were addressed in two phases. The initial phase being a control room-wide review, followed by a second phase entailing a panel-by-panel review. These items included annunciator warning systems and control panel layout. Still other items were evaluated initially component-by-component, followed by overall control room consistency. These items were controls, visual displays, labels and location aids.

The checklist item for control display integration was examined on a panel-by-panel basis. Since the survey was conducted in the control room, the surveyors utilized the availability of control room operators to explain plant specific operational dynamics. Given this information, the surveyors could more readily evaluate control-display integration and functional grouping.

If compliance with a guideline was observed, it was noted by checking the "yes" column of the guideline. An item that received a "yes" response indicates that control room-wide compliance has been observed. In instances of noncompliance, a control room survey HED worksheet was filled out stating the specifics of the noncompliance. The control room survey HED worksheet is illustrated in Figure 5-1.

5.4 CONTROL ROOM SURVEY RESULTS

From the handwritten control room survey HED worksheets, the appropriate information was transferred into the CRDR computer database from which HEDs were written. The results of the control room survey were documented on HEDs 1 through 237.

During the evolution of the CRDR, the NRC requested an in-progress audit to evaluate the progress to date and ensure the direction of the review would result in a comprehensive and successful CRDR. As part of the in-progress audit, the NRC audit team conducted a minisurvey of the control room instrumentation and control characteristics. This survey was conducted in the control room photographic mock-up. Upon completion, a comparison of the audit team results and the CRDR survey results was performed. Approximately 40 individual items were compared. Of the items compared, five HEDs were identified by the audit team that were not found in SCE's survey results. These items and the corresponding HEDs documenting the discrepancies were as follows:

- Alarms on the diesel generator panel direct auxiliary operators to another plant location (NUREG-0700 Section 6.3.1.2.B-1) - This discrepancy was included on HED 169. The description of this HED is a listing of several control alarms which require an operator to leave the control room to obtain additional information, take action, or verify alarm.

- Hi-Lo Annunciators have input from multiple plant parameters (NUREG-0700 Section 6.3.1.2C-1) - This discrepancy was included on HEDs 169, 189, 190, 191 and 192. These HEDs identify the various problems relating to multi-channel and shared alarms.

- First out panel does not have separate annunciator tile for auto first out function (NUREG-0700 Section 6.3.1.3A-2) - This discrepancy was documented as HED 220.

- Control rod lever on J-console is not shielded to prevent accidental activation (NUREG-0700 Section 6.4.1.2B-1) - This discrepancy was documented as HED 219.

- Several controls on the J-console are difficult to operate, e.g., feedwater valves process controllers and the RPS mode selector switch (NUREG-0700 Section 6.4.1.1A-2) - The discrepancies for operational difficulties for controls on the J-console were documented on HEDs 174, 384, 386 and 387.

Figure 5-1
CONTROL ROOM SURVEY
HED WORKSHEET

SAN ONOFRE CR SURVEY HED WORKSHEET		NO:
Reviewer:	Date:	Checklist Number:

EQUIPMENT			
Panel No.	Component No.	System	Component Name

DESCRIPTION

GR6-0098

6.0 SYSTEM FUNCTION AND TASK ANALYSIS

6.1 OBJECTIVE

The objective of the System Function and Task Analysis (SFTA) review was to establish the information and control characteristics required to support operator tasks during emergency operations, and verify that required systems can be efficiently and reliably operated by available personnel.

A system function or subfunction was defined as a process performed by one or more components or systems which may contribute to a larger function or goal. A task was defined as an action performed by an operator that contributes to the accomplishment of a function.

For purposes of the CRDR SFTA, system functions were reviewed in support of emergency plant operation, and the task analysis was limited to control room operator tasks exclusive of automated equipment tasks.

6.2 BACKGROUND

The CRDR review of SONGS 1 system functions and operator tasks was performed by General Physics personnel with technical support and required documentation provided by SCE. The functional process of the SFTA required evaluation of plant specific documents relating to design and operation of SONGS 1 as well as generic publications including the Westinghouse Owners Group (WOG) Emergency Response Guidelines (ERGs) and the WOG Systems Function Task Analysis. The plant specific documentation used to identify systems, subsystems and respective functions included the SONGS 1 System Descriptions, SONGS 1 EOI Bases Documents, and the SONGS 1 Q-List.

Based on the system functions descriptions, General Physics developed accident scenarios designed to exercise system functions and evaluate operator tasks. The methodology employed during the SFTA provided independence of the defined tasks and task elements from the actual instrumentation and controls in the control room.

6.3 METHODOLOGY

6.3.1 Overview

A top-down approach was used to perform the SFTA. This methodology was initiated with a review of systems, subsystems, and their respective functions to assure that all operator tasks are considered. Upon completion, an evaluation was performed to assess the potential effects of design-related performance error on system safety.

The activities which comprised the SFTA are illustrated in Figure 6-1. The SFTA consisted of two primary areas:

- o Identification of systems and system functions

- o Identification and analysis of operator tasks

The systems and system functions were defined through review of plant specific documentation describing SONGS 1 systems and their interrelationships with plant operation. From the system function descriptions and the EOI Procedures Generation Package (PGP) validation scenarios, plant specific scenarios were developed to exercise the system functions in order to evaluate associated operator tasks. Thus the task analysis identified information and control requirements, scenario based operator tasks, operator decision requirements, and residual operator tasks. Verification and validation of control room instrumentation and controls were performed to ensure the availability and suitability of equipment to perform operator tasks. This process was performed by comparison with the Control Room Inventory and conducting scenario walkthroughs with a normal operating crew in the full scale control room photographic mockup. These activities are discussed in Sections 7.0 and 8.0.

6.3.2 Identification of Systems and System Functions

Plant systems and subsystems the operator must access and utilize during emergency operations were identified. This set of identified systems is safety related and is plant specific to SONGS 1. This system set was compared to the generic PWR Westinghouse safety related systems as illustrated in Table 6-1. Differences from the Westinghouse systems are due to the plant specific design of SONGS 1. The SONGS 1 EOI Bases Documents and the System Descriptions were the main source of information used to compile the system set.

Each system function as well as the conditions under which the system is utilized was identified. This information served as a reference base for subsequent task analysis and was used to assist in the operating scenarios selection. Appendix 6A presents the SONGS 1 System Function Description.

A supporting document used to verify the completeness of the system set was the SONGS 1 Q-List. The Q-List is a controlled document which lists systems and portions of systems. The components of the Q-List can be broadly divided into two groups - safety related and non-safety related.

During the CRDR In-Progress Audit in July 1986 the NRC compared the Q-List to the system set used for the SFTA. This comparison resulted in finding some Q-List safety related systems not included in the SFTA system set. To satisfy this discrepancy, each Q-List system was compared to the SFTA system set. Based on this comparison, the following systems were identified to be included in Q-List but not part of the SFTA system set. For each instance, the exclusion of the system from the SFTA was determined to be appropriate based on the reasons stated.

- Primary Plant Sampling System - This system provides a non-safety related function, but most system components are safety related because of containment boundary crossings. No emergency operator actions are required with this system.

- Post Accident Sampling System - Post accident sampling is a manual task not performed from the control room. In addition, sampling is not performed by the operators and therefore does not constitute an operator task.

- Feedwater Sampling System - This system provides a non-safety related function, but some of the components are safety related because of containment boundary crossings. No emergency operator actions are required with this system.

- Turbine Plant Cooling Water System - This system provides a non-safety related function, but some of the components are safety related because of containment boundary crossings. No emergency operator actions are required with this system.

- Dedicated Safe Shutdown System - This system is located outside the control room and will only be used in the event that control room evacuation is necessary. The functions that are safety related are performed from the remote shutdown panel. This panel is included in the Human Engineering Survey, but is not part of the control room.

6.3.3 Identification and Analysis of Operator Tasks

Following the identification of system functions, the operator tasks associated with each function were identified. The instrumentation and equipment required for task performance were analyzed for each event sequence. Because the task analysis is performed to support a human engineering evaluation of control room equipment, the focus was on establishing operator action/decision relationships, and the instrumentation, control, and equipment requirements for action and decision making.

6.3.3.1 Representative Scenarios Defined

The SFTA systems and function descriptions were used to define a set of scenarios that sampled various emergency conditions, and plant systems and functions used in those conditions. A total of five scenarios were developed. The scenarios were carefully selected to exercise virtually all systems and functions identified in Section 6.2.2. A matrix was developed to ensure that the desired systems and the system functions were exercised by the scenarios chosen. This matrix is illustrated in Table 6-2.

A narrative description of each scenario was also 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 scenario descriptions are provided in Appendix 6B and are paraphrased below.

Scenario 1. Anticipated Transient Without Scram With Loss of Coolant - A reactor trip signal is generated from a spurious turbine trip with the failure of all automatic and manual reactor trips. A small break loss of coolant accident (LOCA) develops initiating safety injection. Scenario continues until RHR is placed in service.

Scenario 2. Large Break LOCA - During normal, full power operation at End of Life (EOL), a catastrophic rupture of an reactor coolant system (RCS) hot leg occurs. During transfer to cold leg injection and recirculation, recirculation capacity is lost when the charging pumps trip. Scenario continues until stable plant conditions are achieved and transfer to hot leg recirculation is completed.

Scenario 3. Steam Generator Tube Rupture - Tube rupture occurs at full power. Scenario terminates when plant conditions are stabilized.

Scenario 4. Secondary Break Inside Containment With Loss of Spray - Main steam line rupture inside containment with no containment spray because both refueling water pump motor breakers open with closing springs and charging motors de-energized. Scenario continues until spray is initiated, charging and letdown flows are established and plant equipment is re-aligned for shutdown conditions.

Scenario 5. Loss of All AC Power - Unit trip from full power coincident with failure of the Station Auxiliary Transformer and auto-start failure of both emergency diesel generators. Also, loss of instrument air to the main condenser steam dump valves and relief valves occurs. Scenario terminates when plant conditions are stabilized.

Residual operator tasks from the plant-specific EOIs not covered in the scenarios were analyzed independently for information and control requirements. The analysis of residual tasks was done to ensure all operator interfaces in the EOIs had been examined even if those interfaces were 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 embodied in the emergency scenarios.

6.3.3.2 Task Analysis Worksheet Developed

A Task Analysis Worksheet was developed and used to collect and document task performance data and other information. The worksheet indicated the operational steps required in each scenario, along with the appropriate information and control requirements, means of operation, and instrumentation and controls present on the control boards. The operator tasks were analyzed using the selected plant-specific EOIs as a starting basis and documented in the following manner.

1. Discrete plant-specific EOI steps were recorded in order of performance in the "Procedure Number and Step Number" column. Branching points were noted, depending on the plant transient being analyzed, in the "Scenario Response" column.

2. A brief description of the operator's tasks (in order of procedural steps) was recorded in the "Tasks/Subtasks" column. All tasks, both explicit and implicit, were documented using operations, engineering, and human factors personnel.

3. The operator decisions and actions linked to 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 data set also included EOI branching points that determined the operating sequence outcome.

4. Input and Output requirements for successful task performance were recorded in the System Component Parameter and the Relevant Characteristics columns. These would typically be system component and parameter, relevant characteristics, and procedural information necessary for operators to adequately 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 (i.e., closed-open, off-auto-on) were recorded based on EOIs, EOI Bases documents, and Technical Specifications.

It is important to note that Steps 1 through 4 were completed on the Task Analysis Worksheet by General Physics consultants using independent sources of data other than the actual I&C present in the control room. The remaining task analysis worksheet columns were completed during the Verification and Validation phases discussed in Sections 7.0 and 8.0 of this report.

6.4 RESULTS

Upon completion, the Task Analysis Worksheets served as a complete record of operator tasks, decisions, information and control requirements, and I&C availability and suitability during the selected emergency operating sequences. The task analysis worksheet is illustrated in Figure 6-2 and worksheet field definitions provided in Table 6-3. The principle result of the SFTA was a plant specific consolidated list of instrumentation and control requirements necessary for emergency operation of SONGS 1.

Figure 6-1

SYSTEM FUNCTION REVIEW AND TASK ANALYSIS

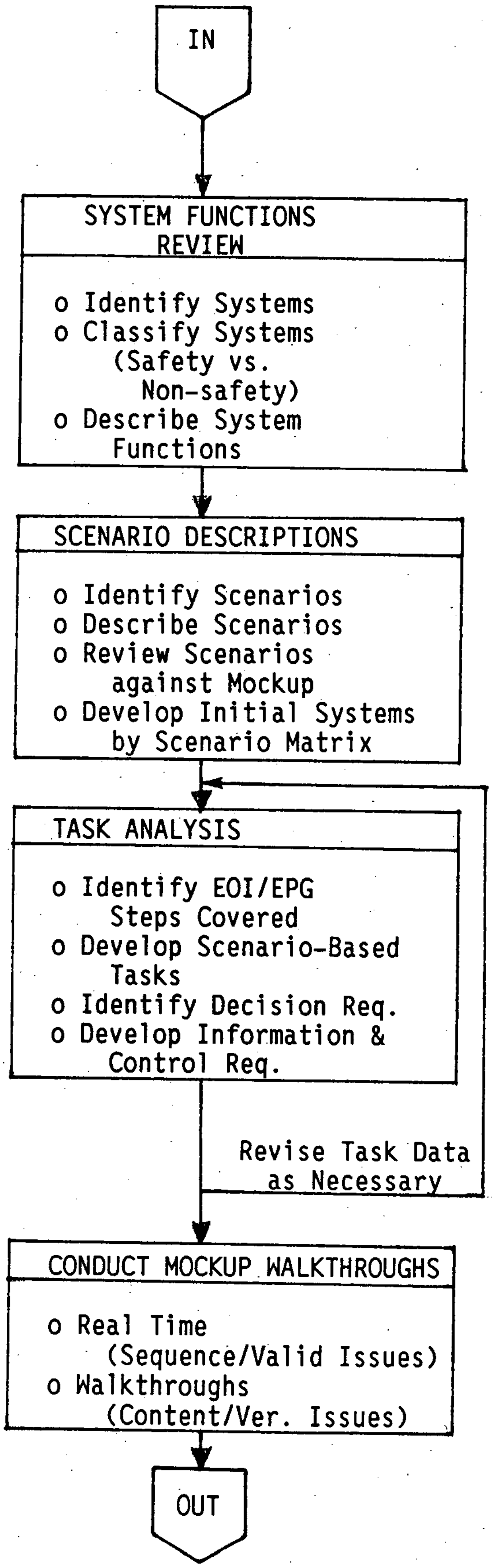
INPUTS

o Training Documents
o System Descriptions
o System Functions Form

o Procedure Generation Package Validation Scenarios
o Scenario Description Form

o Westinghouse Owners Group Emergency Response Guidelines
o EOE Bases Document
o Technical Specs.
o TA Worksheets

o TA Worksheets
o Observation Forms
o Videotape Equipment



PRODUCTS

o System Functions Descriptions

o Scenario Descriptions (5)
o Matrix

o Residual Tasks for Walk-throughs Only
o EOI Steps Covered
o TA Worksheets (1/2 done) Independently Completed
o Identify Missing Instrumentation

o Revised TA Worksheets
o Observation Forms for Validation

Figure 6-2

TASK ANALYSIS WORKSHEET

SCENARIO: 1 ATWS/LOSS OF REACTOR COOLANT (Part 1)

PROC NO. STEP NO.	TASK/SUBTASK	SCEN. RESP	CREW MEMB	LOC	TASK DECISION REQUIREMENTS	TASK ACTION REQUIREMENTS
S01-1.0-10.9B	CHECK RCS PRESSURE - LESS THAN 1170 PSIG	1050 PSIG			TO DETERMINE IF RCS PRESSURE IS LESS THAN 1170 PSIG	IF RCS PRESSURE IS LESS THAN 1170 PSIG, GO TO NEXT TASK. IF NOT GO TO STEP 9D.

SCENARIO: 1 ATWS/LOSS OF REACTOR COOLANT (Part 2)

PROC NO. STEP NO.	SYSTEM COMP PARAM	RELEVANT CHARACTERISTICS	MEANS	NO	PANEL	AVAIL	SUIT	COMP	COMMENTS
S01-1.0-10.9B	RCS NR PRESSURE METER INDICATION	LINEAR, ANALOG, RANGE 100-220 x 10 PSIG, 20 MAJ, 10 INT, 2 MIN							

Table 6-1

COMPARISON OF WOG SYSTEM REVIEW AND TASK ANALYSIS (SRTA)
WITH SAN ONOFRE UNIT ONE SAFETY-RELATED SYSTEMS

System Categories	WOG SRTA System	Equivalent San Onofre Unit One Safety-Related System
Control & Protection Actuation Systems:	Reactor Trip Actuation System	Reactor Protection System
	Engineered Safety Features Actuation System	Safeguard Load Sequencing System
Instrumentation Systems:	Nuclear Instrumentation System	Nuclear Instrumentation System
	Control Rod Instrumentation System	Rod Position Indication System
	Radiation Instrumentation System	Radiation Monitoring System a) Area Radiation Monitoring System (ARMS) b) Operational Radiation Monitoring System (ORMS) c) Post-Accident Radiation Monitoring System (PARMS)
	Containment Instrumentation System	*d) Containment Rad Monitor e) Containment Hydrogen Monitor f) Containment Pressure Monitors
Fluid Systems:	Reactor Coolant System	Reactor Coolant System
	Safety Injection System	Safety Injection System/Main Feedwater System
	Residual Heat Removal System	Residual Heat Removal System
	Chemical and Volume Control System	Chemical and Volume Control System/Boric Acid System RCP Seal Water System
	Component Cooling Water System	Component Cooling Water System
	Service Water System	Salt Water Cooling System

* Containment Instrumentation System Parameters are monitored by the Reactor Protection System, Safeguard Load Sequencing System and the Safety Injection and Containment Spray Systems

Table 6-1 (Continued)

COMPARISON OF WOG SYSTEM REVIEW AND TASK ANALYSIS (SRTA)
WITH SAN ONOFRE UNIT ONE SAFETY-RELATED SYSTEMS

System Categories	WOG SRTA System	Equivalent San Onofre Unit One Safety-Related System
Fluid Systems: (Continued)	Containment Spray System	Containment Spray System/ Containment Recirculation System
	Containment Atmosphere Control System	N/A - Rx Cavity Vent/H_2 Recombiner System CR System HVAC
	Main Steam System	Main Steam System
	Main Feedwater and Condensate System	Main Feedwater and Condensate System Feedwater Pump LO System
	Auxiliary Feedwater System	Auxiliary Feedwater System
	Steam Generator Blowdown System	Steam Generator Blowdown System
Support Systems:	Electrical Power System	Electrical Power System Diesel Generator System
	Pneumatic Power System	Compressed Air System N_2 Gas Systems
Other Systems:	Control Rod Drive Mechanism Cooling System	Control Rod Drive Mechanism Cooling System
	Spent Fuel Storage and Cooling System	Spent Fuel Storage and Cooling System
	Turbine Electro-Hydraulic Control System	Turbine Electro-Hydraulic Control System
	Control Rod Control System	Rod Control System
	Sampling System	Reactor Cycle Sampling System Fuel Handling/Transfer Equipment

Table 6-2

SUMMARY OF SONGS-1 SYSTEM
FUNCTIONS EXERCISED BY SCENARIOS
(Sheet 1 of 3)

System	System Function	SCENARIO EXERCISED 1	2	3	4	5
RPS	o generates auto reactor scram breaker trip signals	X	X	X	X	X
	o provides control room annunciation					
	o prevents exceeding plant design limits	X	X	X	X	X
	o monitors specific nuclear and non-nuclear parameters	X	X	X	X	X
SLSS	o provides proper loading and load sequencing of ESF equipment	X	X	X	X	
	o actuates on specific nuclear and non-nuclear parameters	X	X	X	X	X
	o provides manual actuation/block/reset of SI and/or loss of Offsite Power	X	X	X	X	X
NIS	o monitors/displays core neutron flux and neutron flux leakage	X	X	X	X	X
	o calculates reactor start-up rate		X			
	o provides variable range flux recorder	X	X	X	X	
RPIS	o monitors/displays actual and demanded control rod position	X	X	X	X	X
RMS (ARMS, ORMS, PARMS)	o ARMS detects, indicates, records radiation levels	X	X	X	X	
	o ARMS alarms on increasing radiation levels	X	X	X	X	
	o ORMS detects, computes, indicates, records and alarms radioactivity levels		X	X		
	o ORMS initiates auto actions		X	X		
	o PARMS detects, indicates, annunciates records accident radiation levels	X	X	X		
RCS	o transfer thermal energy from reactor to secondary	X	X	X	X	X
	o reduces thermal neutron leakage					
	o promotes thermal fission					
	o provides solvent for chemical neutron control	X	X	X	X	X
	o provides boundary to contain all modes of reactor coolant	X	X	X	X	X
	o prevents fission product release	X	X	X	X	X

Table 6-2

SUMMARY OF SONGS-1 SYSTEM
FUNCTIONS EXERCISED BY SCENARIOS
(Sheet 2 of 3)

System	System Function	SCENARIO EXERCISED 1	2	3	4	5
SIS	o injects borated water to mitigate core damage	X	X	X	X	
RHR	o removes residual heat from RCS during shutdown and long term post-accident cooling	X		X	X	
CVCS	o adjusts chemical reactivity control	X	X	X	X	X
	o maintains RCS inventory	X	X	X	X	X
	o provides RCP seal water	X	X	X	X	X
	o processes RCS effluent					
	o adjusts chemical corrosive control					
	o maintains RCS activity within limits					
BAS	o provides BORIC ACID to:					
	- RCS via CVCS	X	X			
	- RWST/Spent Fuel Pit					
	- VCT				X	
CCW	o provides intermediate heat removal from potentially radioactive system processes	X	X	X	X	X
SWS	o provides heat removal from CCW	X	X	X	X	X
	o provides back-up to TPCW					X
CSS	o sprays cool water into containment to mitigate peak pressure		X		X	
	o washes down radioactive particulates and airborne fission products		X		X	
CRS	o provide water to reactor core for long term post-accident cooling after RWST is depleted		X		X	
CAR	o limits containment temperature during normal operation only					
MSS	o provides controlled heat removal from RCS via steam generators	X	X	X	X	X
	o provides mainsteam release capability via steam dumps	X	X	X	X	
	o provides steam to (TD) AFW pump	X	X	X	X	

Table 6-2

SUMMARY OF SONGS-1 SYSTEM
FUNCTIONS EXERCISED BY SCENARIOS
(Sheet 3 of 3)

System	System Function	SCENARIO EXERCISED 1	2	3	4	5
MFW	o provides water to steam generator secondary side					X
	o provides safety injection	X	X	X	X	
AFW	o provides coolant to steam generator secondary side during start-up, shut-down, hot standby and loss of MFW	X	X	X	X	X
SGB	o provides steam generator secondary side letdown			X		
EPS	o provides offsite AC power	X	X	X	X	
	o provides onsite emergency AC power	X	X	X	X	
	o provides onsite emergency DC power					X
CAS	o supplies service, instrument, and control air to pneumatic equipment	X	X	X	X	X
CRC	o provides heat removal from CRD mechanisms					
SFP	o controls fuel storage ensuring subcritical configuration					
	o provides heat removal to maintain stored fuel within limits					
	o provides spent fuel pit level instrumentation					
EHC	o controls turbine speed or load					
	o closes turbine stop and control valves on turbine trip signal	X	X	X	X	X
RRS	o controls control rod position manually and/or automatically to control reactor neutron flux	X	X	X	X	X
RCSS	o provides means for sampling primary systems			X	X	

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Table 6-3

TASK ANALYSIS
WORKSHEET FIELD (COLUMN) DEFINITIONS

SCENARIO - operating scenario name and identifier (ID).

PROCEDURE STEP - procedure step number for SONGS 1 EOIs (Emergency Operating Instructions).

TASK/SUBTASK - a description of the crew member task/subtask in the operating sequence.

SCEN. RESP. - a notation designating decision points or branching information needed for correct task execution for the operating scenario (as defined in the operating scenario description).

CREW MEMBER - the crew member who performs the task.

LOC - the location where the task is performed.

TASK DECISION REQUIREMENTS - operator decisions that are linked to task performance.

TASK ACTION REQUIREMENTS - operator action requirements for task performance.

INFORMATION AND CONTROL REQ. - the information and control requirements for successful task performance (derived independently of the actual I&C in the control room). (1) System Component/Parameter (2) Relevant Characteristics (type of component, range, units, positions).

MEANS - the actual means (e.g. switch, meter, etc.) used by operators to perform the task in the control room.

I&C NO. - the actual Instrumentation and Controls (I&C) number identified from the control room inventory.

PANEL NO. - the panel on which the control or instrument is located.

VERIFICATION (AVAIL./SUIT.) - columns that indicate the availability and suitability of the Instrumentation and Controls (I&C) needed for task performance. These columns would contain a "yes" or "no" answer.

COMP - the presence or absence of the I&C and associated characteristics on the FOX 3 and/or SPDS Computer is noted in this column.

COMMENTS - any comments relating to scenario execution, task performance, or the accompanying task requirement columns (the balance of the task analysis worksheet).

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APPENDIX 6A

SONGS 1 SYSTEM FUNCTION DESCRIPTIONS

The SONGS 1 System Function Descriptions are contained in this appendix as follows:

System	Abbreviation	Page
Reactor Protection System	RPS	6A-1
Safeguard Load Sequencing System	SLSS	6A-3
Nuclear Instrumentation System	NIS	6A-4
Rod Position Indicating System	RPI	6A-5
Radiation Monitoring System	RMS	6A-6
Reactor Coolant System	RCS	6A-7
Safety Injection System	SIS	6A-9
Residual Heat Removal System	RHR	6A-10
Chemical and Volume Control System	CVCS	6A-11
Boric Acid System	BAS	6A-13
Component Cooling Water System	CCW	6A-14
Salt Water Cooling System	SWS	6A-15
Containment Spray System	CSS	6A-16
Containment Recirculating System	CRS	6A-18
Containment Air Recirculation System	CAR	6A-19
Main Steam System	MSS	6A-20
Main Feedwater and Condensate System	MFW	6A-22
Auxiliary Feedwater System	AFW	6A-23
Steam Generator Blowdown System	SGB	6A-25
Electrical Power System	EPS	6A-26
Compressed Air Systems	CAS	6A-27
Control Rod Drive Mechanism Cooling System	CRC	6A-28
Spent Fuel Storage and Cooling System	SFP	6A-29
Turbine Electro-Hydraulic Control System	EHC	6A-30
Rod Control System	RRS	6A-31
Reactor Cycle Sampling System	RCSS	6A-32

SYSTEM NAME: Reactor Protection System

SYSTEM ABBREVIATION: RPS

SYSTEM FUNCTION(S):

The Reactor Protection System monitors specified nuclear and non-nuclear parameters to anticipate and prevent exceeding plant design limits by limiting plant transients. The reactor protection system generates automatic reactor scram breaker trip signals and annunciators are provided in the control room to indicate when a reactor trip or permissive signal is activated.

The following trip signals are provided:

- Nuclear Overpower Trip
- High Intermediate Range Start-Up Rate Trip
- Variable Low Pressure Trip (VLPT)
- Single Loop Loss of Flow Trip
- Two Loop Loss of Flow Trip
- Feedwater Flow/Steam Flow Mismatch Trip
- Safeguard Load Sequencing System Trip
- Turbine Trip/Reactor Trip
- Loss of 125 VDC Bus 1 Voltage Trip
- Manual Trip
- Unit Trip

The following permissive signals are provided:

- Overpower Rod Stop Permissive, P-1
- Low Power Cutout of Automatic Rod Withdrawal Permissive, P-2
- Rod Drop Rod Stop Permissive, P-3
- Steam Dump Automatic Mode Cutout Permissive, P-4
- Shutdown Margin Alarm Permissive, P-5
- High Startup Rate Rod Stop Permissive, P-6
- At Power Reactor Trip Defeat Permissive, P-7
- Single Pump Loss of Flow Reactor Trip Defeat Permissive, P-8

CONDITIONS FOR USE: Plant operating or shutdown - anytime the control or shutdown rods are withdrawn.

REFERENCE: Training System Description

REVIEWER: M. Jennex **DATE:** 3/14/86

SYSTEM NAME: Safeguard Load Sequencing System

SYSTEM ABBREVIATION: SLSS

SYSTEM FUNCTION(S):

The Safeguard Load Sequencing System is used to detect and react to low pressurizer pressure, high containment pressure and 4160 V Bus 1C and/or 2C undervoltage signals. It is designed to provide proper loading and load sequencing of emergency safeguard equipment upon actuation in order to mitigate accidents and prevent vital bus overloading. The SLSS actuates in the event of a safety injection signal, loss of offsite power signal, loss of 4160 V Bus 1C/2C signal or a safety injection with loss of offsite power signal. The SLSS also provides for manual actuation of the safety injection and/or loss of offsite power signals, manual blocking of the safe injection signal and manual resetting of the safety injection and/or loss of offsite power signals. This system is used in place of an Emergency Safeguard Features System.

CONDITIONS FOR USE: Required Operable with RCS temperature greater than 200°F

REFERENCE: Training System Description

REVIEWER: M. Jennex

DATE: 3/14/86

SYSTEM NAME: Nuclear Instrumentation System

SYSTEM ABBREVIATION: NIS

SYSTEM FUNCTION(S):

The Nuclear Instrumentation System monitors and displays the neutron flux within the reactor core. It consists of instrumentation that monitors leakage neutron flux outside the reactor vessel. Neutron flux is monitored over the source, intermediate, and power ranges. Startup rate is calculated over the source and intermediate ranges. The NIS includes a neutron flux recorder that can be switched to record different ranges. The source range neutron flux detectors automatically energize when flux decreases below the source range high flux trip setpoint following a reactor trip, permitting the neutron flux recorder to be manually transferred to the source range scale.

CONDITIONS FOR USE: At all times (plant operating or shutdown).

REFERENCE: Training System Description

REVIEWER: M. Jennex **DATE:** 3/14/86

SYSTEM NAME: Rod Position Indicating System

SYSTEM ABBREVIATION: RPI

SYSTEM FUNCTION(S):

The Rod Position Indicating System monitors and displays both the actual and demanded position of each control rod assembly. Analog position indication as well as rod bottom lights are provided as actual position for each rod while demand position for each group in the rod banks is indicated by digital step counters.

CONDITIONS FOR USE: Plant operating or shutdown - at all times except during refueling operations.

REFERENCE: Training System Description

REVIEWER: M. Jennex DATE: 3/14/86

SYSTEM NAME: Radiation Monitoring System

SYSTEM ABBREVIATION: RMS

SYSTEM FUNCTION(S):

The Radiation Monitoring System is comprised of three independent systems, the Area Radiation Monitoring System (ARMS) the Operational Radiation Monitoring System (ORMS) and the Post Accident Radiation Monitoring System (PARMS). ARMS consists of area radiation monitors and emergency radiation monitors and is used to detect, indicate, and record radiation levels and/or alarm to warn individuals of increasing radiation levels. ORMS is used to monitor and prevent the uncontrolled release of radioactivity to the environment from various station process systems. It is designed to detect, compute, indicate, record and alarm radioactivity levels and to initiate automatic actions if needed. PARMS is used to detect, indicate, annunciate, and record radiation levels following an accident involving fuel failure and/or loss of RCS integrity. This system was added as a result of TMI-2. PARMS utilizes some ORMS detectors and some PARMS only detectors.

CONDITIONS FOR USE: At all times (plant operating or shutdown).

REFERENCE: Training System Description

REVIEWER: M. Jennex

DATE: 3/14/86

SYSTEM NAME: Reactor Coolant System

SYSTEM ABBREVIATION: RCS

SYSTEM FUNCTION(S):

The functions of the Reactor Coolant System (RCS) and the reactor coolant contained within, are to:

(1) Transfer the thermal energy generated in the reactor core to the secondary system water in the steam generators.

(2) Reflect neutrons back into the reactor core; thus reducing the amount of thermal neutron leakage.

(3) Moderate (slow down) fast neutrons to thermal energies thus promoting thermal fission.

(4) Act as a solvent for the soluble neutron absorber, boric acid, used in chemical shim control.

(5) Provide a boundary to contain the reactor coolant and accommodate the system pressures and temperatures attained under all expected modes of plant operation or anticipated transients.

(6) Provide a boundary to prevent the release of fission products to the containment atmosphere or secondary plant.

The RCS consists of three identical heat transfer loops, connected in parallel to the reactor vessel, a pressurizer and a pressurizer relief tank. Each loop includes a Reactor Coolant Pump (RCP) and various instrument connections. Loop penetrations are provided for connecting pressurizer, sampling, drain, residual heat removal, and letdown lines. The pressurizer is connected to the loop B hot leg via the pressurizer surge line and the loop A and B cold legs via the pressurizer spray lines. The pressurizer has two Power Operated Relief Valves (PORVs) with associated block valves, two safety valves, and heaters. RCS pressure is controlled by use of the pressurizer where water and steam are maintained in equilibrium through the use of the heaters, water spray, and steam release.

The pressurizer PORVs and safety valves discharge to the pressurizer relief tank where steam discharge is condensed and cooled by mixing with water.

Normal operating pressure of 2085 psig in the RCS is maintained by a pressure control system which automatically energizes heaters and normal spray in the pressurizer as necessary to maintain pressure. Pressurizer PORVs are automatically controlled to open at 2185 and 2200 psig, and the safety valves have a lift pressure of 2485 and 2510 psig.

CONDITIONS FOR USE: At all times (plant operating or shutdown).

REFERENCE: Training System Description

REVIEWER: M. Jennex **DATE:** 3/14/86

SYSTEM NAME: Safety Injection System

SYSTEM ABBREVIATION: SIS

SYSTEM FUNCTION(S):

The Safety Injection System is designed to mitigate core damage following a loss of coolant accident by injecting borated water, i.e., negative reactivity, into the core. Two independent trains of safety injection are provided. Each train consists of a safety injection pump which takes a suction from the Refueling Water Storage Tank (RWST) and provides suction for a main Feedwater Pump. Both trains discharge to a common header which directs flow to each of the three RCS cold legs. Additional flow is provided by the charging pump. This is useful as the charging pump provides injection flow above approximately 1175 psig, the shutoff heat for the main feedwater pumps. This flow is preferably directed to the RCS loop A cold leg but can be directed to any of the cold legs.

CONDITIONS FOR USE: Required Operable with RCS Pressure above 500 psig.

REFERENCE: Training System Description

REVIEWER: M. Jennex **DATE:** 3/14/86

SYSTEM NAME: Residual Heat Removal System

SYSTEM ABBREVIATION: RHR

SYSTEM FUNCTION(S):

The Residual Heat Removal System removes residual heat from the reactor coolant system during plant shutdown operations at low reactor coolant system temperatures and pressures. It also provides long term post-accident cooling.

The RHR system consists of two RHR pumps and two RHR heat exchangers. The RHR system provides normal shutdown heat removal when RCS pressure and temperature are reduced to approximately 400 psig and 350°F. During normal shutdown heat removal operations, the RHR pump suction is aligned to the RCS Loop C hot leg and the RHR system discharge is aligned to RCS Loop A cold leg.

CONDITIONS FOR USE: Plant shutdown with temperature below 350°F and pressure below 400 psig.

REFERENCE: Training System Description

REVIEWER: M. Jennex

DATE: 3/14/86

SYSTEM NAME: Chemical and Volume Control System

SYSTEM ABBREVIATION: CVCS

SYSTEM FUNCTION(S):

The functions for the Chemical & Volume Control System (CVCS) are to:

(1) Adjust RCS Boron concentration for chemical reactivity control.

(2) Maintain proper water inventory in the RCS.

(3) Provide the required seal water flow for the reactor coolant pump shaft seals.

(4) Process reactor coolant effluent for reuse of boric acid and reactor makeup water.

(5) Maintain the proper concentration of corrosion inhibiting chemicals in the reactor coolant.

(6) Reduce the quantity of fission and corrosion products and maintain the reactor coolant activity within limits in accordance with the Technical Specifications.

The CVCS consists of charging and letdown capability for RCS inventory control. Letdown capability is provided by two letdown paths, the letdown line and the lower capacity, alternate excess letdown line.

Charging capability is provided by two 12 stage, centrifugal charging pumps that deliver flow to the RCS through a charging line and RCP seal injection lines. Alternate RCP seal injection is provided by the charging test pump. The RCP seal injection lines deliver to each RCP and provide RCP seal cooling. A single RCP seal return line returns RCP seal leakoff flow to the suction of the charging pumps. The charging pumps can be used as high pressure safety injection pumps. The letdown, charging and RCP seal return lines are automatically isolated on a containment isolation signal.

Suction flow to the charging pumps is provided by the volume control tank (VCT) which is connected to the letdown line or by the refueling water storage tank (RWST) in the SI system. The charging pump suction is normally aligned to the VCT, but is automatically transferred to the RWST on a VCT low-low level signal or SI initiation.

CONDITIONS FOR USE: At all times (plant operating or shutdown).

REFERENCE: Training System Description

REVIEWER: M. Jennex

DATE: 3/14/86

SYSTEM NAME: Boric Acid System

SYSTEM ABBREVIATION: BAS

SYSTEM FUNCTION(S):

The Boric Acid System provides boric acid to the RCS via the CVCS charging pumps, RWST and the spent fuel pool. The BAS consists of a storage tank, batching tank, two transfer pumps, an injection pump, a filter and blending services. System flow to the RCS is normally from the storage tank, through the injection pump, the blending device, and to the suction of the Volume Control Tank (VCT). Boric acid concentration is normally kept at 12 weight percent. In order to keep the acid in solution the BAS is kept at approximately 170°F by heaters and heat tracing.

CONDITIONS FOR USE: Whenever there is fuel in the core.

REFERENCE: Training System Description

REVIEWER: M. Jennex

DATE: 3/14/86

SYSTEM NAME: Component Cooling Water System

SYSTEM ABBREVIATION: CCW

SYSTEM FUNCTION(S):

The component cooling water (CCW) system provides intermediate heat removal from potentially radioactive system processes and equipment to the Salt Water Cooling System (SWS). The following equipment is cooled by CCW:

- Recirculation heat exchange
- RHR heat exchangers
- Seal Water heat exchanger
- Containment fan coolers
- Motor bearing oil coolers and thermal barrier, RHR pump motor bearing and coolers, and charging pump oil coolers
- Excess Letdown Heat Exchanger
- Various Primary Sample heat exchangers
- Waste Gas Compressors

The system consists of three pumps, two heat exchangers, a surge tank, and the Emergency RCP Thermal Barrier Pump.

CONDITIONS FOR USE: At all times (plant operating or shutdown).

REFERENCE: Training System Description

REVIEWER: M. Jennex DATE: 3/14/86

SYSTEM NAME: Salt Water Cooling System

SYSTEM ABBREVIATION: SWS

SYSTEM FUNCTION(S):

The Salt Water Cooling System provides heat removal from the component cooling water system to the ultimate heat sink. It also serves as a back up cooling water source for the Turbine Plant Cooling Water System (TPCW). This system consists of two saltwater cooling pumps separated into two independent trains.

CONDITIONS FOR USE: At all times (plant operating or shutdown).

REFERENCE: Training System Description

REVIEWER: M. Jennex **DATE:** 3/14/86

SYSTEM NAME: Containment Spray System

SYSTEM ABBREVIATION: CSS

SYSTEM FUNCTION(S):

The functions of the Containment Spray System are to:

(1) Limit peak pressure in the containment structure to less than design pressure, 49.4 psig, in the event of a loss-of-coolant accident or steam line break accident inside the containment by spraying cool water into the containment atmosphere.

(2) Wash down radioactive particulate matter and airborne iodine fission products which would be released into the containment atmosphere during a LOCA, and keep them in solution, through the addition of Hydrazine to the spray water.

The containment spray system consists of the two refueling water pumps, two spray flow limiting valves and flow restrictor orifices and the sphere spray nozzles. Normal flow is from the refueling water storage tank, through the refueling water pumps, through the flow limiting valves and flow restricting orifices and out the sphere spray nozzles located in the top of the containment sphere and arranged in four rings. As described in the description of the containment recirculation system the refueling water pumps can also be used to recirculate water form the sphere sump. The system can be actuated both manually and automatically. Automatic actuation occurs on a 2 of 3 high containment pressure, (i.e., 10 psig), a SI signal, and normal voltage on the 4160V Bus 1C and 2C for 10 seconds.

CONDITIONS FOR USE: Required operable with RCS temperature above 200°F.

REFERENCE: Training System Description

REVIEWER: M. Jennex **DATE:** 3/14/86

SYSTEM NAME: Containment Recirculation System

SYSTEM ABBREVIATION: CRS

SYSTEM FUNCTION(S):

The Recirculation System is designed to provide water flow to the reactor core for long term, post accident cooling after the RWST inventory has been discharged into the containment sump following a loss-of-coolant accident. It is manually initiated after the Safety Injection System has reduced the Refueling Water Storage Tank level to 21%. The system consists of two recirculation pumps, recirculation heat exchanger, and associated piping and valves. Normal flow is from the containment sump through the pumps and heat exchanger to the RCS cold legs. Alternate flow paths include directing flow to the Refueling Pumps and then to the RCS cold legs, through the charging pumps and the regeneration heat exchanger to the Pressurizer and into RCS Loop B hot leg, and through the refueling water pumps, the letdown system, the residual heat removal system in reverse direction and into RCS Loop C hot leg.

CONDITIONS FOR USE: Required Operable with RCS Pressure greater than 500 psig.

REFERENCE: Training System Description

REVIEWER: M. Jennex

DATE: 3/14/86

SYSTEM NAME: Containment Air Recirculation System

SYSTEM ABBREVIATION: CAR

SYSTEM FUNCTION(S):

The Containment Air Recirculation system limits the containment temperature during normal operation. It is isolated upon a high radiation in containment signal and is not used during accident conditions.

CONDITIONS FOR USE: At all times (plant operating or shutdown).

REFERENCE: Training System Description

REVIEWER: M. Jennex

DATE: 3/14/86

SYSTEM NAME: Main Steam System

SYSTEM ABBREVIATION: MSS

SYSTEM FUNCTION(S):

The Main Steam System provides controlled heat removal from the reactor coolant system via the steam generators. It consists of separate main steam lines from each steam generator that join to form a common steam header which then splits into two main steam lines leading to the turbine-generator and condenser. The steam generators can be isolated from the main steam header by manual main steam line isolation valves located in each of the main steam lines. The valves can be selectively closed manually to isolate a specific steam line.

Main steam release capability is provided via the condenser and atmospheric steam dumps. The condenser steam dumps use the main steam header and steam dump valves to the condenser. The atmospheric steam dumps uses steam dump valves upstream of the main steamline isolation valves to release steam to the atmosphere.

Each main steam line contains five ASME code safety valves for overpressure protection. A steam line from the main steam header to the turbine-driven AFW pump is provided which includes isolation valves for initiation and isolation of steam supply to the turbine-driven AFW pump.

CONDITIONS FOR USE: Any time RCS temperature 212°F.

REFERENCE: Training System Description

REVIEWER: M. Jennex **DATE:** 3/14/86

SYSTEM NAME: Main Feedwater and Condensate System

SYSTEM ABBREVIATION: MFW

SYSTEM FUNCTION(S):

The Main Feedwater and Condensate System provides water to steam generators secondary side during plant power operation. It consists of separate main feedwater lines to each steam generator that originate from a common main feedwater header. The steam generators can be isolated from the main feedwater header by feedwater flow control valves and bypass valves located in the individual main feedwater lines.

The main feedwater system includes two motor-driven feedwater pumps which double as Safety Injection Pumps. The condensate system includes four motor-driven condensate pumps with a discharge shutoff pressure of approximately 350 psig.

CONDITIONS FOR USE: Plant startup, shutdown and power operation.

REFERENCE: Training System Description

REVIEWER: M. Jennex

DATE: 3/14/86

SYSTEM NAME: Auxiliary Feedwater System

SYSTEM ABBREVIATION: AFW

SYSTEM FUNCTION(S):

The Auxiliary Feedwater system provides feedwater to the steam generators secondary side during plant startup, shutdown, and hot standby operations and for events that result in a loss of main feedwater. The system consists of a motor-driven and a turbine-driven AFW pump that deliver water from the Auxiliary Feedwater Storage Tank (AFST) to each steam generator. Both the pumps are used to supply each of the four auxiliary feedwater flow control valves through independent flow control valve inlet isolation and check valves. From the flow control valves, flow is directed through the flow control valves outlet isolation and check valves to the main feedwater system piping and into the steam generators.

The AFW system automatically initiates if 2 out of 3 steam generator levels are 5% narrow range and decreasing. Each train can be manually initiated by depressing the appropriate AFWS initiate button.

When initiated, AFW flow is initially controlled by the preset auxiliary feedwater flow control valves. Flow can then be controlled by manually operating the AFW flow control valves or the main feedwater bypass regulators.

CONDITIONS FOR USE: Plant startup, shutdown and for events that result in a loss of main feedwater.

REFERENCE: Training System Description

REVIEWER: M. Jennex

DATE: 3/14/86

SYSTEM NAME: Steam Generator Blowdown System

SYSTEM ABBREVIATION: SGB

SYSTEM FUNCTION(S):

The Steam Generator Blowdown System provides blowdown from the secondary side of the steam generators. It consists of separate blowdown lines from each steam generator that join to form a common header that directs (routes) the blowdown to the ocean or to a flash tank (normal flow-path). The steam generators can be isolated by blowdown isolation valves located in the individual blowdown lines.

CONDITIONS FOR USE: At any time (plant operating or shutdown).

REFERENCE: Training System Description

REVIEWER: M. Jennex

DATE: 3/14/86

SYSTEM NAME: Electrical Power System

SYSTEM ABBREVIATION: EPS

SYSTEM FUNCTION(S):

The Electrical Power System provides ac and dc electrical power to equipment as necessary to accomplish respective functions. It consists of an offsite ac power supply and onsite emergency ac and dc power supplies. The emergency ac power supply is a two train system powered by separate diesel generators. The unit dc power supply is a two train system powered by separate battery banks. Two additional dc power trains are provided for backup power to the security system and for backup power to safety injection valve MOV 850C. Vital ac instrument power can be supplied by either the emergency ac power supply or the dc power supply via inverters.

CONDITIONS FOR USE: At all times (plant operating or shutdown).

REFERENCE: Training System Description

REVIEWER: M. Jennex

DATE: 3/14/86

SYSTEM NAME: Compressed Air Systems

SYSTEM ABBREVIATION: CAS

SYSTEM FUNCTION(S):

The Compressed Air Systems consists of the service air system and the instrument air system. These systems supply service, instrument, and control air at approximately 100 psig to equipment that require pneumatic power to accomplish their functions. Equipment in this category include:

- Pressurizer PORVs
- Atmospheric steam dump valves
- Condenser steam dump valves
- Letdown line isolation valves

The air supply to equipment located inside containment is automatically isolated on a containment isolation signal which closes the containment isolation valves in the air supply line(s).

CONDITIONS FOR USE: At all times (plant operating or shutdown).

REFERENCE: Training System Description

REVIEWER: M. Jennex

DATE: 3/14/86

SYSTEM NAME: Control Rod Drive Mechanism Cooling System

SYSTEM ABBREVIATION: CRC

SYSTEM FUNCTION(S):

The Control Rod Drive Mechanism (CRDM) cooling system provides heat removal from the control rod drive mechanisms. It consists of three air handling units used to circulate air through ducts and across the control rod drive mechanisms.

CONDITIONS FOR USE: Any time RCS temperature 150°F.

REFERENCE: Training System Description

REVIEWER: M. Jennex

DATE: 3/14/86

SYSTEM NAME: Spent Fuel Storage and Cooling System

SYSTEM ABBREVIATION: SFP

SYSTEM FUNCTION(S):

The Spent Fuel Storage and Cooling System ensures a subcritical geometric configuration and provides heat removal to maintain stored fuel within specified temperature limits. It includes the level instrumentation for the spent fuel pool.

CONDITIONS FOR USE: Any time spent fuel is in the spent fuel pool.

REFERENCE: Training System Description

REVIEWER: M. Jennex

DATE: 3/14/86

SYSTEM NAME: Turbine Electro-Hydraulic Control System

SYSTEM ABBREVIATION: EHC

SYSTEM FUNCTION(S):

The Turbine EHC System controls the turbine's speed or load as a function of governor valve position. It also allows the turbine stop and control valves to close upon receipt of a turbine trip signal.

CONDITIONS FOR USE: During power operation (any time the turbine is latched).

REFERENCE: Training System Description

REVIEWER: M. Jennex

DATE: 3/14/86

SYSTEM NAME: Rod Control System

SYSTEM ABBREVIATION: RRS

SYSTEM FUNCTION(S):

The Rod Control System controls the position of the control rods in the reactor core to control the fission rate in the reactor. Provisions for manual and automatic control exist.

CONDITIONS FOR USE: Plant operating or shutdown - Anytime the reactor trip breakers are closed. Automatic control only used above 15% power.

REFERENCE: Training System Description

REVIEWER: M. Jennex

DATE: 3/14/86

SYSTEM NAME: Reactor Cycle Sampling System

SYSTEM ABBREVIATION: RCSS

SYSTEM FUNCTION(S):

The Reactor Cycle Sampling System provides means for sampling primary systems. It consists of several trains of sampling lines and associated equipment and is used to sample the RCS, RHR system, CVCS, and Pressurizer. Results of analysis performed on these samples are used to detect fuel element leakage, unusual corrosion or erosion, evaluate demineralizer performance, and regulate RCS boron concentration through the Boron Analyzer and boric acid samples.

CONDITIONS FOR USE: At any time (plant operating or shutdown).

REFERENCE: Training System Description

REVIEWER: M. Jennex

DATE: 3/14/86

APPENDIX 6B

SCENARIO DESCRIPTIONS

The SONGS 1 SFTA Scenario Descriptions are provided in this appendix as follows:

Scenario	Description	Page
1	ATWS/Loss of Reactor Coolant	6B-1
2	Large Break LOCA	6B-3
3	Steam Generator Tube Rupture	6B-5
4	Secondary Break Inside Containment With Loss of Spray Capability	6B-7
5	Loss of All AC Power	6B-9

SCENARIO 1 ATWS/LOSS OF REACTOR COOLANT

Procedures Used:

SO1-1.0-10 Reactor Trip or Safety Injection
SO1-1.1-1 Response to Nuclear Power Generation/ATWS
SO1-1.0-20 Loss of Reactor Coolant
SO1-1.0.22 Post LOCA Cooldown and Depressurization

Initial Conditions:

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

Scenario Sequence:

1. Initialize at HFP
2. Simulate with mockup to fail all automatic and manual reactor trips
3. Inform crew of plant conditions
4. Allow sufficient familiarization time
5. Simulate with mockup turbine trip
6. Simulate with mockup 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 push-buttons, which do not function properly. A transition is made to SO1-1.1-1, from SO1-1.0-10.

Once the crew determines RCS emergency boration has initiated, efforts are made to locally open the reactor trip breakers in the 4KV room.

While opening reactor trip breakers, a SBLOCA develops. Safety injection occurs due to pressurizer low pressure. This causes a return to Step 1 of SO1-1.0-10, and the automatic actions of SI are verified. As plant symptoms are diagnosed, high containment pressure is identified and at Step 23 a transition is made to SO1-1.0-20.

SO1-1.0-20 is followed. In Step 5, the criteria for SI termination are not met. At Step 14 the need for further cooldown and depressurization is established, resulting in a transition to procedure SO1-1.0-22.

The scenario should be continued until RHR has been placed in service.

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

SCENARIO 2 LARGE BREAK LOCA

<u>Procedures Used:</u>	S01-1.0-10	Reactor Trip or Safety Injection
	S01-1.0-20	Loss of Reactor Coolant
	S01-1.0-23	Transfer to Cold Leg Injection and Recirculation
	S01-1.0-1	Critical Safety Function Status Trees
	S01-1.6-2	Response to Low System Inventory
	S01-1.2-1	Response to Inadequate Core Cooling
	S01-1-1.5-3	Response to High Containment Radiation Level
	S01-1.0-24	Transfer to Hot Leg Recirculation

<u>Initial Conditions:</u>

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

<u>Scenario Sequence:</u>

1. Initialize at HFP
2. Implement malfunction to fail both charging 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, return a charging pump to operable status

<u>Expected Response:</u>

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

S01-1.0-10 is immediately entered to verify automatic actuation of Safety Injection. In Step 23 of S01-1.0-10 a transition to S01-1.0-20 is made on abnormal containment conditions. At Step 5, conditions prevent SI termination and at Step 7 containment spray is verified operational. The operating crew continues with SO1-1.0-20 until Step 15a is reached.

At this point, RWST level decreases to less than 21% causing a transition to procedure SO1-1.0-23.

High containment radiation alarm initiates; requiring the operating crew to implement SO1-1.5-3, concurrently with SO1-1.0-23.

During the transfer to cold leg injection and recirculation, recirculation capability is lost when the charging pumps trip. This situation causes a transition to SO1-1.0-1. SO1-1.0-1 directs the crew to SO1-1.6-2. The crew starts the CVCS test pump after both charging pumps trip. At Step 8 during performance of SO1-1.6-2, core exit temperatures have increased to a level sufficient to implement SO1-1.2-2. Upon completion of SO1-1.6-2, a transition is made back to SO1-1.0-23. After SGs are depressurized, recirculation capability is reestablished with a start of a charging pump. The transfer to cold leg injection and recirculation continues to completion.

After the plant is stable on cold leg injection and recirculation, SO1-1.0-24 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 STEAM GENERATOR TUBE RUPTURE

Procedures Used:	SO1-1.0-10 Reactor Trip or Safety Injection SO1-1.0-40 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 operating. An automatic SI occurs as a result of pressurizer pressure decrease, which causes the operating crew to implement SO1-1.0-10. In Step 24a of SO1-1.0-10, abnormal radiation indication from the air ejector causes a transition to SO1-1.0-40. Once in SO1-1.0-40, the ruptured SG is identified while monitoring SG blowdown activity. This action is followed by an RCS cooldown and depressurization to recover pressurizer level. At Step 17 of SO1-1.0-40, SI termination criteria are met. Normal pressurizer spray and PORV cycling are used to control pressurizer level, and at Step 20 of SO1-1.0-40 the RCS is depressurized.

SG level is maintained using one condensate pump. This process is repeated until RCS temperatures are less than 350^{o}F and main steam pressure is less than 350 psig to satisfy requirements for RHR system start.

Scenario Termination Criteria:	Discretion of CRDR Coordinator

SCENARIO 4 SECONDARY BREAK INSIDE CONTAINMENT WITH LOSS OF SPRAY CAPABILITY

Procedures Used:

S01-1.0-10	Reactor Trip or Safety Injection
S01-1.0-30	Loss of Secondary Coolant
S01-1.5-1	Responses to High Containment Pressure
S01-1.0-31	SI Termination Following Loss of Secondary Coolant

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

Scenario Sequence:

1. Initialize at HZP
2. Implement malfunction to fail refueling water motors.
3. Inform crew of plant conditions
4. Allow sufficient familiarization time
5. Implement malfunction for steam line break inside containment.
6. Return refueling water 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 S01-1.0-10.

Containment pressure increases, exceeds the containment spray initiation setpoint, but containment spray does not initiate. Containment pressure is greater than 20 psig, which forces a transition to S01-1.5-1.

An investigation into the containment spray auto start failure is begun. After a brief time, the SRO receives a report that both refueling water 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 SO1-1.0-10 at the completion of SO1-1.5-1.

At Step 22 of SO1-1.0-10, a transition is made to SO1-1.0-30. After the faulted SG is identified, the operating crew continues with SO1-1.0-30.

In SO1-1.0-30, abnormal containment safety injection termination criteria are met. At Step 10 of SO1-1.0-30, the operating crew is directed to SO1-1.0-31 for SI termination criteria.

In SO1-1.0-31, 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 ALL AC POWER

Procedures Used:

SO1-1.0-10	Reactor Trip or Safety Injection
SO1-1.0-60	Loss of All AC Power
SO1-1.0-61	Loss of All AC Power Recovery
SO1-1.3-4	Response to Loss of Steam Dump Valves

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

Scenario Sequence:

1. Initialize at HFP
2. Implement malfunction to fail all Diesel generators
3. Inform crew of plant conditions
4. Allow sufficient familiarization time
5. Implement malfunction for unit trip coincident with unit blackout
6. Implement malfunctions to fail instrument air to the main condenser steam dumps and the steam generator atmospheric relief valves at the direction of the CRDR Coordinator.

Expected Response: A unit trip from full power, EOL, coincident with a failure of the Station Auxiliary Transformer results in a unit blackout. SO1-1.0-10 is implemented by the operating crew, but the auto-start failure of all emergency diesel generators forces an immediate transition to SO1-1.0-60.

Procedure S01-1.0-60 is implemented to restore AC power. At Step 26, manual start attempts are successful on one diesel generator. A transition is made to Step 28 to verify at least one diesel generator is operating, and in Step 38 a transition is made to S01-1.0-61.

During this recovery, the instrument air to the main condenser steam dump valves and the SG atmospheric relief valves is lost and cannot be reestablished. This provides an opportunity for the operators to implement S01-1.3-4.

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

7.0 VERIFICATION OF TASK PERFORMANCE CAPABILITIES

7.1 OBJECTIVE

The objective of the Verification of Task Performance Capabilities was to systematically verify that instrumentation and controls identified in the Task Analysis as being required by the operator were:

- Present in the Control Room

- Effectively designed to support correct task performance

The focus of this verification effort was on control room instruments and equipment, not on operator skills or knowledge. The premise is that the control room should provide all information and control capabilities called for by the operator task action requirements in a manner suitable to assure minimum potential for human error.

7.2 METHODOLOGY

The Verification of Task Performance Capabilities used a two-phase approach to achieve the objective stated above. In the first phase, the availability (i.e., presence or absence) of the instrumentation and controls necessary to implement operator tasks as required in the Task Analysis was confirmed. The second phase evaluated the suitability (i.e., effectiveness or usability) of the man-machine interfaces provided by the displays, controls and other control room features to support operator task accomplishment.

General Physics personnel performed both phases of the verification effort with support from SCE as necessary. The General Physics personnel who performed the verification were also involved in the System Function and Task Analysis review. This provided a high level of continuity between the SFTA and verification activities.

7.2.1 Verification of Availability

This task required utilization of the control room inventory of instrumentation and controls (previously discussed in Section 4.0) and the operator task specifications as defined in the System Function and Task Analysis (Section 6.0).

A comparative process was performed of the instrumentation and control input-output requirements specified in the task analysis to the input-output capabilities shown in the inventory. The task analysis worksheets developed in Section 6.0 provided the documentation of the results of the comparison.

Once the tasks, decision requirements, and information and control requirements were specified and documented on the task analysis worksheets, the existing instrumentation and controls the operator used or could use for each procedural step were documented based on the control room inventory. All I&C needed or available to either (1) initiate, maintain, or remove a system from service, (2) confirm that an appropriate system response has or has not occurred, i.e., feedback, or (3) make a decision regarding plant or system status, was listed in the "Means", "I&C No." and "Panel" columns of the task analysis worksheets. The "Means" column referred to how the information and control requirements are presented on the existing control boards (e.g., switch, meter, etc.). The "I&C No." column provided the specific control or instrument identification number. The "Panel" column provided the specific panel number the control or instrument was located on.

The presence or absence of the required Instrumentation and Controls was noted by a "Yes" or "No" in the "Availability" column on the task analysis worksheets. When required Instrumentation and Controls were not available to the operator, such occurrences were identified as a HED and documented accordingly on a HED form. Also identified were those excess or abandoned instruments and controls. These were likewise documented as HEDs.

7.2.2 Verification of Suitability

The objective of the verification of suitability is to identify interface problems that may effect performance of operator tasks but may not be evident when control room components are examined without reference to specific task use (as in the control room survey).

The methodology for performing the suitability verification of required instrumentation and control was to compare the equipment to a set of acceptance criteria as functionally illustrated in Figure 7-1. These criteria establish appropriate verification that the equipment is suitable to meet the demands of emergency contingencies. The acceptability of the equipment is based on the ability of the instrumentation and controls to provide all of the following:

- Appropriate information to complete operator task
- Direct system status information
- Usable equipment (i.e., proper scale, control range, etc.)

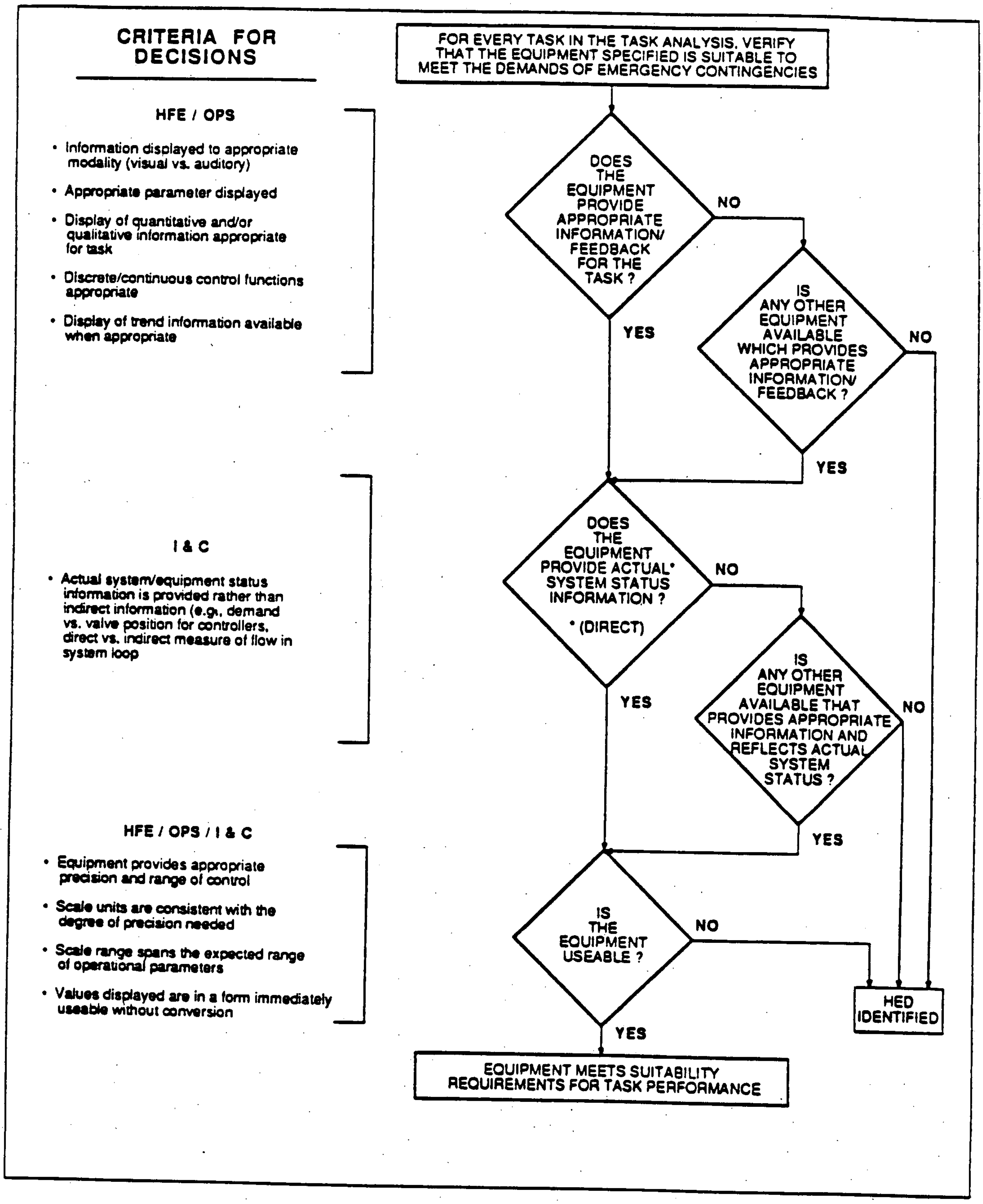
If the instrumentation and controls satisfied these criteria, it was noted by a "yes" in the "Suitability" column on the task analysis worksheets. For instances of non-compliance with this criteria, a "no" was recorded and the problem was documented on a HED form.

7.3 VERIFICATION RESULTS

The results of the verification of operator task performance capabilities is a consolidated listing of control room instrumentation and controls requirements including availability and suitability of existing control room equipment. Instances of nonconformance with the availability and suitability criteria were documented as HEDs 300 through 350.

Figure 7-1

FLOWCHART OF DECISION PROCESS FOR
VERIFYING EQUIPMENT SUITABILITY



8.0 VALIDATION OF CONTROL ROOM FUNCTIONS

8.1 OBJECTIVE

The objective of the validation process was to determine whether the functions allocated to the control room operating crew could be accomplished effectively within the structure of the SONGS 1 plant specific EOIs, and the control room design as it exists.

The process of validation incorporated a dynamic aspect to the overall control room design review process. The previously described aspects of the CRDR analyzed the control room equipment and design from a "static" perspective. The validation process exercises the control room man-machine interface and considers function execution with respect to operator task performance during interactive operations. The validation process essentially evaluates the integration of control room design configuration, operator skills and training, and requirements of the EOIs through the scenarios.

The primary emphasis of validation was placed on the operator's ability to ascertain and evaluate plant status during transients with the existing control room display systems. The number of process parameters displayed, the format of the displayed data, and the dynamic response needed from instruments, displays, and indicators are all factors considered in the validation of control room functions.

8.2 METHODOLOGY

The validation process was performed by having a normal control room operating crew walk through the event scenarios developed in the system function review and task analysis. The scenario walk-throughs illustrate performance dynamics and permit assessment of operating crew interaction with each other and the control room work stations. Functional relationship of instrumentation, operator workload, and feasibility of task completion can be assessed in the context of operating sequences that exercise the functions allocated to the control room operating crew. Other control room aspects evaluated by the validation include visual and communication links, manning levels, and operator traffic patterns.

The scenario walk-throughs were performed in the SONGS 1 full-scale control room photographic mock-up. A normal control room operating crew that recently completed their annual simulator retraining was selected to perform the walk-throughs. All activities relating to the scenario walk-throughs were video taped by General Physics for documentation and subsequent post-walk-through reviews.

The five event scenarios developed as part of the system function review and task analysis established the basis for the operator walk-throughs. These scenarios are discussed in Section 6.0 and provided in Appendix 6B. Prior to walk-through of the scenarios, the participating control room personnel were briefed on the purpose and objectives of the walk-throughs, and on how they would be performed. The operators were informed that individual performances would not be graded in any way and to react as they normally would under the given conditions.

Each event scenario was described prior to initiation of the walk-through. The operating scenario assumptions and plant conditions were defined. Only one scenario was performed at a time. Three phases were performed for each scenario.

8.2.1 Real-Time Walk-Throughs

Operators initially performed each walk-through in simulated real-time. Since the walk-throughs were performed in a photographic mock-up, real-time sequencing of events was estimated based on plant specific transient documentation. An SCE training instructor acted as the scenario orator providing changes in plant status as dictated by the scenario sequence of events, and providing plant parameter indication as requested by the operators. A human factors engineer and an operations expert observed the crew performing their activities and noted validation issues.

8.2.2 Real-Time Debriefing

Following the real-time walk-throughs, General Physics human factors personnel held a debriefing with the operating crew. The operators were asked to discuss the scenario and indicate any errors or problems relating to the objectives of the validation that were encountered in the walk-through. Errors or problems that were identified were explored to clarify the area of the problem, associated circumstances, significance, and interrelationships with operator tasks. All information gathered was documented for investigation of potential HEDs.

8.2.3 Slow-Time Walk-Throughs

Subsequent to the real-time walk-throughs and debriefings, a slow-time walk-through of each scenario was performed. During these walk-throughs the operators described the actions they were taking, identified information sources including annunciators for decision points, how information was used, what controls were used, the expected system response, verification of responses, and contingency actions if responses were not obtained. The human factors personnel conducting these walk-throughs utilized the ability of stopping the scenario to explore scenario variations and potential impact of changes in control room features or configurations that might improve operational dynamics and task accomplishment. All information gathered was documented for investigation of potential HEDs.

8.3 VALIDATION RESULTS

Upon completion of the validation process all problems relating to the operational dynamics of the control room including sequencing of operator tasks in emergency conditions were documented. The video-taping of all phases of the scenario walk-throughs allowed detailed review of potential problems, and an effective method of analyzing the real-time walk-throughs during which there was a multitude of concurrent operator tasks. Any dynamic performance problems that were identified during the validation phase were documented as HEDs 351 through 391.

9.0 HED ASSESSMENT

9.1 OVERVIEW

The CRDR processes described in Sections 3.0 through 8.0 were performed to evaluate all aspects of the control room man-machine interface. In the course of evaluating these aspects, situations of less than optimal design were documented as human engineering discrepancies (HEDs). These discrepancies required analysis and interpretation to establish their potential safety significance. The HED Assessment phase of the control room design review established a means of correcting or minimizing the effects of HEDs by identifying cost-effective solutions that provide the necessary design improvements. This section describes the methodology for assessing human engineering discrepancies and selecting control room design improvements.

9.2 OBJECTIVES

The objective of the HED Assessment phase was to systematically evaluate all HEDs identified through the CRDR Review Phase in a consistent manner in order to establish a framework for implementation of modifications to enhance operator effectiveness. Through the HED assessment process, priority ratings were assigned to individual, and in some instances generic, control room human engineering discrepancies. These priority ratings established the degree of safety significance associated with the discrepancy and provided a key input to the implementation process described in Section 10.0.

9.3 METHODOLOGY

In order to provide a thorough and comprehensive review of all control room HEDs, a program was developed to organize discrepancies in a systematic format. This program established an organization structure of HEDs based on characteristics of the various discrepancies. The criteria for group selection were based on the interrelationships of control room panel location and problem category. These HED packages were specifically organized to account for interaction between HEDs that resulted in cumulative effects due to the concurrent existence of several HEDs impacting the same operating activity or panel, which could make a less than optimum operating situation much worse.

Once the organization structure for HED grouping was established, the HED packages were processed by the HED Assessment Team, the HED Evaluation Team, and the Plant Modification and Review Committee (PMRC). The organization of these groups and their functional responsibilities in HED processing is illustrated in Figure 9-1.

The procedure for HED assessment is in accordance with the process described in Section 9.3.1.

9.3.1 <u>Procedure For Processing HEDs</u>

9.3.1.1 HED Assessment Team

Following the identification of HEDs as documented on HED forms, the HED Assessment Team performed the initial evaluation of the HEDs. The Assessment Team received an orientation of the overall assessment process and the objectives and responsibilities of the Assessment Team.

The Assessment Team was composed of the CRDR Team Leader and six other members with expertise in the following areas:

- Nuclear Licensing
- Instrumentation and Controls Engineering
- Human Factors Engineering
- Senior Reactor Operator
- Nuclear Engineering
- Balance-of-Plant (BOP) Engineering

All members of the team had an equal vote, and simple majority rule prevailed.

The HED Assessment Team processed the discrepancies through the following steps:

A. Utilizing the HED assessment criteria, as defined in Table 9-1, verify that the discrepancy described on the HED form constitutes a human engineering discrepancy.

B. If the HED was deemed not to be a control room man-machine interface problem, the rationale was documented on the HED form in the "Recommendations" section and no priority rating was assigned (i.e., N/A was entered on the HED form).

C. For all other HEDs, the procedure described in Table 9-2 was implemented to prioritize the HED. This categorization considered probability of error, safety significance, and Technical Specification conformance. The criteria described in Tables 9-3 and 9-4 provided a reference for evaluating these considerations.

D. Formulate and describe alternative conceptual design improvements for correcting the HED.

E. Select the preferred conceptual design alternative.

Meeting minutes were prepared to document all Assessment Team meetings held to discuss, resolve or make recommendations on HEDs.

Upon completion of the Assessment Team activities, all HEDs were forwarded to the HED Evaluation Team for the second phase in the assessment process.

Prior to initiation of the second phase, an inconsistency in the format of the HED form was identified. As previously determined during the SONGS 1 CRDR In-Progress Audit, the format of the HED form should be revised to require determination of priority prior to recommended solution. The concern associated with the existing format was that with the "recommended solution" identified prior to "determination of priority", the priority rating might be biased by the recommendation. It was planned to reverse the order in which these items appear on the HED form. However, this format revision did not occur prior to the completion of Assessment Team activities. To alleviate the potential concern, the Assessment Team performed a verification effort to ensure consistent prioritizations were applied and that no biased ratings were given. The results of this verification were that prioritizations had been assigned in a consistent manner and that no biasing was evident.

Subsequent to the verification effort, the NRC was informed that the format revision to the HED form did not occur, and that a verification process had been completed and determined that all priority ratings had been assigned appropriately.

9.3.1.2 HED Evaluation Team

The second phase in the processing of HEDs was the responsibility of the HED Evaluation Team. As with the Assessment Team, an orientation was provided for the Evaluation Team describing the overall assessment process and the objectives and responsibilities of the Evaluation Team.

The Evaluation Team was composed of the CRDR Team Leader and eight other members from the following areas:

- Station Management
- Station Operations
- Station Training
- Shift Technical Advisor
- Instrumentation and Controls Engineering
- Nuclear Engineering
- Nuclear Licensing
- Human Factors Engineering

All members of the team had an equal vote, and simple majority prevailed.

It is noted that the human factors consultant, although a member of the Evaluation Team, participated in an "off-line" capacity only. That is, the HF consultant did not attend Evaluation Team meetings but performed in a review capacity only. Any changes to the recommended solutions or priority ratings assigned by the Assessment Team were reviewed by the human factors consultant to verify human factors considerations were maintained.

The HED Evaluation Team processes the HEDs through the following steps:

A. Review the HED documentation for validity and verification of prioritization.

B. Review the recommended conceptual design improvement and consider the practical acceptability of the recommended correction including the degree of difficulty of the recommended solution. Also, evaluate the impact of the proposed modification in relation to introduction of new or different HEDs.

C. Endorse the recommendation of the assessment team or discuss and resolve disagreements. If agreement cannot be reached, the evaluation team has the prerogative to document the disagreement and make its own recommendation on the HED form.

Meeting minutes were prepared to document all Evaluation Team meetings held to discuss, resolve or make recommendations on HEDs. The minutes document the date of the meeting, the names of personnel involved and their job functions, the serial numbers of the HEDs discussed, and highlights of any unusual considerations or pertinent discussions that would not ordinarily be documented on the HED report form.

Upon completion of the Evaluation Team activities, the HED forms provided documentation of the results of the HED Assessment and Evaluation Teams' decisions and recommendations. A numerical listing of completed HEDs is provided in Appendix A located in Volume 2 of this report.

Once the HED Evaluation Team documented all decisions and recommendations, the HEDs were forwarded to the CRDR Team Leader for coordination of the remaining HED Assessment activities. The CRDR Team Leader initiated an effort to prepare documentation to support presentation of CRDR recommended modifications to the Plant Modification Review Committee. This documentation consisted of descriptions of the proposed modifications and control panel drawings illustrating panel changes resulting from the recommended modifications. The purpose of this effort was to functionally illustrate the impact of the recommended modifications to the PMRC members without the burden of reviewing individual HEDs. In this manner, the PMRC was able to readily conceptualize the scope of the recommended modifications. It is noted that although this effort was intended to streamline the PMRC review, all HED Evaluation Team recommendations were submitted to the PMRC.

The documentation generated in this effort is provided in Appendix B located in Volume 3 of this report. It is noted that this documentation provides explicit detail of the HED Evaluation Team recommendations but does not reflect the results of the PMRC decisions including rejections or alterations to the recommendations. These decisions are documented in Section 9.4.

9.3.1.3 Plant Modification Review Committee

The final phase in the HED assessment process was to provide a management level review of all proposed CRDR modifications to ensure meaningful control room improvements will be provided. The Plant Modification Review Committee (PMRC) provided a medium to incorporate technical organization management considerations. The PMRC was composed of the following members:

- Station Manager
- Operations Manager
- Station Technical Manager
- Project Manager
- Maintenance Manager

The methodology for performing the PMRC review was for the PMRC members to assign individuals from their respective organizations to independently review all proposed control room modifications. The review criteria was based on safety, compliance, technical specifications, cost benefit, operation, and ALARA. The PMRC had authority to accept or reject proposed modifications based on scope, priority and budget category.

Upon completion of the independent reviews, meetings were held between the various disciplines to discuss results of the review and resolve discrepancies. Subsequently, the PMRC held a formal meeting to document the results of their independent reviews including approval, rejection and alternatives to the modifications proposed by the HED Evaluation Team. These decisions are discussed in Section 9.4.

9.4 HED ASSESSMENT RESULTS

The results of the HED Assessment and Evaluation Teams decisions and recommendations are provided on the individual HED forms. A numerical listing of the HEDs is provided in Appendix A located in Volume 2 of this report. Figure A-1 located in Volume 2 illustrates the distribution of priority ratings for the HEDs. As can be seen in this figure, the SONGS 1 control room contains a relatively few number of high and medium priority discrepancies and a large number of lower priority problems. This type of distribution is considered appropriate for SONGS 1 based on the significant amount of operating experience from which significant control room deficiencies would have previously been identified and corrected. Conversely, implementation of control room modifications throughout the operating history of SONGS 1 were not necessarily performed in a manner consistent with the current human factors engineering guidelines, and have thus resulted in a large number of minor control room inconsistencies.

In regards to the decisions of the Plant Modification Review Committee, the majority of the recommendations were approved as proposed by the HED Evaluation Team. Section 9.4.1 summarizes the CRDR modifications that were approved for implementation. Sections 9.4.2 and 9.4.3 provide discussions of the recommendations that were not approved or modified by the PMRC.

9.4.1 Summary of CRDR Modifications

The control room changes identified and approved for implementation through the various CRDR processes include surface enhancements and various types of hardware modifications intended to improve the man-machine interface in the control room. The following is a brief summation of the control room changes to be implemented. Although not inclusive of all approved modifications, the following presents an overview of the significant control room changes that will be implemented as a result of the CRDR.

Enhancements

A. Provide functional system demarcation of the control panels by repainting the panels and all instrument bezel color coding by system.

B. Implement full labeling scheme to provide a five-level labeling hierarchy with clear, concise and consistent information; relocate the labels to the top of the instruments; and replace pushbutton labeling wherever required.

C. Provide scale coding and re-scaling for indicators and recorders to show key operating information, legibility, engineering units and proper ranging.

D. Prioritize annunciator system by the use of colored windows and replace the legends to improve size, consistency and clarity of characters.

E. Address glare problems by utilizing non-glare paint and non-glare lenses where indicated.

Modifications

A. Redesign of the Emergency Diesel Generator Control Panels C41 and C42 to improve control/display integration.

B. Significant design modifications to the Nuclear Control Auxiliary Panel C09 including over forty (40) I&C component relocations.

C. Deletion of 16 abandoned controls and indicators on various panels.

D. Component relocations on the Remote Shutdown Panel C38, Auxiliary Equipment Control Panel C13, and Recorder Panel C05.

E. Replacement of several indicators, recorders, and controls judged unsuitable.

F. Annunciator system upgrades including elimination of boric acid heat trace nuisance alarms via addition of new alarm points, color code prioritization, tile replacement to improve legibility, consistency and accuracy, and elimination of abandoned points.

G. Miscellaneous actions to complete in the areas of Communications, Environmental, and Procedures.

H. Installation of protective hinge covers and barriers.

I. Installation of additional indicators and controls.

9.4.2 HEDs Modified by the PMRC

During the PMRC review of the HED Evaluation Team recommendations, the proposed modifications to correct the following HEDs were modified. The alternate corrective actions were reviewed by the HED Evaluation Team and were determined to be acceptable.

Hed No. 307

Description of Proposed Modification - This HED identified that the SLSS load group lights are normally illuminated and upon receipt of an abnormal condition go out. This configuration is opposite from stereotype. The HED

Evaluation Team recommended to reverse the illumination and simplify the load group display to essential information only.

Alternative - The PMRC determined that the existing illumination configuration was acceptable. With the existing configuration routine operability testing of the load group lights is not necessary. Further, the operators are aware of the illumination configuration. Consequently, only a modification to simplify the load group display to essential information was approved for implementation.

HED No. 430

Description of Proposed Modification - This HED identified the need for sump pump controls in the control room. The HED Evaluation Team recommended to provide these controls in the control room.

Alternative - The PMRC determined that local control of these pumps at the breaker power supply would be sufficient for operator needs. Therefore, in lieu of pump controls in the control room, local controls will be provided.

9.4.3 HEDs Not Approved by the PMRC

During the PMRC review of the HED Evaluation Team recommendations, the proposed modifications to correct the following HEDs were not approved for implementation. The HED Evaluation Team reviewed these HEDs to evaluate possible alternatives that would be satisfactory to the PMRC. No such alternatives were approved.

Description	HED No.	Priority Level
1. Sequentially number K01-C annunciator windows and provide legend label with alarm details.	13	9

Description	HED No.	Priority Level
2. Differentiate switch light bezels from indicating lights.	114	8
3. Provide ringback for selected annunciator windows.	127	7
4. Increase audible annunciator alarm level to recommended level above ambient.	142	8
5. Revise Diesel Generator panel annunciator flashing rate to less than 5/sec.	144	8
6. Resolve problem of more than 50 annunciator tiles in several matrices.	147	8
7. Install intercom between control room and other plant areas.	168	9
8. Relocate 8 indicators on panel C13 for control display integration.	*	
9. Rearrange annunciator titles for functional grouping and location.	183	7
10. Provide recording as necessary for multi-channel alarms.	190	8

* This proposed modification was part of a group of proposed changes relating to control display integration and functional grouping. Only this modification was not approved, all others were approved.

	Description	HED No.	Priority Level
11.	Split windows on TG first-out alarm panel and add sequencer generated LOP alarm on RP first-out alarm panel.	220	8
12.	Relocate CIS valves and controls on C13A and 13 to provide logical valve ordering.	311	8
13.	Relocate ZLC-1105 indicating light from panel C13 to proximity of controller on panel C03.	324	9
14.	Relocate PCV-430 C and H indicating lights from panel C09 to proximity of controllers on panel C03.	334	9
15.	Provide ammeters for the refueling water pump controls.	366	9
16.	Improve controllability of main feedwater controls.	387	7
17.	Provide background music in the control room.	410	9
18.	Provide boric acid heat trace recorder on a control room back panel.	424	5**

** This modification was proposed in order to provide control room verification capability of a nuisance alarm for boric acid heat trace temperature. This nuisance alarm will be corrected by HED 466. Consequently, the heat trace recorder is no longer required in the control room.

	Description	HED No.	Priority Level
19.	Modify or replace reheater warm-up control RMC-3.	426	8
20.	Relocate feedwater flow integrators and steam dump elapsed timer from panel C09 to a back panel.	447	9
21.	Provide LOVATS completion alarm and LOVATS failure to function in the time limit specified alarm.	464	8
22.	Relocate steam dump loss of MWe turbine trip and safety injection annunciators and blank over existing tiles.	468	8

Figure 9-1
ORGANIZATION CHART FOR HED PROCESSING

CRDR Team Member	Identifies HED Prepares HED Form
<u>HED Assessment Team</u> (4 of 7) CRDR Team Leader - J. Prickett (Chair)/J. Ibarra SCE Nucl Licensing - M. Thomas/J. Ibarra SCE Instr & Controls Engr - A. Hernandez/ E. Siemion Human Factors Consultant - L. Schroeder/M. Dawson Senior Reactor Operator - J. Kroeger/W. McGhee Nuclear Engr - J. Ibarra/M. Thomas BOP Engr - E. Siemion/D. Snuggs	Performs Review Determines if HEd should be HED Establishes Categorization and Prioritization Formulates Alternative Corrections Recommends Corrective Action
<u>HED Evaluation Team</u> (5 of 9) SCE Sta/Mgmt - J. Tate (Chair)/R. Krieger SCE Sta/Operations - W. McGhee/J. Kroeger SCE Sta/Training - M. Kirby/J. Kroeger SCE Shift Tech Advisor - M. McKinley/ J. Reynoso CRDR Team Leader - J. Prickett/J. Ibarra SCE Instr & Controls Engr - A. Hernandez SCE Nucl Engr - J. Ibarra/M. Thomas SCE Nucl Licensing - M. Thomas/J. Ibarra Human Factors Consultant - L. Schroeder*	Reviews Assessment Findings Confirms/Rejects Assessment Recommendation Endorses Recommendation/ Resolves Differences/Overrides Recommendation
<u>Plant Modifications Review Committee</u> Station Manager - H. E. Morgan Maintenance Manager - D. E. Shull Operations Manager - R. W. Krieger Station Technical Manager - J. Reilly Project Manager - J. J. Wambold	Approves/Rejects Recommendations Authorizes/Rejects Corrective Action Considers Cost/Schedule Factors
<u>SCE Executive Approval</u> Vice President and Site Manager - H. B. Ray Vice President Nuclear Engineering, Safety and Licensing - K. P. Baskin Vice President Engineering and Construction - R. Dietch	Overall Budget/Cost Approval

* General Physics provides 100% onsite coverage in the assessment phase and 100% offline coverage for the Evaluation Phase. Team Conference participation will occur only if the Evaluation Team decides to change a prioritization or recommendation for corrective action.

TABLE 9-1
HED ASSESSMENT CRITERIA

1. Could the HED cause a unit trip or loss of equipment availability?

2. Could the HED result in personnel injuries?

3. Could the HED cause confusion, create difficulty for the operator, or cause him a problem?

4. Could the HED increase the operator's mental workload or distract him from his duties?

5. Could the HED hamper the operator's ability to see or read accurately or to hear clearly?

6. Could the HED cause a delay or degrade signal or information feedback to the operator?

7. Could the HED contribute to or make stressful situations worse?

8. Could the HED lead to the inadvertent activation or deactivation of controls?

9. Does the HED seem likely to cause a specific type of error?

10. Could the HED detract from the operator's ability to correctly or effectively manipulate the controls?

11. Will the HED contribute to operator discomfort or fatigue?

12. Could the HED degrade control room personnel performance?

13. Can the HED actually be considered a defect?

14. Is the HED one of a larger group of similar HEDs that could have an adverse cumulative effect?

15. Does the HED violate conventions or practices followed in control rooms or by the nuclear industry?

Table 9-2

CATEGORIZING AND ESTABLISHING PRIORITIES FOR CORRECTION OF HEDS
(Sheet 1 of 3)

Described below is a method of evaluating the importance or significance of individual HEDs and, based upon this, assigning a priority for their correction. This method is an approved, published means for normalizing random but rated variables, and is adapted from D. Meister's Human Factors Theory and Practice dated 1971.

Three factors (i.e., the "W" factors as defined below) were considered for relative importance and corresponding numeric value weightings were assigned using the following comparison matrix:

1.	Potential for Error (W_1)	1
2.	Degree of Safety Importance (W_2)	1
3.	Potential for Unsafe Condition or Technical Specification Violation (W_3)	3

		First Pass Relative Weight	Readjusted Relative Weight
1.	Potential for Error	2/3 = 0.667	0.555 (W_1)
2.	Degree of Safety	0/3 = 0.000	0.167 (W_2)
3.	Potential for Unsafe Condition or Technical Specification Violation	1/3 = 0.333	0.278 (W_3)
		1.000	1.000

A scale to assign a relative magnitude (i.e., the "M" factor) to each individual "W" factor was established as follows:

0	1	2	3	4	5
None	Very Low	Low	Moderate	High	Very High or Documented

The relative importance of each HED was then determined by employing the following formula:

Relative Weight of Factor 1 = W_1 = 0.555
Relative Weight of Factor 2 = W_2 = 0.167
Relative Weight of Factor 3 = W_3 = 0.278
Scale Magnitude of Factor 1 = M_1 = Variable *
Scale Magnitude of Factor 2 = M_2 = Variable *
Scale Magnitude of Factor 3 = M_3 = Variable *

HED Point Value = $(W_1)\ (M_1) + (W_2)\ (M_2) + (W_3)\ (M_3)$

Where the higher the point value of the HED, the more critical is the need for correction.

* The Assessment Team held meetings to assess each HED using a 0 to 5 scale to determine the Scale Magnitudes of Factor 1, 2 & 3 (M_1, M_2, and M_3). Tables 9-3 and 9-4 served as guidance in this process. Each member had an equal vote and simple majority rule prevailed.

Example:

An HED that has resulted in a documented error of low safety importance, and having resulted in a documented Technical Specification violation would have the following calculated point value:

$$M_1 = 5$$
$$M_2 = 2$$
$$M_3 = 5$$

$$(0.555)\ (5) + (0.167)\ (2) + (0.278)\ (5) = 4.499$$
$$[(W_1)\ \ (M_1) + (W_2)\ \ (M_2) + (W_3)\ \ (M_3) = \text{HED Point Value}]$$

Table 9-2 (cont.)
(Sheet 3 of 3)

By reference to the following table of HED point value ranges, this example HED with a HED Point Value of 4.499 would be placed in Priority Level 2 requiring prompt correction. This is the equivalent of the NUREG-0801 Category IB, which is also a Priority 2 prompt correction HED.

Based upon the HED point value totals, nine priority levels for correction were established in an approximate correspondence to the NRC's total number of categories as follows:

Priority Level for Correction	NRC Equivalent Category	HED Point Value Range	Category for Modification*
1	IA	4.667 to 5.0	Prompt
2	IB	4.334 to 4.666	Prompt
3	IC	4.0 to 4.333	Prompt
4	IIA	3.5 to 3.999	Near-Term
5	III	3.0 to 3.499	Near-Term
6	IIB	2.5 to 2.999	Near-Term
7	ID	2.0 to 2.499	Near-Term
8	IIC	1.0 to 1.999	Long-Term (Mandatory)
9	IV	0 to 0.999	Long-Term (Optional)

* As discussed in Section 10.0 the actual implementation schedule of HED corrections will be established in accordance with the Integrated Implementation Schedule (IIS).

TABLE 9-3
HED POTENTIAL FOR ERROR ASSESSMENT CRITERIA
(Modified from NUREG-0800)

To what extent do you agree with the following?

1. This discrepancy will cause undue operator fatigue.
2. This discrepancy will cause operator confusion.
3. This discrepancy will cause operator discomfort.
4. This discrepancy presents a risk of injury to control room personnel.
5. This discrepancy will increase the operator's mental workload (for example, by requiring interpolation of values, remembering inconsistent or unconventional control positions, etc.).
6. This discrepancy will distract control room personnel from their duties.
7. This discrepancy will affect the operator's ability to see or read accurately.
8. This discrepancy will affect the operator's ability to hear correctly.
9. This discrepancy will degrade the operator's ability to communicate with others (either inside or outside the control room).
10. This discrepancy will degrade the operator's ability to manipulate controls correctly.
11. This discrepancy will cause delay of necessary feedback to the operator.
12. Because of this discrepancy the operator will not be provided with positive feedback about control tasks(s).
13. This discrepancy violates control room conventions or practices.
14. This discrepancy violates nuclear industry conventions.
15. This discrepancy violates societal stereotypes.
16. Operators have attempted to correct this discrepancy themselves (by self-training, temporary labels, "cheaters," "helper" controls, compensatory body movements, etc.).
17. Tasks in which this discrepancy is involved will be highly stressful (i.e., highly time constrained, of serious consequence, etc.).

TABLE 9-3
HED POTENTIAL FOR ERROR ASSESSMENT CRITERIA (Cont.)

18. This discrepancy will lead to inadvertent activation or deactivation of controls.

19. If this discrepancy causes a specific error, it is probable that another error of equal or more serious consequences will be committed.

20. This discrepancy is involved in a task which is usually performed concurrently with another task (e.g., watching water level while manipulating a throttle valve control).

TABLE 9-4
HED PLANT IMPACT ASSESSMENT CRITERIA

To what extent do you agree with the following:

1. This discrepancy involves controls or displays that are used by operators while executing emergency procedures.

2. It is likely that the error caused by this HED would result in:

   a. A violation of a technical specification, safety limit or a limiting condition for operation.

   b. The unavailability of a safety-related system needed to mitigate transients or system needed to safely shut down the plant.

3. This discrepancy involves controls or displays that are part of an engineered safety function or are associated with a reactor trip function.

4. This discrepancy involves control or display problems that would not be readily identified or corrected by alarms, interlocks or other instruments.

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10.0 IMPLEMENTATION OF CRDR MODIFICATIONS

10.1 OBJECTIVE

The objective of the Implementation phase of the CRDR process is to establish a program for scheduling the approved CRDR control room modifications. In accordance with NUREG-0700, the emphasis of this program is on prompt correction of discrepancies with significant safety consequences. SONGS 1 currently has an Integrated Implementation Schedule (IIS) for determining the implementation schedule for capital projects at San Onofre Unit 1. CRDR control room modifications will be scheduled in accordance with the IIS.

10.2 INTEGRATED IMPLEMENTATION SCHEDULE

10.2.1 Development of the IIS

SCE developed the IIS in order to establish a program for implementation of San Onofre Unit 1 capital modifications in a stable, controlled manner. The goal of the IIS is to determine the priority of projects according to their ability to enhance safe operation of the plant.

The IIS is based on the Westinghouse Analytical Ranking Process. This process determines the relative potential safety contribution of each modification. The safety ranking is then used as a priority criterion in scheduling the project.

The IIS program reflects limited outage time, and financial and manpower resources, while at the same time implementing those modifications determined necessary for enhanced plant safety. The plan provides for integration of all future identified work into one comprehensive schedule and has built in mechanisms for changes to the schedule when new modifications are identified or when key program milestones cannot be achieved due to considerations beyond the control of SCE.

10.2.2 IIS Methodology

The initial step in the implementation process of scheduling CRDR modifications is to develop a systematic organization of HEDs for input to the IIS. This is necessary due to the significant number of HEDs identified through the various CRDR processes. The IIS ranking process is not structured to allow processing of a large number of individual items (e.g. 250 HEDs) in an efficient manner. Consequently, the HEDs will be grouped into categories based on functional relationship and HED priority rating. This grouping will allow scheduling of CRDR modifications in an efficient manner. The content of each HED group is outlined as follows:

1) Relocate Instruments and Controls on panel C09 - This HED modification group will consist of individual HEDs regarding component grouping and control display integration on panel C09. The panel modifications include relocation of RCS, CVCS, pressurizer, and shaft seal leakoff instrumentation and controls.

2) Protective Switch Covers and Barriers - This HED modification group will consist of HEDs regarding equipment protection from inadvertent actuation. The control room modifications include installation of protective hinge covers, push button barriers, and a guard rail and lip edge for the J-console.

3) Surface Enhancements - This HED modification group will consist of HEDs regarding control room color coding, labeling, demarcation, scale coding, and shape coding. The control panel modifications include color coding, system demarcation and hierarchical labeling for the entire control room, switch handle shape coding, scale coding for instruments and recorders, and formalizing the existing color dot system for instrument and control power supply and failure mode.

4) Communications/Miscellaneous - This HED modification group will consist of HEDs regarding communication between the control room and other plant areas. The only control room modification in this group is installation of a phone jack in a central location in the control room. Plant modifications include installation of quiet booths inside containment and plant high noise areas. Also included are measures to improve overall communication between the control room and containment.

5) Re-Design of Diesel Generator Panels - This HED modification group will consist of HEDs regarding component grouping, control display integration, anthropometrics and component suitability of equipment located on the diesel generator panels. The modifications to this panel include relocation of instruments and controls and simplification of the SLSS load group display to essential information only.

6) Remove Safety Injection Test Switches - This HED modification group will consist of only one HED regarding removal of equipment no longer necessary in the control room. The control room modification is to remove the safety injection test switches on panel C09. If actuated, these test switches would initiate safety injection.

7) Relocate Control Room Instrumentation and Controls - This HED modification group will consist of HEDs regarding functional grouping, control display integration, and geographic orientation. The control panel changes include relocation of instruments and controls on panels C09, C03, C05 and the remote shutdown panel.

8) Unsuitable Equipment - This HED modification group will consist of HEDs relating to instrumentation and controls that require modifications to reduce potential for operator error and increase plant safety.

Control panel modifications include instrument range extensions, replacement of controls and modification to recorders for improved readability.

9) Install New Equipment - This HED modification group will consist of HEDs regarding instrumentation and controls not currently in the control room. The equipment to be installed by these modifications includes charging pump lockout lights, remote sphere sump pump controls (not in control room proper), indication of feedwater pump miniflow to condenser, and safety related narrow range containment pressure indication.

10) Control Room Annunciators - This HED modification group will consist of HEDs regarding annunciator labeling, missing annunciators, and unsuitable annunciators. The modifications include providing legible, descriptive and consistent tile legends, providing panel radiation alarms, RCP high vibration alarm, test controls for reactor plant permissive panel, and eliminating the radiation area entry nuisance alarm.

11) Remove Abandoned Equipment - This HED modification group will consist of HEDs regarding removal of instrumentation and controls that are no longer used or in service. This equipment includes the 10-90% toggle switch on the J-console, push button controls for each train of containment spray, high pressure scram setpoint indicators on C09, tsunami gate controls, and an abandoned recorder on C03.

12) Unsuitable Equipment - This HED modification group will consist of HEDs relating to instrumentation and controls that require modifications to reduce potential for operator error and increase plant safety. Modifications include replacing the RPS mode selector switch, correct SI vent system controls, and replace SI flow meters for EOI range and low end scale resolution.

13) Control Room Annunciators - This HED modification group will consist of HEDs relating to annunciator color coding, multi-point alarms, and a nuisance alarm. The modifications include alarm prioritization, separating three multi-channel alarms, and eliminating the boric acid nuisance alarm.

The IIS ranking process employs a comparative analysis technique by which individual items requiring prioritization are compared to other IIS projects in each of four criteria. The criteria for the comparison are based on safety and nonsafety considerations. A detailed discussion of the IIS ranking process and the methodology of the IIS were provided in SCE's letter to the NRC dated September 2, 1983. The groups of HEDs will be processed in accordance with the procedure described in that letter.

10.3 IIS RESULTS

Once the CRDR modifications and other IIS projects are ranked they will then be evaluated using normal scheduling methods to determine how long they will take to implement. The projects ranked highest will first be evaluated to determine whether they can be implemented during the next scheduled refueling outage. Projects continue to be selected from the top of the ranked lists and scheduled for the earliest outage in which implementation constraints of a three month outage have not been exceeded. The results of the IIS ranking and scheduling of the HED groups will be provided in the next IIS update to the NRC scheduled for April 1988.

11.0 COORDINATION AND INTEGRATION OF NUREG-0737, SUPPLEMENT 1 INITIATIVES

The CRDR process was an integral part of an overall program to provide control room improvement and control room Emergency Response Capability (ERC). Effective control room emergency operations are dependent on a complete analysis of all control room functions and operator needs during an accident. Emergency drills exercise the different elements of ERC and provide a gauge to measure the integration of the elements. SONGS Unit 1 has successfully completed drills that have been judged acceptable by both the NRC and FEMA. Therefore, SCE is confident that good integration exists in the ERC present configuration, and SCE will continue to provide a high degree of integration in the new systems and plant modifications.

By letter dated April 23, 1985 SCE provided to the NRC an integrated plan to respond to the NUREG-0737, Supplement 1 initiatives. The CRDR is an integral part of these initiatives, and as such, the Supplement 1 activities will interface with the CRDR as appropriate. The integrated plan addressed the implemented initiatives of the Emergency Response Facilities (ERFs), the Emergency Operating Instructions (EOIs) and operator training. In addition, the ongoing efforts in the areas of the Safety Parameter Display System (SPDS) and Regulatory Guide 1.97 were discussed. These activities were integrated by following the NUREGs and Regulatory Guides that NUREG-0737, Supplement 1 referenced.

In addition to the Supplement 1 activities, several plant modifications that have an impact on CRDR and the other Supplement 1 activities are scheduled for implementation. These modifications have been integrated into the plan and will incorporate human factors considerations in accordance with the other Supplement 1 initiatives. Based on this integration approach, all known control room modifications have been coordinated consistent with the overall control room design objective.

An additional mechanism to ensure integration of NUREG-0737, Supplement 1 initiatives and planned control room modifications is provided by the designation of several CRDR team members to act as the responsible engineers for their respective disciplines in the design and implementation stages of related initiatives. This mechanism allows coordination of control room modifications with proper consideration to the related initiatives.

A discussion for each Supplement 1 initiative in relation to integration and coordination of the initiative with CRDR and related initiatives in addition to future plant modifications is provided below.

11.1 EMERGENCY OPERATING INSTRUCTIONS (EOIs)

SCE provided to the NRC by letter dated April 12, 1985 the Procedures Generation Package (PGP) required by Supplement 1. Upgraded EOIs have been implemented at San Onofre Unit 1 and provisions have been established to ensure upgrades to the EOIs are made as necessitated by resolution of other Supplement 1 initiatives. Upgraded EOIs are an important component in the overall effort to enhance the operator's ability to comprehend and cope with abnormal plant conditions. The coordination of the procedure upgrade effort and the CRDR is therefore especially important. It is partially accomplished by the inherent commonality of the EOIs and the CRDR system function and task analysis. The task analysis provides a consistency check for the EOIs and, when compared with the control room inventory, assures that all necessary information and controls are available to the operator in the control room. The CRDR and EOI efforts must also be coordinated with respect to control room design changes being incorporated into the EOIs and EOI training.

The EOIs, in conjunction with the Westinghouse Owners Group (WOG) Emergency Response Guidelines (ERGs) provide the basis for the CRDR system function and task analysis. The WOG ERGs provide the top-down systems function approach to identifying necessary operator tasks during post-accident mitigation situations. Recovery strategies, safety function status checks, and resource assessment guidelines are developed for both event oriented and functional recovery procedures. In the CRDR task analysis, the EOIs are broken into individual operator tasks and subtasks required for implementation of each procedure guideline. For each task and subtask, the operator information requirements and control action requirements are defined. These requirements represent everything that must be available to the operator in the control room to accomplish the given task. The task analysis methodology and results are presented in Section 6.0. The information and control requirements are compared to a control room inventory (Section 4.0) to assure that all information and controls required by the operator are available in the control room. The independent task analysis provides a verification of the development process for San Onofre Unit 1 EOIs from the generic ERGs. The dynamic aspect of the task analysis provides assurance that task loadings of the operation crew during abnormal conditions is acceptable. The task analysis is the basis for validation of the EOI upgrade effort and integration with the CRDR effort.

It is equally important that implementation of CRDR HED recommendations that result in control room changes are considered for their impact on station EOIs. This is extended to include all control room changes whether or not they are initiated from the CRDR. The method for accomplishing this is documented in the PGP in place at San Onofre Unit 1. The upgrade process section of the PGP describes in detail the procedure for upgrading EOIs. Any item that may affect procedure revision is forwarded to the Operations Procedure Group (OPG). This includes proposed facility changes, design change packages (DCPs), temporary change notices (TCNs), feedback from operating experience, etc. An additional source of input for OPG consideration is Configuration Control Procedure SO123-XIV-3.1 "Configuration Document Change Control." All CRDR HEDs are reviewed by station personnel as part of the CRDR HED Evaluation Team. Potential EOI impact from CRDR HEDs will be considered during this evaluation by HED Evaluation Team members including individuals from station operations and station technical. Actual implementation of control room design changes will be forwarded to the OPG through one of the above existing design change mechanisms such as the DCP process. Upon receiving input of a design change, the OPG considers it according to the guidelines of the PGP.

If a revision to the EOIs is warranted, Administrative Procedures SO123-VI-1.0, "Station Orders, Procedures and Instruction - Preparation, Revision, Review, Approval and Publication," and SO123-VI-1.0.5, "Unit 1 EOIs - Preparation, Revision and Validation" are employed to effect the revision development and implementation. A complete discussion of this method is described in the PGP.

The control room changes resulting from the CRDR must also be incorporated into EOI training. This is discussed in Section 11.5 concerning the coordination of the CRDR with operator training.

In summary, the CRDR and EOI upgrade activities are well coordinated. The derivation of the system functions and task analysis from generic ERGs provides a strong procedural link to the CRDR effort. The results of the task analysis are used to verify both the control room design/inventory and the station EOIs. Administrative procedures exist that provide a systematic OPG assessment of control room design changes with possible development and implementation of revised EOIs. It is therefore concluded that the CRDR effort has been properly coordinated and integrated with the EOI upgrade activities.

11.2 SAFETY PARAMETER DISPLAY SYSTEM (SPDS)

The conceptual design for the San Onofre Unit 1 SPDS was provided to the NRC by letter dated March 17, 1987. The conceptual design included SPDS criteria development and conceptual hardware/software design. The second phase in the design of an SPDS requires development of the SPDS Functional Criteria. This information will be developed and provided to the NRC subsequent to determining which of the four viable options for the SPDS will be implemented. All HEDs relating to digital computer design will be evaluated and incorporated, as appropriate, into the design of the SPDS. The functional location of the SPDS within the control room will be determined with consideration given to control room work space and environment, and optimal integration with operator tasks as determined in review of the video taped scenarios (Section 8.0). In response to NRC concerns regarding the inclusion of Radioactivity Control as an SPDS "top level" display, it is SCE's intention to incorporate into the design of the SPDS a top level display for this parameter. The remaining top level displays will be as identified in our letter of March 17, 1987. A detailed discussion of the verification and validation program for the SPDS will be provided to the NRC with the Functional Criteria.

11.3 REGULATORY GUIDE 1.97

SCE reviewed the recommendations of Regulatory Guide 1.97 and provided a plant specific submittal by letter dated December 16, 1985. The plant specific submittal identifies the scope of instrumentation necessary to provide information to allow operators to (1) take the necessary preplanned actions to accomplish safe shutdown of the plant, (2) ensure accomplishment of critical plant safety functions, and (3) monitor the release of radioactive materials and implement the radiological dose assessment actions of the offsite emergency plan. By letter dated December 22, 1986, the NRC provided SCE with a draft TER which requested additional information regarding apparent deviations from the Regulatory Guide 1.97 recommendations. This additional information was provided by SCE letter dated May 29, 1987.

SCE recognizes that coordination of the CRDR with the Regulatory Guide 1.97 compliance effort differs from coordination with other control room initiatives in that it is essentially a one-time effort. Coordination of the CRDR with training and EOI development are continuing processes. The primary impact that review of Regulatory Guide 1.97 and the CRDR have on each other is their effect on the characteristics of control room instrumentation. Control room instrumentation changes resulting from Regulatory Guide 1.97 concerns will be coordinated and integrated with the overall control room design. In implementing control board changes originating from Regulatory Guide 1.97 concerns, human engineering principles of the CRDR will be used.

An additional aspect of coordination between the CRDR and Regulatory Guide 1.97 evaluation efforts is provided by the involvement of the same key SCE personnel in both projects.

11.4 EMERGENCY RESPONSE FACILITIES (ERFs)

The ERFs at San Onofre Unit 1 consist of the Technical Support Center (TSC), the Operations Support Center (OSC) and the Emergency Operations Facility (EOF). The status of regulatory compliance for each of these facilities is described below.

11.4.1 Technical Support Center (TSC)

The requirements for the TSC were initiated by Item 2.2.2.b of NUREG-0578, and later revised in NUREG-0737 and NUREG-0696. SCE committed to the implementation of an onsite technical support center in our letter of October 17, 1979. We updated this commitment in our letter of January 17, 1980, which provided the NRC with details of what technical data would be available in the TSC via the technical data display system. We informed the NRC in our letter of July 1, 1981 that the TSC would be completed by October 1, 1982.

The TSC, as a facility, meets the requirements of Supplement 1. The technical data display and acquisition capability, as installed, meets the commitments in our letter of June 1, 1981. Currently there are no detailed plans to upgrade this system until the resolution of our plans for the SPDS initiative. Information regarding the SPDS upgrade was provided to the NRC by our letter of March 17, 1987.

11.4.2 Operation Support Center (OSC)

The requirements for the OSC were initiated by Item 2.2.2.c of NUREG-0578 and later revised in NUREG-0737 and NUREG-0696. We committed to implementing an OSC in our letter of October 17, 1979 and informed the NRC that our OSC was operational in our letter of July 1, 1981. The OSC meets the criteria of Supplement 1.

11.4.3 Emergency Operations Facility (EOF)

The requirements of the EOF were initiated by NUREG-0660 and later revised by NRC letter dated February 18, 1981. The functional criteria for the EOF is contained in NUREG-0696. In our letter dated July 1, 1981, we informed the NRC of our conceptual design and scheduled implementation date of October 1, 1982. The EOF, as a facility, meets the criteria of Supplement 1, with the exception of the distance to the backup EOF previously discussed in our letter of December 7, 1982. The technical data display and acquisition capability, as installed, meets what was committed to in our letter of July 1, 1981. Currently there are no plans to consider upgrade to this data display and acquisition capability until the resolution of the SPDS initiative.

11.4.4 Conclusion

The two primary areas requiring coordination between the control room and the ERFs are access to plant parameters and safety status, and communications. The information requirements for the technical support personnel in the ERFs will be augmented by the implementation of the SPDS. This information is independent of control room information requirements and panel design. Adequacy of information provided to the technical personnel in the ERFs will be verified periodically during station emergency preparedness tests. Any additional information requirements identified for the EOF or TSC during these tests would be incorporated into the SPDS design since neither facility has control board-type indicators. As a result, little coordination is required between the CRDR and the TSC and EOF with respect to technical personnel information needs. The coordination that does exist is through the SPDS as discussed in Section 11.2.

The communication links between the TSC and EOF and the control room are another important aspect of coordination between the control room and emergency response facilities. The telephone communication provisions of the SONGS 1 TSC and EOF include dedicated telephones connected to the control room and meet regulatory requirements. There are no HEDs resulting from the CRDR that impact the current EOF and TSC communications system. Thus, there are no coordination activities required in this area.

Based on the information presented, little coordination is necessary between the TSC and EOF and the CRDR. The coordination provided in the technical support personnel information needs area by the SPDS effort is sufficient.

11.5 OPERATING STAFF TRAINING

A well-established operator training and requalification program exists at San Onofre Unit 1. An important part of that program is in the area of EOI training. EOI training enables operators to understand the structure, format and technical bases of the EOIs. It also provides them with a working knowledge of the technical content of the EOIs and enables them to use EOIs under operational conditions. The EOI training program consists of both classroom lectures and discussions, and simulator training. The simulator training consists of EOI walk-throughs and event scenarios. Although not a training requirement, the event scenarios often include multiple and sequential failures. Written and simulator evaluations of each student are conducted after the initial EOI training is completed. Retraining occurs as part of the Operator Requalification Program (ORP).

Simulator training is conducted at Commonwealth Edison Company's Zion simulator. This facility provides a simulation of SONGS 1 for normal, abnormal and emergency scenarios.

The coordination of the CRDR effort with operating staff training involves several areas. The operator questionnaire and interviews performed as part of the CRDR include specific questions intended to identify deficiencies in the area of operator training. In addition, the Nuclear Training Administrator was part of the HED Evaluation Team, which ensures HED's affecting operator training were dispositioned appropriately. Finally, control room changes resulting from the CRDR will be identified to the Nuclear Training Department (NTD) prior to implementation for evaluation and incorporation into training programs as appropriate.

Plant changes and control room changes must be evaluated for their impact on operator training. This must include changes that have occurred or will occur as the result of the implementation of CRDR HEDs.

Maintenance of the content of training materials (lesson plans, course objectives, etc.) is coordinated by the SONGS Action Item Management System (AIMS). Through this procedure, all station DCPs, PFCs etc. are received and evaluated by the NTD. Appropriate items are forwarded to the SONGS 1 Operations Training Administrator. Each item will be evaluated for its impact on training by the training administrator. If a change is determined to have an impact on training, it is dispositioned to the individuals whose training sessions are impacted. The instructors are responsible for making appropriate changes to lesson plans, etc. This activity is documented in SO123-XXI-4.400, Training Implementation - Instructor Responsibility. The revised materials are approved and then incorporated into the training program. This is accomplished by the master file system procedure.

Another source of changes relating to the control room is the CRDR. All CRDR HEDs were evaluated by an evaluation team as documented in Section 9.0 A station training representative on this team will evaluate HEDs for their impact on training. This provides further coordination of the CRDR with the operator training program.

11.6 OTHER MODIFICATIONS

Several plant modifications that will involve control room changes have been identified by the CRDR team. Each of these modifications has been reviewed to determine the ability for these changes to resolve HEDs as well as the potential for these changes to introduce new HEDs. Human factors considerations will be incorporated into all resultant control room changes consistent with the overall control room design objective. In addition, any impact of these modifications on the remaining Supplement 1 initiatives will be evaluated and accounted for as appropriate. Each modification is identified below with the current implementation schedule.

11.6.1 Foxboro Rack Replacement

Replace the existing reactor control and protection, chemical and volume control, and main feedwater control instrumentation racks with new Foxboro Spec 200 micro racks. Control room modifications include replacement of all associated Foxboro controllers with new Spec 200 micro controllers. This modification is scheduled for implementation in Cycle 11 refueling outage.

11.6.2 Inadequate Core Cooling (ICC)

Upgrade the existing Core Exit Thermocouple (CET) instrumentation to meet NUREG-0737 requirements. Control room modifications include installation of two new seismic qualified Foxboro Spec 200 CET displays. This modification is scheduled for implementation in Cycle 11 refueling outage.

11.6.3 Nuclear Instrumentation System (NIS)

Replace the existing NIS with a new system including detectors, signal processing units, cabling and containment penetrations. Control room modifications include replacement and relocation of the two existing axial offset indicators and 12 NIS indicators to a location outside the control room proper. This modification is scheduled for implementation in Cycle 10 refueling outage.

11.6.4 Anticipated Transients Without Scram (ATWS)

Provide an automatic turbine trip diverse from the reactor protection system. Control room modifications include installing system controls, indication and alarms as necessary. This modification is scheduled for implementation in Cycle 11 refueling outage.

11.6.5 Auxiliary Feedwater System (AFW)

Provide a third AFW pump including two AFW flow control valves, and a venturi/orifice downstream of the flow control valves in each of the three AFW lines. Control room modifications including providing controls, indication and alarms as necessary. This modification is scheduled for implementation in Cycle 10 refueling outage.

11.6.6 Turbine Generator Vibration and Expansion Monitoring

Provide permanent installation of field racks and weather proofing for the housing on the turbine deck. Control room modifications include replacement of the existing Westinghouse instrumentation with new instrumentation including a host computer. This modification is scheduled for implementation in Cycle 10 refueling outage.

11.6.7 Recent Control Room Modifications

In addition to the future control room modifications identified in the above sections, the CRDR Team evaluated control room modifications that have been implemented subsequent to the development of the Control Room Inventory and performance of the Control Room Human Engineering Survey. These modifications were evaluated to determine potential introduction of new HEDs to the control room, and possible impact on existing HEDs. Each modification is discussed below. In each instance, the CRDR Team concluded that no significant discrepancies were associated with the change. Modifications to correct existing generic control room HEDs establish provisions for correcting the minor discrepancies associated with these changes.

- DSD Charging Pump Switch - As part installation of the Dedicated Safe Shut Down System a push button switch was installed on the J-console to ensure availability of one charging pump during a DSD situation. The switch has a protective cover and is located in close proximity to the charging pump controls. Implementation of control room surface enhancements will ensure consistent labeling, color coding and demarcation are provided as appropriate.

- Installation of 24-Hour Digital Clock - The existing 12-hour analog clock was replaced with a 24-hour (military time) digital clock. Based on the current station practice of recording time measurements for control room surveillances in military time, the CRDR Team concluded that the replacement clock was acceptable.

- Steam Generator Blowdown Switch - A push button control was installed on the Auxiliary Feedwater Panel (C71) to provide indication and manual initiation capability to isolate steam generator blowdown on actuation of auxiliary feedwater. Implementation of control room surface enhancements will ensure consistent labeling, color coding and demarcation are provided as appropriate.

- Installation of Three Alarms on Auxiliary Feedwater Panel - Three new alarms were installed on the Auxiliary Feedwater Annunciator Panel for the Office of Emergency Services - Automated Alert System. Implementation of generic control room annunciator modifications will ensure consistent tile legend content.

- 4KV Busses Paralleled Alarm - Two new alarms were installed on the Electrical Power System Annunciator Panel to indicate paralleling of 4KV Busses. In addition a Vital Buss alarm was revised to provide further clarification of alarm and alarm inputs. Implementation of generic control room annunciator modifications will ensure consistent tile legend content with all other tile legends.

- Pressurizer Pressure/Level Control Transfer Labels - Two labels were provided for the pressurizer pressure and level control transfer switches to distinguish between the two controls as they are adjacent. The label content, color and size will be revised consistent with the overall control room standard.

- Diesel Generator Start/Stop Controls - Labels were installed on the DG start/stop controls to indicate control function. The label content, color and size will be revised consistent with the overall control room standard.

- Fire Protection Panel - A Fire Protection panel system status board was installed. The status board is an operator aid that will be regulated by administrative controls as part of the temporary label and tagout procedures.

11.7 SONGS 1 CONTROL ROOM STANDARDS DOCUMENT

11.7.1 Objective

SCE recognizes that the objective of performing a CRDR is to evaluate and correct existing control room man-machine interface problems. SCE also recognizes the potential for future plant modifications requiring changes to the control room instrumentation and controls. In order to assure these changes are performed in a manner consistent with the objective of the CRDR, SCE proposes to develop a "Control Room Standards Document" that defines human factors criteria for the SONGS 1 control room. The objective of this document will be to establish an administrative control whereby future control room changes are designed, reviewed and implemented in accordance with a control room standard as established by the CRDR. The CRSD is scheduled to be released in May 1988.

11.7.2 Methodology

The Control Room Standards Document (CRSD) will be developed to act as a human factors engineering design guide. The CRSD will establish criteria in accordance with NUREG-0700, Section 6.0 for any type of control room panel changes. In addition, all design changes affecting the control room or control panels will be reviewed for verification that human factors considerations have been incorporated. The responsibility of this review and qualifications necessary to perform the review will be established in the CRSD.

In the event that significant modifications to the control room become necessary in the future, the CRSD will establish provisions for a multi-disciplined group, similar to the CRDR team, to evaluate the changes for compliance with the CRDR and potential impact on related areas including Training, EOIs, ERFs, etc.

Once implemented, the CRSD will minimize the potential for introduction of new man-machine interface problems in the control room and maintain SCE's commitment to control room enhancement throughout the service life of SONGS 1.

AFTER

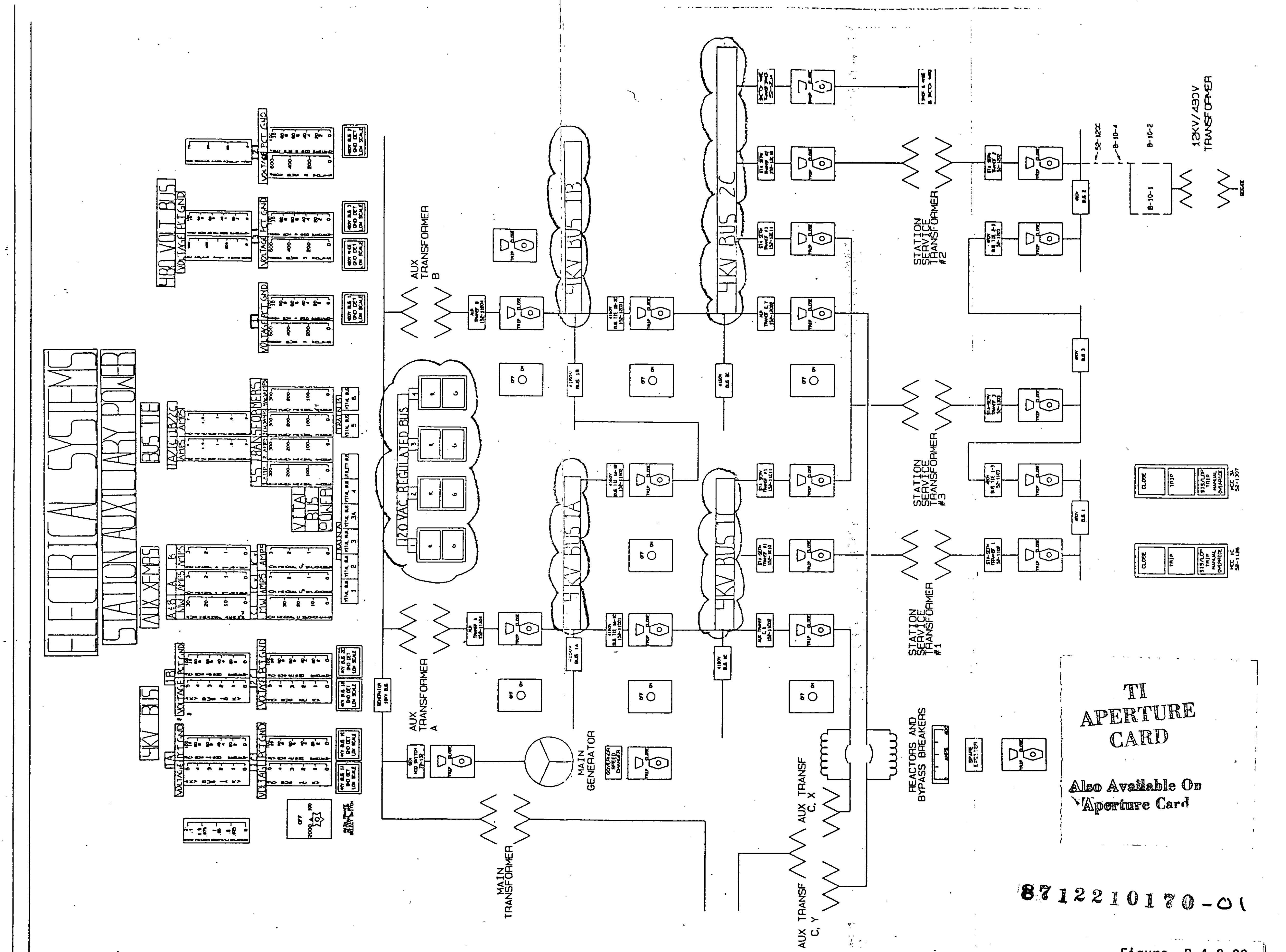

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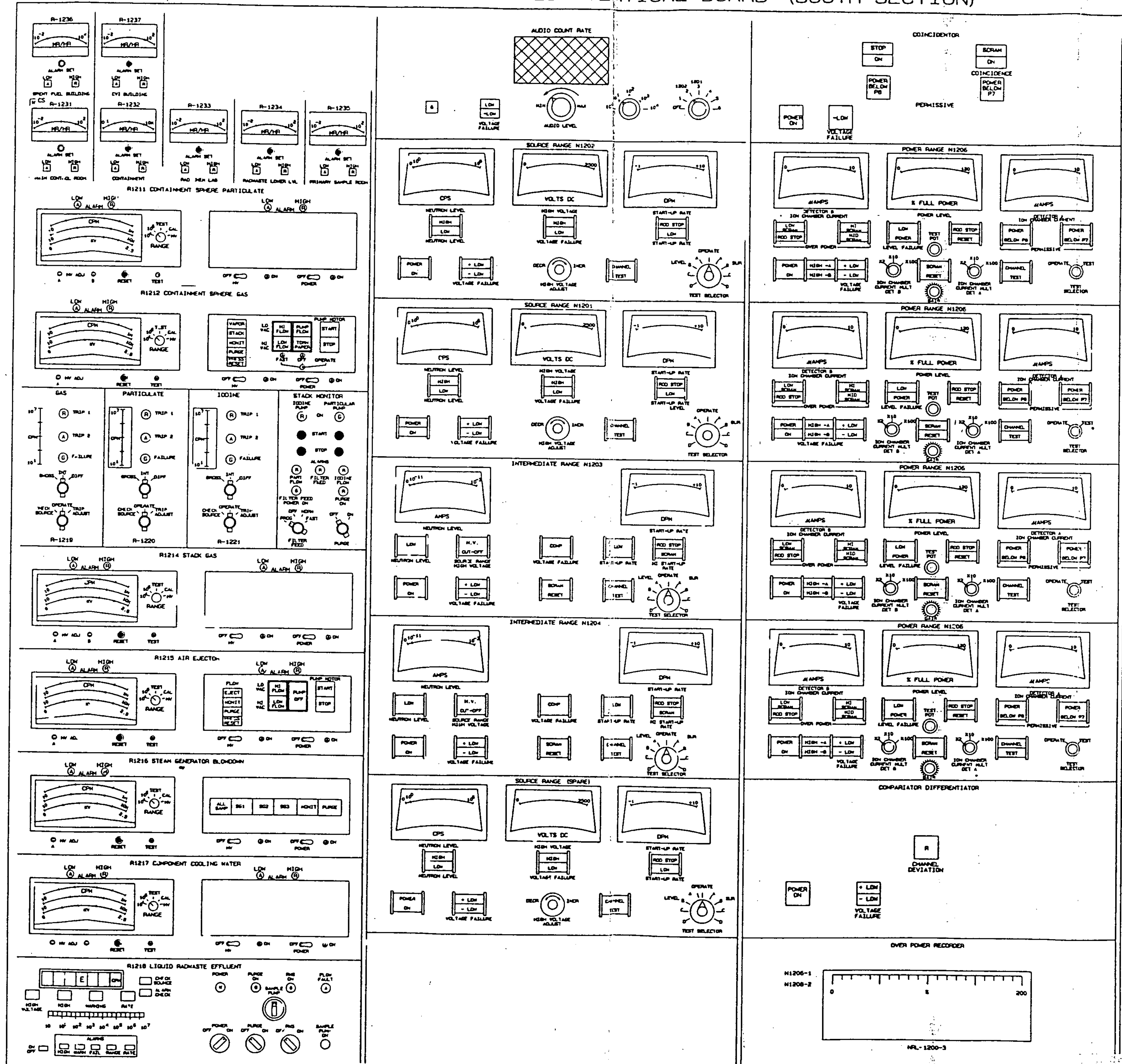
Figure B.4.2-20

South Vertical Board, East Section, C12, After

WEST VERTICAL BOARD (SOUTH SECTION)

BEFORE



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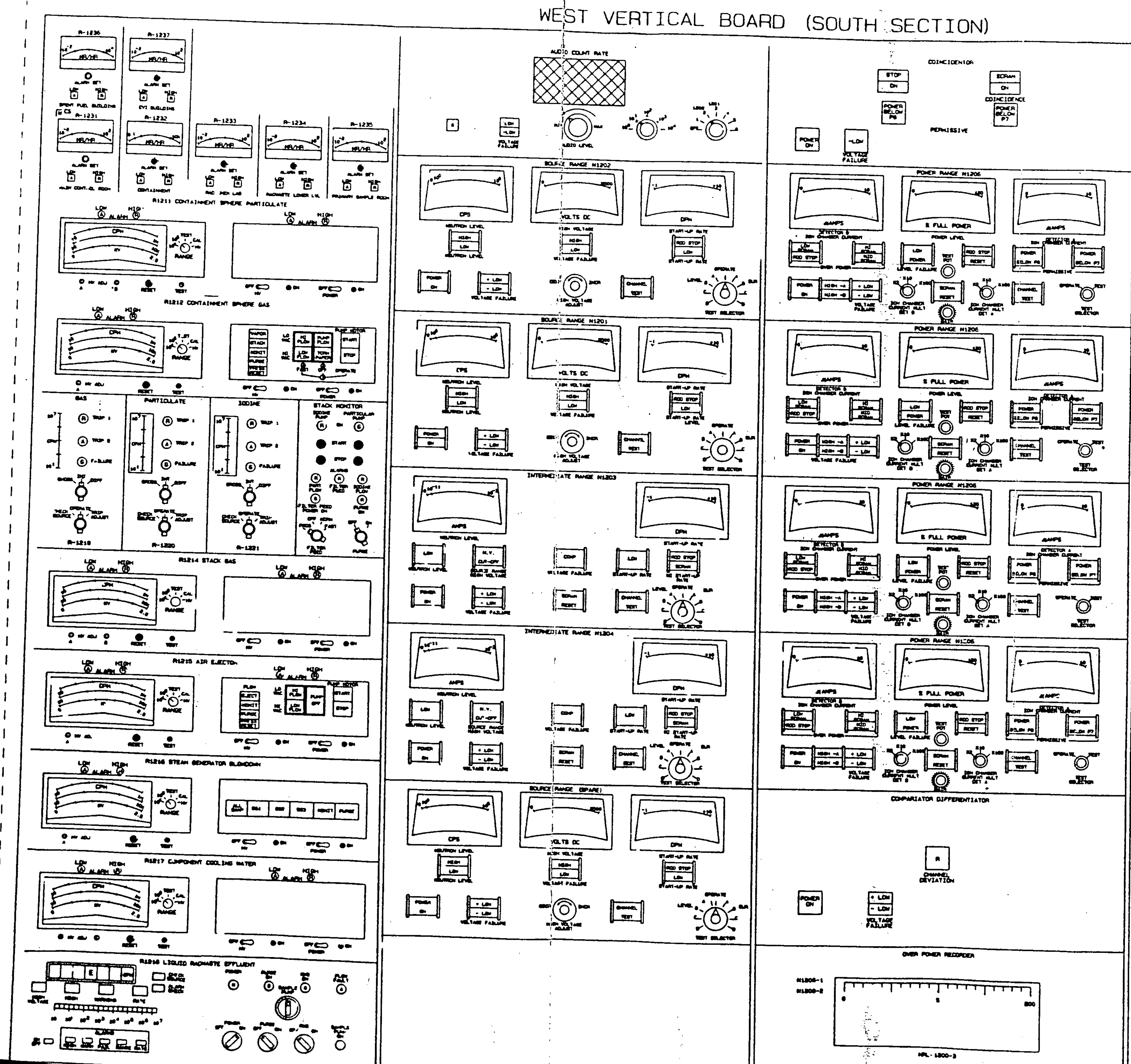
Also Available On Aperture Card

8712210170-02

Figure B.4.2-21

Nuclear Instrumentation System, C10A/10B, Before



BEFORE

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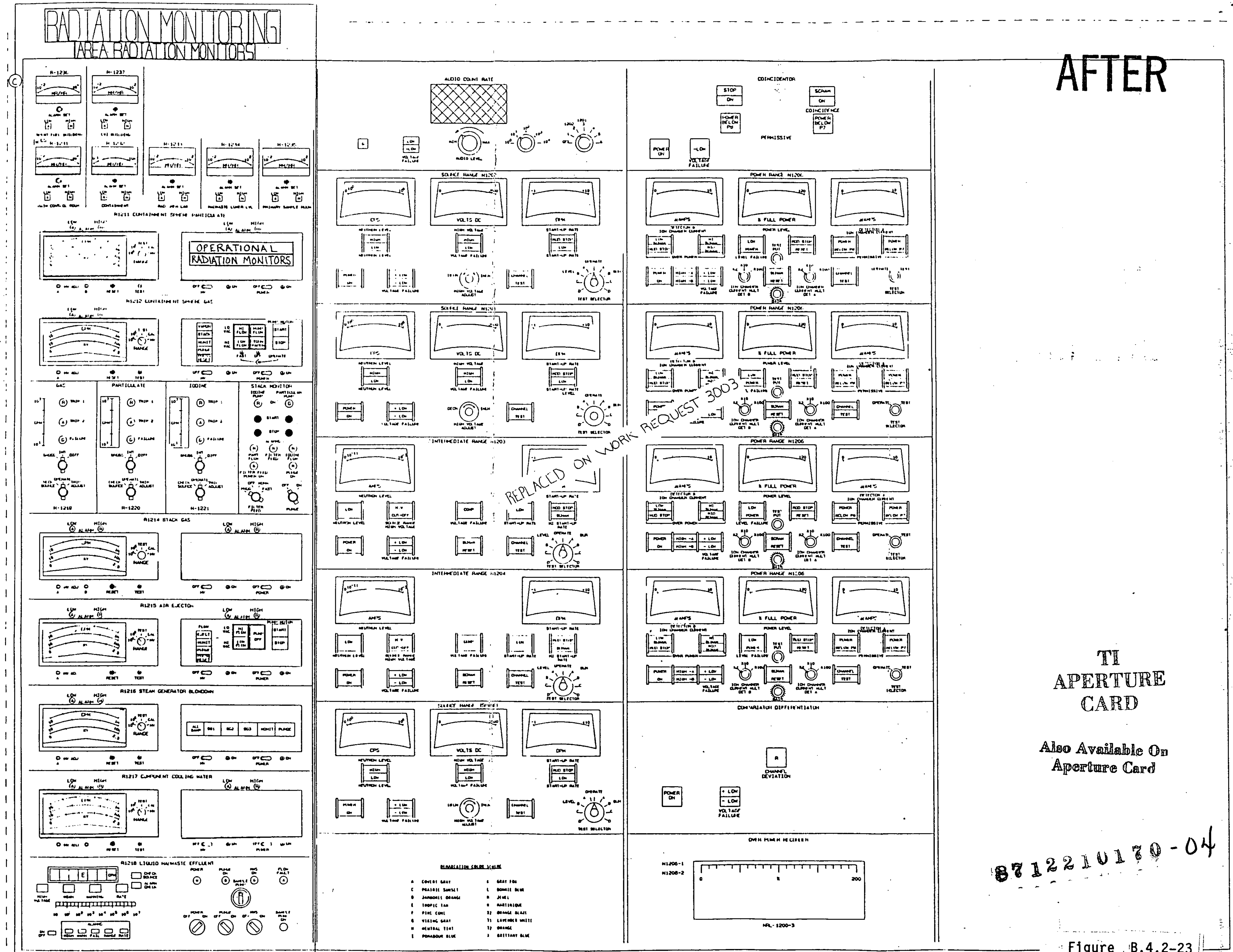
8712210170-03

Figure B.4.2-22

Radiation Monitoring System Panel C11.



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Also Available On
Aperture Card

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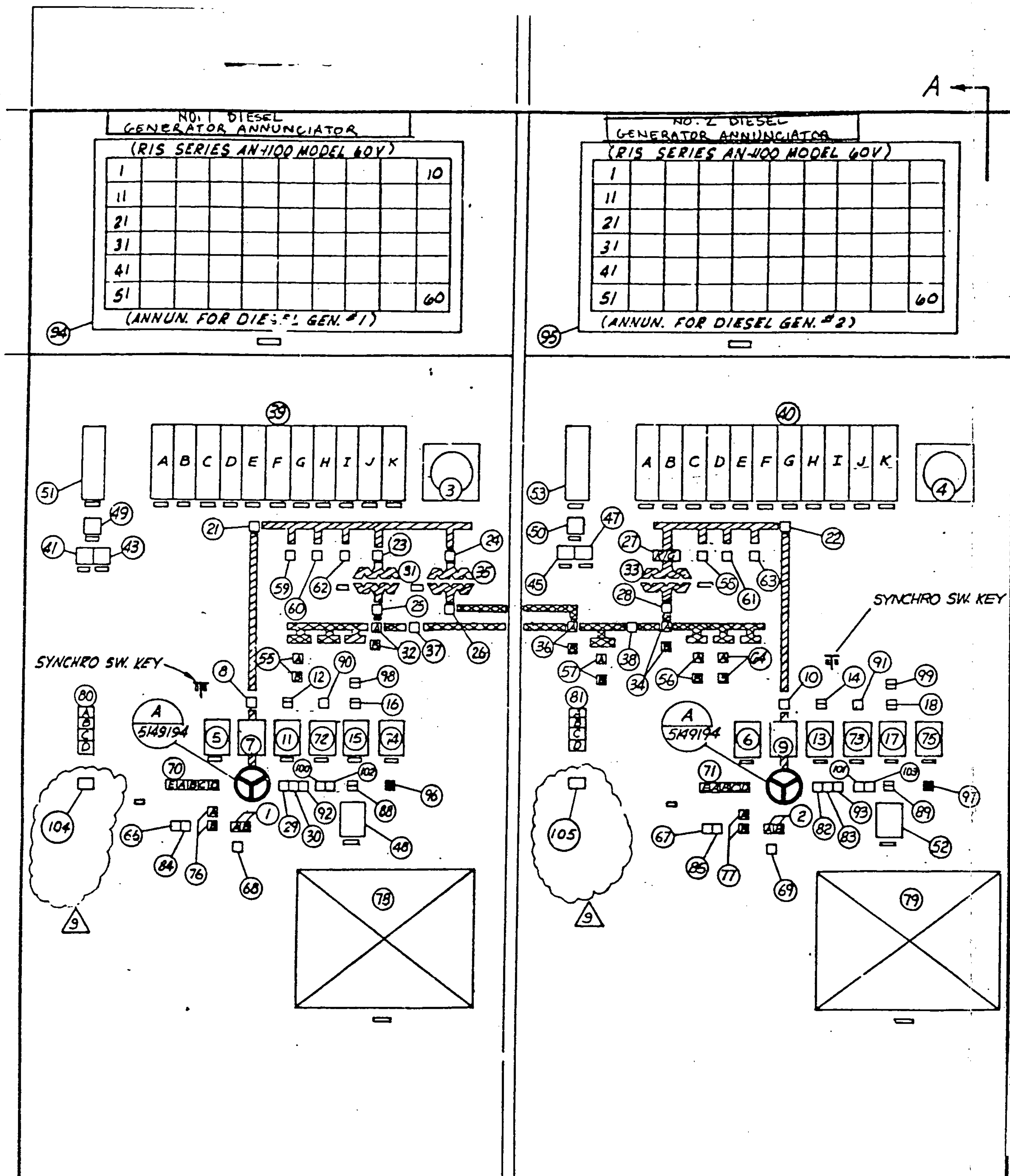
Figure B.4.2-23

Radiation Monitoring System Panel, C11, After

BEFORE

TI APERTURE CARD

Also Available On Aperture Card



EQUIPMENT SCHEDULE
SPA-ECCS. CONTROL INSTRUMENTATION

ITEM NO.	TAG NO.	SERVICE DESCRIPTION
1	HS-100A, HS105	DIESEL ENGINE #1 START-STOP PUSHBUTTON
2	HS-140A, HS145	DIESEL ENGINE #2 START-STOP PUSHBUTTON
3	SI-101	DIESEL GENERATOR SYNCHROSCOPE
4	SI-111	DIESEL GENERATOR SYNCHROSCOPE
5	HS-116	SYNCHRO SWITCH (KEY OPERATED)
6	HS-155	SYNCHRO SWITCH (KEY OPERATED)
7	HS-111	DIESEL GENERATOR #1 CONTROL & LOCKOUT SWITCH
8	—	DIESEL GENERATOR #1 CONTROL & LOCKOUT SWITCH
9	HS-151	DIESEL GENERATOR #2 CONTROL & LOCKOUT SWITCH
10	—	DIESEL GENERATOR #2 CONTROL SW STATUS LIGHT
11	HS-103A	DIESEL GENERATOR #1 GOVERNOR SPEED CONTROL
12	—	DIESEL GENERATOR #1 GOVERNOR SPEED LIMIT LIGHT
13	HS-143A	DIESEL GENERATOR #2 GOVERNOR SPEED CONTROL
14	—	DIESEL GENERATOR #2 GOVERNOR SPEED LIMIT LIGHT
15	HS-109A	DIESEL GENERATOR #1 AUTO. VOLTAGE ADJUST
16	—	DIESEL GENERATOR #1 VOLTAGE ADJUST LIGHT
17	HS-149A	DIESEL GENERATOR #2 AUTO VOLTAGE ADJUST
18	—	DIESEL GENERATOR #2 VOLTAGE ADJUST LIGHT
21	—	4160V BUS 1-C LIGHT
22	—	4160V BUS 2-C LIGHT
23	—	CKT BKR 11-C-10 LIGHT (4160V)
24	—	CKT BKR 11-C-11 LIGHT (4160V)
25	—	CKT BKR. 11-0-2 LIGHT (480V)
26	—	CKT BKR 13-0-2 LIGHT (480V)
27	—	CKT BKR 12-C-10 LIGHT (4160V)
28	—	CKT BKR 1202 LIGHT (480V)
29	—	JACKET WATER KEEP WARM PUMP OPERATING LIGHT GEN. 1
30	—	LUBE OIL KEEP WARM PUMP OPERATING LIGHT GEN. 1
31	—	STATION SERVICE TRANSF. #1
32	HS-118	480V SWGR #1 FEEDER LOCKOUT INDIC. LIGHT & RESET P.B.
33	—	STATION SERVICE TRANSF. #2
34	HS-156	480V SWGR #2 FEEDER LOCKOUT INDIC. LIGHT & RESET P.B.
35	—	STATION SERVICE TRANSF #3
36	HS-160	480V SWGR #3 FEEDER LOCKOUT INDIC. LIGHT & RESET P.B.
37	—	480V BUS TIE 1-3 LIGHT
38	—	480V BUS TIE 2-3 LIGHT
39	—	DIESEL GEN. #1 INDICATING INSTRUMENTS (SEE NOTE 8)
39A	JI-100	DIESEL GEN. #1 INDICATING INSTRUMENTS (SEE NOTE 8)
39B	II-101A	DIESEL GEN #1 INDICATING INSTRUMENTS (SEE NOTE 8)
39C	EI-102A	DIESEL GEN #1 INDICATING INSTRUMENTS (SEE NOTE 8)
39D	JI-101	DIESEL GEN #1 INDICATING INSTRUMENTS (SEE NOTE 8)
39E	II-100A	DIESEL GEN #1 INDICATING INSTRUMENTS
39F	II-100B	DIESEL GEN #1 INDICATING INSTRUMENTS
39G	II-100C	DIESEL GEN #1 INDICATING INSTRUMENTS
39H	EI-100A	DIESEL GEN #1 INDICATING INSTRUMENTS
39I	SI-102	DIESEL GEN #1 INDICATING INSTRUMENTS
39J	EI-101	DIESEL GEN #1 INDICATING INSTRUMENTS
39K	SI-103	DIESEL GEN #1 INDICATING INSTRUMENTS (SEE NOTE 8)
40	—	DIESEL GEN #2 INDICATING INSTRUMENTS
40A	JI-110	DIESEL GEN #2 INDICATING INSTRUMENTS (SEE NOTE 8)
40B	II-121A	DIESEL GEN #2 INDICATING INSTRUMENTS (SEE NOTE 8)
40C	EI-122A	DIESEL GEN #2 INDICATING INSTRUMENTS (SEE NOTE 8)
40D	JI-111	DIESEL GEN #2 INDICATING INSTRUMENTS (SEE NOTE 8)
40E	II-120A	DIESEL GEN #2 INDICATING INSTRUMENTS
40F	II-120B	DIESEL GEN #2 INDICATING INSTRUMENTS
40G	II-120C	DIESEL GEN #2 INDICATING INSTRUMENTS
40H	EI-120A	DIESEL GEN #2 INDICATING INSTRUMENTS
40I	SI-112	DIESEL GEN #2 INDICATING INSTRUMENTS
40J	EI-121	DIESEL GEN #2 INDICATING INSTRUMENTS
40K	EI-113	DIESEL GEN #2 INDICATING INSTRUMENTS (SEE NOTE 8)
41	HS-1A	FUEL TRANSFER PUMP G74A MAN AUTO SELECTOR
104		SEE NOTE 10
105		SEE NOTE 10

9

EQUIPMENT SCHEDULE
SPA-ECCS CONTROL INSTRUMENTATION

ITEM NO.	TAG. NO.	SERVICE DESCRIPTION
42	—	
43	HS-1C	FUEL TRANSFER PUMP G74B MAN AUTO SELECTOR
44	—	
45	HS-2A	FUEL TRANSFER PUMP G75A MAN AUTO SELECTOR
46	—	
47	HS-2C	FUEL TRANSFER PUMP G75B MAN AUTO SELECTOR
48	HS-110A	
49	HS-3A	FUEL TRANSFER PUMP G74A & G74B SELECTOR
50	HS-3C	FUEL TRANSFER PUMP G75A & G75B SELECTOR
51	LI-14A	MAIN FUEL TANK D-23 LEVEL INDICATOR
52	HS-150A	D-G #2 FIELD CURRENT SHUT OFF SWITCH
53	LI-15A	MAIN FUEL TANK D-24 LEVEL INDICATOR
54	—	
55	HS-112	MCC #1 FEEDER LOCKOUT INDICATING LIGHT & RESET P.B.
56	HS-161	MCC #2 FEEDER LOCKOUT INDICATING LIGHT & RESET P.B.
57	HS-162	MCC #3 FEEDER LOCKOUT INDICATING LIGHT & RESET P.B.
58	—	CHARGING PUMP (NORTH) LIGHT
59	—	CHARGING PUMP (SOUTH) LIGHT
60	—	SAFETY INJECTION PUMP (WEST) LIGHT
61	—	SAFETY INJECTION PUMP (EAST) LIGHT
62	—	FEED WATER PUMP (WEST) LIGHT
63	—	FEED WATER PUMP (EAST) LIGHT
64	HS-152	MCC #2A FEEDER LOCKOUT INDIC. LIGHT & RESET P.B.
65	—	
66	HS-122A	DIESEL GENERATOR #1 ENGINE LOCKOFF SWITCH
67	HS-166A	DIESEL GENERATOR #2 ENGINE LOCKOFF SWITCH
68	—	DIESEL GENERATOR #1 SYSTEMS AVAILABLE LIGHT
69	—	DIESEL GENERATOR #2 SYSTEMS AVAILABLE LIGHT
70	—	DIESEL GENERATOR #1 ENGINE STATUS
71	—	DIESEL GENERATOR #2 ENGINE STATUS
72	HS-123	CURRENT LIMITING REACTOR BREAKER
73	HS-167	CURRENT LIMITING REACTOR BREAKER
74	HS-109E	GENERATOR #1 PHASE VOLTAGE SELECTOR SWITCH
75	HS-149E	GENERATOR #2 PHASE VOLTAGE SELECTOR SWITCH
76	—	CONTROL POWER ON #1
77	—	CONTROL POWER ON #2
78	—	REMOTE SURVEILANCE PANEL SEQUENCER NO. 1
79	—	REMOTE SURVEILANCE PANEL SEQUENCER NO. 2
80	—	ANNUN-ACKNOWLEDGE, TEST
80A	HS-125A	ANNUN-ACKNOWLEDGE, TEST
80B	HS-126A	ANNUN-ACKNOWLEDGE, TEST
80C	HS-127A	ANNUN-ACKNOWLEDGE. TEST
80D	HS-128A	ANNUN-ACKNOWLEDGE, TEST
81	—	ANNUN-ACKNOWLEDGE, TEST
81A	HS-168A	ANNUN-ACKNOWLEDGE, TEST
81B	HS-169A	ANNUN-ACKNOWLEDGE, TEST
81C	HS-170A	ANNUN-ACKNOWLEDGE, TEST
81D	HS-171A	ANNUN-ACKNOWLEDGE, TEST
82	—	JACKET WATER KEEP WARM PUMP OPERATING LIGHT. GEN. 2
83	—	LUBE OIL KEEP WARM PUMP OPERATING LIGHT GEN. 2
84	—	DG #1 ENGINE LOCKOFF SW. LIGHT & TEST P.B.
85	—	DG #2 ENGINE LOCKOFF SW. LIGHT & TEST P.B.
86	—	
87	—	
88	—	DG #1 FIELD CURRENT SW LIGHTS & TEST P.B.
89	—	DG #2 FIELD CURRENT SW LIGHTS & TEST P.B.
90	—	CURRENT LIMITING REACTOR BREAKER INDICATING LIGHT & TEST P/B
91	—	CURRENT LIMITING REACTOR BREAKER INDICATING LIGHT & TEST P/B
92	—	D.C. STANDBY FUEL OIL PUMP RUNNING INDICATING LIGHT (D-G #1)
93	—	D.C. STANDBY FUEL OIL PUMP RUNNING INDICATING LIGHT (D-G #2)
94	UA-1A	DG #1 ANNUNIATOR
95	UA-2A	DG #2 ANNUNIATOR
96	HS-117	TRAIN 1 LOAD MANUAL LOCKOUT SW (MCC'S & SWGR)
97	HS-157	TRAIN 2 LOAD MANUAL LOCKOUT SW (MCC'S & SWGR)
98		MAN-AUTO VOLT. REG. STATUS LIGHT & TEST P.B.-DG #1
99		MAN-AUTO VOLT REG. STATUS LIGHT & TEST P.B.-DG. #2
100		FIELD RELAY RESET STATUS LIGHT & TEST P.B. (B.M. # 218 ON CI01)
101		FIELD RELAY RESET STATUS LIGHT & TEST P.B (B.M. # 218 ON CI01)
102		VOLT. REG. SETPOINT LIGHT & TEST P.B. (B.M. # 255 ON CI01)
103		VOLT. REG. SETPOINT LIGHT & TEST P.B. (B.M. # 255 ON CI01)

Figure B.4.2-24

Diesel Generator Control Panels, C41/42, Before

8712210170-05

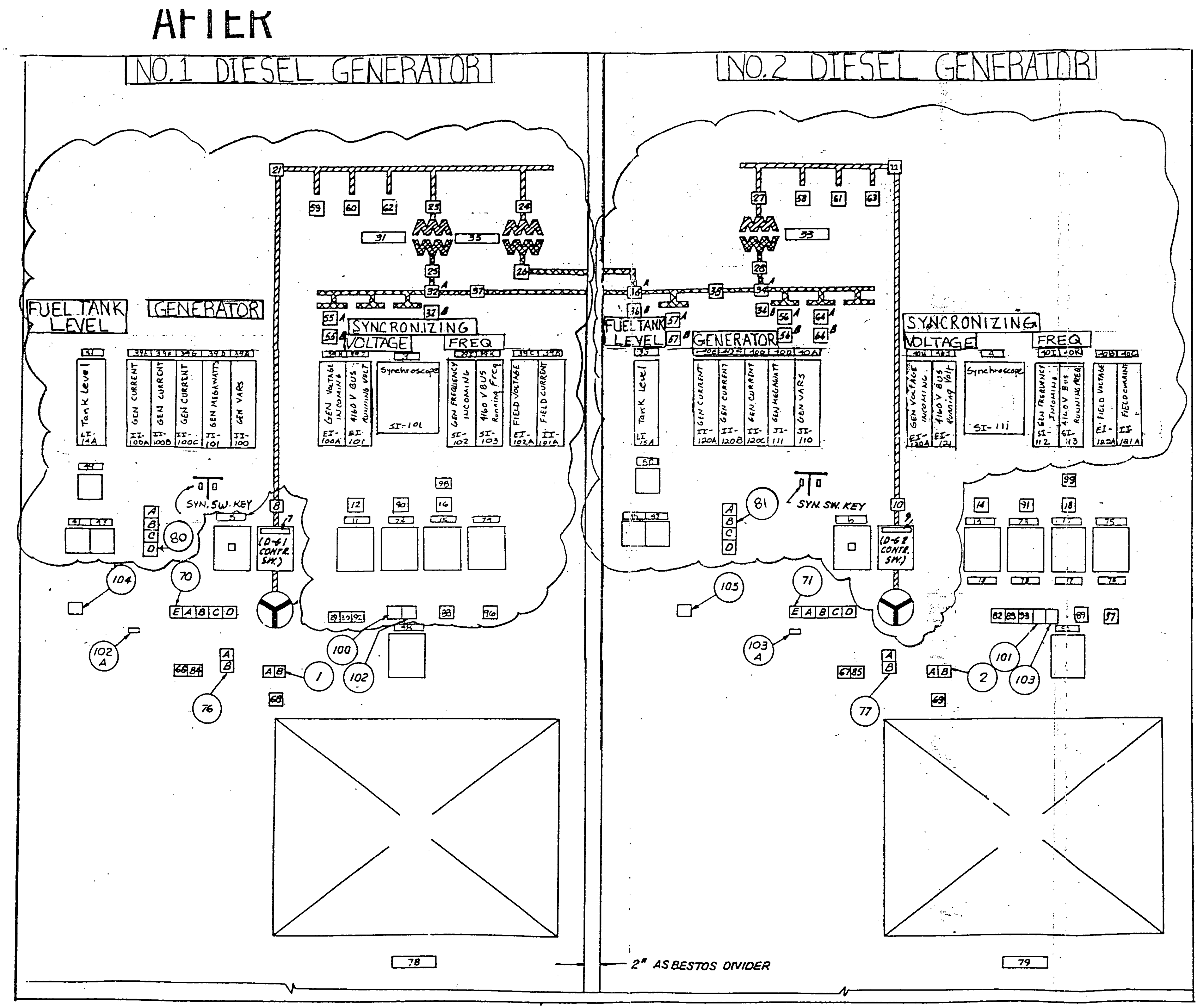
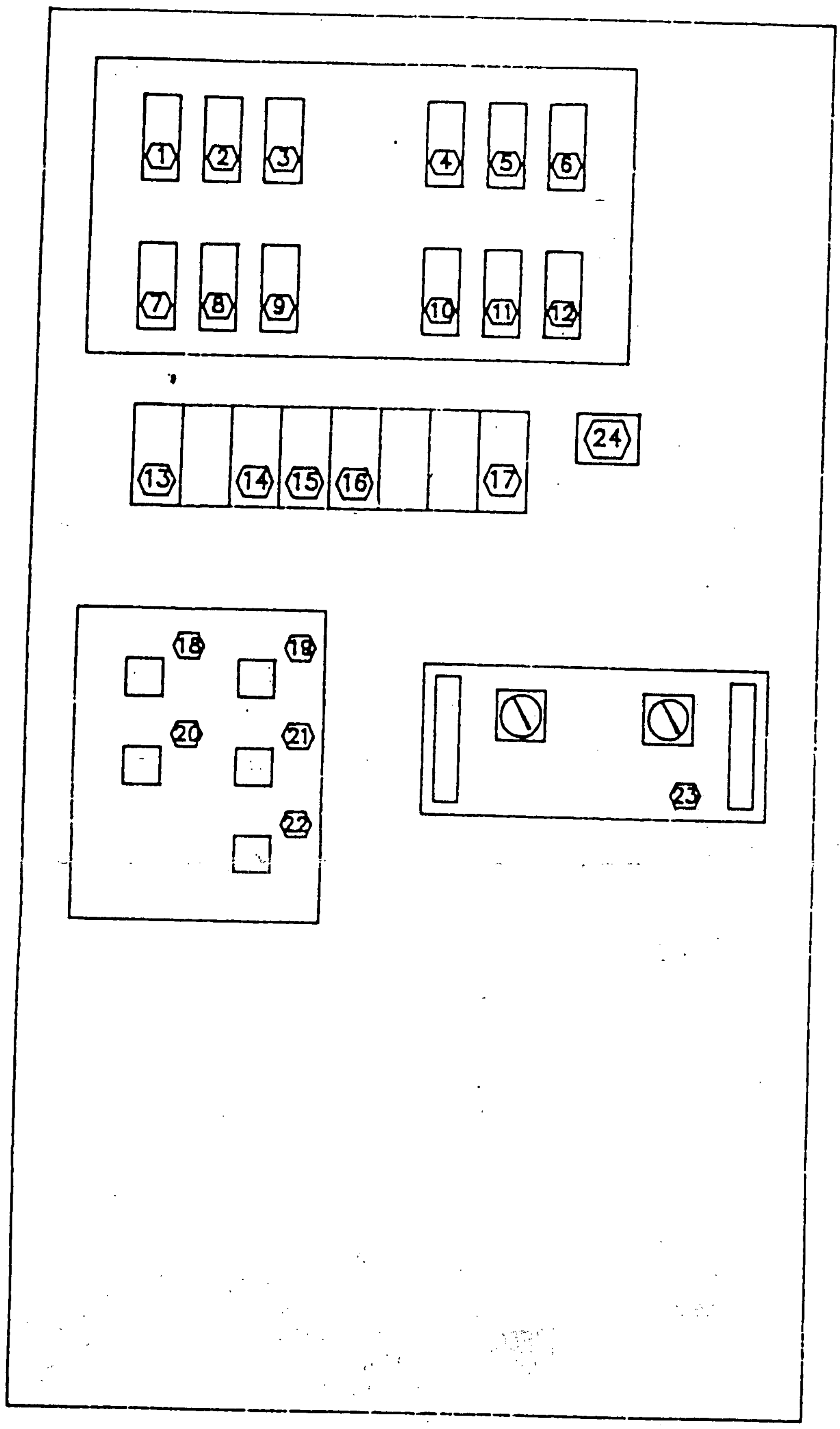


Figure B.4.2-25

Diesel Generator Control Panels, C41/42, After

BEFORE

Remote Shutdown Panel



1–RCS Cold Leg Temp (Loop A)
2–RCS Cold Leg Temp (Loop B)
3–RCS Cold Leg Temp (Loop C)
4–RCS Hot Leg Temp (Loop A)
5–RCS Hot Leg Temp (Loop B)
6–RCS Hot Leg Temp (Loop C)
7–Steam Generator Level (Loop A)
8–Steam Generator Level (Loop B)
9–Steam Generator Level (Loop C)
10–Spare
11–Pressurizer Level
12–Pressurizer Pressure
13–Atmospheric Steam Dump Contr
14–CV 142 Control
15–CV 144 Control
16–CV 143 Control
17–Charging Flow Control
18–Steam Dump Control Transfer
19–CV 530 & 546 Control
20–Charging Flow Control Transfer
~~21–Feedwater Flow Control Transfer~~
22–Condenser Steam Dump Control
23–Log Power Channel
24–FCV–5051 Indication

Remote Shutdown Panel, C38, Before

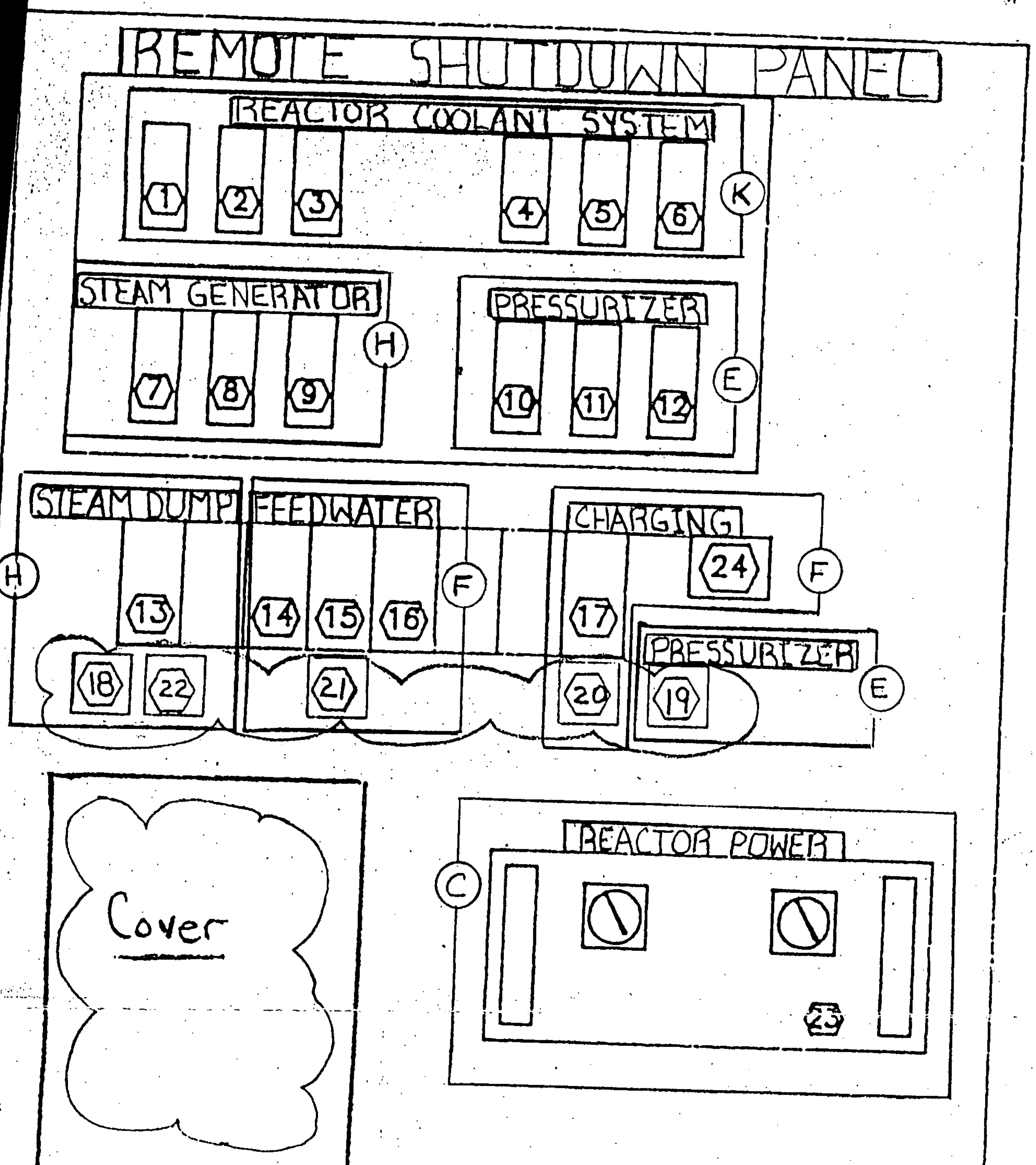
Figure B.4.2-26

8712210170-07

Also Available On Aperture Card

TI APERTURE CARD

AFTER



1—RCS Cold Leg Temp (Loop
2—RCS Cold Leg Temp (Loop
3—RCS Cold Leg Temp (Loop
4—RCS Hot Leg Temp (Loop
5—RCS Hot Leg Temp (Loop
6—RCS Hot Leg Temp (Loop
7—Steam Generator Level (L
8—Steam Generator Level (L
9—Steam Generator Level (L
10—Spare
11—Pressurizer Level
12—Pressurizer Pressure
13—Atmospheric Steam Dum
14—CV 142 Control
15—CV 144 Control
16—CV 143 Control
17—Charging Flow Control
18—Steam Dump Control Tra
19—CV 530 & 546 Control
20—Charging Flow Control Tr
21—Feedwater Flow Control T
22—Condenser Steam Dump
23—Log Power Channel
24—FCV—5051 Indication

DEMARCATION COLOR SCHEME

A	COVERT GRAY	K	GRAY FOX
C	PRAIRIE SUNSET	L	BONNIE BLUE
D	JAMBOREE ORANGE	M	JEWEL
E	TROPIC TAN	N	MARTINIQUE
F	PINE CONE	S2	ORANGE BLAZE
G	VIKING GRAY	T1	LAVENDER WHITE
H	NEUTRAL TINT	T2	ORANGE
I	POMADOUR BLUE	X	BRITTANY BLUE

Remote Shutdown Panel, C38, After

Figure B.4.2-27

8712210170-08

Also Available On Aperture Card

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BEFORE

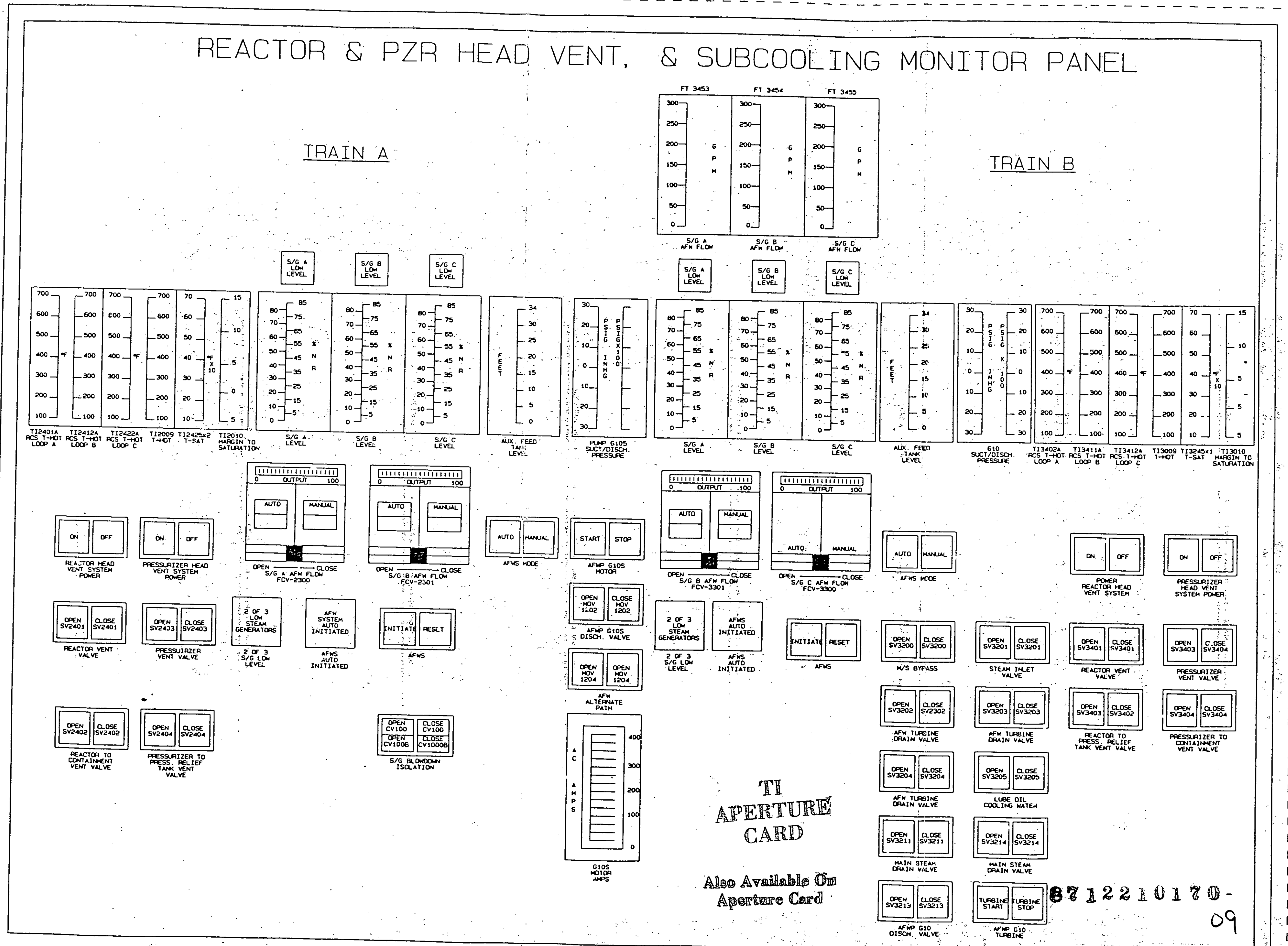


Figure B.4.2-28

Auxiliary Feedwater Panel C71 Before

AFTER

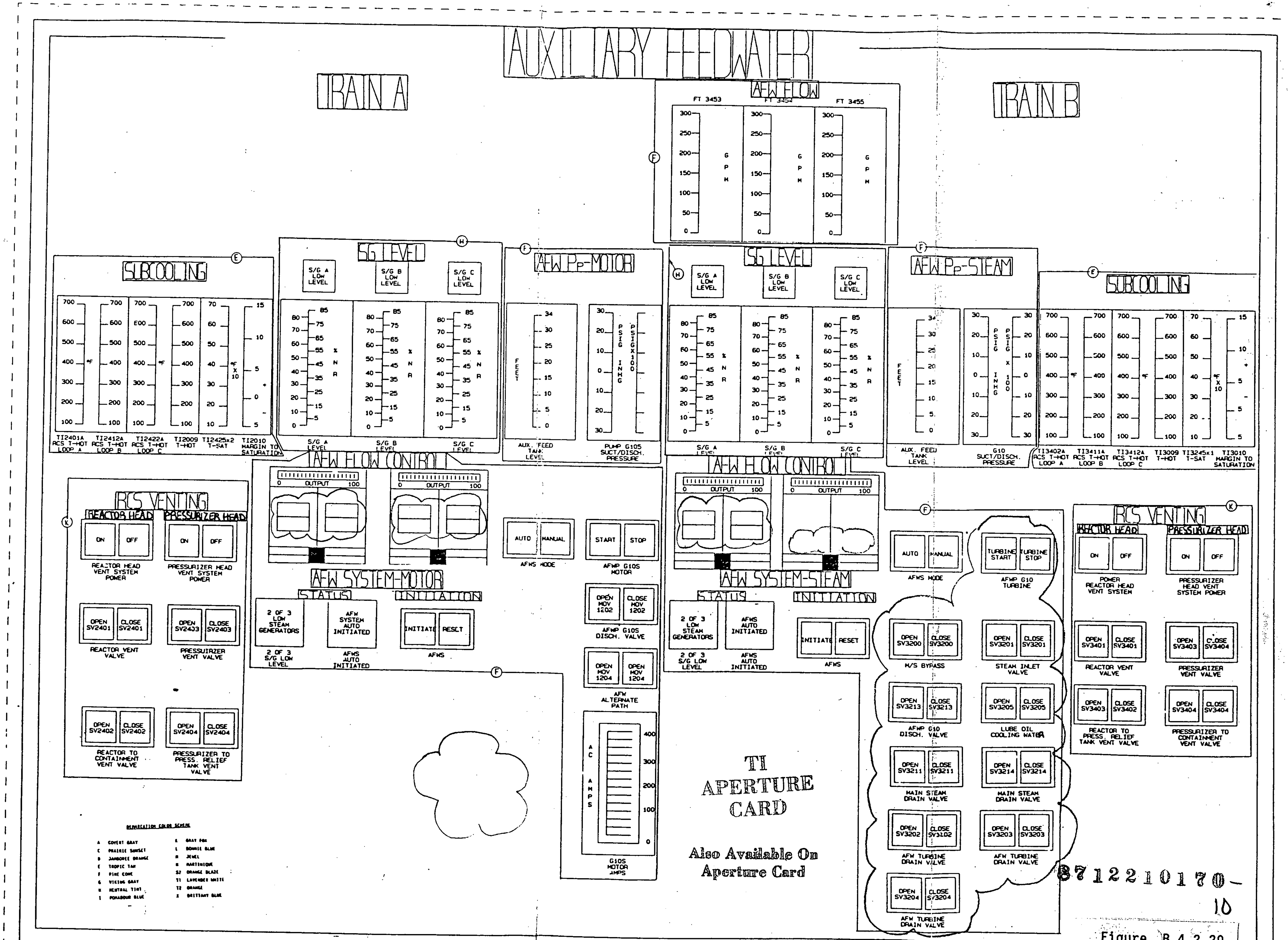
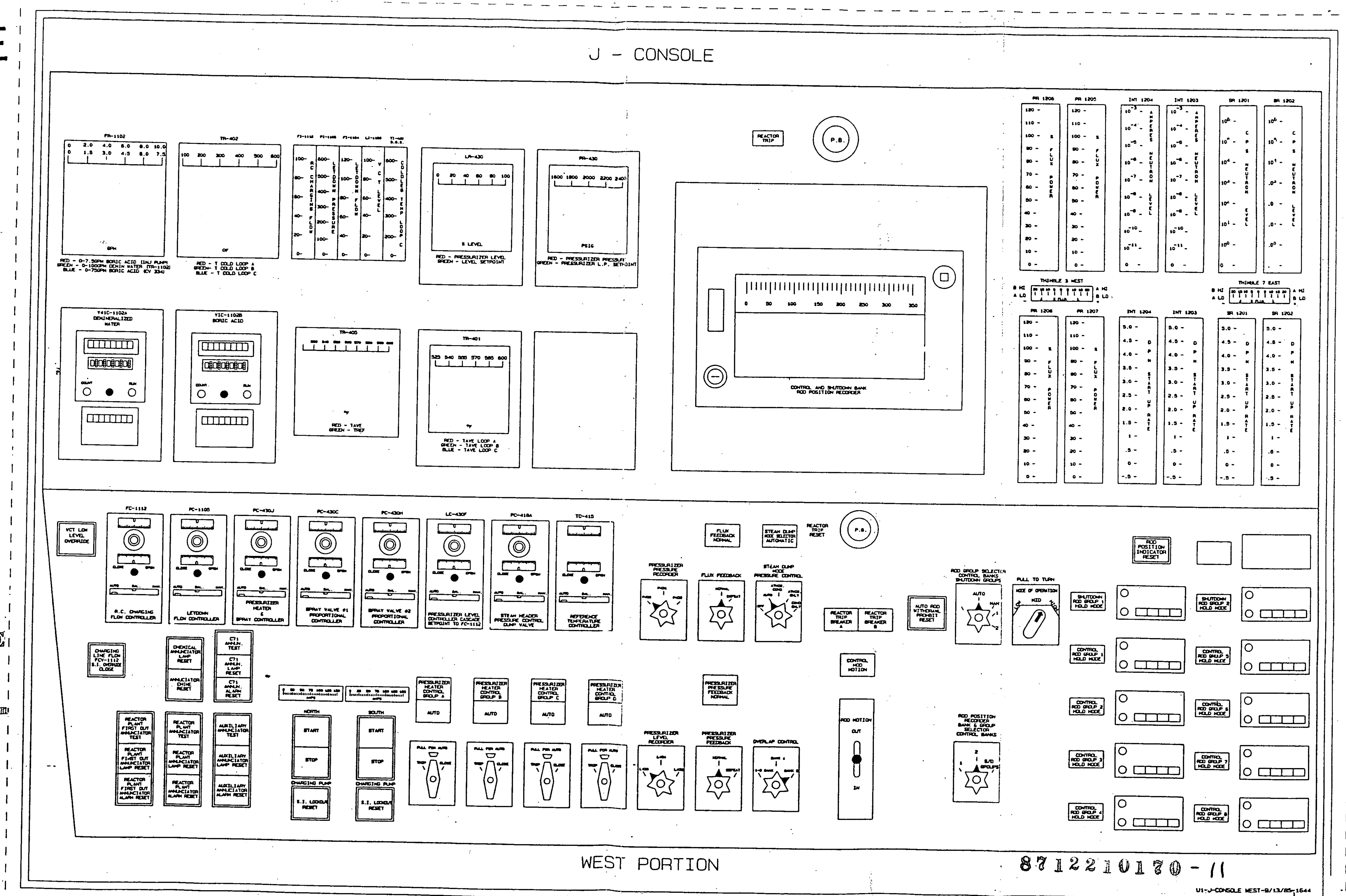


Figure B.4.2-29
Auxiliary Feedwater Panel: C71- Aft

Work Package No.

BEFORE



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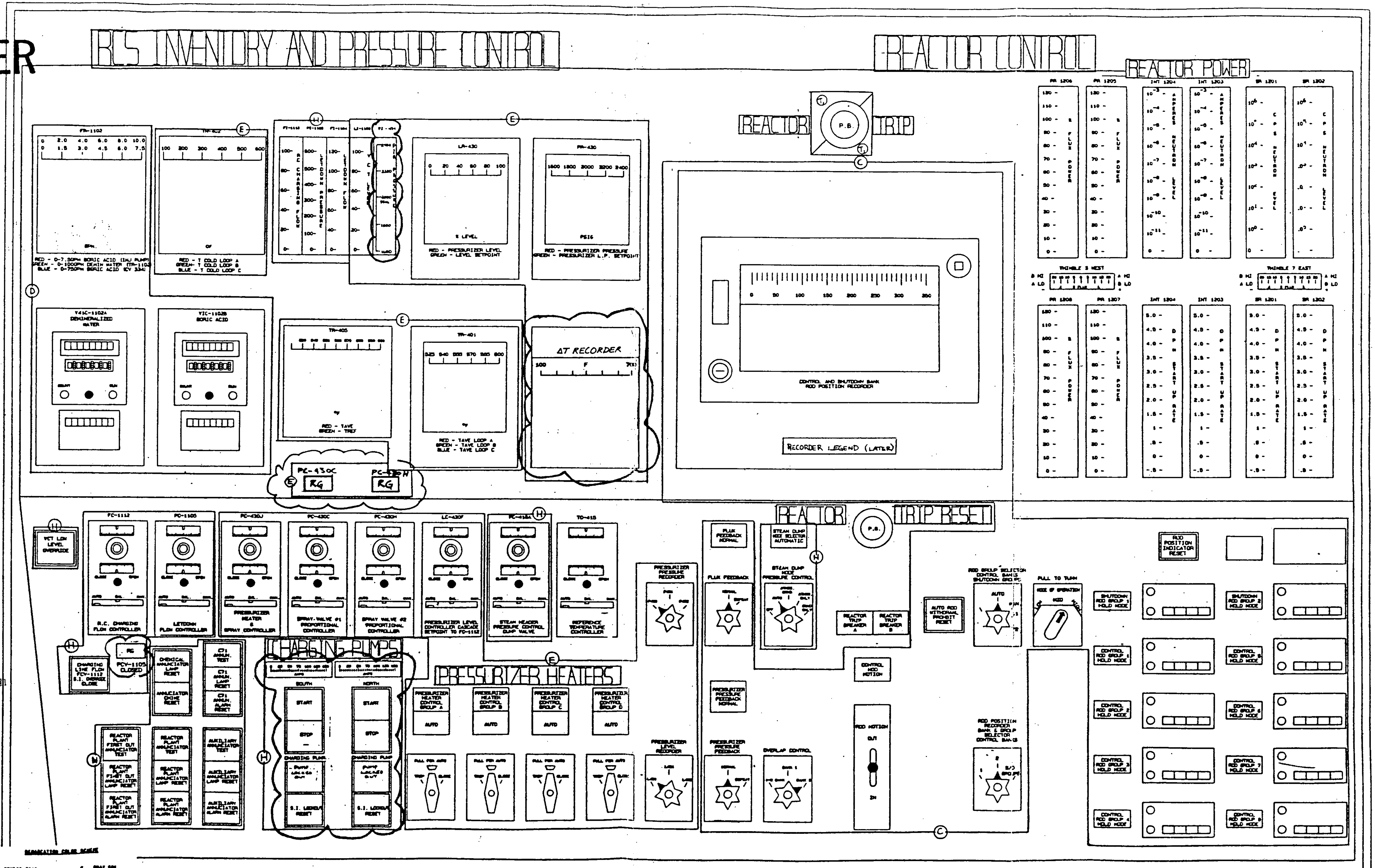
Also Available On Aperture Card

Figure B.4.2-1

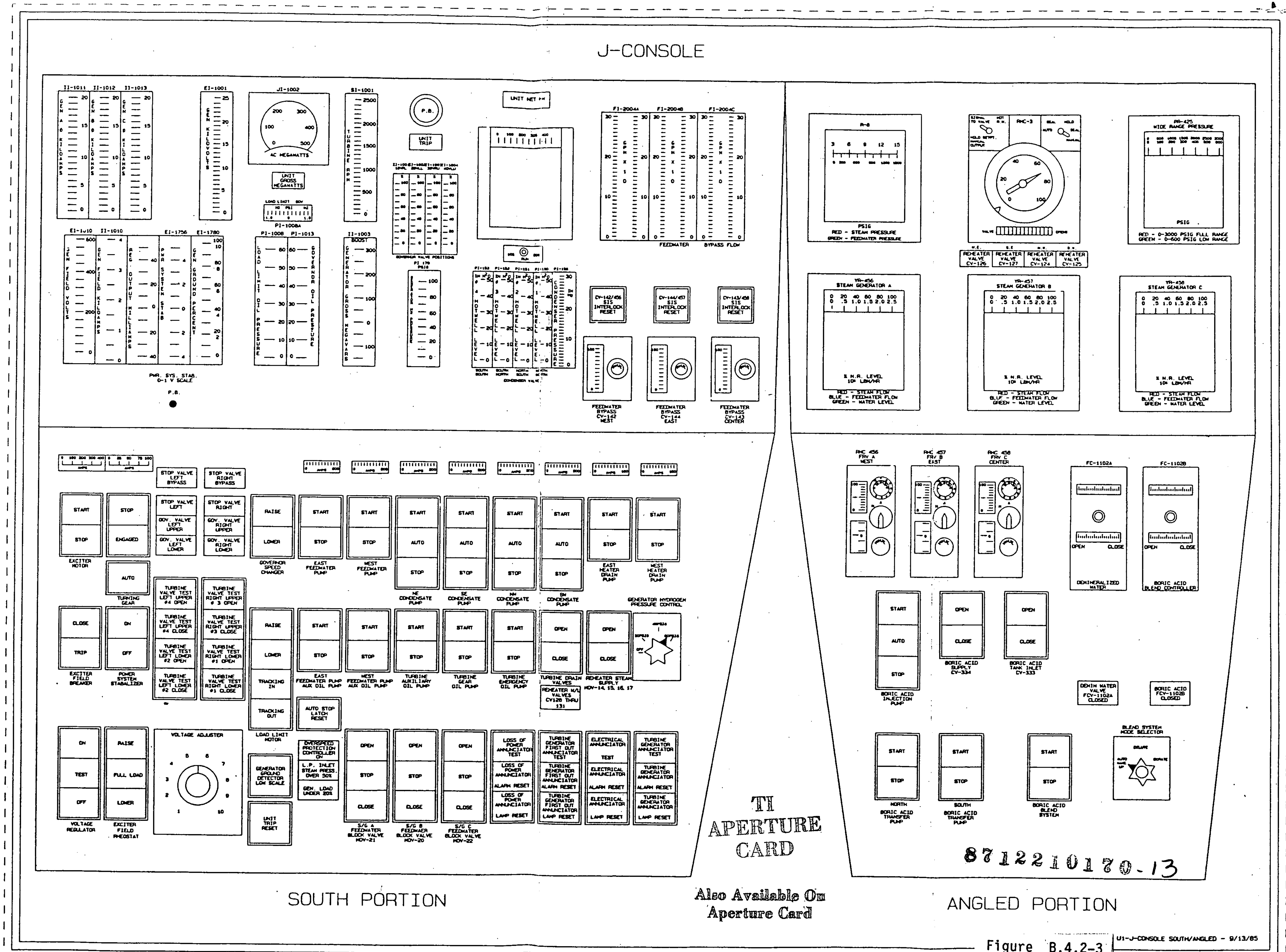
Operator J-Console C03 Before

● AFTER



87122101 70 -12
Figure B.4.2-2
Operators J-Console C03 After

BEFORE



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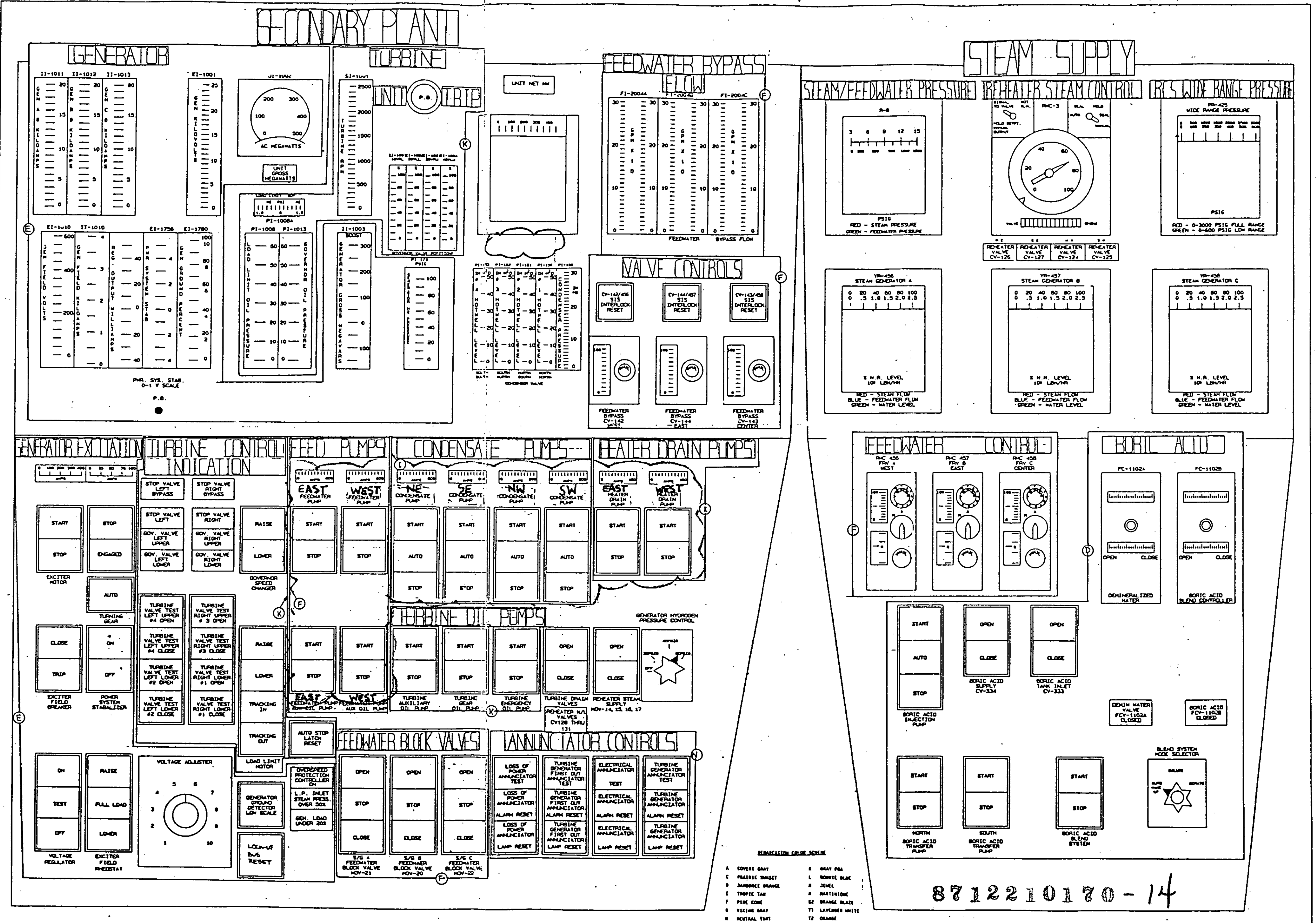
Also Available On Aperture Card

8712210170-13

Figure B.4.2-3 Operators J-Console, C04, Before

Work Package No1

AFTER



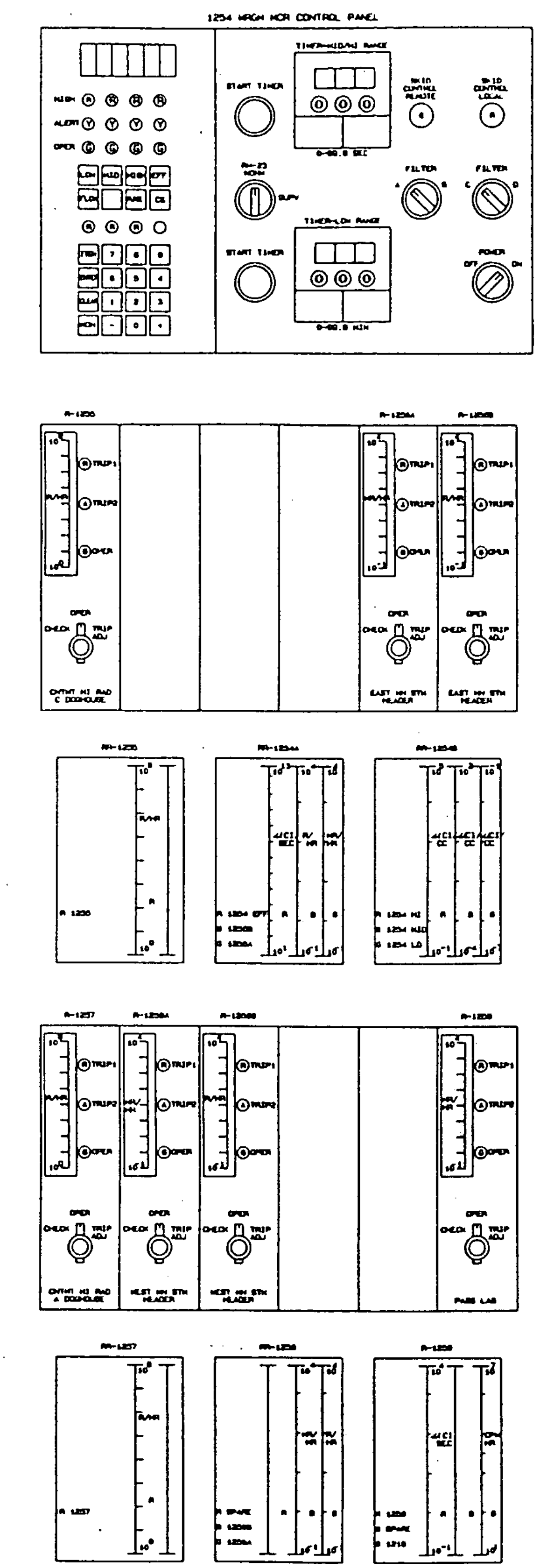
8712210170 - 14

Figure B.4.2-4

Operators J-Console, C04, After

BEFORE

WORK PACKAGE NO. 7



TI APERTURE CARD

Also Available On Aperture Card

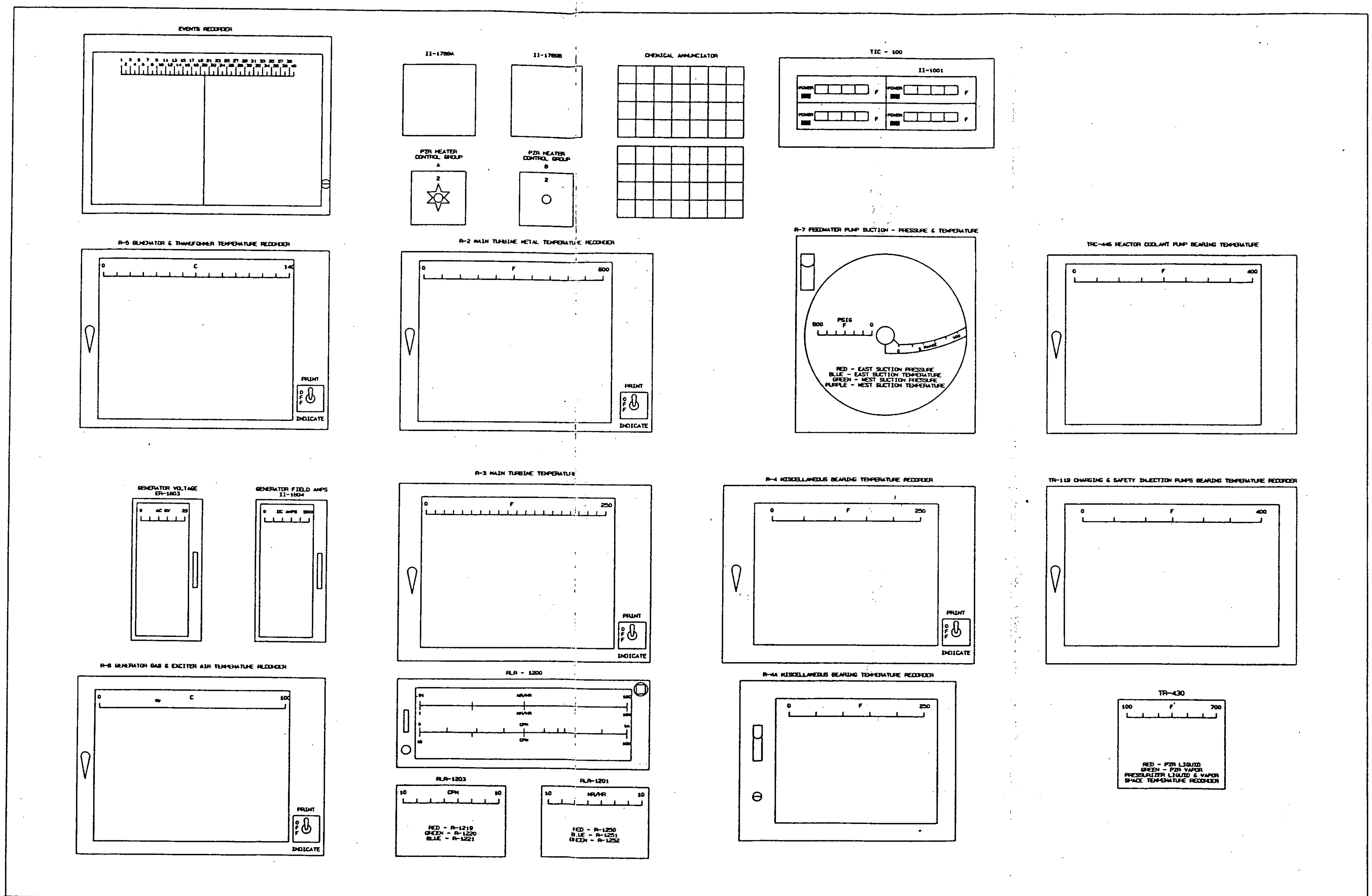
8712210170-15

Figure B.4.2-5 SH 1

Recorder Board C05, Before

BEFORE

WEST VERTICAL BOARD (MID SECTION)



TI APERTURE CARD

Also Available On Aperture Card

8712210170-16

Figure B.4.2-5 SH 2
Recorder Board, C05, Before

AFTER

WORK PACKAGE NO.7

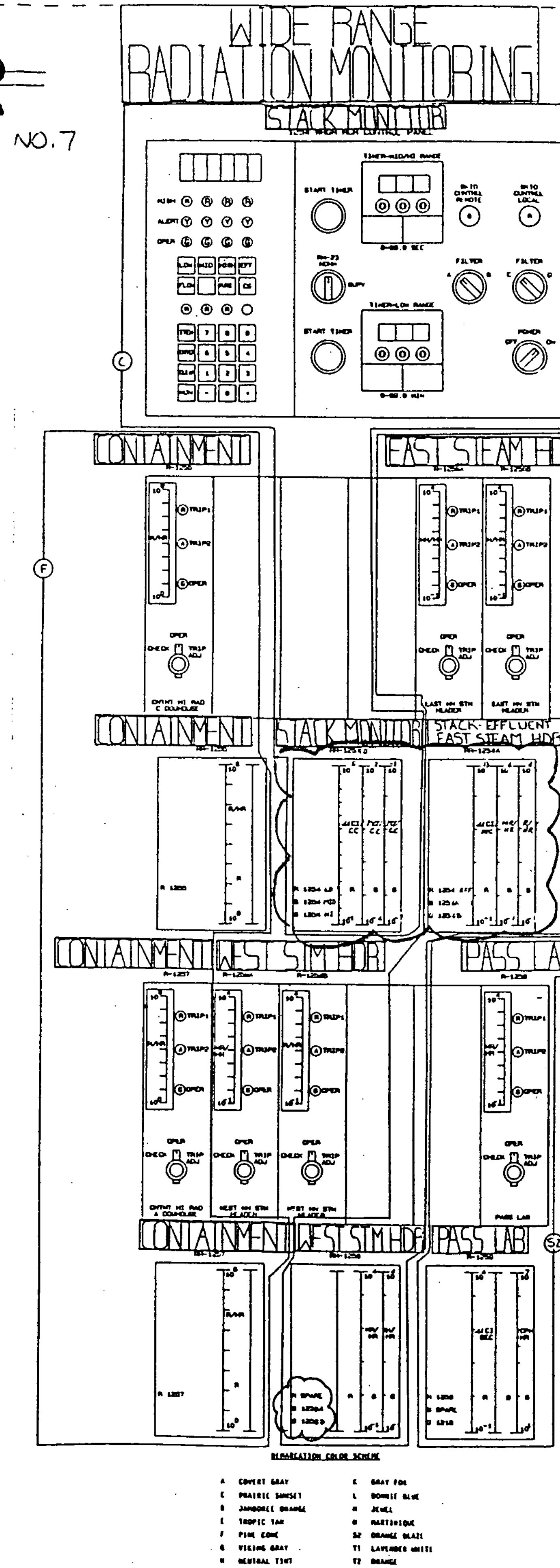
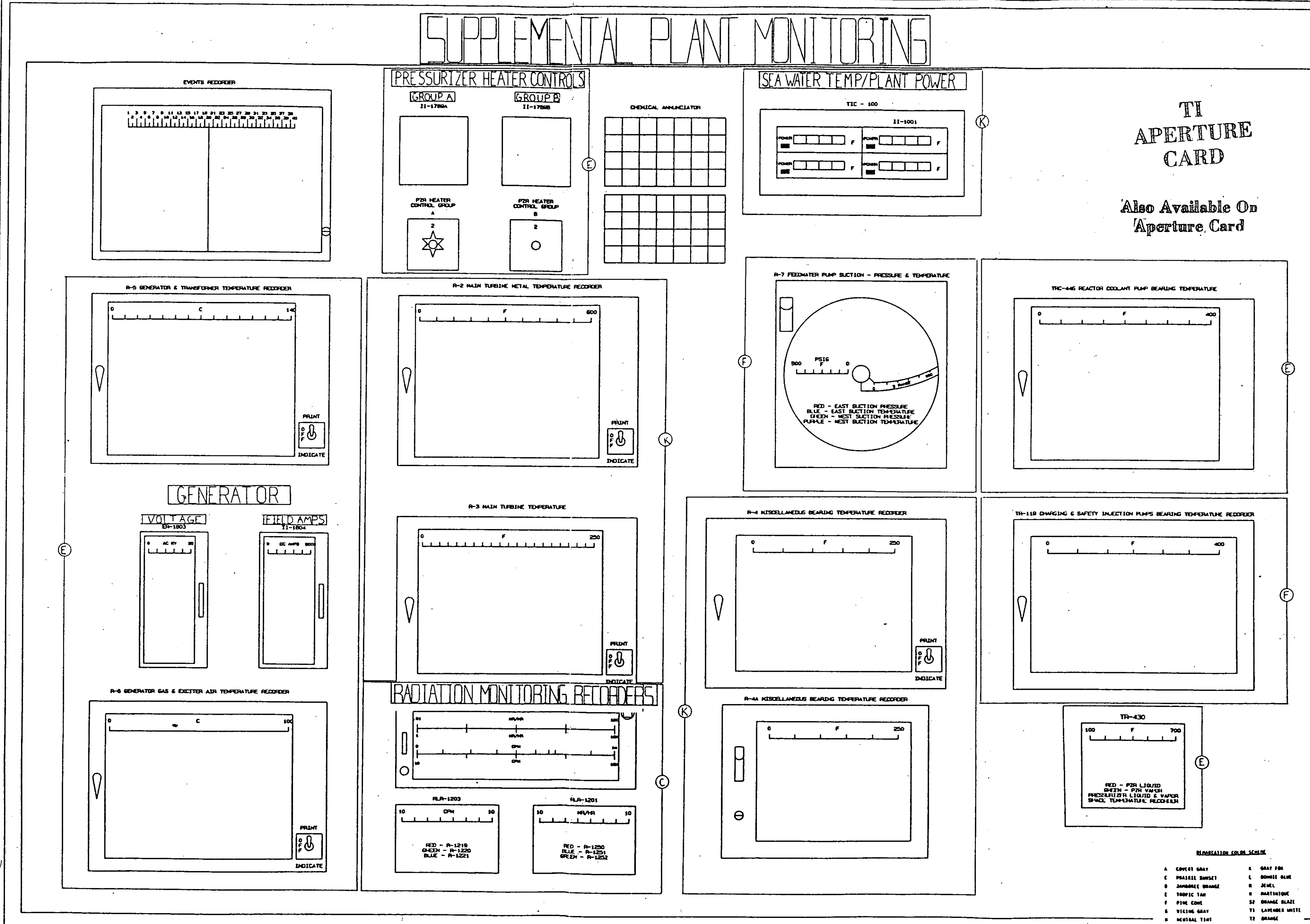


Figure B.4.2-6 SH 1

AFTER



TI
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CARD

Also Available On
Aperture Card

8712210170-18

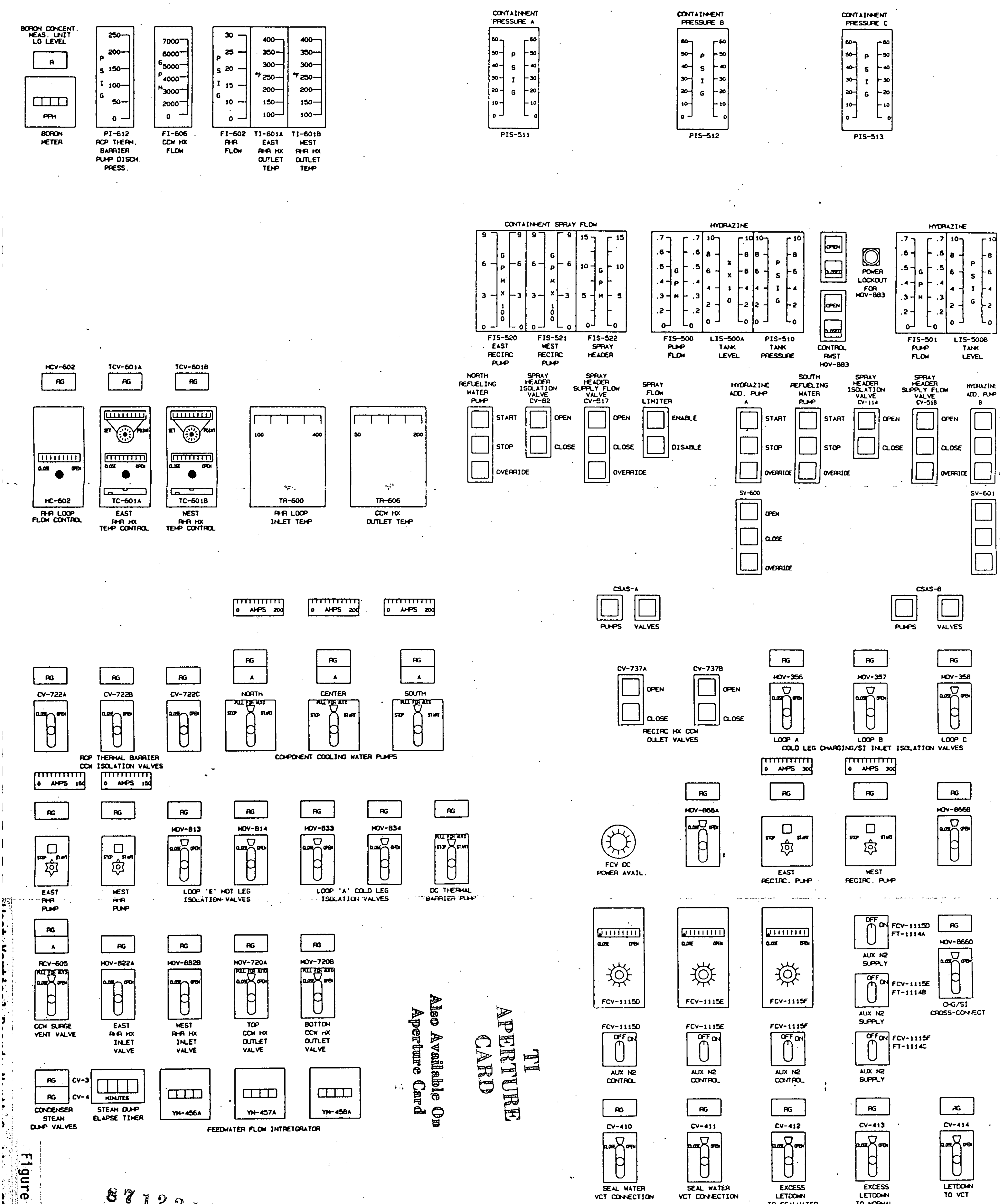
Figure B.4.2-6 SH 2

Recorder Board, C05, After

WORK PACKAGE NO. 3

BEFORE

WEST VERTICAL BOARD
(NORTH SECTION)



TI APERTURE CARD

Also Available On Aperture Card

8712210170-19

Figure B.4.2-7

TI
APERTURE
CARD

Also Available On
Aperture Card

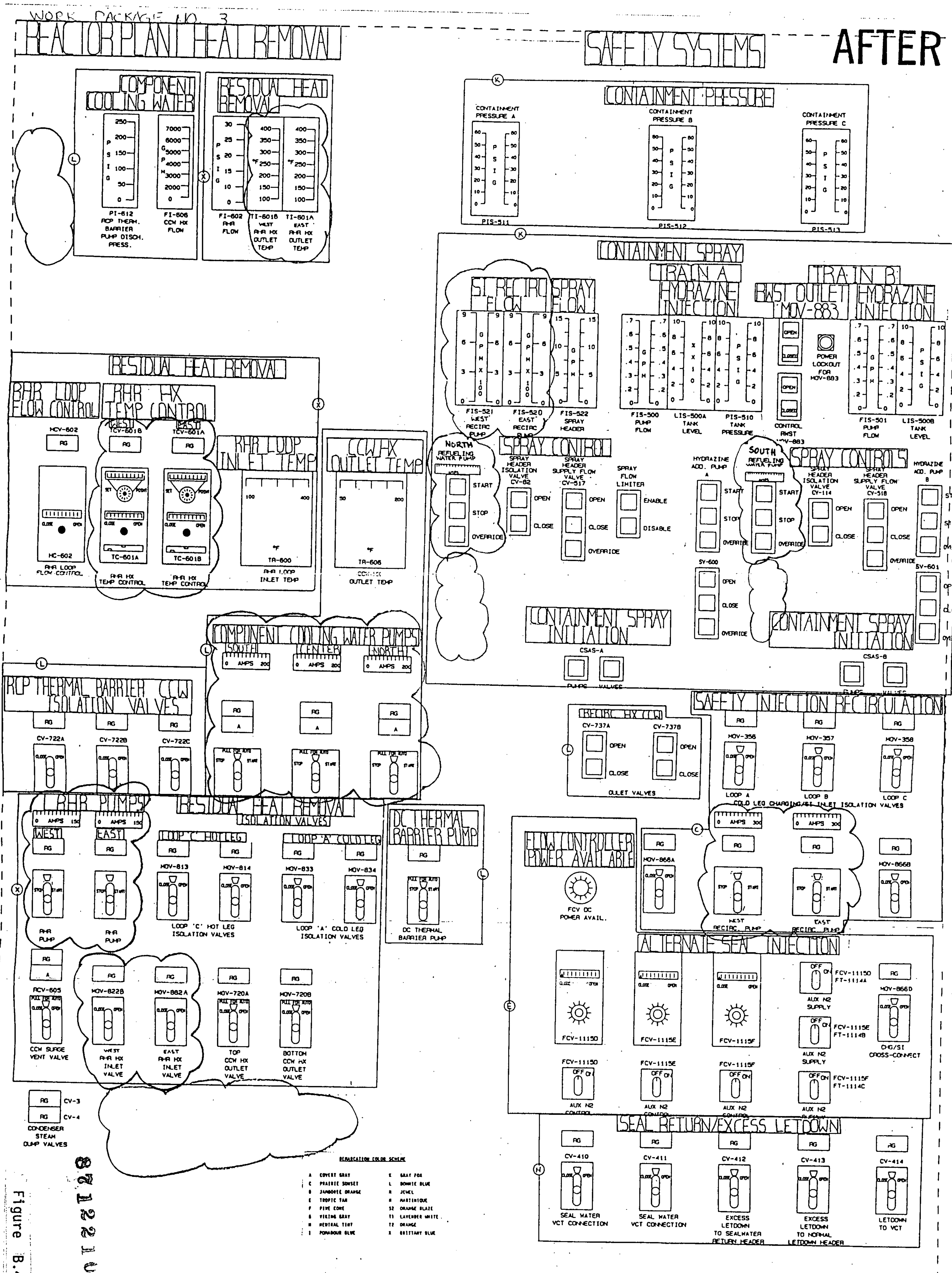
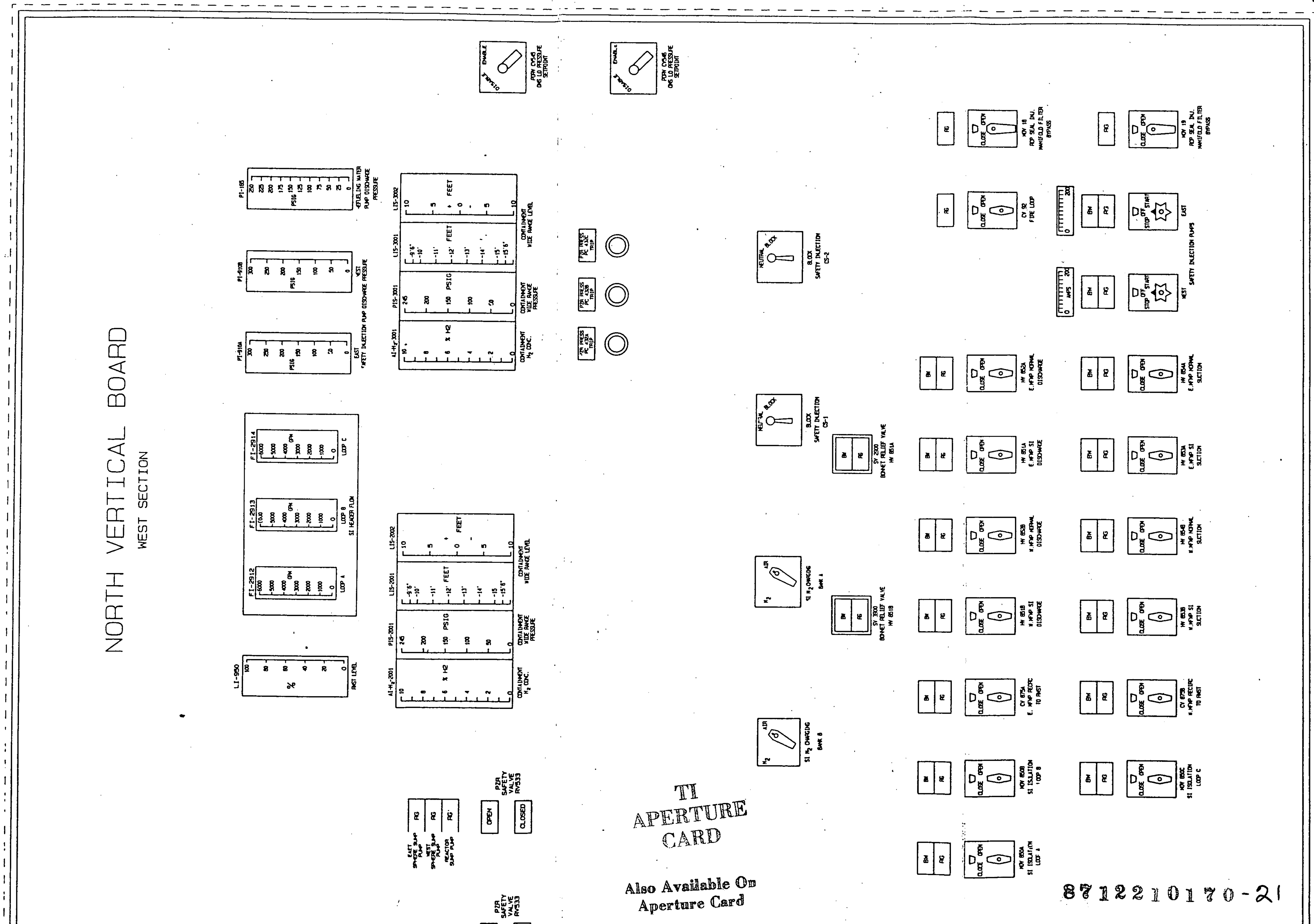


Figure B.4.2-8 West Vertical Board North Section

8712210170-20

WORK PACKAGE NO. 13

BEFORE



TI APERTURE CARD

Also Available On Aperture Card

8712210170-21

Figure B.4.2-9

North Vertical Board, West Section, C09, Before

AFTER

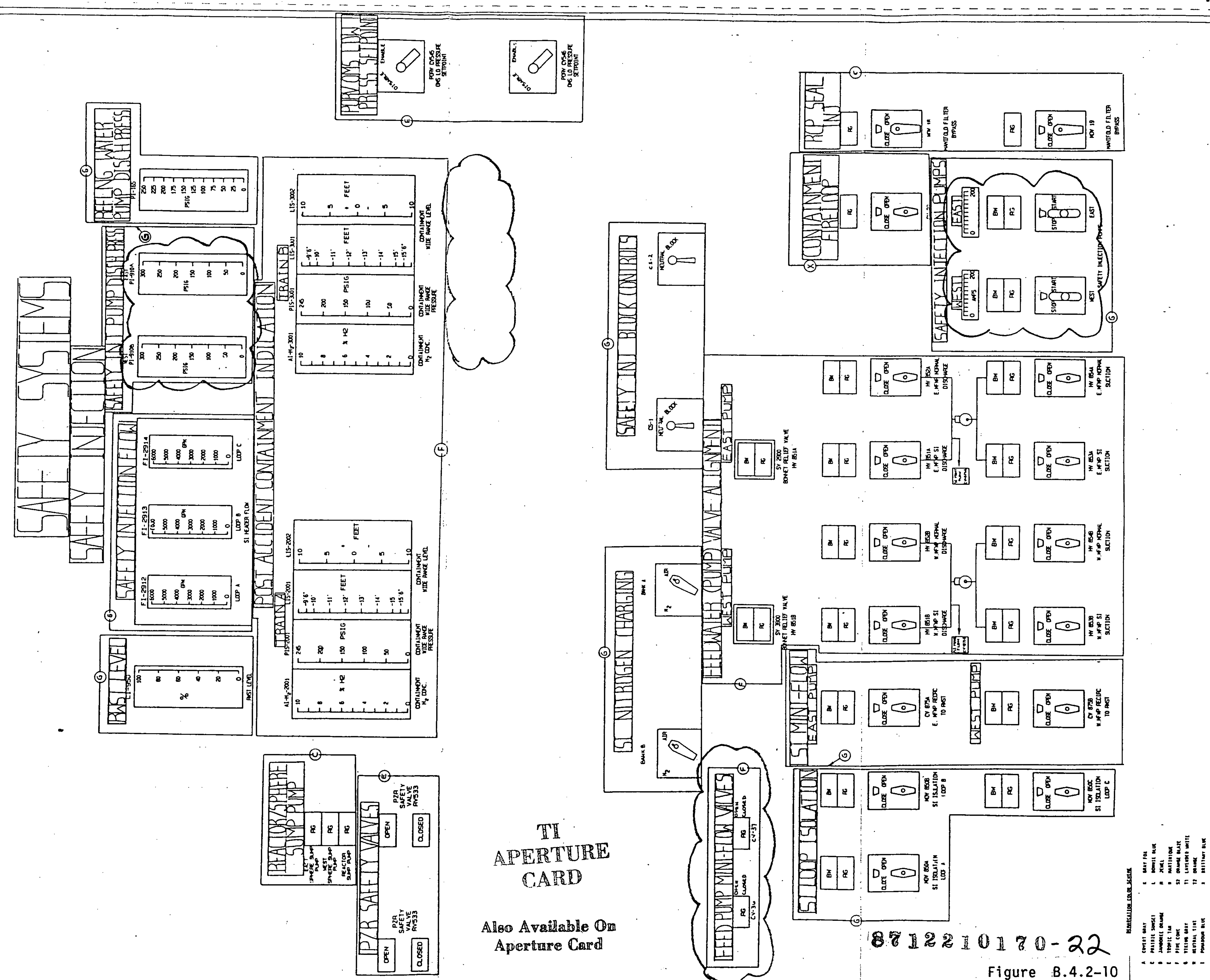


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Figure B.4.2-10

North Vertical Board, West Section, C09, After

BEFORE

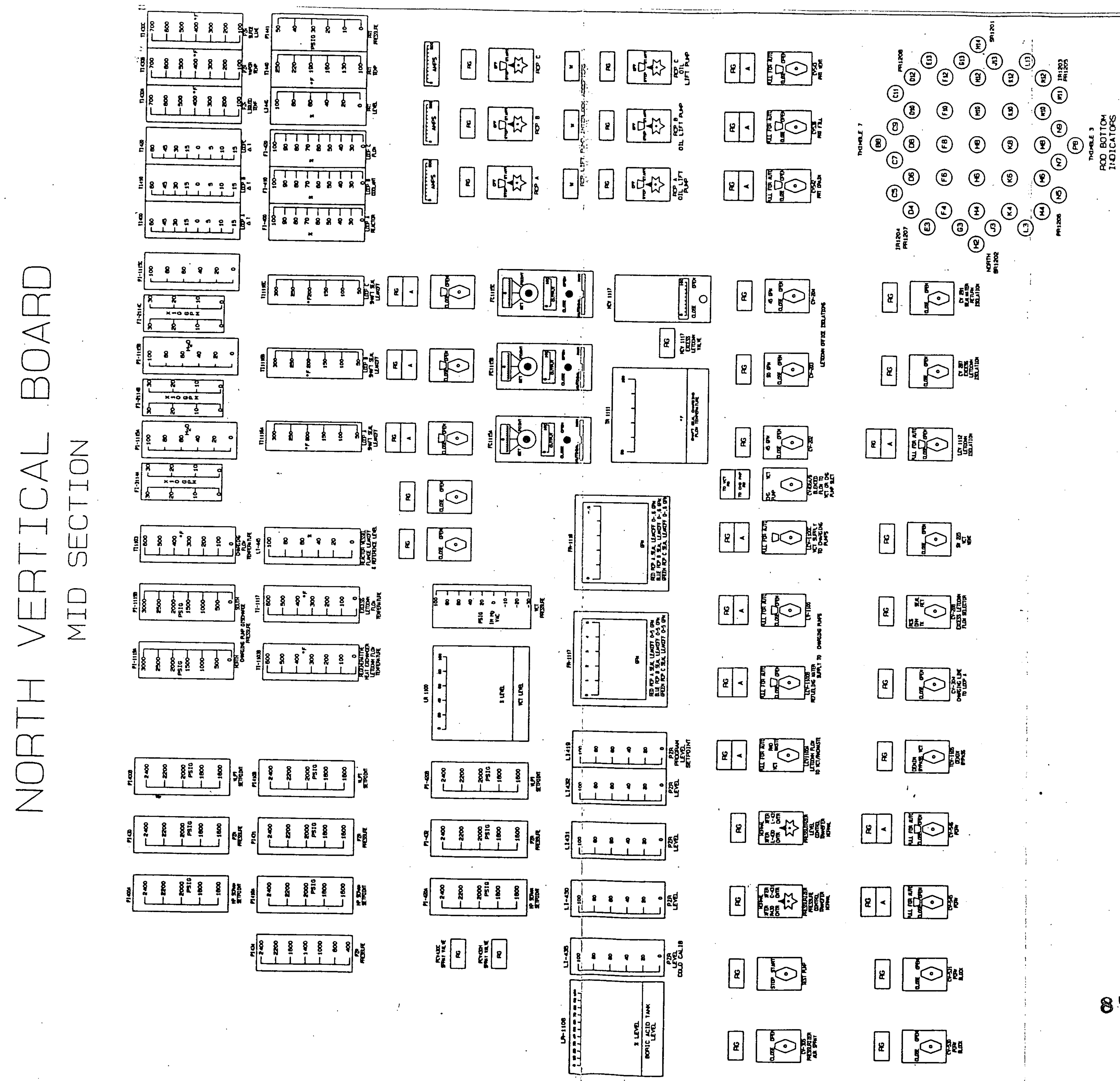

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Figure B.4.2-11
North Vertical Board, Mid Section, C09, Before

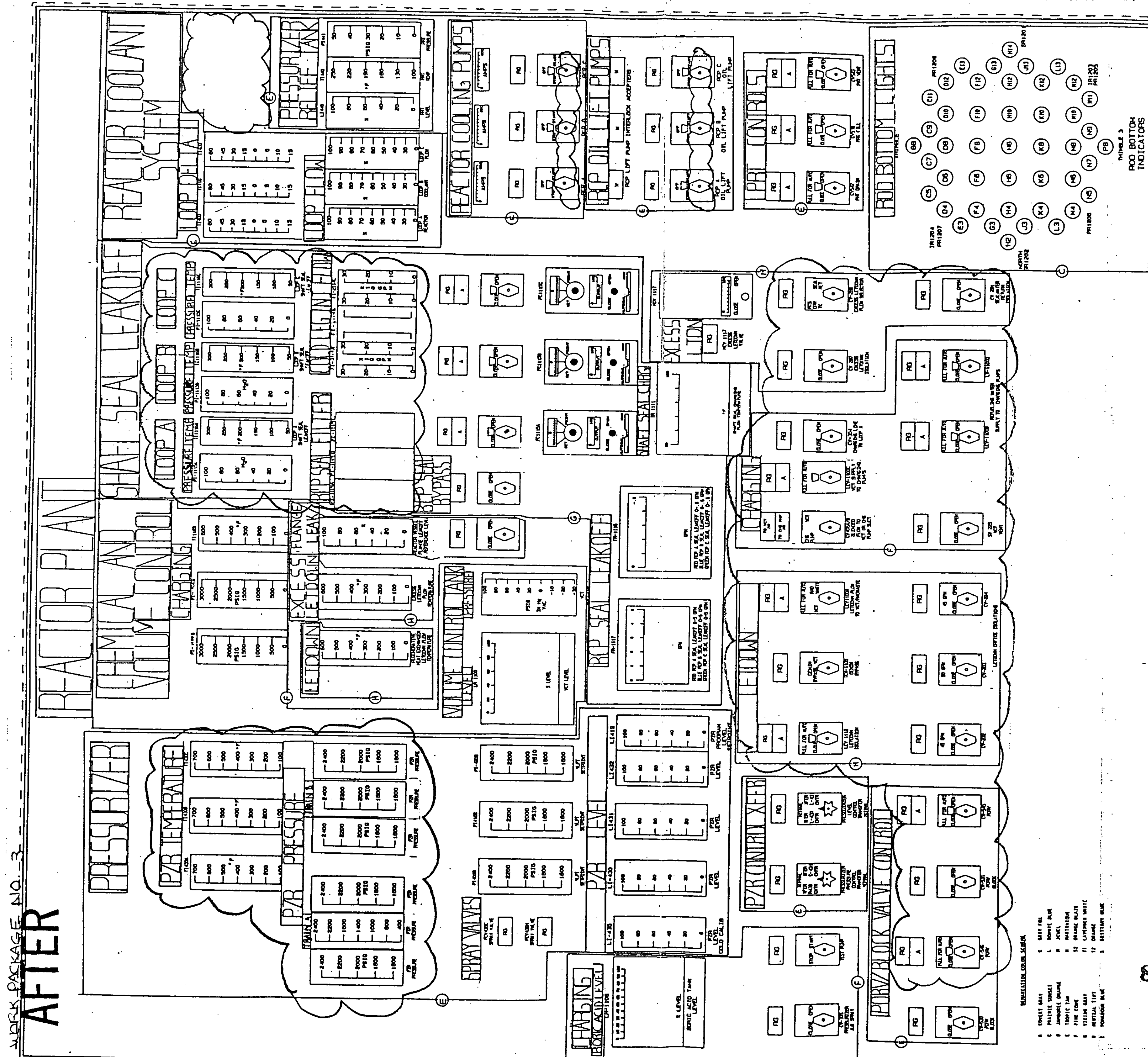

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Figure B.4.2-12

North Vertical Board, Mid Section, C09, After

BEFORE

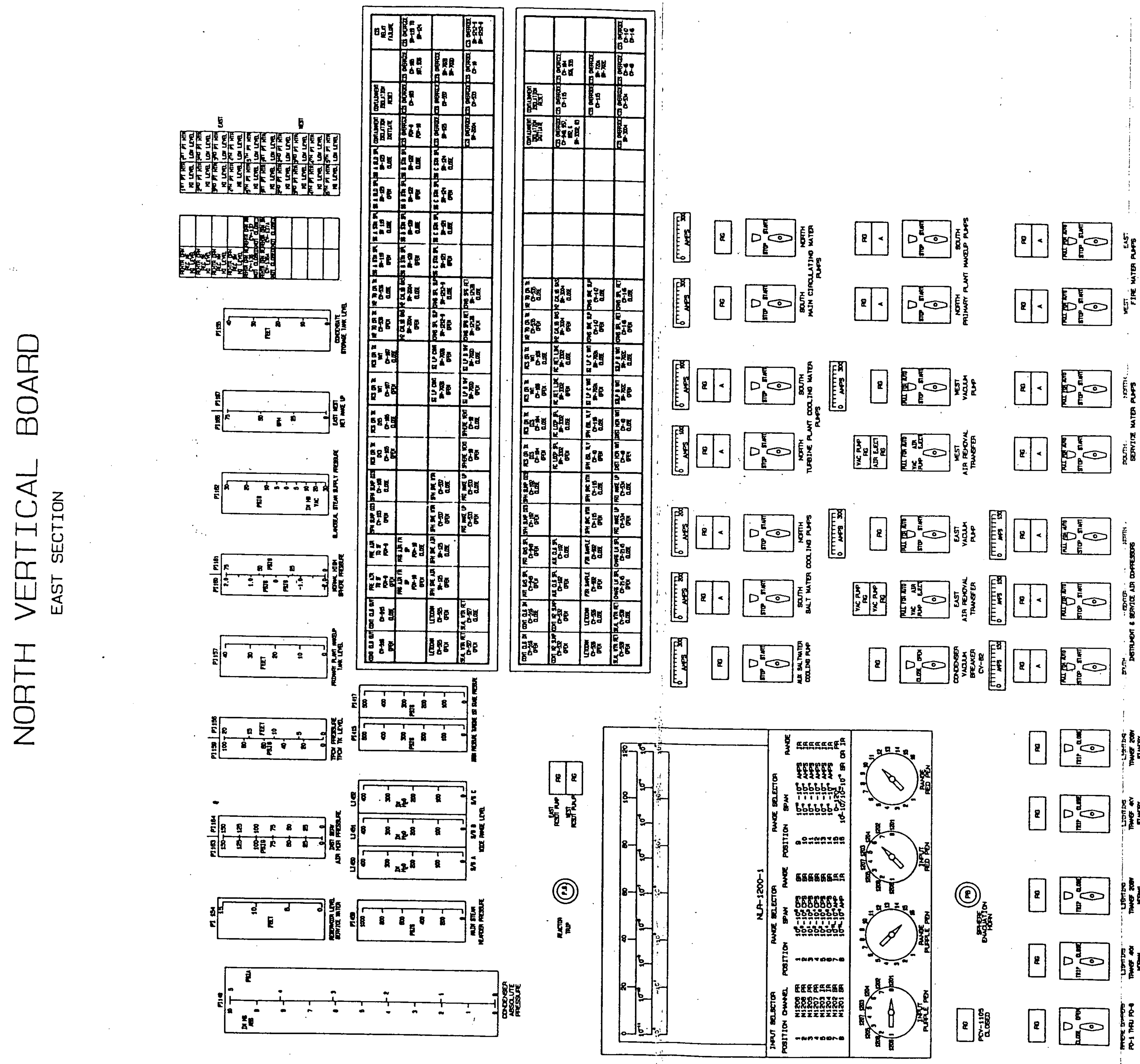

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Figure B.4.2-13

North Vertical Board, East Section, C13, Before

AFTER

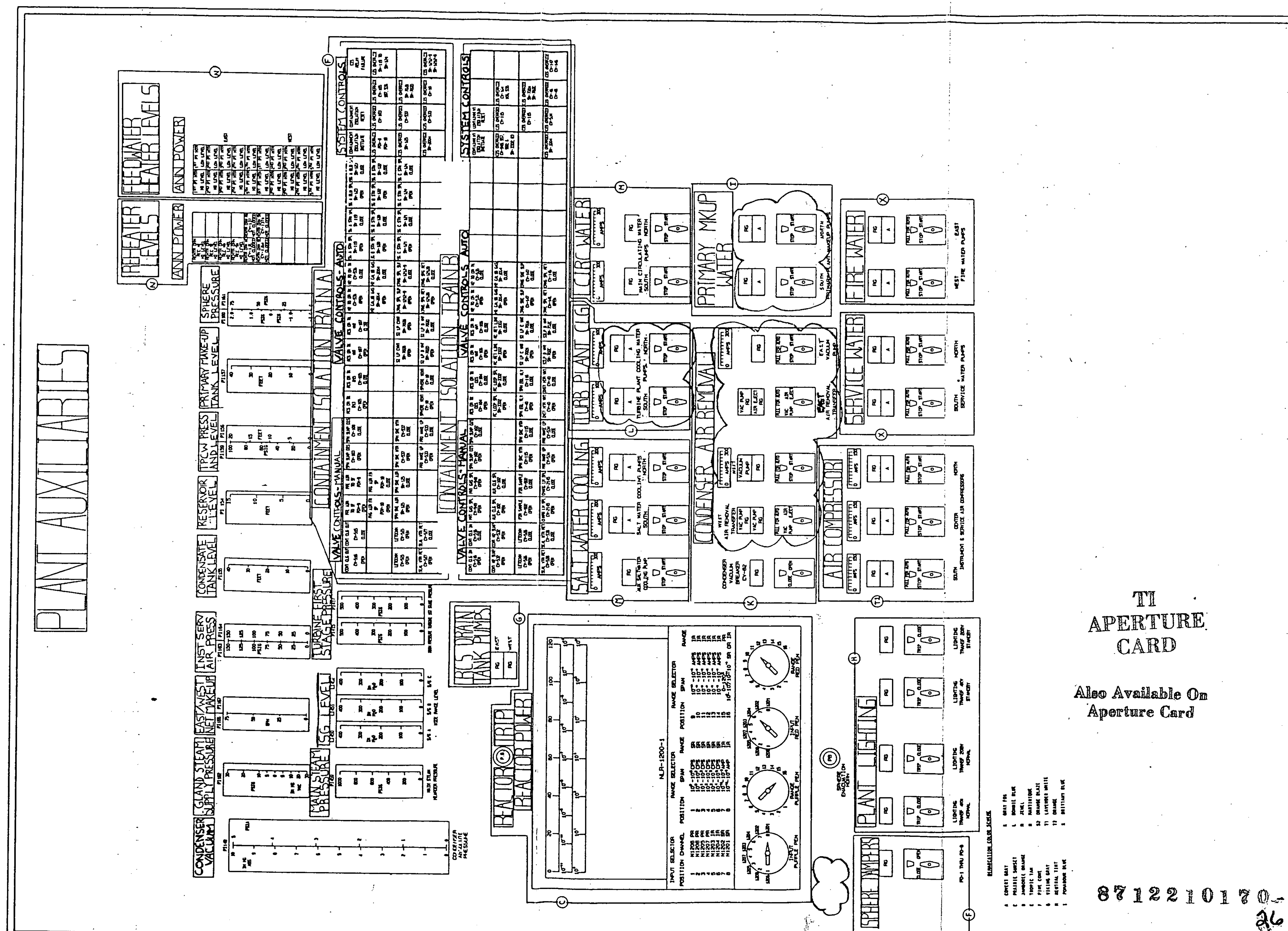

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Figure B.4.2-14

North Vertical Board East Section C13 After

BEFORE

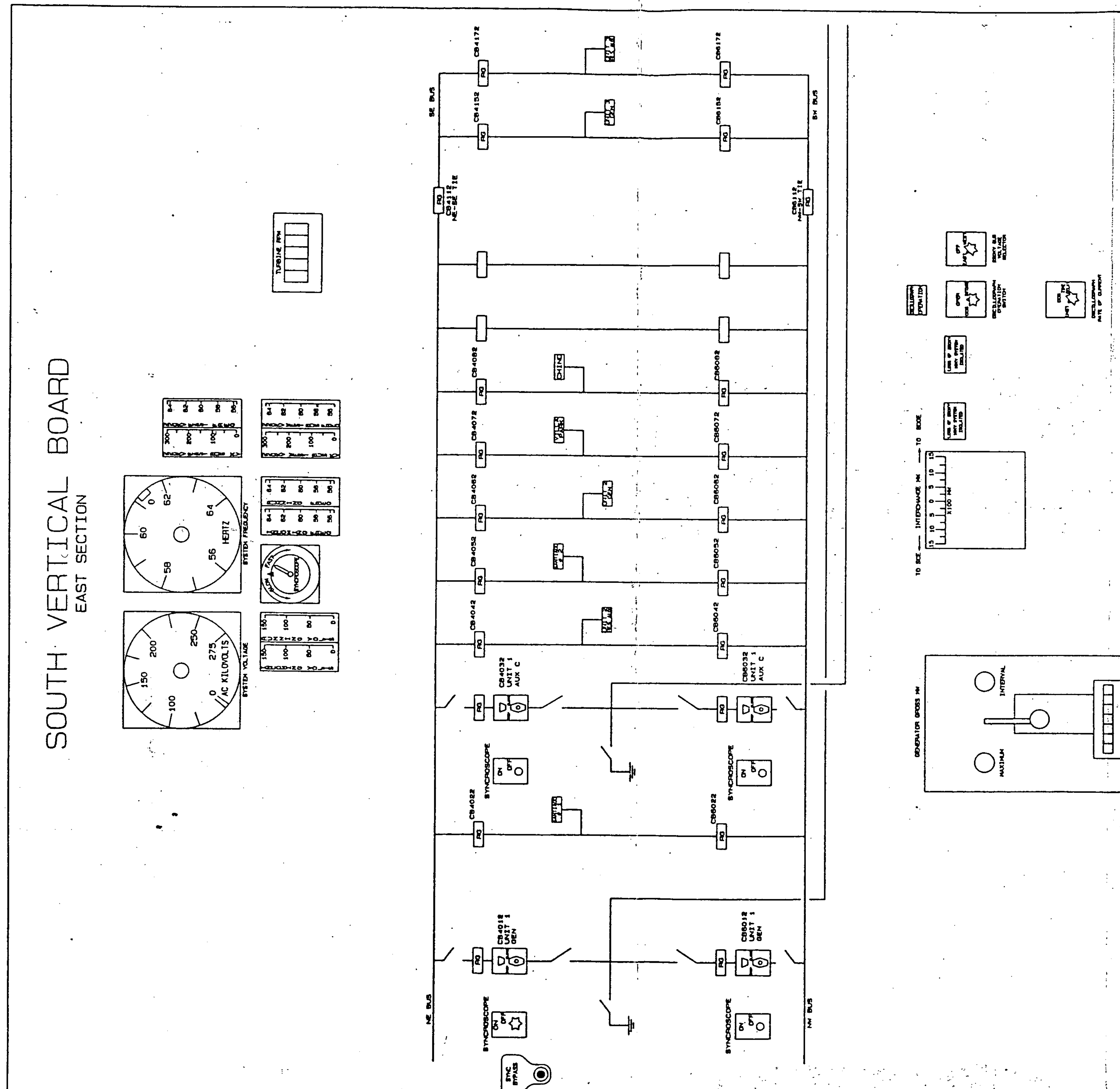

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Figure B.4.2-15

AFTER

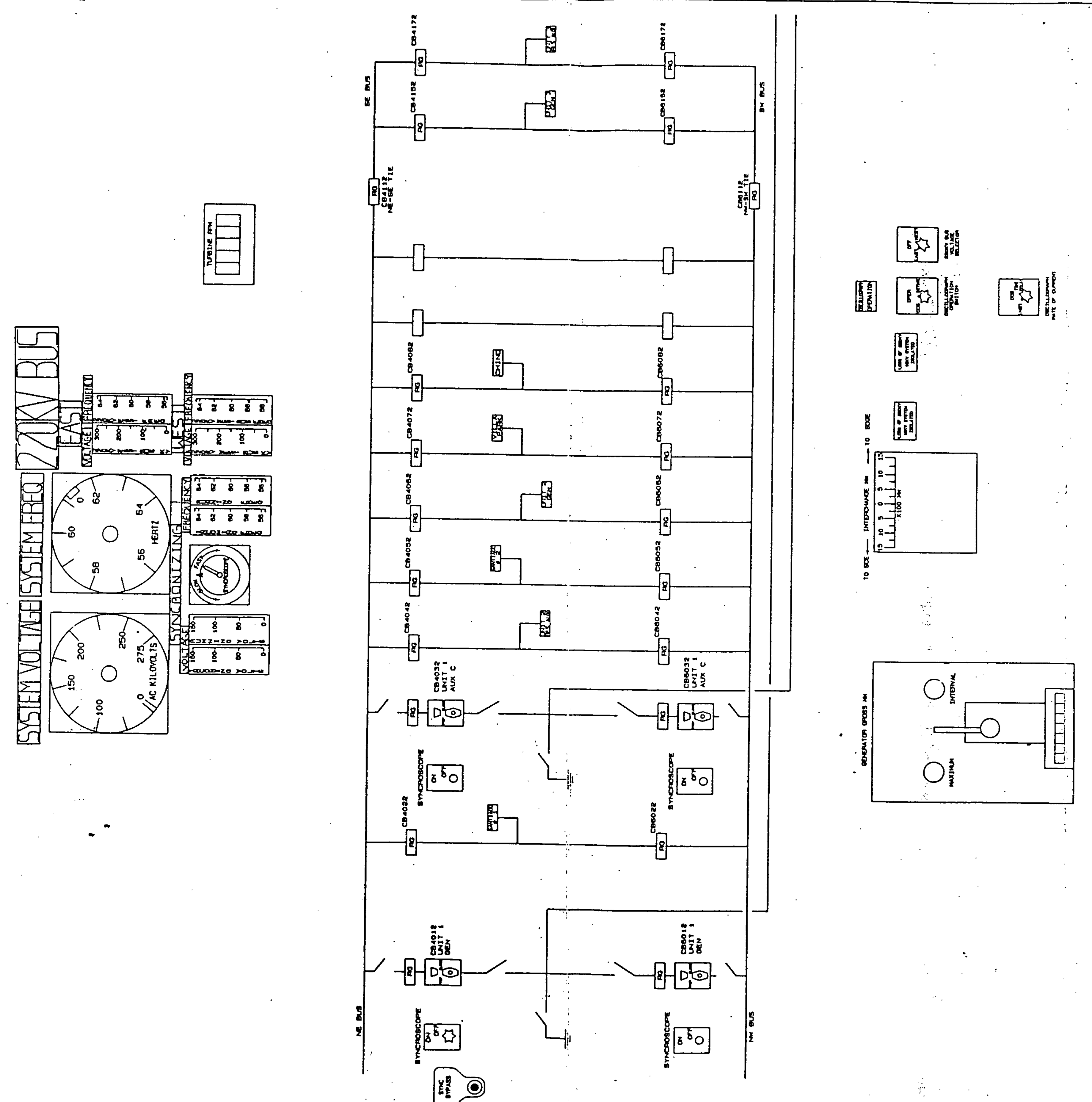

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Figure B.4.2-16

South Vertical Board, West Section, CT2, After

BEFORE

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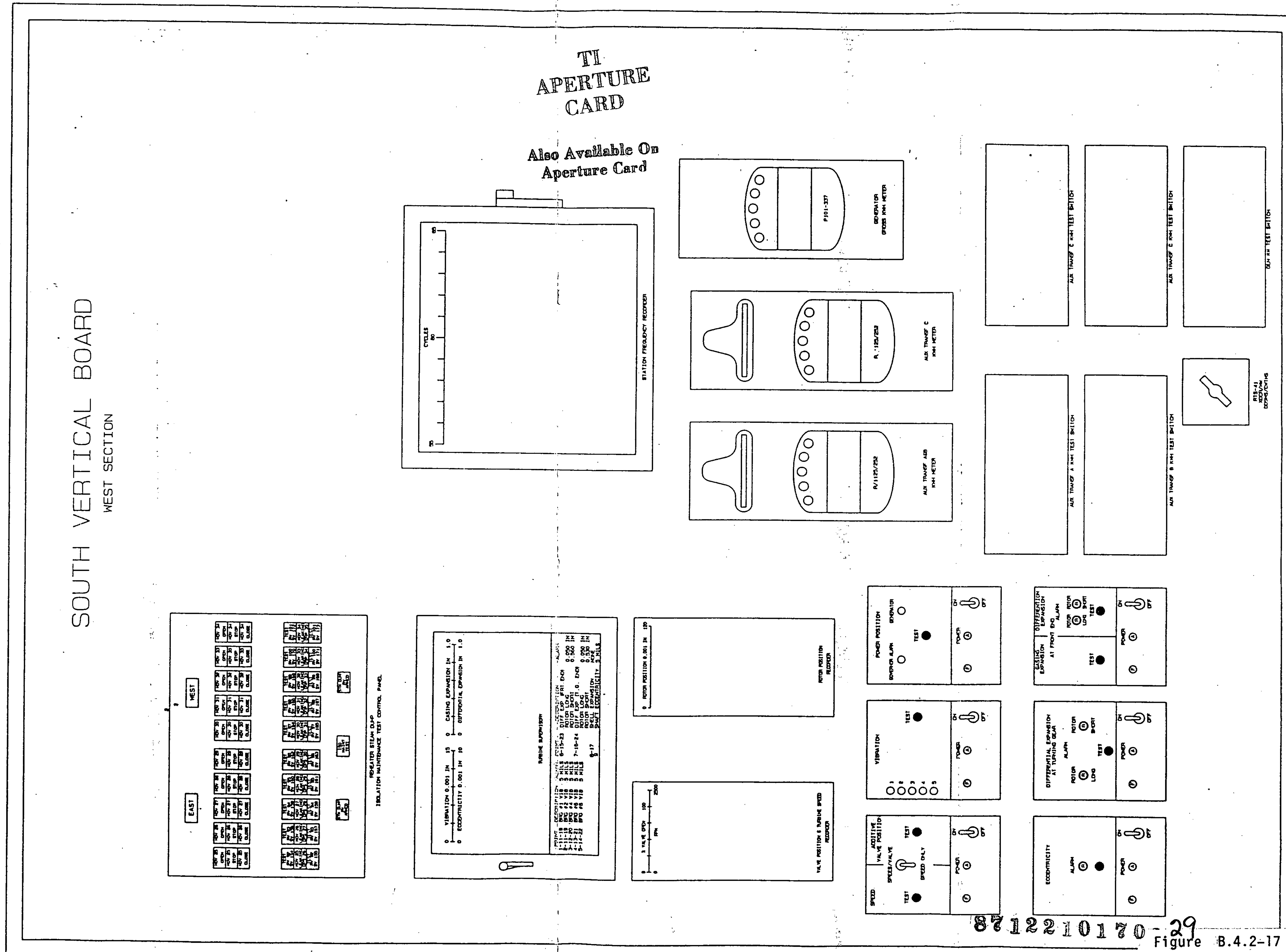

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Figure B.4.2-17

South Vertical Board, Mid Section, C12, Before

AFTER

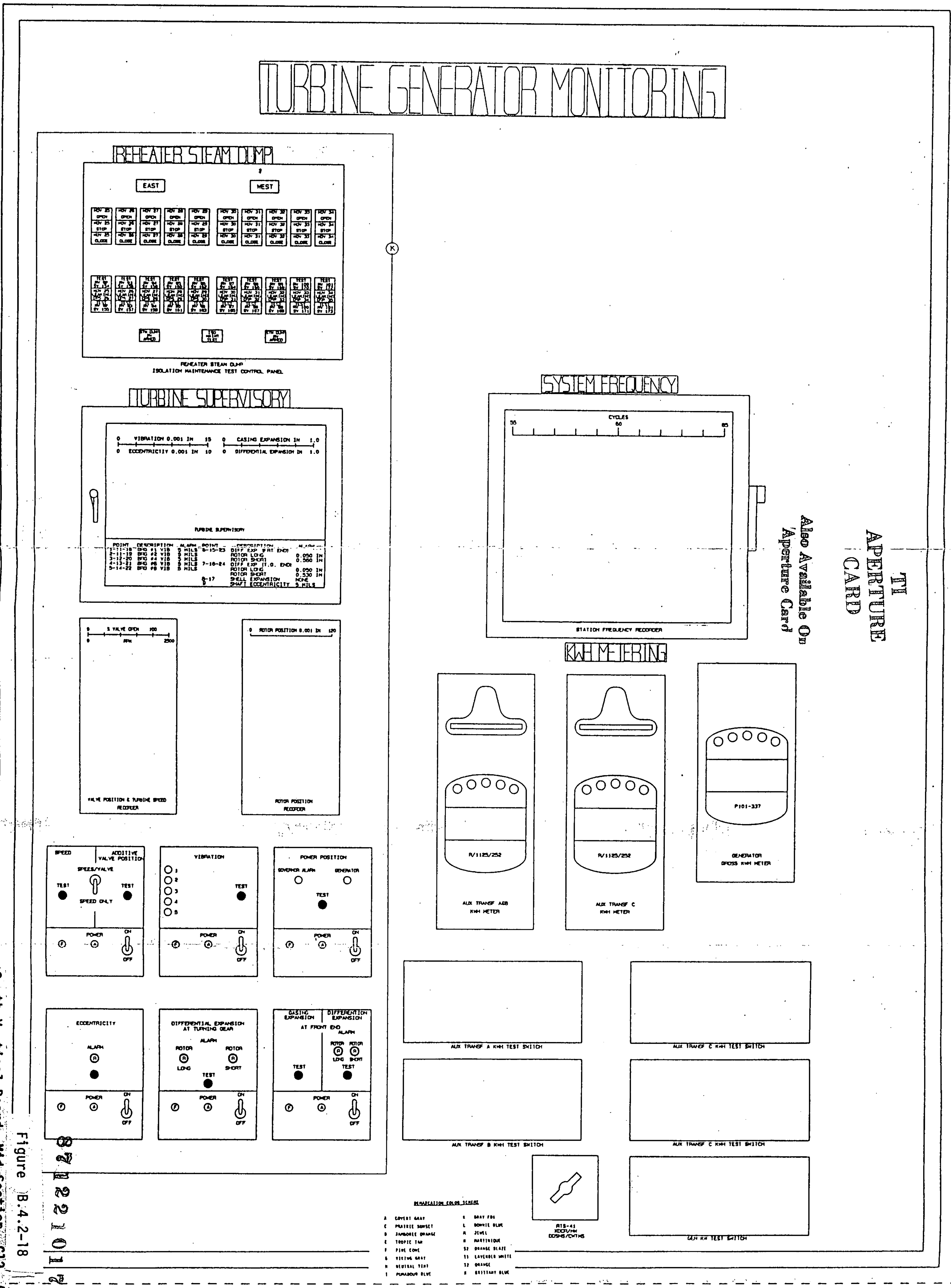

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Figure B.4.2-18 South Vertical Board, Mid Section, C12, After

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BEFORE

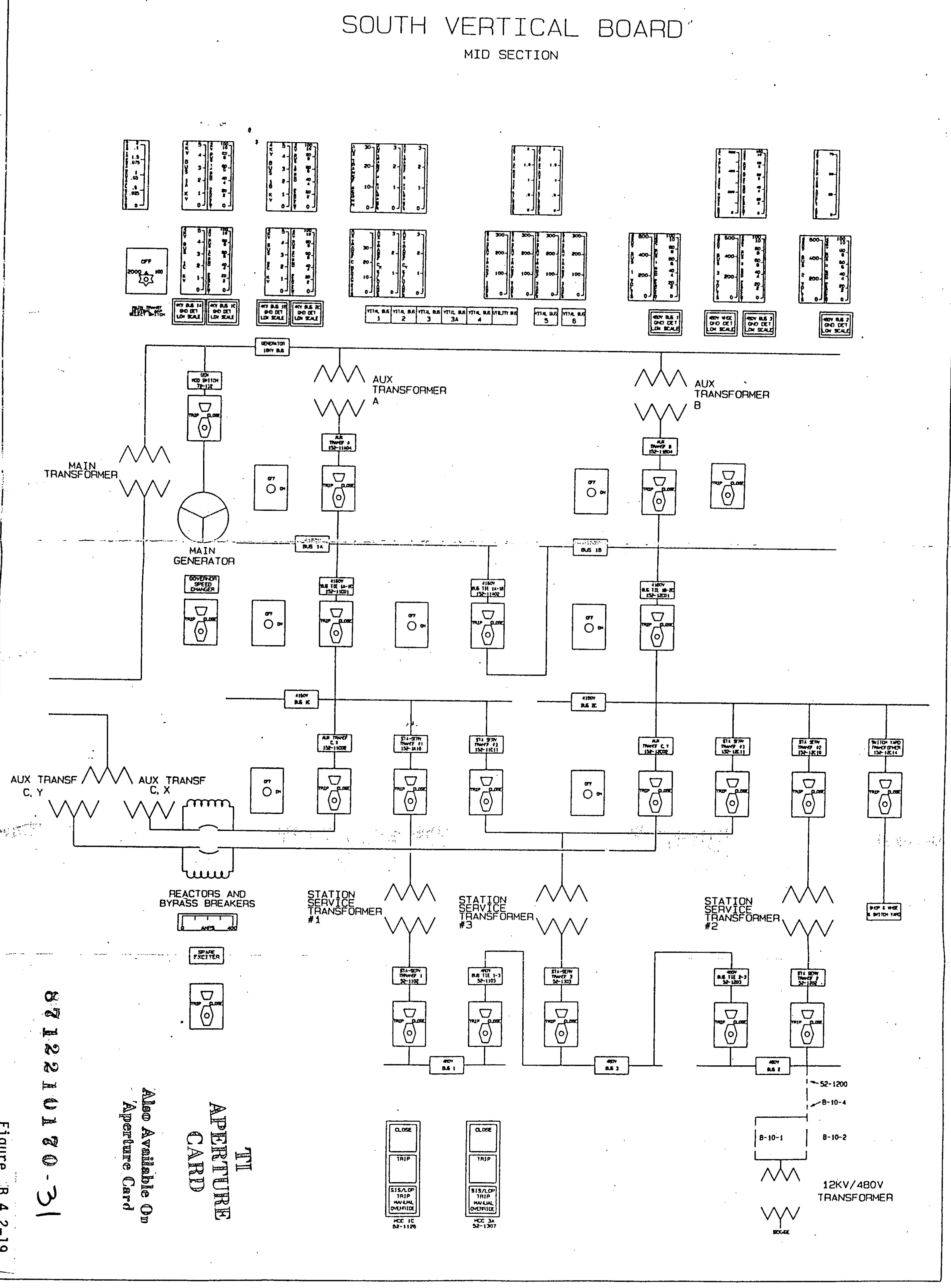

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Figure B.4.2-19

South Vertical Board, East Section, C12, Before