VIRGINIA ELECTRIC AND POWER COMPANY
RICHMOND, VIRGINIA 23261

W. L. STEWART
VICE PRESIDENT
NUCLEAR OPERATIONS

April 15, 1983

Mr. Harold R. Denton
Office of Nuclear Reactor Regulation
Attn: Mr. D. G. Eisenhut, Director
Division of Licensing
U.S. Nuclear Regulatory Commission
Washington, D. C. 20555

Serial No. 006
NO/JBL:ogw
Docket Nos. 50-280
50-281
50-338
50-339
License Nos. DPR-32
DPR-37
NPF-4
NPF-7

Gentlemen:

RESPONSE TO SUPPLEMENT 1 TO NUREG-0737
REQUIREMENTS FOR EMERGENCY RESPONSE
CAPABILITY (GENERIC LETTER NO. 82-33)
NORTH ANNA UNITS 1 AND 2
SURRY UNITS 1 AND 2

VEPCO has recently developed a comprehensive Five-Year Integrated Plan for all capital projects at its nuclear power stations to assist management in more effectively utilizing the fiscal, engineering, construction and human resources of the company. Attachment 1 is a description of the plan and presents the integrated schedule of all capital projects for VEPCO's nuclear power stations during 1983-1987. The integration of the ERC resource requirements with other station projects has, in part, provided the basis for the schedules provided in attachment 2. Periodic updates of the plan will form the basis for schedule revisions of the Emergency Response Capability (ERC) projects. The plan methodology incorporates requirements, priorities and restraints to optimize the scheduling of projects over a five year period. The plan is dynamic and is updated over the life of each project. The impact of new projects that appear during the five year period can be evaluated and appropriate action taken if plan revision is necessitated.

Attachment 2 furnishes the data requested in Generic Letter No. 82-83 as clarified during the NRC regional workshops conducted during January and February 1983. Where the dates in this letter conflict with previous VEPCO commitments to NUREG-0737 items, this letter takes precedence. Attachment 3 provides VEPCO's Integrated Implementation Plan Guidelines for Emergency Response Capability projects. These guidelines will be incorporated into procedures to ensure overall integration and coordination of the ERC projects.

Very truly yours,

W. L. Stewart

W. L. Stewart

A003
s
1/40

ADD:
W. Paulson

Attachments
cc: Mr. Leon B. Engle
Mr. J. Don Neighbors
Mr. J. P. O'Reilly

COMMONWEALTH OF VIRGINIA)
)
CITY OF RICHMOND)

The foregoing document was acknowledged before me, in and for the City and Commonwealth aforesaid, today by W. L. Stewart, who is Vice President-Nuclear Operations, of the Virginia Electric and Power Company. He is duly authorized to execute and file the foregoing document in behalf of that Company, and the statements in the document are true to the best of his knowledge and belief.

Acknowledged before me this 15th day of April, 1983.

My Commission expires: 2-26, 1985.

Ann C. McRae
Notary Public

(SEAL)

M2/004

RESPONSE TO SUPPLEMENT 1 TO NUREG 0737 POST TMI - REQUIREMENTS

VIRGINIA ELECTRIC AND POWER COMPANY

Vepco

8304210250

FIVE-YEAR PLAN
NUCLEAR OPERATIONS DEPARTMENT

ATTACHMENT 1

Vepco

VIRGINIA ELECTRIC AND POWER COMPANY

INTRODUCTION

On February 1, 1983, a special Vepco Committee was established to develop an integrated plan for capital projects for North Anna and Surry. The Integrated Five-Year Plan Committee was chartered to develop a plan which would help to reduce cost, establish logical completion dates for projects, provide a framework for budgeting and provide a framework for managing human resources. The Committee included personnel representing the Nuclear Operations, Power Station Engineering and Power Station Construction Departments.

The initial activity of the Integrated Plan Committee involved basic research into the programs used by similar large corporations for developing long-range planning programs. The Committee explored with several consultants the possibilities of support for development of the Vepco Five-Year Plan for Nuclear Operations. Since the Committee had included in its planning and scheduling model the essential elements necessary for an integrated program, Vepco found that it was not necessary to acquire outside assistance for the project.

As mandated in the charter, Vepco has been developing the methodology that is the foundation of a computer based long-range planning and scheduling model. The model includes the requirements, priorities and restraints that must be addressed before either an active or anticipated project can be entered into the computer model. The formats for presenting the output reports and schedules have been developed based on the the needs of management regarding content and substance of the reports.

PLAN METHODOLOGY

The principles of the Five-Year Plan are symbolized in the diagram shown as Exhibit A. At the core or center of the diagram is the computer based planning and scheduling model. The Project Resource Evaluation and Management Information System (PREMIS) software package was chosen to manage the project data, since it can be used to generate an interacting network for all capital projects for the nuclear plants. In addition, changes can be made to activities associated with each project and the total network is automatically updated to show impact on other projects.

The following input data and output information derived from the computer is based on specific elements that are unique to the Nuclear Operations Department:

Input --

- Requirements
  - License or legal commitments
  - Engineering and construction
  - Surveillance and maintenance
  - Training
  - Future projects
- Priorities
  - Safety
  - Legal
  - ALARA
  - Operations/Maintenance Improvements

- Restraints
  - -- Imposed deadlines
  - -- Corporate goals
  - -- NOD goals
  - -- Budgets (current/forecast)
  - -- Outages (forced/routine)
  - -- Manpower resources

Output --

- Reports
  - -- Management
  - -- Line organizations

- Schedules
  - -- Milestone
  - -- Detailed

PLANNING PROCESS

One of the key elements involved in the planning process was to establish a data base representing the current status of active and anticipated projects. A Data Survey Questionnaire was devised to obtain information from those individuals most knowledgeable about specific project activities to generate information for the model. The questionnaire requested data pertinent to project budget, schedule, commitments, priorities and human resources.

A supplemental guideline titled, Project Priority Classification Matrix (Exhibit B), was used to assign project priorities. The classification matrix was utilized to compare the relative importance of one project with another, such that a systematic meaningful prioritization system could be implemented. The three primary variables employed in the matrix are the Requirement Classification Category, Priority Group and the Cost/Benefit factor. The Requirement Classification Category has been designed to categorize each project according to its importance to safe generation of electrical power at the plants. Three Priority Groups were established to define whether a project could be deferred and what impact a deferral would have on the plant. A limited cost/benefit analysis was performed on each project to ascertain the overall benefit, based on cost, to Vepco of implementing a given project. Each three digit number in the matrix represents the relative priority for accomplishment of the project. The project matrix code numbers were rank-ordered by the computer to establish the hierarchy for activity scheduling of each project. An example of the use of the matrix in the assignment of a code number is the Technical Support Center (TSC) installation project. The TSC is a non-deferrable, well defined NRC requirement to satisfy the Emergency Response Capabilities issue as specified in Supplement 1 to

NUREG-0737. In the opinion of Vepco, the installation of this facility represents a low cost/benefit to the Company in terms of an overall improvement to the plants. Therefore, the TSC project was given the matrix code number of 038 which means that the project is non-deferrable, a legal requirement well defined and has minimal benefit to Vepco.

One of the most important aspects of the planning process is the appraisal of the quality of the data base. A confidence factor was assigned to each project based on the extent to which budget and schedule data was available. For example, project budgets that were based on fixed price contracts or written contractor estimates were assigned high confidence ratings whereas project budgets based only on preliminary data or past experience at another site were assigned low confidence ratings. Therefore, each project has identified with it a good, fair or poor confidence factor which is based on the guideline in Exhibit C.

A Masterfile Input Data Report (MIDR) was formulated to capture a total work list of all capital active and anticipated projects for North Anna and Surry, (Exhibit D). The MIDR includes imposed dates, predecessor activities and resource requirements that are necessary to implement a project. The design and access to the MIDR provides for the maintenance of the data base as a living schedule which can be updated whenever there is a significant change to a project. A regular MIDR updating activity is necessary to provide sufficient schedule definition to gain management confidence to commit to specific dates for projects.

At the nucleus of the computer model used for the Five-Year Plan is the PREMIS program which utilizes a critical path method processor for scheduling projects. Once the known durations and resources, logical relationships to other activities in the project and imposed commitment dates were placed in the main frame computer, PREMIS generated a network for each project. All projects in the MIDR are shown with beginning and ending dates for each project in the Time Scheduling Project Bar Chart (TSPBC) as shown as a sample in Exhibit E. After the first network is generated in TSPBC, changes may be made to any activity parameters and the network would be updated with the project dynamics.

ANALYSIS OF RESULTS

The quality of the assumptions and estimates made in the data base are reflected in the confidence factors in the Five-Year Integrated Schedule, (Exhibit F). Action has been taken to improve the quality of the data base and the procedures for upgrading the individual project estimations throughout the life of the project will further enhance the utility of the plan. The Administrative and Procedures Manual will provide detailed guidance for the submission of data required to support the plan. Items from Supplement 1 to NUREG-0737 have been identified on Exhibit F by a single asterisk (*). Additionally, the Nuclear Operations Department's <u>most</u> significant projects have been identified by double asterisks (**).

PLAN OF ACTION

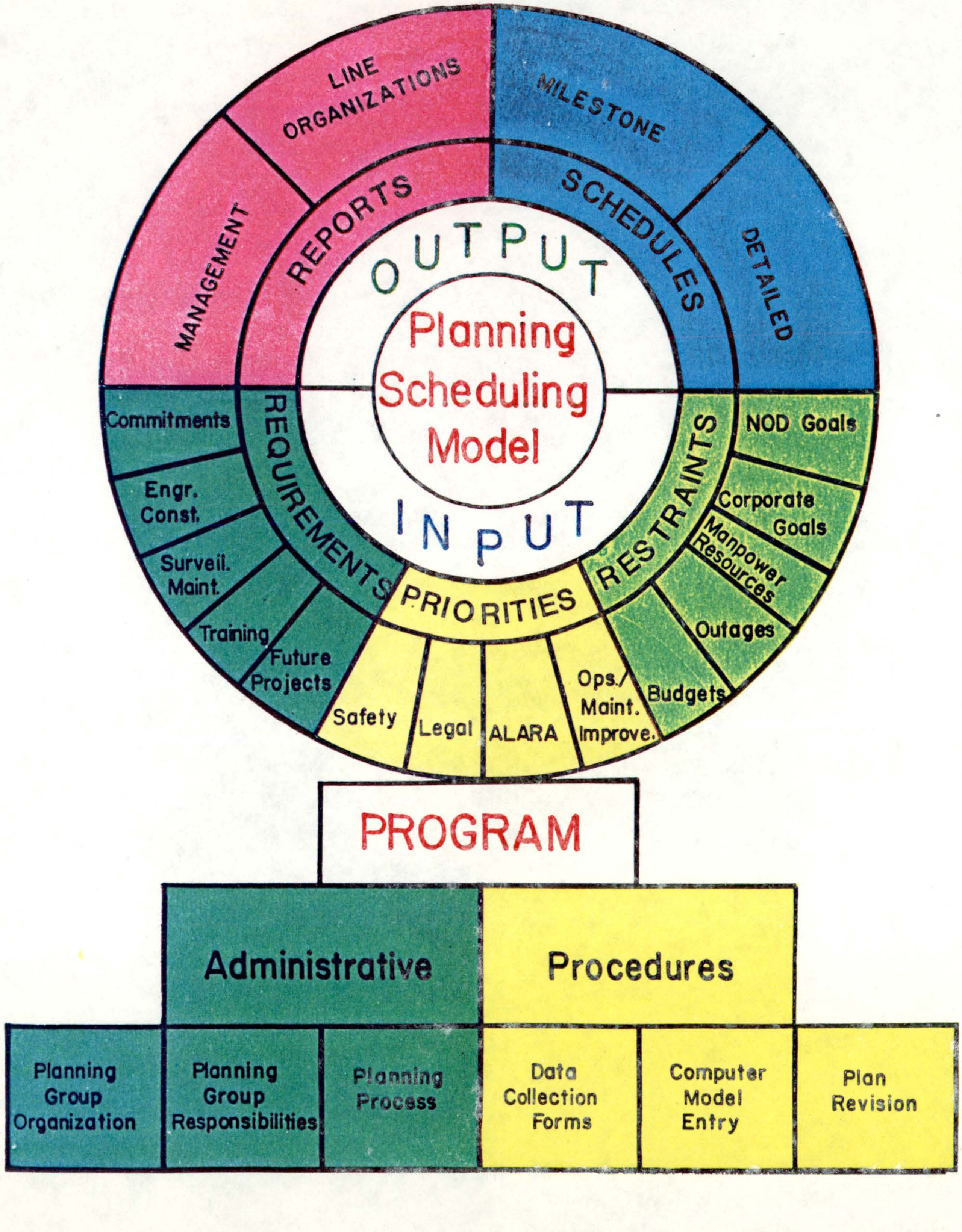
The Integrated Plan Committee was chartered to develop a long-range computer-based planning program for the Nuclear Operations Department (NOD). This has been accomplished and one of the major recommendations of the Committee involved the establishment of a planning group within the department to coordinate planning and scheduling activities to support the maintenance of the dynamic Five-Year Plan.

The goal of the evolutionary planning process (Exhibit G) will be to attain at least a ±15% margin in the estimates for engineering and construction by the completion of the Final Design stage of most of the projects. In an effort to achieve the ±15% estimate goal on present projects by January 1, 1984, interface procedures will be established between the stations, engineering/construction, and the NOD Planning Group.

The procedures will ensure that the Five-Year Plan computer model is updated during and after each development phase of a project. The uncertainties surrounding the nuclear power industry dictate that the planning and scheduling model be designed such that changes to requirements, priorities and restraints can be reflected in the model at any time. Therefore, the planning and scheduling model will be maintained and controlled in a dynamic (living) manner.

SPECPR/ogw/ds5

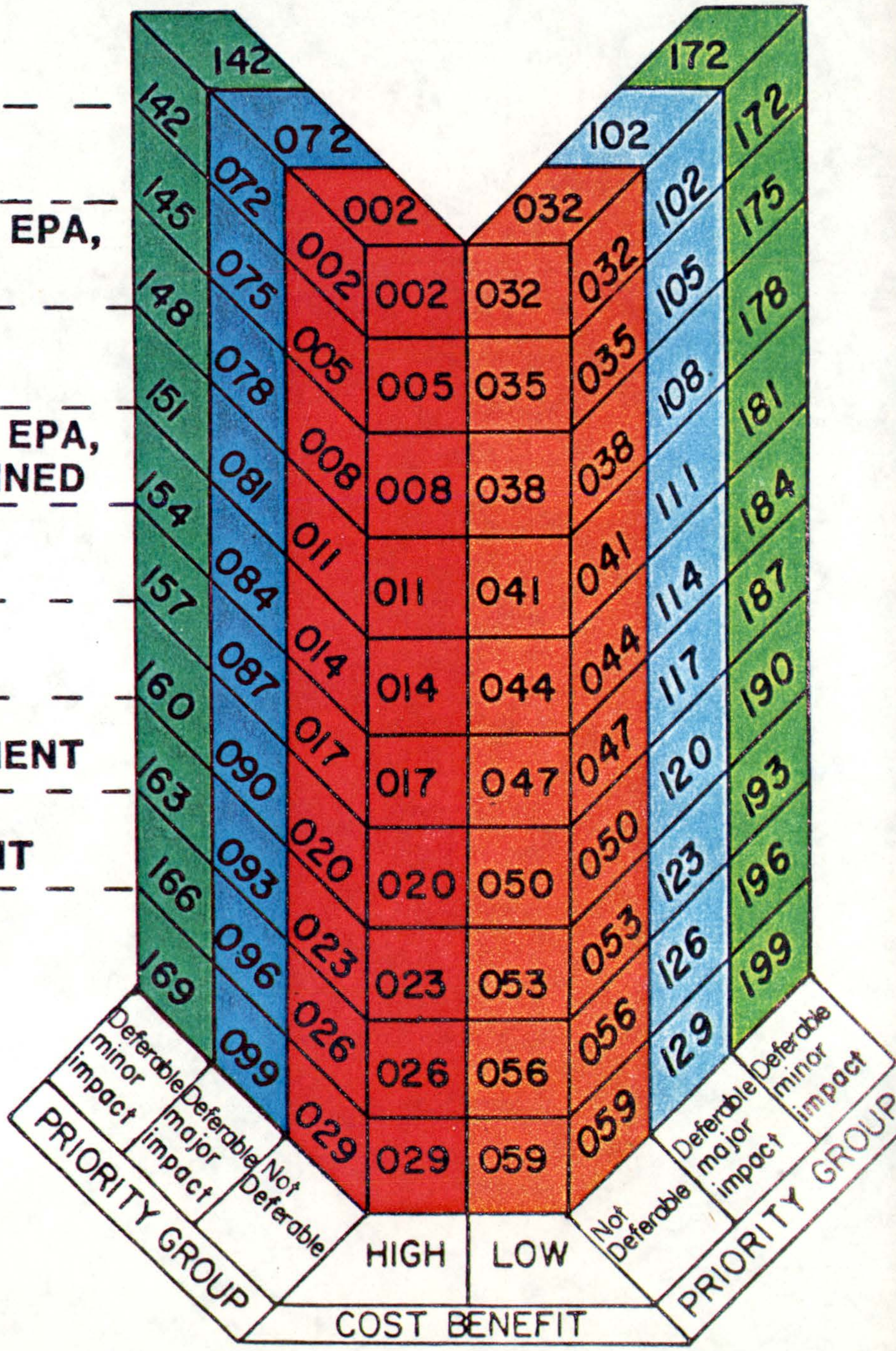
DYNAMIC 5-YEAR PLANNING SCHEDULING MODEL



PROJECT PRIORITY CLASSIFICATION MATRIX

REQUIREMENT CLASSIFICATION CATEGORY

1. **MAINTAIN SAFE ELECTRICAL PRODUCTION**
2. **LICENSE COMMITMENT**
3. **LEGAL REQUIREMENTS (NRC, EPA, OSHA, NFPA) WELL DEFINED**
4. **ALARA**
5. **LEGAL REQUIREMENTS (NRC, EPA, OSHA, NFPA) NOT WELL DEFINED**
6. **PLANT MAINTENANCE**
7. **REQUIRED SURVEILLANCE**
8. **PLANT EFFICIENCY IMPROVEMENT**
9. **PLANT SUPPORT IMPROVEMENT**



032283

CONFIDENCE FACTOR GUIDELINES

C.F.	INPUT BASED ON
1.	ACTIVE PROJECT-MAJORITY OF ACTIVITIES ARE BEING PERFORMED ON A FIXED PRICE BASIS.
2.	ACTIVE PROJECT-ACTIVITIES ARE BEING PERFORMED BY OUTSIDE ORGANIZATIONS UNDER TIME AND EXPENSE CONTRACTS OR BY VEPCO PERSONNEL BASED ON DEFINED SCOPES OF WORK.
3.	ACTIVE OR PROJECTED PROJECT-ESTIMATES BASED ON SAME WORK PERFORMED ON OTHER UNIT AT SAME STATION.
4.	ACTIVE OR PROJECTED PROJECT-ESTIMATES BASED ON SAME WORK PERFORMED AT OTHER STATION.
5.	ACTIVE OR PROJECTED PROJECT-ESTIMATES BASED ON ITEMIZED MATERIAL AND MANPOWER PROJECTS FOR FIRST OF A KIND WORK.
6.	ACTIVE OR PROJECTED PROJECT-ESTIMATES BASED ON SAME WORK PERFORMED AT ANOTHER UTILITY OR ON SIMILAR WORK IF VEPCO.
7.	ACTIVE OR PROJECTED PROJECT-PRELIMINARY ESTIMATE.

ATTACHMENT I
Exhibit D

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VP   5Y                                                                 83/04/13          15.06.56    905
SITE : FI\  .EAR FORECAST                   MASTERFILE INPUT DATA REPORT                        PAGE        13
                                            ----------------------------
UNIT : OUTAGE SCHEDULE                               REPORT  MP09                    RUN DATE          :  13APR83
                                                     ------------                    TIME NOW DATE     : S01JAN83
SEL ACT(2) EQU 'R' PAGE ON AP                                                        PROJECT START DATE :  01JAN83
SORT ON AP                                                                           PROJECT END DATE  :  30DEC89
--------------------------------------------------------------------------------------------------------------------------
 ACTIVITY                                  IMPOSED DATES           C   *****PREDECESSORS******  *******RESOURCES*************
  IDENT.    CODES    DURATION  CAL TAC TYPE    DATE   ACTUAL DATES C   ACT. IDENT. LAG CAL TYP  LAG  DUR CODE AMOUNT TY  RAC  PR.
--------------------------------------------------------------------------------------------------------------------------
                     REM.   15                                          SR6084C1     90      S
            3 TYP3   80-49D  TSC STRUCTURAL INSTALLATION

SR6084C5    1 PSC    ORIG.                                              SR6084E5    300      F                                   35
                     REM.                                               SR6084C1     90      S
            3 TYP3   80-49E  TSC ELECTRICAL INSTALLATION

SR6084E1    1 PSE    ORIG. 180       B                                                                                           35
                     REM.  180
            3 TYP3   80-49A 1&2 TECH SUPPORT CENTER FAC REM

SR6084E2    1 PSE    ORIG. 180       B                                                          130  VPE    542   W              35
                     REM.  180                                                                  130  VPD    136   W
            3 TYP3   80-49B  TSC RADIATION MONITOR                                          43   87  NVE   1364   W
                                                                                            43   87  NVD    682   W
SR6084E3    1 PSE    ORIG. 180       B                                                          130  VPE    542   W              35
                     REM.  180                                                                  130  VPD    136   W
            3 TYP3   80-49C  TSC FACILITIES INSTALLATION                                    43   87  NVE   1364   W
                                                                                            43   87  NVD    682   W
SR6084E4    1 PSE    ORIG. 180       B                                                          130  VPE    542   W              35
                     REM.  180                                                                  130  VPD    136   W
            3 TYP3   80-49D  TSC STRUCTURAL INSTALLATION                                    43   87  NVE   1364   W
                                                                                            43   87  NVD    682   W
SR6084E5    1 PSE    ORIG. 180       B                                                          130  VPE    542   W              35
                     REM.  180                                                                  130  VPD    136   W
            3 TYP3   80-49E  TSC ELECTRICAL INSTALLATION                                    43   87  NVE   1364   W
                                                                                            43   87  NVD    682   W
SR6084E6    1 PSE    ORIG. 243       E                                  SR6084E2             N      173  VPE   2608   W              35
                     REM.  243                                          SR6084E3             N      173  VPD    652   W
            3 TYP3           TSC ENGINEERING SUPPORT                    SR6084E4             N      173  NVE   3562   W
                                                                        SR6084E5             N      173  NVD   1781   W
                                                                                               194   43  MTD   1580   W
SR6109C1    1 PSC    ORIG.  84       E    FNL 15JUL83                   SR6109E1             N                                   35
                     REM.   84
            3 TYP3   80-96 REACT. VES. LEVEL INDIC. #2

SR6109E1    1 PSE    ORIG. 122       B                                                           86  VPE    192   W              35
                     REM.  122                                                                   86  NVE   4774   W
            3 TYP3   80-96 REACT. VES. LEVEL INDIC. #2                                           86  VPD     48   W
                                                                                                 86  NVD   2387   W
SR6116C1    1 PSC    ORIG.           E    FNL 16APR83                   SR6116E1             N                                   35
                     REM.
            3 TYP3   80-29 REACT. HD & PRZ VENT #1

SR6116E1    1 PSE    ORIG. 122       B                                                           86  VPE    192   W              35
                     REM.  122                                                                   86  NVE   1056   W
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ATTACHMENT I
Exhibit E

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VP   5Y                                                                                83/04/13          15.06.56   563
                                                                                                                    141

-----------1981                1982                                               1983
ACTIVITY    AUG SEP  OCT NOV DEC  JAN FEB MAR  APR MAY JUN  JUL AUG SEP  OCT NOV DEC  JAN FEB MAR  APR MAY JUN  JUL AUG  SEP OCT NOV
  IDENT.   20112001230122011200123012201120112301220112001230122011200122012201120012201120012001230122011200122012201123012201120
-----------952962963074184185296306307307430741741852962963074184185185296307307418529529629632963063074185185296307307417418529 62
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SR6084C4                                                                        *
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SR6084C5                                                                        *
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                                                                                *
                                                                                *
SR6084E1                                                                        XXXXXXXXXXXXXXXXXXXXXXXXXXX------------------
                                                                                *
                                                                                *
SR6084E2                                                                        XXXXXXXXXXXXXXXXXXXXXXXXXXX------------------
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SR6084E3                                                                        XXXXXXXXXXXXXXXXXXXXXXXXXXX------------------
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SR6084E4                                                                        XXXXXXXXXXXXXXXXXXXXXXXXXXX------------------
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SR6084E5                                                                        XXXXXXXXXXXXXXXXXXXXXXXXXXX------------------
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SR6084E6                                                                        *                         XXXXXXXXXXXXXXXXXXX
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ATTACHMENT
Exhibit F

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VP    5Y                                                  83/04/13       15.06.56    626
                                                                                     206

-----------1983   1984                                      1985                                                1986
ACTIVITY   NOV DEC JAN FEB  MAR APR MAY  JUN JUL AUG  SEP OCT  NOV DEC JAN  FEB MAR APR MAY  JUN JUL  AUG SEP OCT  NOV DEC JAN  FEB
  IDENT.   012301220112001220122011200123012201120012201120112301220112001230122012201120012201120112301220112001230122011200122 01
-----------96307418418518529741841852963063074185185295296307417418529629630630763073074185295296307417418418529630630741851852952

SR6084C4            XXX-----------------------------------------------------------------------------------------------------

SR6084C5              ------------------------------------------------------------------------------------------------------

SR6084E1   ---------------------------------------------------------------------------------------------------------------------

SR6084E2   ---------------------------------------------------------------------------------------------------------------------

SR6084E3   ---------------------------------------------------------------------------------------------------------------------

SR6084E4   ---------------------------------------------------------------------------------------------------------------------

SR6084E5   ---------------------------------------------------------------------------------------------------------------------

SR6084E6   XXXXXXXXXXXXXXXX-----------------------------------------------------------------------------------------------------

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ATTACHMENT I
Exhibit E

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VP    5Y                                                  83/04/13          15.06.56    689
                                                                                        271

-----------1986                            1987                                           1988
ACTIVITY     MAR APR  MAY JUN JUL  AUG SEP OCT  NOV DEC  JAN FEB MAR APR  MAY JUN JUL  AUG SEP  OCT NOV DEC  JAN FEB MAR  APR MAY
  IDENT.   12011200123012201120012301220112001220112011230122011201120012201220112001220112001230122011200123012201120012301220112
-----------96529629630741841852963063073074185295296307417418418541851852963073074185295296296307418418529630630730742963063074185

SR6084C4   ----L-----------------------------------------------------------------------------------------------------------------

SR6084C5   --------------------------------------------------------------------------------------------------------------------------

SR6084E1   --------------------------------------------------------------------------------------------------------------------------

SR6084E2   -------------------------------------------------------------------------------------------------------------L

SR6084E3   -------------------------------------------------------------------------------------------------------------L

SR6084E4   -------------------------------------------------------------------------------------------------------------L

SR6084E5   -------------------------------------------------------------------------------------------------------------L

SR6084E6   --------------------------------------------------------------------------------------------------------------------------

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ATTACHMENT I
EXHIBIT F

*ERC Items
**Vepco Most Significant Issues

FIVE-YEAR INTEGRATED SCHEDULE

Nuclear Operations Department

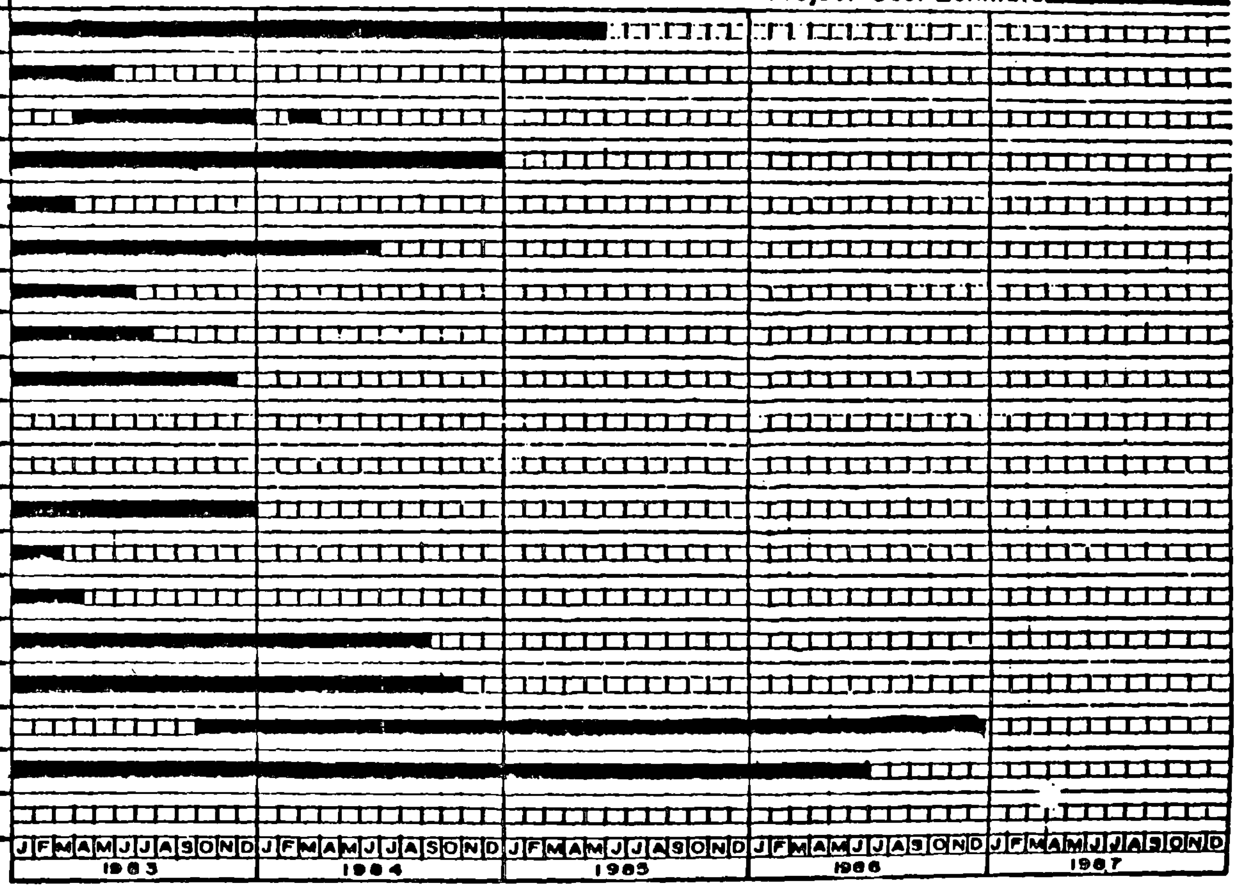
Power Station NORTH ANNA

Unit 1 and 2

Date 4-15-83

Project Cost Estimate ≥ $100,000

	m	Project Description	[illegible]	[illegible]	[illegible]	[illegible]	[illegible]
**	4865	SERVICE WATER UPGRADE & CONNECTION	PSEC	1	1	H	P
	7043	MOD ISO PHASE BUS DUCT	PSEC	1	1	H	G
**		GDC-17 REANALYSIS	PSE	1	1	H	P
	4281	LONG RANGE EFFLU MONITOR	PSC	1	3	H	P
**	4994	SPENT FUEL HANDLING	PSC	1	4	H	P
	7027	ADD TURB BLDG. SECURITY LIGHTING	PSEC	1	5	H	P
	4280	P A LONG TERM CONT SAMPLE SYSTEM	PSEC	1	2	L	P
	4283	POST ACCIDENT SHIELDING TMI	PSE	1	2	L	G
	6141	SECONDARY PROTECT ELECT PENETRATION	PSEC	1	2	L	G
	4876	MECH SAFETY EQUIPMENT QUAL STUDY	PSEC	1	3	L	P
	4877	SEISMIC & DYNAMIC QUAL SAFETY EQUIPMENT	PSEC	1	3	L	P
	4878	NUREG 0612 HEAVY LOADS	PSEC	1	3	L	G
	6106	RVLIS DWG UPDATE UNIT 1	PSE	1	3	L	G
	6107	RVLIS DWG UPDATE UNIT 2	PSE	1	3	L	G
	6153	EQUIP QUAL UNIT 1	PSEC	1	3	L	G
	6154	EQUIP QUAL UNIT 2	PSEC	1	3	L	G
*	6157	INSTALL TSC (0696)	PSEC	1	3	L	G
*	6157	SPDS INSTALL	PSEC	1	3	L	G



LEGEND:

Priority Group -
1-Not Deferable
2-Deferable (major impact)
3-Deferable (minor impact)

Requirement Classification Category -
1- Maintain Safe Electrical Production
2- License Commitment
3- Legal Requirement (well defined)
4- ALARA
5- Legal Requirement (not well defined)
6- Plant Maintenance
7- Required Surveillance
8- Plant Efficiency Improvement
9- Plant Support Improvement

Cost/Benefit -
H - High Benefit
L - Low Benefit

Confidence Factor -
G-Good F-Fair P-Poor

ATTACHMENT I
EXHIBIT F

FIVE-YEAR INTEGRATED SCHEDULE

Nuclear Operations Department

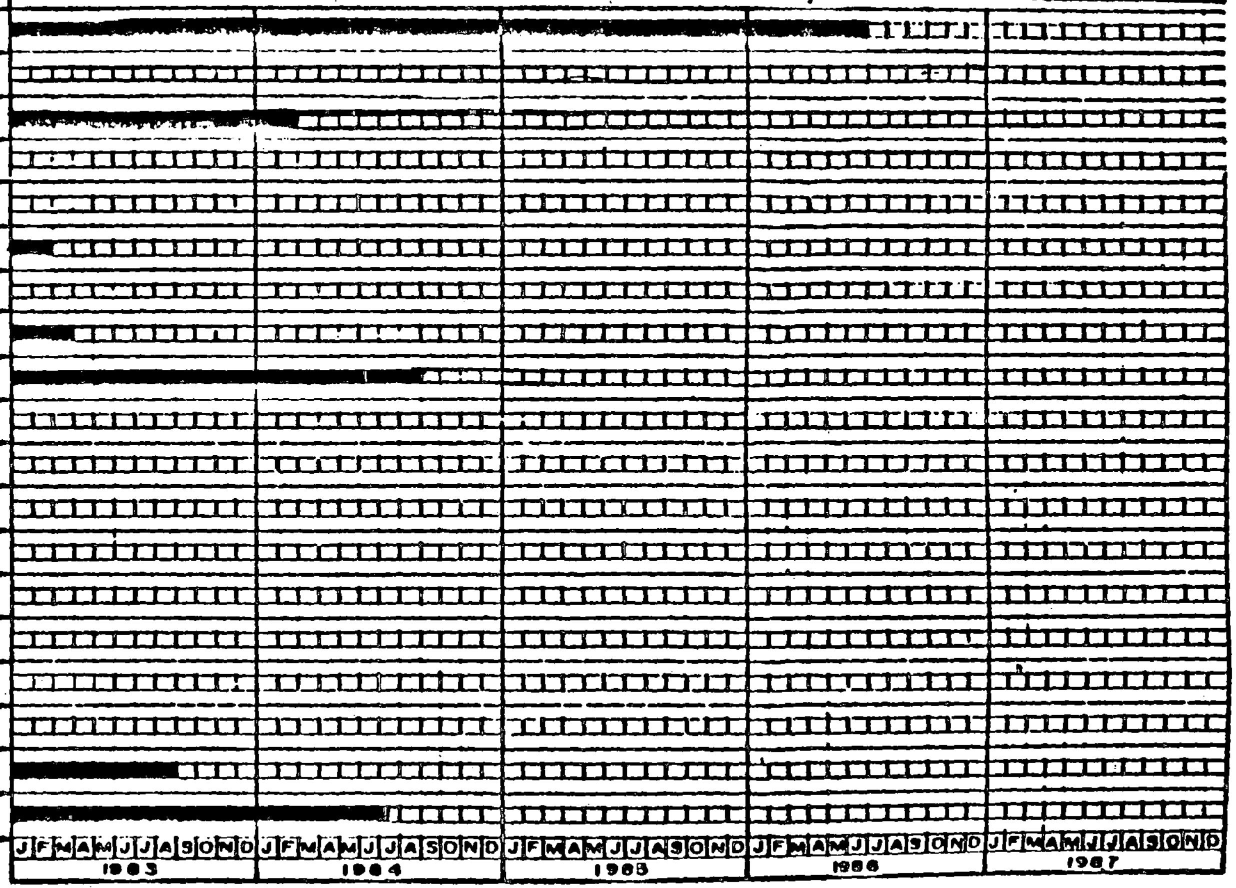
Power Station NORTH ANNA

Unit 1 and 2

Date 4-15-83

Project Cost Estimate ≥ $100,000

	[illegible]	Project Description	[illegible]	[illegible]	[illegible]	[illegible]	[illegible]
*	·6082	TSC FACILITIES	PSEC	1	3	L	G
*	·4827	INSTALL LOCAL EOP	PSEC	1	5	L	P
*	·4874	CONT RM UPGRADE NUREG 0700	PSEC	1	5	L	P
	4237	RECIRC SPRAY SUMP PUMP	PSEC	1	6	L	P
**	T112	S/G LOOSE PARTS MONITOR	SITE	2	1	H	P
	4883	SC FEEDPUMP WARMUP BYPASS	SITE	2	1	H	P
	4913	RAD MEASURING EQUIPMENT	SITE	2	1	H	P
	4999	RMV PRZR SRG LINE THERMAL SLV	PSEC	2	1	H	G
	6165	NEUTRON ABSORBENT FUEL RAC	PSEC	2	1	H	C
	7041	SPARE ELEC GENERATOR	SITE	2	1	H	P
	N044	REPLACE FUEL XFER DRIVE SYSTEM IN UNIT 2	PSEC	2	6	H	P
	N045	REPLACE FUEL XFER DRIVE SYSTEM	SITE	2	4	H	P
	4987	TURBINE BLDG ROOF ACC & STNDPIP	PSEC	2	5	H	P
	4856	REBUILD U1 LP TURB RTR	SITE	2	6	H	P
	4901	MOD MAIN XFMR OIL PIT	PSEC	2	6	H	P
	4904	CONT PERSONNEL AIR SUPPLY	PSEC	2	6	H	P
	4985	PRESS PORV N2 UPGRADE	SITE	2	6	H	P
**	4884	SC BLOWDOWN RECOVERY SYS	PSEC	2	8	H	P
	4995	PRESSURE UPRATING	PSEC	2	8	H	G



LEGEND:

Priority Group-
1-Not Deferable
2-Deferable (major impact)
3-Deferable (minor impact)

Requirement Classification Category-
1- Maintain Safe Electrical Production
2- License Commitment
3- Legal Requirement (well defined)
5-Legal Requirement (not well defined)
6-Plant Maintenance
7-Required Surveillance

Cost/Benefit-
H-High Benefit
L-Low Benefit

ATTACHMENT I
EXHIBIT F

FIVE-YEAR INTEGRATED SCHEDULE
Nuclear Operations Department

Power Station NORTH ANNA

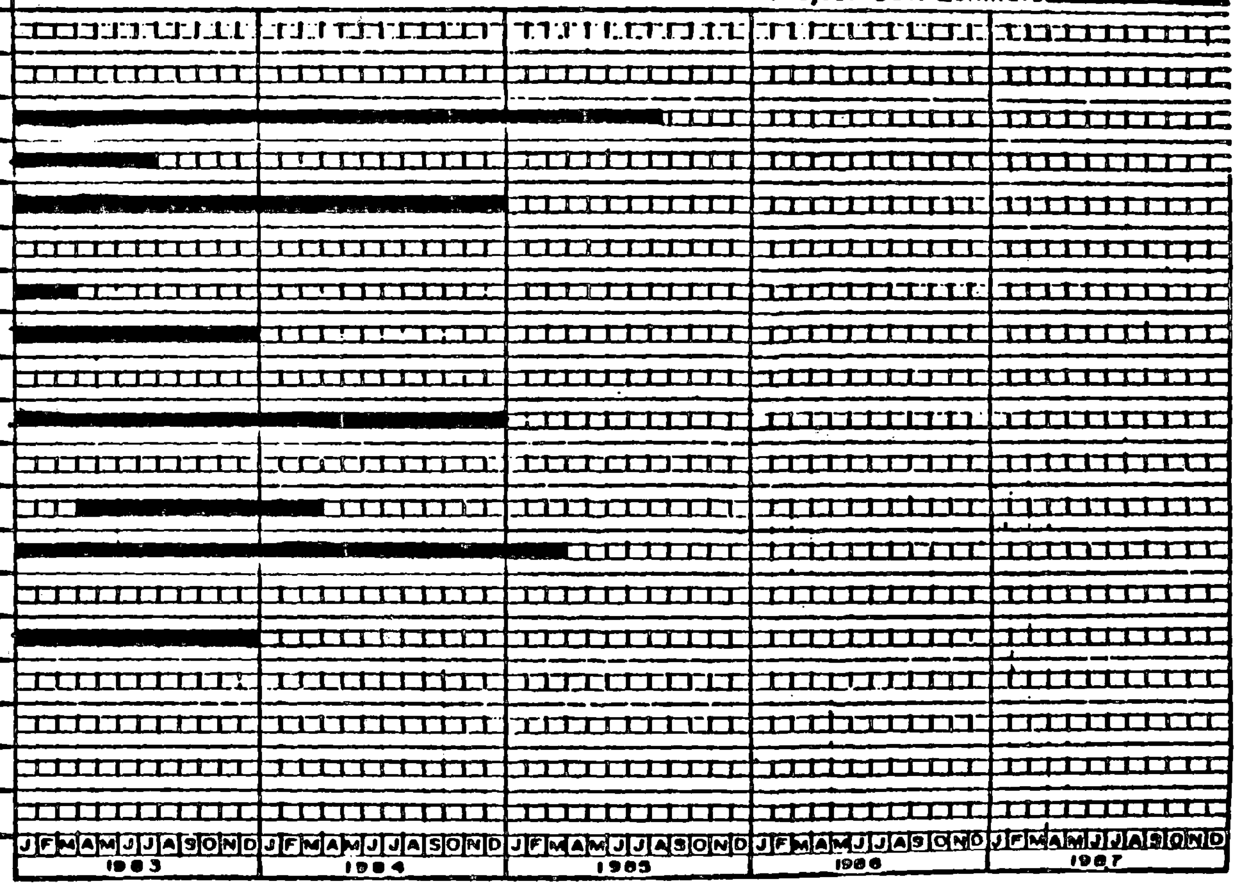
Unit 1 and 2

Page 3 of 9

Date 4-15-83

Project Cost Estimate ≥ $100,000

	[illegible]	Project Description	[illegible]	[illegible]	[illegible]	[illegible]	[illegible]
	7011	INSTALL MSR BYPASS LINE	PSEC	2	8	H	P
	7022	AUX BOILER CONT SYS MOD	SITE	2	8	H	P
	7025	CORE UPGRADE MOD	PSEC	2	8	H	P
**	7029	DEEP BED COND POLISHING	PSEC	2	8	H	P
	4121	RAD DEMIN WASTE SYS	PSEC	2	9	H	P
	4907	AUX FD PUMP OIL SUMP LVL ALARM	SITE	2	9	H	P
	6133	LOW LVL WASTE HANDLING FACILITY	PSEC	2	9	H	P
	7020	NA3 SITE IMPROVEMENT	PSEC	2	9	H	P
	4909	REPLACE 1-MOV-RS-155 A&B	PSEC	2	1	L	P
	6083	NORMAL TO EMERGENCY BUS TIE UNIT 2	PSEC	2	1	L	P
	N006	INSTALL BRANCH FLOW ORIFICE	PSEC	2	2	L	P
	4892	FUEL BUILDING BLOCK WALL	PSEC	2	3	L	P
**	6186	GDC - 17 MODS	PSEC	2	3	L	P
	T805	REDUCE SOLID RADWASTE VOLUME	SITE	2	4	L	P
*	4870	REG GUIDE 1.97	PSEC	2	5	L	P
	4871	MS DISCHARGE STUDY	PSEC	2	5	L	P
	4872	MS DISCHARGE MONITOR STUDY	PSEC	2	5	L	P



LEGEND:

Priority Group-
1-Not Deferable
2-Deferable (major impact)
3-Deferable (minor impact)

Requirement Classification Category-
1- Maintain Safe Electrical Production
2- License Commitment
5-Legal Requirement (not well defined)
6-Plant Maintenance

Cost/Benefit-
H-High Benefit
L-Low Benefit

ATTACHMENT I
EXHIBIT F

Power Station NORTH ANNA

Unit 1 and 2

FIVE-YEAR INTEGRATED SCHEDULE

Nuclear Operations Department

Date 4-15-83

Project Cost Estimate ≥ $100,000

	m	Project Description	[illegible]	[illegible]	[illegible]	[illegible]	[illegible]
	4880	FW RECIRC LINE LEAK MOD	PSEC	2	6	L	P
	7001	GUIDE TUBE DISPOSAL	PSE	2	6	L	P
	4426	REDUNDANT STATION AIR COM	PSE	2	9	L	P
**	4826	BORIC ACID CONCENT STUDY	SITE	2	9	L	P
	4885	PG WATER STORAGE TANK	SITE PSC	2	9	L	P
	4832	INSTALL FUEL XFER BLIND FLANGE UNIT 1	PSEC	3	4	H	P
	4835	INSTALL FUEL XFER BLIND FLANGE UNIT 2	PSEC	3	4	H	P
	4914	TRAINING LAB EQUIPMENT	SITE	3	9	H	P
	7014	RESPIRATOR DECON REPAIR FACILITY	SITE	3	1	L	P
	4886	FP HEAT/SMOKE DETECTORS	PSEC	3	3	L	P
		CHARLESTON EARTHQUAKE STDY	NOD	3	5	L	P
	7012	SEAL IMPROVEMENT PACKAGE	SITE	3	8	L	P
**	4895	UPGRADE LAUNDRY FACILITY	PSEC	3	9	L	P
	4903	FLASHDEMIN REGEN SYS UPGRADE	PSEC	3	9	L	P
**	4905	REFURBISH OLD MACHINE SHOP	SITE	3	9	L	P
	4910	INSTALL SANGAMO DIST ANALYZER	PSEC	3	9	L	P
	4911	INSTALL DIESEL GEN RM DOOR	PSEC	3	9	L	P

J F M A M J J A S O N D — 1983 | J F M A M J J A S O N D — 1984 | J F M A M J J A S O N D — 1985 | J F M A M J J A S O N D — 1986 | J F M A M J J A S O N D — 1987

LEGEND:

Priority Group-
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2-Deferable (major impact)
3-Deferable (minor impact)

Requirement Classification Category-
1- Maintain Safe Electrical Production
2- License Commitment
3- Legal Requirement (well defined)
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6-Plant Maintenance
7-Required Surveillance
8-Plant Efficiency Improvement

Cost/Benefit-
H-High Benefit
L-Low Benefit

Confidence Factor-

ATTACHMENT I
EXHIBIT F

Power Station NORTH ANNA

Unit 1 AND 2

FIVE-YEAR INTEGRATED SCHEDULE

Nuclear Operations Department

Date 4-15-83

Project Cost Estimate ≤ $100,000

IR	Project Description	[illegible]	[illegible]	[illegible]	[illegible]	[illegible]
7031	ASDP WIDE RANGE TEMP IND	PSEC	1	3	H	P
4986	VCT LEVEL CONTROL CIRCUIT MOD	SITE	1	6	H	P
4916	SEC COMPUTER LOSS OF EP	PSEC	1	9	H	P
A008	EXPANSION JOINT LEAK DETECT	SITE	1	3	L	P
4842	RELIABLE DG PRELUBE PUMP PWR UNIT 2	PSEC	1	3	L	G
6114	REACTOR HEAD & VENT SYSTEM	PSEC	1	3	L	P
6115	VENT SYSTEM UNIT 2	PSEC	1	3	L	P
7004	APPENDIX R MODIFICATIONS	PSC	1	3	L	P
4827N	NRC AUDIT	SITE	1	5	L	P
4875	PZR PORV & LOOP SEAL DRAIN UNIT 2	PSEC	1	5	L	P
6118	INSTALL EARLY WARNING SYSTEM	PSE	1	5	L	P
A002	EMERG DIESEL GEN CHANGE	SITE	2	1	H	P
A003	(CVCS)NI TO PT RTD CHANGE	SITE	2	1	H	G
A004	(RC) NI TO PT RTD CHANGE	SITE	2	1	H	G

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1983 1984 1985 1986 1987

LEGEND:

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7- Required Surveillance
8- Plant Efficiency Improvement
9- Plant Support [illegible]

Cost/Benefit-
H-High Benefit
L-Low Benefit

Confidence Factor-

ATTACHMENT I
EXHIBIT F

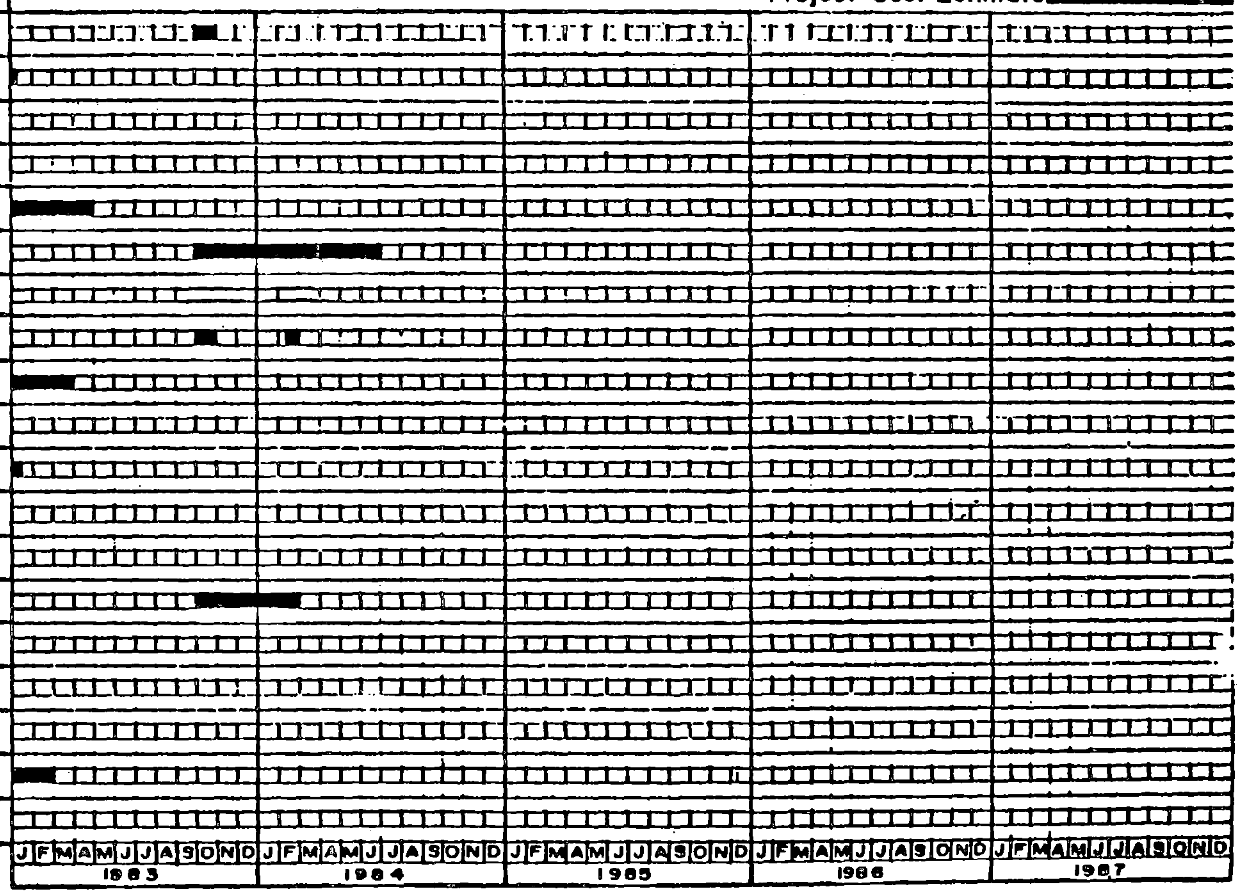
Power Station NORTH ANNA
Unit 1 AND 2

FIVE-YEAR INTEGRATED SCHEDULE

Nuclear Operations Department

Date 4-15-83
Project Cost Estimate < $100,000

IR	Project Description	[illegible]	[illegible]	[illegible]	[illegible]	[illegible]
A005	(RHR) NI TO PT RTD CHANGE	SITE	2	1	H	G
A015	ADD AUX OIL PUMP THROTTLE VALVE	SITE	2	1	H	G
A022	ADD P-13 CKT LEAD/LAG CARD	SITE	2	1	H	P
A024	H2 RECOMBINER COVER STANDOFFS	SITE	2	1	H	P
A026	CHARGE PUMP RECIRC VLV CIRCUIT	SITE	2	1	H	P
A027	PRT N2 PRESS REQ SENSE LINE	SITE	2	1	H	P
A030	GAS STRIPPER MOISTURE SEPARATOR	PSEC	2	1	H	P
4882	NITROGEN SUPPLY TO PORV MOD	PSC SITE	2	1	H	P
4898	RTD BYPASS FLOW INDICATION	PSEC	2	1	H	P
7013	REPLACE STATION BATTERIES	SITE	2	1	H	P
7030	ASDP REDUNDANT PZR LVL INDICATOR	PSC	2	1	H	P
N045	REPLACE FUEL XFER DRIVE SYSTEM	SITE	2	1	H	P
7016	INSERVICE INSPECTION PROG	PSC	2	5	L	P
A021	MODIFY FT-CC-115 A, B, C PIPE	SITE	2	6	H	P
7028	RELOCATE CORE COOLING MONITOR	PSEC	2	6	H	P
N007	INSTALL PORT IODINE FILTERS	SITE	2	8	H	P
7023	DIONEX ION CHROM	SITE	2	8	H	P
7024	RWST CHEM ADD TANK PROBES	SITE	2	8	H	P



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Cost/Benefit-
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L - Low Benefit

Confidence Factor-

ATTACHMENT I
EXHIBIT F

Power Station NORTH ANNA
Unit 1 AND 2

FIVE-YEAR INTEGRATED SCHEDULE
Nuclear Operations Department

Date 4-15-83
Project Cost Estimate < $100,000

ID	Project Description	[illegible]	[illegible]	[illegible]	[illegible]	[illegible]
7026	REPLACE GEN H2 PURITY METER	SITE	2	8	H	P
A029	CONT SUMP PUMP CHANGE	PSEC	2	9	H	P
4993	HP & OUTAGE IMPROVEMENTS	PSEC	2	9	H	P
7010	CONT BOLT TENS STRESS MIC	SITE	2	1	L	P
4284	P A CONTAINMENT MONITOR	PSC	2	3	L	P
4331	I.E. BULLETIN 79-14	PSEC	2	3	L	P
4896	CONTROL RM A/C MOD	SITE	2	8	L	P
5600	NON REGN HT XCHNGER DRAIN SEP	PSEC	2	8	L	P
T055	CONTAINMENT WORK TEMP POWER	PSEC	2	9	L	P
4426	REDUNDANT STATION AIR COMPRESS	PSEC	2	9	L	P
4830	ADD SECURITY TURNSTILE DWG UPDATE	PSE	2	9	1	P
4887	ANNUNCIATOR MODS	PSEC	2	9	L	P
A002	EMERG DIESEL GEN CHANGE	SITE	3	1	H	G
A006	RAD MON PUMP 152 & 259 MOD	SITE	3	1	H	P
A011	TURB DECK SPRKLR LVL ALARM	SITE	3	1	H	F
A013	PRC CABINET SETUP/ROD SP	SITE	3	1	H	G
A014	34.5 KV BRKR RLY MULTI PWRE	SITE	3	1	H	G

J F M A M J J A S O N D 1983 | J F M A M J J A S O N D 1984 | J F M A M J J A S O N D 1985 | J F M A M J J A S O N D 1986 | J F M A M J J A S O N D 1987

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Cost/Benefit-
H - High Benefit
L - Low Benefit

Confidence Factor-

ATTACHMENT I
EXHIBIT F

Power Station NORTH ANNA
Unit 1 AND 2

FIVE-YEAR INTEGRATED SCHEDULE
Nuclear Operations Department

Page 8 of 9
Date 4-15-83
Project Cost Estimate < $100,000

IR	Project Description	[illegible]	[illegible]	[illegible]	[illegible]	[illegible]
A016	AUX OIL PUMP PRESS SW & IND	SITE	3	1	H	P
A009	ABV GRD OIL STO TANK FOAM INJ	SITE	3	5	H	P
A012	RM-P-a59 A & B ELEC SEPARATION	SITE	3	6	H	P
A017	LO TC ANNUN PROTECTION REQ'D	SITE	3	6	H	P
A018	FIFTH PT DRAIN COOL VLV REPL	SITE	3	6	H	P
A019	TURB TRIP TO LP HTR DRAIN PUMP	SITE	3	6	H	P
7017	ROTATING EQUIPMENT ANALYSIS	PSEC	3	6	H	P
A007	ADD CO ANALYZER BOTTLED AIR	SITE	3	8	H	P
A025	INSTALL TI & VLV TO SS OF BR	SITE	3	8	H	P
A001	HI RANGE PROT RM CH 129 & 229	SITE	3	9	H	P
A020	INSTALL RCS PRESS RECORDER	SITE	3	9	H	P
N015	INSTALL PERM EP AT TURBINE	SITE	3	9	H	P
N017	INSTALL TEMP ON H & J MCCS UNIT 1	SITE	3	9	H	P
N018	INSTALL TEMP ON H & J MCCS UNIT 2	SITE	3	9	H	P
4882	INSTALL LIQUID NITROGEN SYSTEM	SITE	3	9	H	P
4894	MULTI FREQ & EDDY CURRENT EQPT	SITE	3	4	L	P
4906	VENT FOR CHEM ADD TANKS	PSEC	3	6	L	P

J F M A M J J A S O N D — 1983 | J F M A M J J A S O N D — 1984 | J F M A M J J A S O N D — 1985 | J F M A M J J A S O N D — 1986 | J F M A M J J A S O N D — 1987

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L-Low Benefit

Confidence Factor-

ATTACHMENT I
EXHIBIT F

Power Station NORTH ANNA

Unit 1 AND 2

FIVE-YEAR INTEGRATED SCHEDULE

Nuclear Operations Department

Page 9 of 9

Date 4-15-83

Project Cost Estimate < $100,000

	m	Project Description	[illegible]	[illegible]	[illegible]	[illegible]	[illegible]
	7018	PREMS UPGRADE	PSEC	3	8	L	P
	N024	INSTALL GEN RTD TO COMPUTER	SITE	3	9	L	P
	4825	OUTSIDE CONTAINMENT IA COMP	PSEC	3	9	L	P
	4900	UPGRADE P-250 COMPUTER UNIT 1	PSEC	3	9	L	P
	4908	UPGRADE P-250 COMPUTER UNIT 2	PSEC	3	9	L	P
	6083	GEN BRKR FDN REMOVAL UNIT 2	PSEC	3	9	L	P
**	7015	CONTAIN ACCESS TRAINING FAC	PSEC	3	9	L	P
**	7021	CONTROL ROOM REMODELING	PSEC	3	9	L	P
	0084	MISC. STAT. PROJECTS	SITE	-	-	-	P
	0085	MISC. STAT. PROJECTS	SITE	-	-	-	P
	0086	MISC. STAT. PROJECTS	SITE	-	-	-	P
	0087	MISC. STAT. PROJECTS	SITE	-	-	-	P
	0101	BLANKET IMPROVE.	SITE	-	-	-	P

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1983 1984 1985 1986 1987

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Confidence Factor-

ATTACHMENT I
EXHIBIT F

*ERC Items
**Vepco Most Significant Issues

FIVE-YEAR INTEGRATED SCHEDULE

Nuclear Operations Department

Power Station SURRY

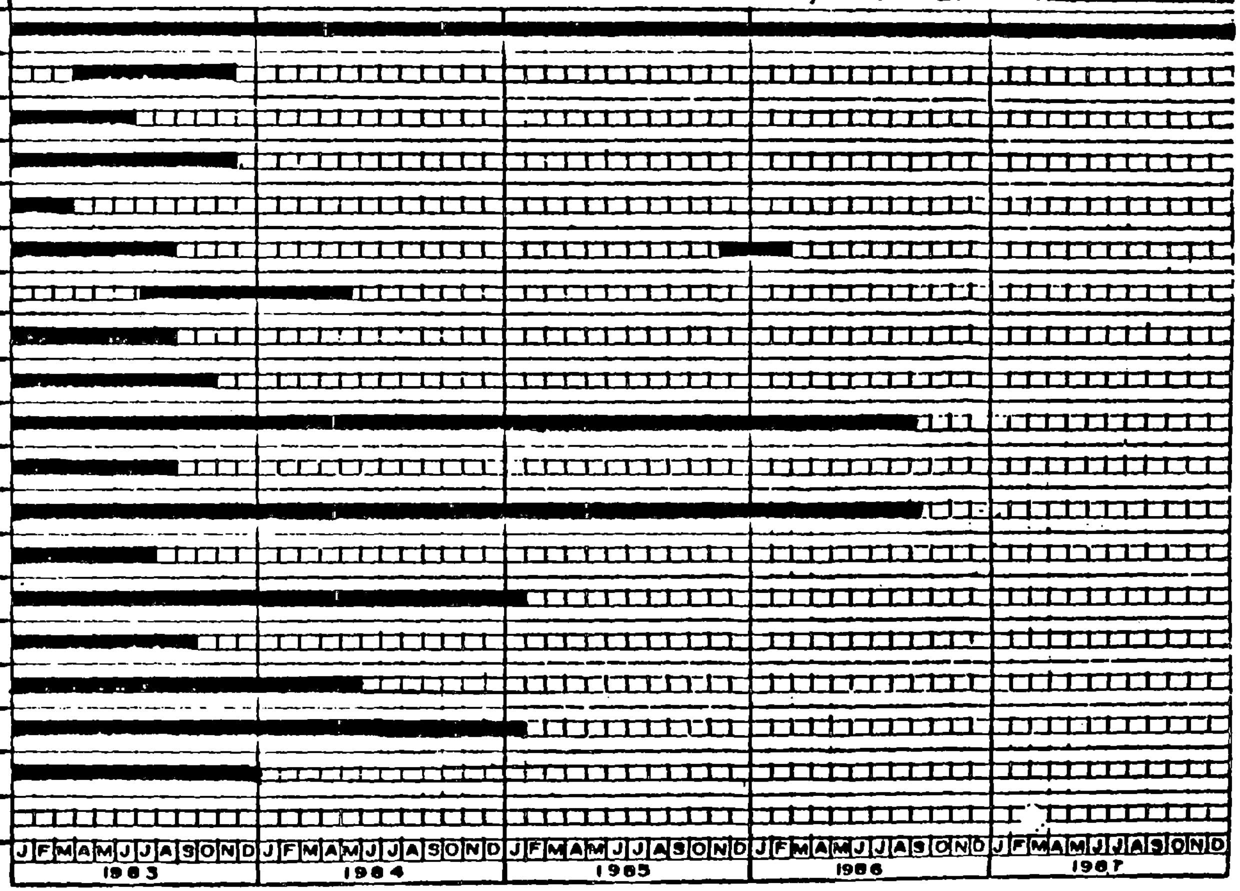
Unit 1 and 2

Page 1 of 9

Date 4-15-83

Project Cost Estimate ≥ $100,000

	[illegible]	Project Description	[illegible]	[illegible]	[illegible]	[illegible]	[illegible]
**	6097	Spent Fuel Stor Install	PSEC	1	1	H	P
**		GDC-17 Reanalysis	PSE	1	1	H	P
	7002	Incore Replacement	PSC	1	3	H	P
**	0201	Spent Fuel Handling Mods	PSEC	1	5	H	P
**	6124	Cond Retubing	PSEC	1	6	H	P
	7032	S/G BD Trip Valves	Site PSC	1	6	H	G
	4920	Drawing Updates	Site	1	2	L	P
	4983	CHG Pump/SV Water, Pump Repl	PSEC	1	2	L	P
	6072	IE Bulletin 79-14	PSEC	1	2	L	P
*	6086	TSC Facility	PSEC	1	2	L	P
	6109	React Ves.Level Indication	PSEC	2	3	L	G
**	6131	GDC-17 Mods	PSEC	1	2	L	G
	6142	Post Accident Sample Sys	PSEC	1	2	L	P
	6144	Cont HI Range RAD Monitor	PSEC	1	2	L	P
	6148	Cont Acc. Mon	PSEC	1	2	L	G
	6155	Environ. Qual. Unit 1	PSEC	1	2	L	G
	6156	Environ. Qual. Unit 2	PSEC	1	2	L	G
**	6164	Control Room Habit.	PSEC	1	2	L	G



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ATTACHMENT I
EXHIBIT F

Power Station SURRY

Unit 1 and 2

FIVE-YEAR INTEGRATED SCHEDULE

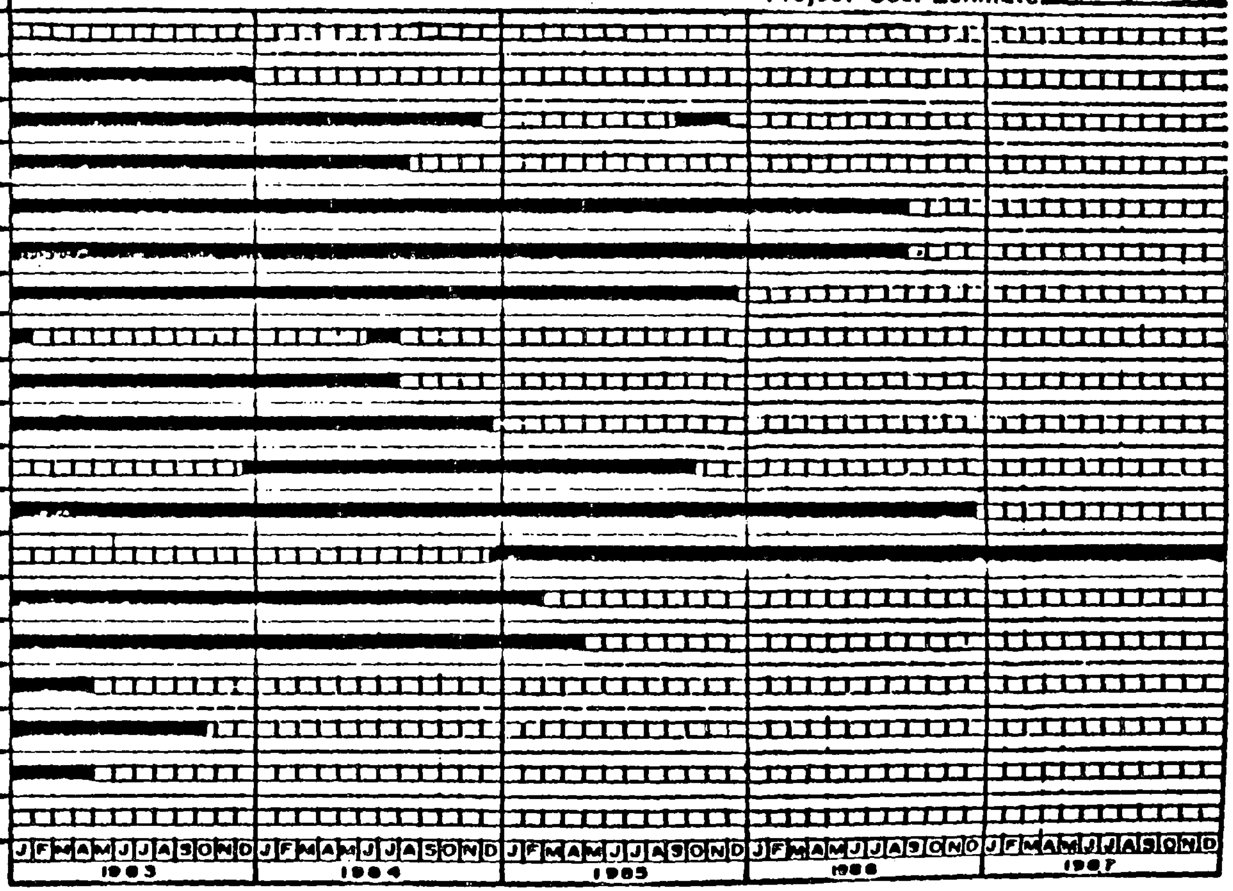
Nuclear Operations Department

Page 2 of 9

Date 4-15-83

Project Cost Estimate ≥ $100,000

	[illegible]	Project Description	[illegible]	[illegible]	[illegible]	[illegible]	[illegible]
*	4805	Local EOF Facility	PSEC	1	3	L	P
	4923	ISI Monit. Equip. Install	Site PSC	1	3	L	P
	6143	Heat Tracing System	PSEC	1	4	L	C
	4932	PORV & Safety Valve Mods.	PSEC	1	5	L	P
*	6158	I & C NUREG 0696	PSEC	1	5	L	P
*	6158M	SPDS Install.	PSEC	1	5	L	P
	T862	Incore Thermocouple Repl.	PSEC	2	5	H	P
	4292	HP HTR Drain Pump Motor	Site	2	6	H	P
	4303	SER & INST AIR	Site	2	6	H	P
**	7040	Feedwater Htr Replacement 5-6 Pts	PSEC	2	6	H	P
**	T179	Administration Building Addition	PSEC	2	8	H	P
	T953	Liquid Radwaste Proc.	PSEC	2	8	H	P
	T854	Solid Radwaste Vol. Red.	PSEC	2	8	H	P
**	4967	Sta. Personnel Facility Renov.	Site	2	8	H	P
	4956	Fuel Bldg. Block Wall	PSEC	2	3	L	P
	7037	Sec. Security Access Build. Imp.	PSEC	2	3	L	P
	4809	Main Stm. Monoball Support Mod.	PSEC	2	6	L	C
	4289	Rad. Count Room Computer Upgrade	Site	2	7	L	P



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ATTACHMENT I
EXHIBIT F

Power Station SURRY

Unit 1 and 2

FIVE-YEAR INTEGRATED SCHEDULE

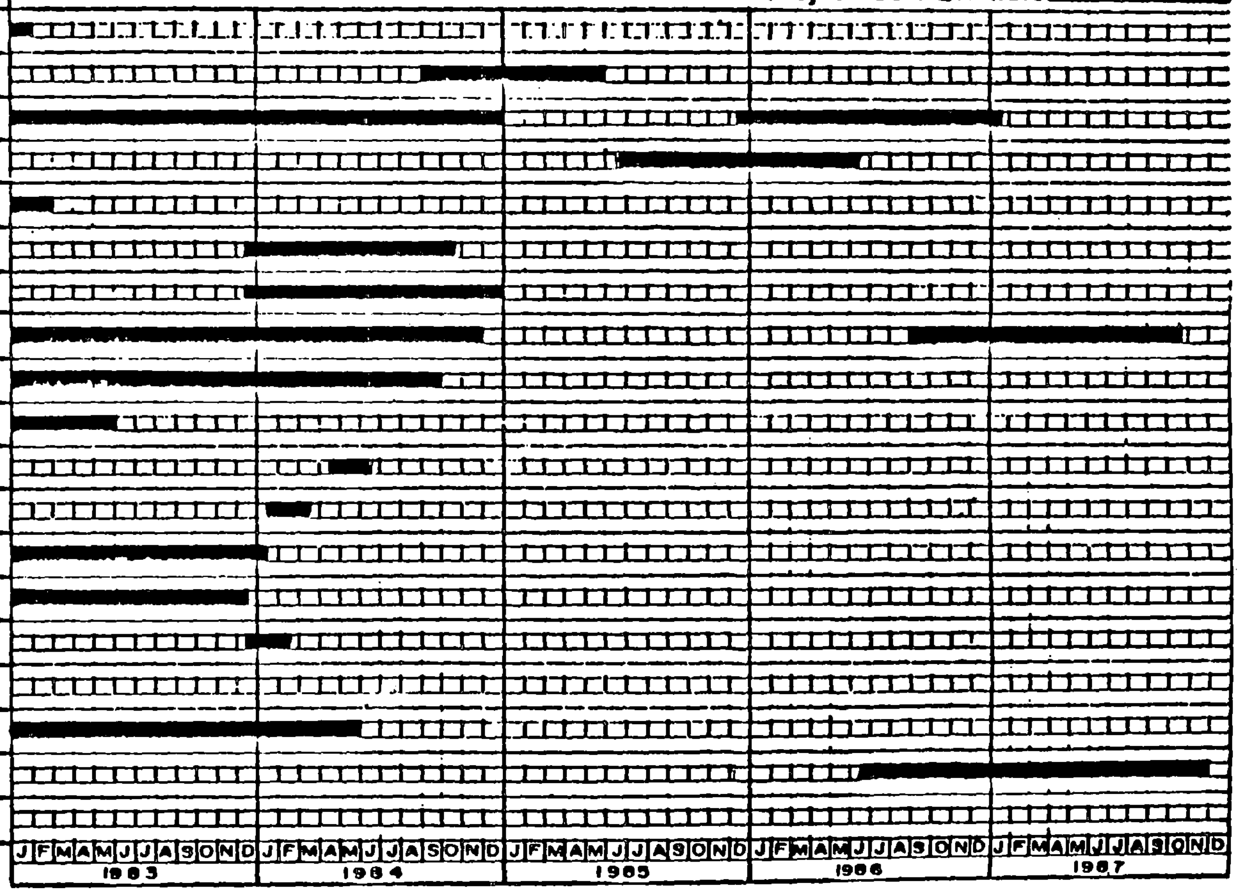
Nuclear Operations Department

Page 3 of 9

Date 4-15-83

Project Cost Estimate ≥ $100,000

	ID	Project Description	[illegible]	[illegible]	[illegible]	[illegible]	[illegible]
	T166	S. G. Prim, Man Han. Device	Site	3	4	H	P
	T164	Chg. Pmp. Cool SW Strainer	Site	3	6	H	P
**	T851	Feedwater Heater Repl 1 & 4 Pts	PSEC	3	6	H	P
	T860	Hi Level Int Screen Repl	Site	3	6	H	P
	0393	Fuel Transfer Sys Mods	PSEC	3	6	H	P
	4950	Plant Computer Interim Upgrade	Site PSEC	3	6	H	P
	0296	Auto Control For Bypass	PSEC	3	8	H	P
	4977	Core Uprating Mods	PSEC	3	8	H	P
**	7039	Sec. Plant Perf. Imp. MSPH	Site	3	8	H	P
	T863	Spare RCP Mtr. Assembly	Site	3	9	H	P
	0210	Regenerative Heat Exchanger	Site	3	9	H	P
	4288	Liquid Nitrogen Storage	Site	3	9	H	P
	4937	Contaminated Laundry Equip	Site	3	9	H	G
	4968	Transformer Fire Protection	PSEC	3	3	L	P
	0392	RCP Quick Disconnects	Site	3	4	L	P
**	T157	Replace SST with Auto LTC	Site	3	8	L	P
	T857	Inst. of 4th Emerg Diesel	PSEC	3	8	L	P
**	T858	Inst of 4th RSS Trans.	PSEC	3	8	L	P



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Confidence Factor-

ATTACHMENT I
EXHIBIT F

Power Station SURRY
Unit 1 and 2

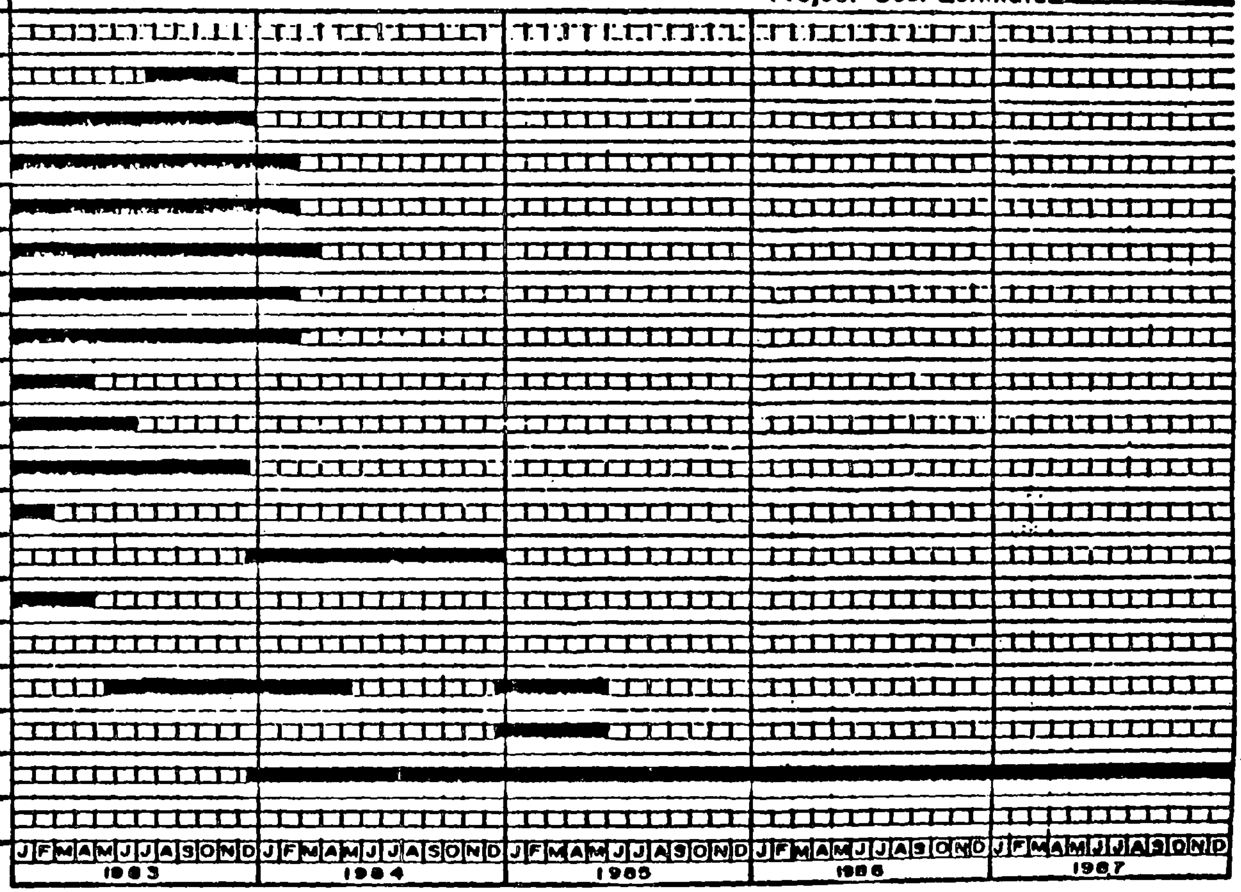
FIVE-YEAR INTEGRATED SCHEDULE

Nuclear Operations Department

Date 6-15-83

Project Cost Estimate ≥ $100,000

IR	Project Description	[illegible]	[illegible]	[illegible]	[illegible]	[illegible]
	Charleston Earthquake Stdy	NOD	3	5	L	P
4819	Communications Audibility	PSEC	3	5	L	P
4926	Main Steam Discharge Flow	PSEC	3	5	L	P
4927	Imp. Range Main Stm. Dish Mon.	PSEC	3	5	L	P
4930	Mech Safe Rel Equi Env. Qual.	PSEC	3	5	L	P
4933	Seismic & Dynamic Qual Elec & Me	PSEC	3	5	L	P
4935	RCS Support	PSEC	3	5	L	P
4975	Rx Vessel Thermal Shock	PSEC	3	5	L	P
7035	Simulator Comp. Upgrade	Site	3	5	L	P
T165	Spare RCP Oil Collection System	Site	3	6	L	P
4802	Containment Personnel Hatch Modification	Site	3	6	L	P
4816	Spare RHR Pump and Motor	Site	3	6	L	P
4944	Replace 12/6" Pathon Snubb	PSEC	3	6	L	P
4855	Aux Boiler Sys Mods	PSC Site	3	8	L	P
** T181	Training Center Expansion, Phase 2	Site	3	9	L	P
** 4818	Training Center Expansion	Site	3	9	L	P
** 4821	Warehouse Expansion	PSEC	3	9	L	P
4943	P-250 Computer Replacement - Unit 1	PSEC	3	9	L	P



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Confidence Factor -

ATTACHMENT I
EXHIBIT F

Power Station SURRY

Unit 1 and 2

FIVE-YEAR INTEGRATED SCHEDULE

Nuclear Operations Department

Page 5 of 9

Date 4-15-83

Project Cost Estimate ≥ $100,000

[illegible]	Project Description	[illegible]	[illegible]	[illegible]	[illegible]	[illegible]	1983	1984	1985	1986	1987
4949	P-250 Computer Replacement Unit 2	PSEC	3	9	L	P					
UN15	Unidentified Projects Allow	Site	3	9	L	P					

Months for each year: J F M A M J J A S O N D

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Confidence Factor -

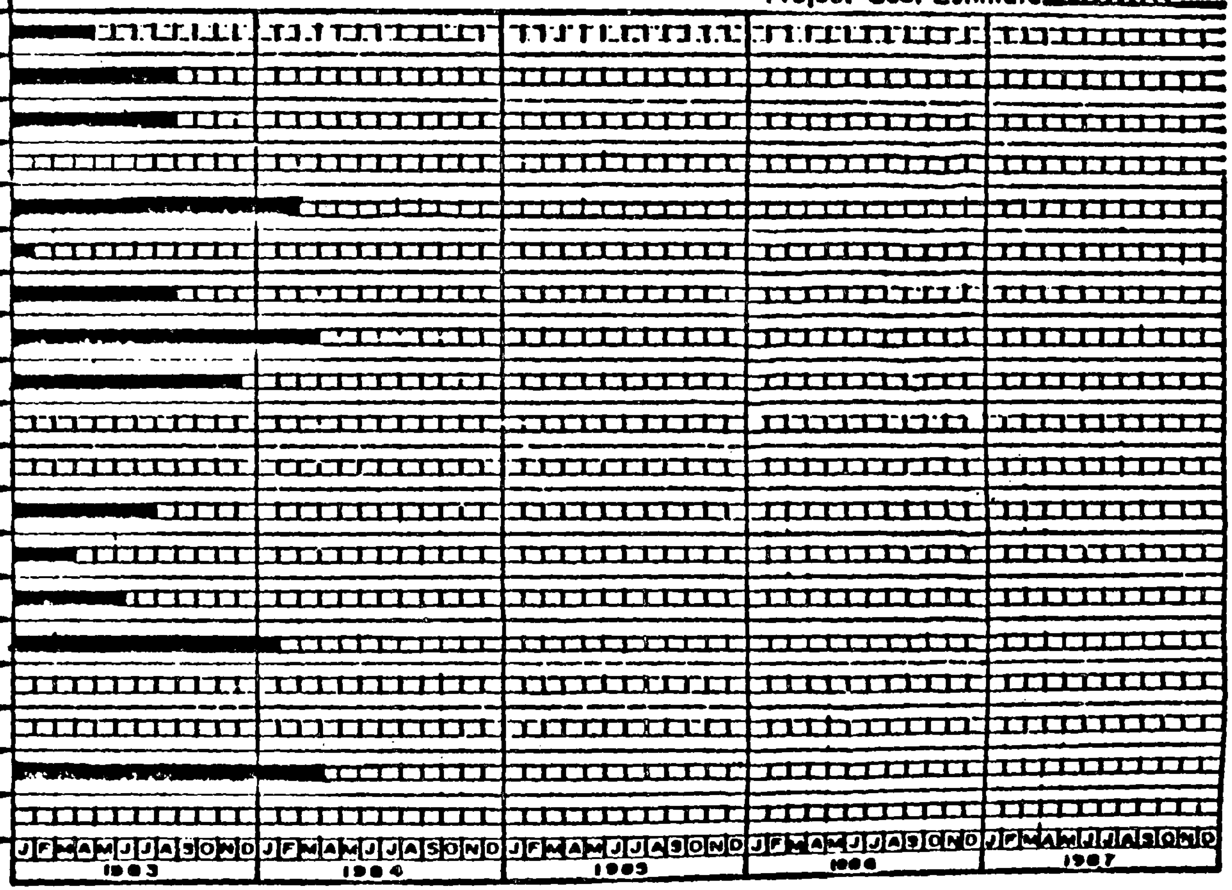
ATTACHMENT I
EXHIBIT F

Power Station SURRY
Unit 1 and 2

FIVE-YEAR INTEGRATED SCHEDULE
Nuclear Operations Department

Page 8 of 9
Date 4-13-83
Project Cost Estimate <$100,000

	[illegible]	Project Description	[illegible]	[illegible]	[illegible]	[illegible]	[illegible]
	6116	React NO & PRZ Vent #1	PSEC	1	2	L	P
	6117	React NO & PRZ Vent #2	PSEC	1	2	L	P
	6146	SW Rad Mont Pump Motor	PSEC	1	2	L	C
*	4929	Control Room Review & Upgrade	PSC	1	5	L	P
	4301	Main Station Battery Charger Rep.	Site	1	6	L	P
	4971	Electrical Penetration Replacement	PSEC	1	6	L	P
	4990	RSST Primary Cable Replacement	T & D	1	6	H	P
	4867	Inservice Inspection System	Site	2	5	H	P
	4942	Rep Switch Cable Low Level Intake	T & D	2	6	H	P
**	6940	Chemistry Instrumentation	Site	2	9	H	P
	4991	Appendix R Fire Protection	PSEC	2	2	L	P
	4167	Auto Extrace Stm. S.	Site	2	3	L	P
	6108	Reactor Vessel Level Indication #1	PSEC	2	3	L	P
*	4925	Reg Guide 1.97	PSEC	2	5	L	P
*	6135	MET Monit Upgrade	NOD	2	5	L	P
	4938	Repl. Charge Pump Disch Val. R	Site PSC	2	6	L	P
*	4807	Interim Contr. Room & Simulator Upgrade	PSEC	2	9	L	P



LEGEND:

Priority Group-
1-Not Deferable
2-Deferable (major impact)
3-Deferable (minor impact)

Requirement Classification Category-
1- Maintain Safe Electrical Production
2- License Commitment
3- Legal Req...
5-Legal Requirement (not well defined)
6-Plant Maintenance
7-...

Cost/Benefit-
H-High Benefit
L-Low Benefit

ATTACHMENT I
EXHIBIT F

FIVE-YEAR INTEGRATED SCHEDULE

Nuclear Operations Department

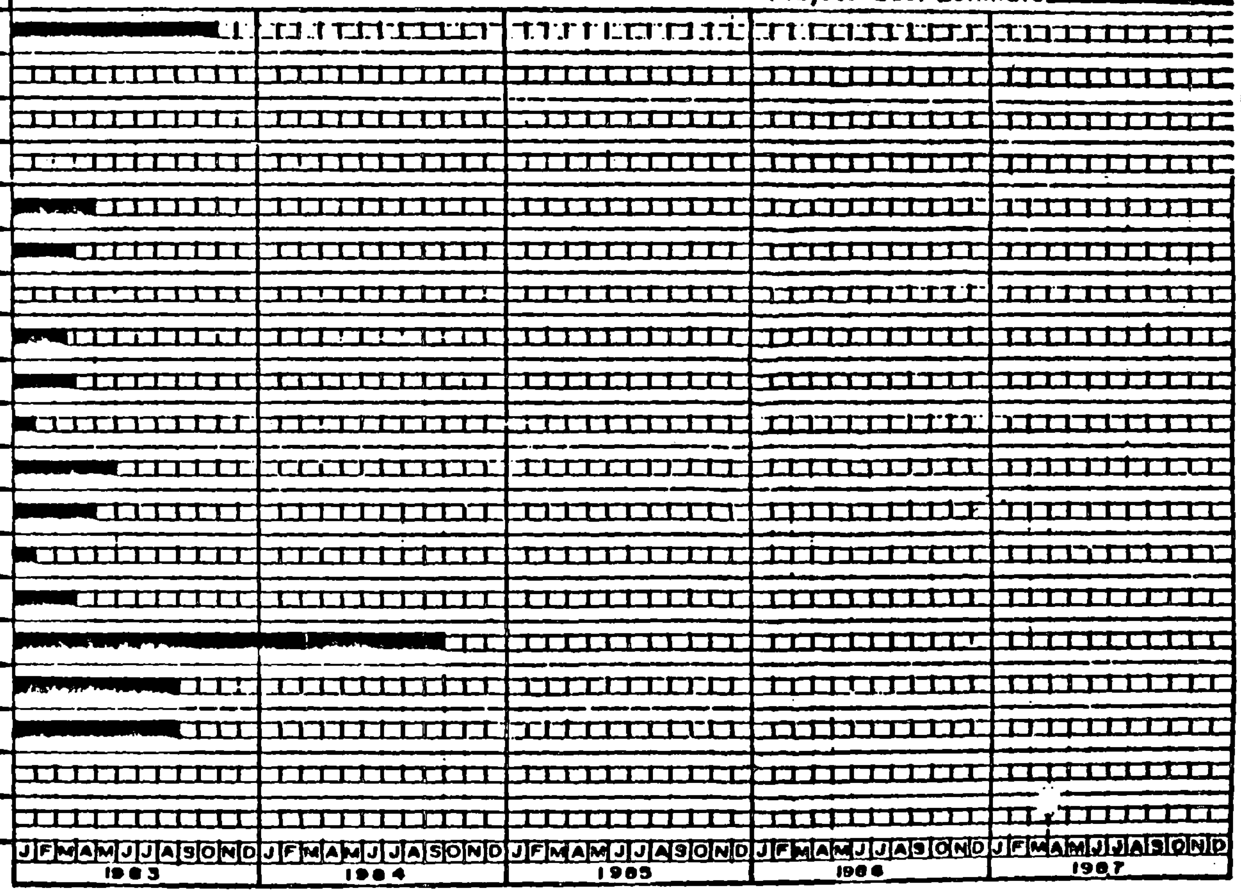
Power Station SURRY

Unit 1 and 2

Date 4-15-83

Project Cost Estimate ≤$100,000

	#	Project Description	[illegible]	[illegible]	[illegible]	[illegible]	[illegible]
	6119	Early Warning System	Comm	2	9	L	P
	D001	Boric Acid Stor. Tk Level Trans	Site	3	4	H	P
	D005	RCP Stator RTD Rewire	Site	3	4	H	P
	D010	Aux Stm Tank Vent Mod.	Site	3	4	H	P
	4947	Fuel Transf Tube Blind Flg.	Site	3	4	H	P
	D007	Serv. Air Compressor Tie-in	Site	3	6	H	C
	D008	S/G Blowdown Tank Removal	Site	3	8	H	P
**	D009	S/G Blowdown Sample System Mod.	Site	3	8	H	P
	D004	Liquid Waste Pump Cross Connect	Site	3	4	L	P
	4302	Reactor Vessel Stud Tool	Site	2	4	L	P
	D012	EDG Air Start Pipe Support	Site	3	5	L	C
	0211	Rad Waste Modification	PSEC	3	5	L	P
	4815	S/G Level Reference Leg Insulation	PSE	3	5	L	P
	7036	Diesel Sequencing	PSEC	3	5	L	P
	4185	Cont SOF Related Equipment	Site	3	6	L	P
	4869	Upgrade Low Level Intake Screen	Site	3	6	L	P
	4941	Major Roof Replacement	Site	3	6	L	P
	4960	Valve Surface Grinder	Site	3	6	L	P



LEGEND:

Priority Group -
1- Not Deferable
2- Deferable (major impact)
3- Deferable (minor impact)

Requirement Classification Category -
1- Maintain Safe Electrical Production
2- License Commitment
3- Legal Requirement (well defined)
4- ALARA
5- Legal Requirement (not well defined)
6- Plant Maintenance
7- Required Surveillance
8- Plant Efficiency Improvement
9- Plant Support [illegible]

Cost/Benefit -
H - High Benefit
L - Low Benefit

Confidence Factor -

ATTACHMENT I
EXHIBIT F

Power Station SURRY
Unit 1 and 2

FIVE-YEAR INTEGRATED SCHEDULE

Nuclear Operations Department

Page 8 of 9
Date 4-15-83
Project Cost Estimate ≤$100,000

[illegible]	Project Description	[illegible]	[illegible]	[illegible]	[illegible]	[illegible]
7033	Cont Vac. Pumps	Site PSE	3	6	L	F
D002	Main Steam Sampling	Site	3	7	L	P
4951	Annunciator Addition	PSEC	3	7	L	P
4955	Cont Gas and Part Monitors	Site	3	7	L	P
4963	Eddy Current Equipment	Site	3	7	L	P
** 4965	Cation Conductivity Sys	Site	3	7	L	P
D003	Spent Fuel Crane Radio Control	Site	3	8	L	P
7034	Hogger Upgrade	PSEC	3	8	L	P
D006	Station Perimeter Gates Mod	Site	3	9	L	P
4341	Speed Sensing Relay	A&C	3	9	L	P
4813	Panasonic TLD Reader	Site	3	9	L	P
4936	Respirator Clean & Test Equipment	Site	3	9	L	P
4952	New Portal Monitors	Site	3	9	L	P
4953	Contaminated Waste Monitor	Site	3	9	L	P
4964	Hot Lab Renovation	Site	3	9	L	P
4978	Motor Storage Building	Site	3	9	L	P
4984	Security Duct Bank Flooding	PSEC	3	9	L	P
** T153	Recreation Facility	Site	3	9	L	P

J F M A M J J A S O N D J F M A M J J A S O N D J F M A M J J A S O N D J F M A M J J A S O N D J F M A M J J A S O N D
1983 1984 1985 1986 1987

LEGEND:

Priority Group–
1-Not Deferable
2-Deferable (major impact)
3-Deferable (minor impact)

Requirement Classification Category–
1- Maintain Safe Electrical Production
2- License Commitment
3- Legal Requirement (well defined)
4- ALARA
5- Legal Requirement (not well defined)
6- Plant Maintenance
7- Required Surveillance
8- Plant Efficiency Improvement
9- Plant Support Improvement

Cost/Benefit–
H-High Benefit
L-Low Benefit

Confidence Factor–
G-Good F-Fair P-Poor

ATTACHMENT I
EXHIBIT F

Power Station SURRY

Unit 1 and 2

FIVE-YEAR INTEGRATED SCHEDULE

Nuclear Operations Department

Date 4-15-83

Project Cost Estimate ≤$100,000

[illegible]	Project Description	[illegible]	[illegible]	[illegible]	[illegible]	[illegible]
4947	Fuel Transf Tube Bld Flange	Site				G

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LEGEND:

Priority Group -
1-Not Deferable
2-Deferable (major impact)
3-Deferable (minor impact)

Requirement Classification Category -
1- Maintain Safe Electrical Production
2- License Commitment
3- Legal Requirement (well defined)
4- ALARA
5- Legal Requirement (not well defined)
6- Plant Maintenance
7- Required Surveillance
8- Plant Efficiency Improvement
9- Plant Support Improvement

Cost/Benefit -
H - High Benefit
L - Low Benefit

Confidence Factor -

ATTACHMENT I
EXHIBIT F

Power Station SYSTEM
Unit

FIVE-YEAR INTEGRATED SCHEDULE

Nuclear Operations Department

Page 1 of 1
Date 4-15-83
Project Cost Estimate N/A

	IR	Project Description	[illegible]	[illegible]	[illegible]	[illegible]	[illegible]
**.	4831	Simulator/Trn Facility NA					P
*.	4917	CEOF	PSE	1	3	L	P
	4919	Commitment Tracking System					P
*.	4921	Emergency Communications					P
	7038	Press. Upgrade					P

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1983 1984 1985 1986 1987

LEGEND:

Priority Group-
1-Not Deferable
2-Deferable (major impact)
3-Deferable (minor impact)

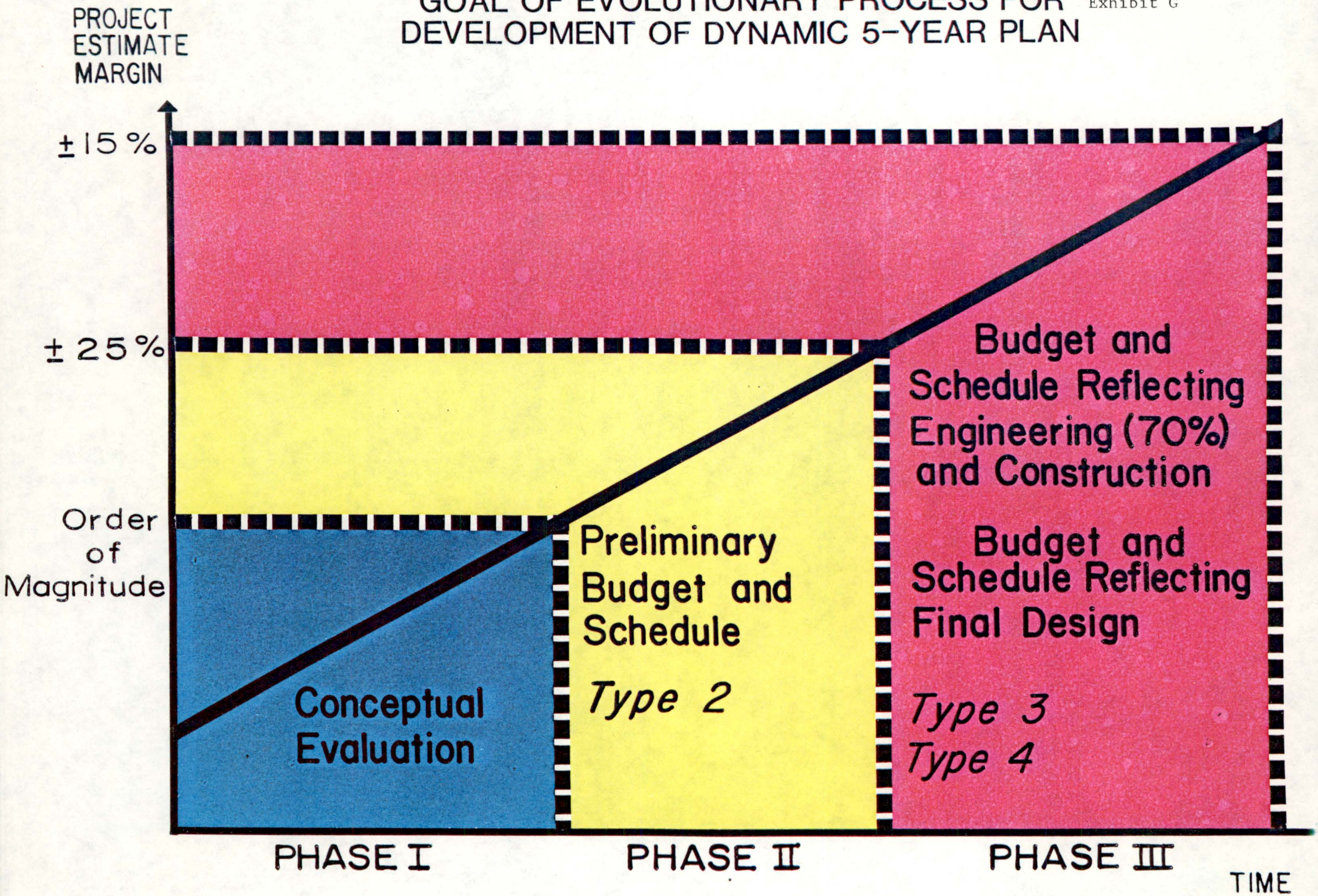
Requirement Classification Category-
1- Maintain Safe Electrical Production
2- License Commitment
3- Legal Requirement (well defined)
4- ALARA
5- Legal Requirement (not well defined)
6- Plant Maintenance
7- Required Surveillance
8- Plant Efficiency Improvement
9- Plant Support Improvement

Cost/Benefit-
H-High Benefit
L-Low Benefit

Confidence Factor-
G-Good F-Fair P-Poor

Attachment 1
Exhibit G

GOAL OF EVOLUTIONARY PROCESS FOR DEVELOPMENT OF DYNAMIC 5-YEAR PLAN



1. SAFETY PARAMETER DISPLAY SYSTEM (SPDS)

NRC Requirements

"a. The SPDS should provide a concise display of critical plant variables to the control room operators to aid them in rapidly and reliably determining the safety status of the plant. Although the SPDS will be operated during normal operations as well as during abnormal conditions, the principal purpose and function of the SPDS is to aid the control room personnel during abnormal and emergency conditions in determining the safety status of the plant and in assessing whether abnormal conditions warrant corrective action by operators to avoid a degraded core. This can be particularly important during anticipated transients and the initial phase of an accident.

"b. Each operating reactor shall be provided with a Safety Parameter Display System that is located convenient to the control room operators. This system will continuously display information from which the plant safety status can be readily and reliably assessed by control room personnel who are responsible for the avoidance of degraded and damaged core events.

"c. The control room instrumentation required (see General Design Criteria 13 and 19 of Appendix A to 10 CFR 50) provides the operators with the information necessary for safe reactor operation under normal, transient, and accident conditions. The SPDS is used in addition to the basic components and serves to aid and augment these components. Thus, requirements applicable to control room instrumentation are not needed for this augmentation (e.g., GDC 2, 3, 4 in Appendix A; 10 CFR Part 100; single-failure requirements). The SPDS need not meet requirements of the singlefailure criteria and it need not be qualified to meet Class 1E requirements. The SPDS shall be suitably isolated from electrical or electronic interference with equipment and sensors that are in use for safety systems. The SPDS need not be seismically qualified, and additional seismically qualified indication is not required for the sole purpose of being a backup for SPDS. Procedures which describe the timely and correct safety status assessment when the SPDS is and is not available, will be developed by the licensee in parallel with the SPDS. Furthermore, operators should be trained to respond to accident conditions both with and without the SPDS available.

"d. There is a wide range of useful information that can be provided by various systems. This information is reflected in such staff documents as NUREG-0696, NUREG-0835, and Regulatory Guide 1.97. Prompt implementation of an SPDS can provide an important contribution to plant safety. The selection of specific information that should be provided for a particular plant shall be based on engineering judgement of individual plant licensees, taking into account the importance of prompt implementation.

"e. The SPDS display shall be designed to incorporate accepted human factors principles so that the displayed information can be readily perceived and comprehended by SPDS users.

"f. The minimum information to be provided shall be sufficient to provide information to plant operators about:

(i) Reactivity control

(ii) Reactor core cooling and heat removal from the primary system

(iii) Reactor coolant system integrity

(iv) Radioactivity control

(v) Containment conditions

The specific parameters to be displayed shall be determined by the licensee."

Vepco Response

a. Status of SPDS

In January, 1981, Vepco initiated an engineering project to provide an SPDS. In order to provide a reliable Data Acquisition System (DAS), it was decided that an Intelligent Remote Multiplex System (IRMS) and an Emergency Response Computer System (ERCS) would be used. The Emergency Response Computer System is composed of the Data Communications Processor (DCP) and the Emergency Response Facility Input/Output Processor (ERFI/O). The IRMS is composed of multiplexers which will receive those inputs required to support the development of the SPDS. The multiplexers through buffers, submultiplexers and subbuffers, feed the Master receivers. Two master receivers feed all inputs into the DCP.

Purchase of the IRMS is over 95% complete and engineering required for installation of the multiplexers and the wiring of their inputs is over 85% complete. Installation of the multiplexers and wiring of the inputs is over 40% complete at Surry and 7% complete at North Anna. Engineering to install the remainder of the IRMS is continuing.

The ERCS is composed of the DCP and ERFI/O which completes the SPDS DAS. The ERFI/O computers were ordered in June 1982. The purchase order includes a color graphic development software package and man machine interface (MMI) which will be used to develop the SPDS displays. To date, all basic software and 47% of the color graphic and MMI have been delivered.

A request for proposal for the DCP was offered for bids on July 19, 1982 but award of the contract was delayed due to a review of the design of the TSC. The contract bidding was reopened and on March 10, 1983 new bids were obtained. The purchase orders for the DCPs was placed in late March 1983.

A review of industry documents related to SPDS design has been performed and this information is being considered during development of displays. Vepco intends to visit several sites where SPDS is operational to review capabilities and displays.

Verification of the SPDS design will be performed using a program which is being developed by a consultant and being reviewed by Vepco. The task analysis prepared by the Westinghouse Owners Group (WOG), which is basic to the generic emergency response guidelines and the generic technical guidelines, will also be used in the design.

Human factors design criteria will be used in the design of CRT displays and a human factors engineering review of the displays may be performed.

b. SPDS Safety Analysis

Based on the scope of work as presently defined, the SPDS safety analysis will be prepared and submitted to the NRC by February 1, 1984.

c. SPDS Operation and Operator Training

The installation of the computers which drive the SPDS in their area of the TSC is planned for Fall of 1984. The computer system and all other portions of the DAS will be tested after installation, and a six month availability analysis of the computers will be performed. During this period operators will be provided training on the system prior to the system becoming operational in the control room.

Because the installation of the SPDS CRTs in the control room will require modifications to the present control room facilities and may require the replacement of the existing computer console, the installation of the SPDS in the control room can only be accomplished during a refueling outage. Since the computers will not complete their availability test until the middle of 1985 and, placing untested displays in the control room could have negative long term impact on the use of SPDS and on operator confidence in the system, Vepco presently plans to have the SPDS installed in the control room at Surry during the first refueling outage of each unit after July 1, 1985, and at North Anna after the refueling in 1986.

This date is predicated upon:

(1) Favorable results from a study currently underway to determine if the vital power buses will require modification to support the additional load of the data acquisition system.

(2) The results of the SPDS Safety Analysis do not require data for which sensors do not currently exist.

If either the vital bus requires modification or additional data points are required to support the SPDS, then the above installation date is subject to modification.

d. Vepco does not desire a pre-implementation review of the SPDS.

e. Integrated Plan Guidelines for the SPDS with the other emergency response capability items is presented in attachment 3.

2. DETAILED CONTROL ROOM DESIGN REVIEW

NRC Requirements

"a. The objective of the control room design review is to "improve the ability of of nuclear power plant control room operators to prevent accidents or cope with accidents if they occur by improving the information provided to them" (from NUREG-0660, Item I. D. 1). As a complement to improvements of plant operating staff capabilities in response to transients and other abnormal conditions that will result from implementation of the SPDS and from upgraded emergency operating procedures, this design review will identify any modifications of control room configurations that would contribute to a significant reduction of risk and enhancement in the safety of operation. Decisions to modify the control room would include consideration of long-term risk reduction and any potential temporary decline in safety after modifications resulting from the need to relearn maintenance and operating procedures. This should be carefully reviewed by persons competent in human factors engineering and risk analysis.

"b. Conduct a control room design review to identify human engineering discrepancies. The review shall consist of:

(i) The establishment of a qualified multidisciplinary review team and a review program incorporating accepted human engineering principles.

(ii) The use of function and task analysis (that had been used as the basis for developing emergency operating procedures Technical Guidelines and plant specific emergency operating procedures) to identify control room operator tasks and information and control requirements during emergency operations. This analysis has multiple purposes and should also serve as the basis for developing training and staffing needs and verifying SPDS parameters.

(iii) A comparison of the display and control requirements with a control room inventory to identify missing displays and controls.

(iv) A control room survey to identify deviations from accepted human factors principles. This survey will include, among other things, an assessment of the control room layout, the usefulness of audible and visual alarm systems, the information recording and recall capability, and the control room environment.

"c. Assess which human engineering discrepancies are significant and should be corrected. Select design improvements that will correct those discrepancies. Improvements that can be accomplished with an enhancement program (paint-tape-label) should be done promptly.

"d. Verify that each selected design improvement will provide the necessary correction, and can be introduced in the control room without creating any unacceptable human engineering discrepancies because of significant contribution to increased risk, unreviewed safety questions, or situations in which a temporary reduction in safety could occur. Improvements that are introduced should be coordinated with changes resulting from other improvement programs such as SPDS, operator training, new instrumentation (Reg. Guide 1.97, Rev. 2), and upgraded emergency operating procedures."

Vepco Response

a. Status of Detailed Control Room Design Review (DCRDR)

Vepco has reviewed available documents regarding the Detailed Control Room Design Review (DCRDR) and determined that, until the design of SPDS displays and location of SPDS equipment is specified, the implementation of new emergency operations procedures (EOPs) is complete, and the determination of which, if any, Reg. Guide 1.97 related instrumentation needs to be installed, the DCRDR cannot be completed.

Vepco is presently preparing a paint, tape, and label upgrade for Surry and has previously performed such an upgrade at North Anna. The Surry upgrade program should be completed by March 31, 1984.

Vepco has been actively involved in the INPO NUTACs on CRDR and Emergency Response Capability (ERC) and intends to begin work on the CDCR plan in the fourth quarter of 1983.

b. Date for CRDR Program Plan

Vepco plans to submit its CRDR Program Plans by March 1, 1984.

c. Date for Summary Report

The scope of the control room design review and hence the schedule for completing the review are impacted by the CRDR Program Plan and the control room modifications required to meet Regulatory

Guide 1.97. Vepco will complete Regulatory Guide 1.97 studies in December of 1983 and know at that time the extent of 1.97 modifications required in the control room. As stated above, the CRDR Program Plans will be submitted by March 1, 1984. The CRDR Program Plan submittal will define the date for completing the CRDR and submitting the Summary Report.

d. Integrated Plan Guidelines for the DCRDR with other ERC efforts is given in attachment 3.

3. REGULATORY GUIDE 1.97 - APPLICATION TO EMERGENCY RESPONSE FACILITIES

NRC Requirements

"a. Functional Statement

Regulatory Guide 1.97 provides data to assist control room operators in preventing and mitigating the consequences of reactor accidents.

"b. Control Room

Provide measurements and indication of Type A, B, C, D, E variables listed in Regulatory Guide 1.97 (Rev. 2). Individual licensees may take exceptions based on plant-specific design features. BWR incore thermocouples and continuous offsite dose monitors are not required pending their further development and consideration as requirements. It is acceptable to rely on currently installed equipment if it will measure over the range indicated in Regulatory Guide 1.97 (Rev. 2), even if the equipment is presently not environmentally qualified. Eventually, all the equipment required to monitor the course of an accident would be environmentally qualified in accordance with the pending Commission rule on environmental qualification.

Provide reliable indication of the meteorological variables (wind direction, wind speed, and atmospheric stability) specified in Regulatory Guide 1.97 (Rev. 2) for site meteorology. No changes in existing meteorological monitoring systems are necessary if they have historically provided reliable indication of these variables that are representative of meteorological conditions in the vicinity (up to about 10 miles) of the plant site. Information on meteorological conditions for the region in which the site is located shall be available via communication with the National Weather Service. These requirements supersede the clarification of NUREG-0737, Item III.A.2.2.

"c. Technical Support Center (TSC)

The Type A, B, C, D and E variables that are essential for performance of TSC functions shall be available in the TSC.

(i) BWR incore thermocouples and continuous offsite dose monitors are not required pending their further development and consideration as requirements.

(ii) The indicators and associated circuitry shall be of reliable design but need not meet Class 1E, single-failure or seismic qualification requirements.

"d. Emergency Operations Facility (EOF)

(i) Those primary indicators needed to monitor containment conditions and releases of radioactivity from the plant shall be available in the EOF.

(ii) The EOF data indications and associated circuitry shall be of reliable design but need not meet Class 1E, single-failure or seismic qualification requirements."

Vepco Response

a. Status of Regulatory Guide 1.97

Vepco, through a contractor, is in the process of comparing the instrumentation specified by Reg. Guide 1.97 to that currently installed at Surry and North Anna.

b. Regulatory Guide 1.97 Report

Vepco will provide a report to the NRC by January 31, 1984 with the results of a comparison and schedule for upgrades and modifications necessitated as a result of the comparison.

c. Vepco plans to satisfy the requirement to provide necessary Reg. Guide 1.97 variables in the TSC and EOF using the SPDS data base and, if necessary, by adding points to a separate data base within the DAS system to provide needed information.

d. Integrated Plan Guidelines for Regulatory Guide 1.97 with other ERC efforts is included in attachment 3.

4. UPGRADE EMERGENCY OPERATING PROCEDURES (EOPs)

NRC Requirements

"a. The use of human factored, function oriented, emergency operating procedures will improve human reliability and the ability to mitigate the consequences of a broad range of initiating events and subsequent multiple failures or operator errors, without the need to diagnose specific events.

"b. In accordance with NUREG-0737, Item I.C.1, reanalyze transients and accidents and prepare Technical Guidelines. These analyses will identify operator tasks, and information and control needs. The analyses also serve as the basis for integrating upgraded emergency operating procedures and the control room design review and verifying the SPDS design.

"c. Upgrade EOPs to be consistent with Technical Guidelines and an appropriate procedure Writer's Guide.

"d. Provide appropriate training of operating personnel on the use of upgraded EOPs prior to implementation of the EOPs.

"e. Implement upgraded EOPs".

Vepco Response

a. Emergency Operating Procedures Upgrade Status

The emergency operating procedure upgrade at Surry is 20% complete using the WOG Technical Guidelines and Emergency Operating Procedures Implementation Assistance (EOPIA) documentation. Although Surry used Rev. 0 of the WOG Generic Technical Guidelines, the procedures will be compared with Rev. 1 of the WOG Generic Technical Guidelines for possible additional upgrade of the emergency operating procedures. North Anna will use Rev. 0 of the WOG Guidelines for initial procedure upgrade.

b. Rev. 0 of the WOG Generic Technical Guidelines have been submitted to the NRC for review and approval.

c. The Procedures Generation Package will be submitted to the NRC by July 1, 1983.

d. The EOPs will be validated using the validation program included in the July 1, 1983 submittal. The training program description will also be included in the July 1, 1983 submittal. Since the development and preparation for implementation is a high priority, Vepco does not plan to delay training for three months at Surry after its July 1, 1983 submittal. Vepco plans to train its operators on the validated procedures and implement the procedures at Surry by October 1, 1983 and at North Anna by March 1, 1984.

e. Integrated Plan Guidelines for EOPs with other ERC efforts is given in attachment 3.

5. INTEGRATED TRAINING REQUIREMENT

NRC Requirement

"The design of the Safety Parameter Display System (SPDS), design of instrument displays based on Regulatory Guide 1.97 guidance, control room design review, development of function oriented emergency operating procedures, and operating staff training should be integrated with respect to the overall enhancement of operator ability to comprehend plant conditions and cope with emergencies."

Vepco Response

The integrated training plan will be completed by the first quarter of 1984. Training will be completed by the first quarter of 1986 except for items, presently undefined, which may extend beyond this date e.g., Control Room design deficiencies remaining to be corrected, Reg. Guide 1.97 modifications not completed, etc.

6. EMERGENCY RESPONSE FACILITIES

Technical Support Center (TSC)

NRC Requirements

"a. The TSC is the onsite technical support center for emergency response. When activated, the TSC is staffed by predesignated technical, engineering, senior management, and other licensee personnel, and five pre-designated NRC personnel. During periods of activation, the TSC will operate uninterrupted to provide plant management and technical support to plant operations personnel, and to relieve the reactor operators of peripheral duties and communications not directly related to reactor system manipulations. The TSC will perform EOF functions for the Alert Emergency class and for the Site Area Emergency class and General Emergency class until the EOF is functional.

"The TSC will be:

"b. Located within the site protected area so as to facilitate necessary interaction with control room, OSC, EOF and other personnel involved with the emergency.

"c. Sufficient to accommodate and support NRC and licensee predesignated personnel, equipment and documentation in the center.

"d. Structurally built in accordance with the Uniform Building Code.

"e. Environmentally controlled to provide room air temperature, humidity and cleanliness appropriate for personnel and equipment.

"f. Provided with radiological protection and monitoring equipment necessary to assure that radiation exposure to any person working in the TSC would not exceed 5 rem whole body, or its equivalent to any part of the body, for the duration of the accident.

"g. Provided with reliable voice and data communications with the control room and EOF and reliable voice communications with the OSC, NRC Operations Centers and state and local operations centers.

"h. Capable of reliable data collection, storage, analysis, display and communication sufficient to determine site and regional status, determine changes in status, forecast status and take appropriate actions. The following variables shall be available in the TSC:

(i) the variables in the appropriate Table 1 or 2 of Regulatory Guide 1.97 (Rev. 2) that are essential for performance of TSC functions; and

(ii) the meteorological variables in Regulatory Guide 1.97 (Rev. 2) for site vicinity and National Weather Service data available by voice communication for the region in which the plant is located.

Principally those data must be available that would enable evaluating incident sequence, determining mitigating actions, evaluating damages and determining plant status during recovery operations.

"i. Provided with accurate, complete and current plant records (drawings, schematic diagrams, etc.) essential for evaluation of the plant under accident conditions.

"j. Staffed by sufficient technical, engineering, and senior designated licensee officials to provide needed support, and be fully operational within approximately 1 hour after activation.

"k. Designed taking into account good human factors engineering principles."

Operational Support Center (OSC)

NRC Requirements

"a. When activated, the OSC will be the onsite area separate from the control room where predesignated operations support personnel will assemble. A predesignated licensee official shall be responsible for coordinating and assigning the personnel to tasks designated by control room, TSC and EOF personnel.

"The OSC will be:

"b. Located onsite to serve as an assembly point for support personnel and to facilitate performance of support functions and tasks.

"c. Capable of reliable voice communications with the control room, TSC and EOF."

Emergency Operations Facility (EOF)

NRC Requirements

"a. The EOF is a licensee controlled and operated facility. The EOF provides for management of overall licensee emergency response, coordination of radiological and environmental assessment, development of recommendations for public protective actions, and coordination of emergency response activities with Federal, State and local agencies.

When the EOF is activated, it will be staffed by predesignated emergency personnel identified in the emergency plan. A designated senior licensee official will manage licensee activities in the EOF.

Facilities shall be provided in the EOF for the acquisition, display and evaluation of radiological and meteorological data and containment conditions necessary to determine protective measures. These facilities will be used to evaluate the magnitude and effects of actual or potential radio-active releases from the plant and to determine dose projections.

"The EOF will be:

"b. Located and provided with radiation protection features as described in Table 1 (previous guidance approved by the Commission) and with appropriate radiological monitoring systems.

"c. Sufficient to accommodate and support Federal, State, local and licensee predesignated personnel, equipment and documentation in the EOF.

"d. Structurally built in accordance with the Uniform Building Code.

"e. Environmentally controlled to provide room air temperature, humidity and cleanliness appropriate for personnel and equipment.

"f. Provided with reliable voice and data communications facilities to the TSC and control room, and reliable voice communication facilities to OSC and to NRC, State and local emergency operations centers.

"g. Capable of reliable collection, storage, analysis, display and communication of information on containment conditions, radiological releases and meteorology sufficient to determine site and regional status, determine changes in status, forecast status and take appropriate actions. Variables from the following categories that are essential to EOF functions shall be available in the EOF:

(i) variables from the appropriate Table 1 or 2 of Regulatory Guide 1.97 (Rev. 2), and

(ii) the meteorological variables in Regulatory Guide 1.97 (Rev. 2) for site vicinity and regional data available via communication from the National Weather Service.

"h. Provided with up to date plant records (drawings, schematic diagrams, etc.), procedures, emergency plans and environmental information (such as geophysical data) needed to perform EOF functions.

"i. Staffed using Table 2 (previous guidance approved by the Commission) as a goal. Reasonable exceptions to goals for the number of additional staff personnel and response times for their arrival should be justified and will be considered by NRC staff.

"j. Provided with industrial security when it is activated to exclude unauthorized personnel and when it is idle to maintain its readiness.

"k. Designed taking into account good human factors engineering principles."

Vepco Response

Technical Support Centers (TSC)

a. TSC Status

Engineering of the new Technical Support Center at North Anna and Surry has been in progress since early 1980. The construction for the facility at Surry was in progress when in April 1982 the project was placed on hold to allow a review of the layout of the operational area. Due to the generic deadline for completion imposed by NUREG 0737, design and construction had proceeded in parallel until it became apparent that the original design might not support the functional requirements of the facility. The review, which resulted in some structural changes and in a total redesign of the layout of the operational area, was completed in December, 1982 when a human factors engineering review approved the revised layout. These modifications will require changes to the design and some reengineering. The TSC facility at North Anna and Surry have been designed in accordance with the BOCA Code, not the Uniform Building Code. VEPCO's plans to design the TSCs in accordance with the BOCA Code were outlined in VEPCO's June 1981 submittal, Serial No. 312. The requirement to design the TSCs in accordance with the Uniform Building Code was not included in NUREG-0696. VEPCO intends to complete construction of the TSCs, based on current design, as paragraph 3.7 of Supplement 1 to NUREG-0737 allows for previous work done in good faith.

The Reg. Guide 1.97 variables needed to perform the functions of the TSC will be provided by displays generated using the DAS and the ERCS installed to support the SPDS.

b. The TSCs will be habitable and available for use without full data communication capability during the fall of 1984. The TSC will become fully operational July 1, 1985 or the first refueling thereafter based on present plans for Surry and after the refueling in 1986 for North Anna. This can be accomplished provided the results of the review of the TSC function does not require data for which sensors do not exist.

Operational Support Centers

The Operational Support Centers presently in use at North Anna and Surry will not be modified with the exception of communications facilities that may be modified when other ERFs are upgraded.

Emergency Operations Facilities

The Emergency Operations Facilities presently in use at Vepco proved adequate during previous exercises and Vepco has previously submitted its plans with regards to EOFs. The plan submittals are being revised to provide a hardened local Emergency Operations Facility with protection factors which exceed the NRC proposed requirements and a central Emergency Operations Facility located at the corporate headquarters. This revision will be provided for the Commissioners approval. Within sixty days of receipt of NRC approval, VEPCO will provide a projected completion date for the EOFs.

INTEGRATED IMPLEMENTATION PLAN
GUIDELINES
FOR
VEPCO ERC PROJECTS

Attachment 3

Generic Letter 82-33 dated December 22, 1982, requested a description of plans for phased implementation and integration of emergency response activities. Vepco personnel have participated in the INPO NUTAC for Emergency Response Capability. The NUTAC has developed a draft "Guideline for an Integrated Implementation Plan." The draft "Guidelines for an Integrated Implementation Plan" is based on a classical approach with all activities initiated at the same time. Since Vepco has already initiated emergency response upgrades, the INPO document was used as guidance for developing an "Integrated Implementation Plan Guidelines for VEPCO ERC Projects" which is included as part of this attachment. A summary of Vepco's plan for integration follows.

SPDS

As indicated by the current status given in attachment 2, data acquisition and processing equipment have been procured and basic software development for the SPDS has begun. Actual definition of the SPDS CRT displays and selecting a location of the SPDS displays in the control room are currently under way. The Technical Guidelines and work performed to date on Emergency Operating Procedures will be utilized to define data to be displayed on the SPDS. A human factors consultant has been actively involved in the SPDS development and has already reported the acceptability of hardware and software thus far developed for the SPDS. Input from the human factors consultant, the SPDS users and the critical Safety Functions Safety Analysis will be used to complete the design of the SPDS. The hardware, CRT, installed in the control room will continuously monitor the five critical safety functions defined in Supplement 1 to NUREG 0737, and at the same time have the capability to display other plant data for day-to-day use by the control room personnel.

DCRDR

The program plan to be developed for the DCRDR will fully develop the integration with other ERC efforts. The timing of the actual DCRDR will be such that; SPDS display design is complete and can be verified by the review, modifications to the control room as a result of Reg. Guide 1.97 can be included in the review, and the generation of EOP's that will be used with these modifications are utilized in the control room survey. The intent of the DCRDR review is to verify that all modifications made in the control room satisfy the objective of improving the operators ability to cope with an accident should one occur. The schedule for implementing modifications required to address HED's, if any, will be negotiated with the NRC after submittal of the Summary Report at completion of the DCRDR.

Regulatory Guide 1.97 Application to Emergency Response Facilities

The review conducted in order to prepare the Regulatory Guide 1.97 Report will compare instrumentation installed in the plant with that specified by Regulatory Guide 1.97. When discrepancies are identified, the Technical Guidelines, Emergency Operating Procedures and Plant design features will be reviewed to determine if the missing variables is required to enhance the operators ability to monitor the course of an accident. Any additional data that should be displayed in the control room will be compared with the SPDS

design to determine if the SPDS could satisfy requirements. Data displayed on the SPDS - CRT in the control room (as part of the SPDS or a supplemental display utilizing SPDS hardware), or on continuous indicators added to the control board, will be reviewed as part of the DCRDR prior to finalizing the SPDS and display instrumentation location. Those 1.97 variables necessary to perform TSC and EOF functions will be displayed in these facilities.

EOP's

As stated above the EOP's or technical guidelines will provide input to the SPDS design, Reg. Guide 1.97 review and the DCRDR. The EOP's will be developed based on the Westinghouse Owners Group's Generic Technical Guidelines, the task analysis that was inherent in the development of these guidelines and a writers guide. EOP training and update programs will be defined in the Procedures Generation Package submittal. The package will also define the methods used to develop plant specific EOP's from the Generic Technical Guidelines and the verifications and validation process for the procedures. Procedure validation may be performed as part of the DCRDR or provide input to the DCRDR.

Emergency Response Facilities

Upgraded Emergency Response Facilities will be provided to meet Regulations and meet the functional requirements provided in Supplement 1 to NUREG 0737. Appropriate voice communication links between the TSC, OSC, EOF, State and local Governments, the NRC and the National Weather Service will be provided in the upgraded facilities. The data acquisition and processing system provided for the SPDS will also collect and process those variables necessary for performance of the TSC and EOF functions. Human factors principles, review of Emergency Plans and Emergency Plan Implementing Procedures will be utilized to define the data needed in the facilities. Those Regulatory Guide 1.97 variables necessary to perform TSC and EOF functions will be displayed in these facilities. Any new or existing data displayed in the control room that is necessary to perform TSC and EOF functions will be provided in these facilities.

FOREWORD

This Integrated Implementation Plan outline for ERC Projects has been developed using the guidance document developed by the INPO NUTAC on ERC.

The basic concepts will be applied to develop a complete integrated plan for Vepco projects. It should be noted that some diagrams developed here may require revision to better satisfy integration requirements.

SUMMARY

The purpose of this document is to outline an integrated implementation plan for Vepco's ERC projects. Detailed development of the plan will be performed as a part of each ERC project. The guidance provided in this document addresses the major elements involved in the implementation process and their interactions. The guidance developed by the INPO NUTAC on ERC has been utilized to develop this document. The NUTAC Guidance has been applied as it appropriately relates to the current status of each Vepco project.

For simplicity of presentation, the integrated implementation plan has been divided into the following sections:

- o Emergency Operating Procedures (EOPs)
- o Control Room Design Review (CRDR)
- o Regulatory Guide 1.97 (R.G. 1.97)
- o Safety Parameter Display System (SPDS)
- o Emergency Response Facilities (ERF)
- o Implementation

Each section describes the elements involved and how these elements interface. More importantly each section addresses the interactions between the EOPs, CRDR, R.G. 1.97, SPDS, and ERF that should be considered to ensure proper integration of all ERC elements.

SECTION 1

INTRODUCTION

1.1 SCOPE

This guidance identifies and describes the major elements and interfaces that will be considered when preparing a plan and schedule for the implementation of the integrated Emergency Response Capabilities provisions (NUREG-0737, Supplement 1).

1.2 BACKGROUND

Realizing the impact of Supplement 1 to NUREG-0737, an industry review group recognized the need for an ad hoc committee to provide guidance to the industry in implementing the provisions of Supplement 1 to NUREG-0737. Therefore, an industry meeting was held September 26, 1982 to determine whether there was sufficient utility interest and a recognized need to pursue an industry program to develop such guidance. The INPO NUTAC on ERC was formed and guidance documents prepared by the NUTAC are utilized here to develop guidelines for a Vepco intergrated plan.

1.3 OVERVIEW

The NRC document (Supplement 1 to NUREG-0737) defines various provisions that are directed toward enhancing the control room operator's ability to deal with emergency conditions. These provisions include installing a safety parameter display system, conducting a control room design review, upgrading plant-specific emergency operating procedures, implementing R.G. 1.97 instrumentation provisions, and providing emergency response facilities. As specified in Supplement 1 to NUREG-0737, coordination and integration must occur during the development and implementation of these provisions. This document provides guidance to assist in the development of an integrated implementation program.

The method developed by the NUTAC for the integration of the provisions of Supplement 1 to NUREG-0737 is shown in Figure 1-1. Figure 1-1 was drawn to illustrate the five major provisions (EOPs, CRDR, R.G. 1.97, SPDS, ERF) and the basic elements and interfaces for each provision that should be considered in the development of an integrated implementation plan. Each element and its relationship to other elements is discussed in the following sections.

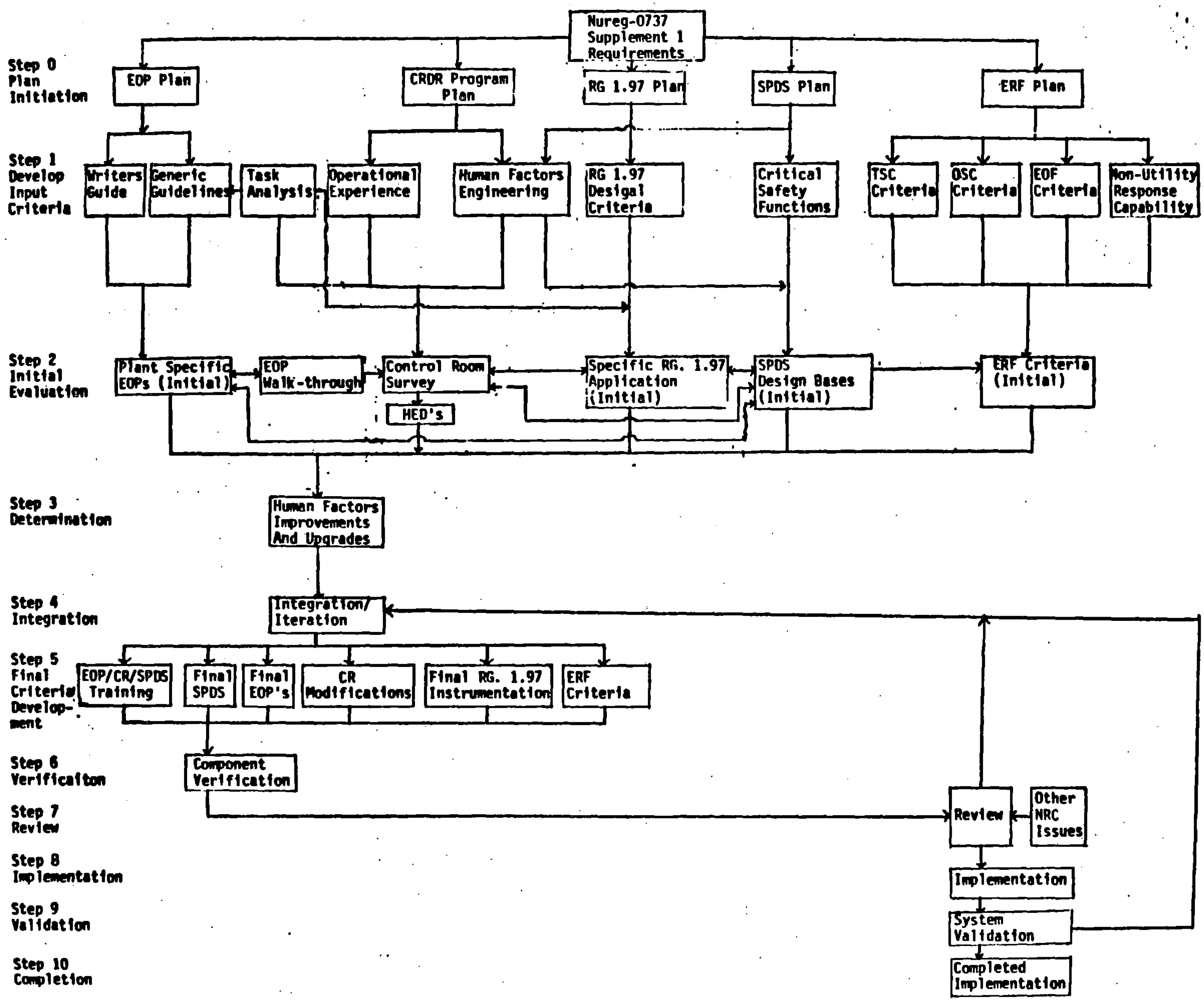


Figure 1-1
Integration method for supplement 1 to NUREG 0737 provisions

1.3.1 EOP Element

The Emergency Operating Procedure (EOP) Plan consists of tasks that will provide a documented method for developing, utilizing, revising, and controlling EOPs. The product will provide a framework that enhances an operator's ability to avoid degraded conditions.

These tasks should include defining source documents, determining manpower requirements, establishing a schedule, and specifying a method of document control. The plan should also define the interfaces with other elements of Supplement 1 to NUREG-0737 to ensure complete integration.

Initial plant-specific EOPs are developed for the purpose of mitigating the consequences of a broad range of initiating events and subsequent multiple failures, without the need to diagnose a specific event. These procedures are function-oriented and include human factors considerations that enhance human reliability. These initial EOPs are developed based upon a writer's guide and NSSS vendor generic technical guidances. NSSS generic technical guidance was prepared based on re-analysis of transients and accidents and included generic task analysis.

EOPs should be checked for completeness, understandability, technical accuracy, usability, and compatibility with the control room design. In order for operators to have confidence in the EOPs, all these criteria must be met. A walk-through of the initial EOPs provides a method for evaluating these criteria.

The EOP walk-through may be performed in the control room, in a simulator, using a mock-up of the control room, or any combination of the three. Prior to making this decision, resource availability and advantages versus disadvantages of each available evaluation method should be determined. Although Figure 1-1 indicates only one EOP evaluation (walk-through), this process should be conducted following any major modifications to the EOPs.

The EOP walk-through will provide input to the DCRDR.

EOPs may be refined or revised based on the impact of control room human engineering discrepancies (HEDs), specific application of R.G. 1.97 recommendations and SPDS design bases.

1.3.2 Control Room Design Review (CRDR) Element

The CRDR Plan is the first step toward performing a Control Room Design Review and provides methodology for performing the entire review. The objective of the Control Room Design Review is to provide an environment that supports and enhances the operator's ability to operate within the EOP framework.

The Control Room Survey is used for identification and documentation of existing equipment. This task can be done as part of the EOP walk-through.

The Operating Experience Review is performed to identify any operational problems resulting from design inefficiencies or identify any modifications to the control room which would enhance the ability of an operator to respond to an emergency condition, with consideration given to the importance of safety, impact to plant and personnel, and economics.

In performing the CRDR, accepted human factors principles should be used. Good human engineering practices should be incorporated in any control room design since the operator must interface with this equipment under abnormal, as well as normal, conditions.

The Control Room Survey should include EOP walk-through or utilize results from the EOP walk-through, and operating experience data, as well as human engineering criteria, to uncover any control room design problems. This survey should include, among other things, an assessment of control room layout, the control room environment, the usefulness of audible and visual alarms, the readability of displays, the adequacy of instrumentation, and the information recording and recall capabilities. This survey is essential to ensure that all human engineering discrepancies (HEDs) are found.

The operator's tasks and information requirements are validated by the EOP walk-through and provide input criteria to the control room survey process. The walk-through may be performed during the CRDR Survey.

Control room additions associated with the SPDS and incorporation of selected R.G. 1.97 modifications will be considered in the CRDR, along with changes resulting from other programs. Supplement 1 to NUREG 0737 also states that the CRDR should be used to verify SPDS parameters.

1.3.3 R.G. 1.97 Element

The R.G. 1.97 plan provides the administrative guidance needed to assess and document all aspects of R.G. 1.97 consideration. The objective of the R.G. 1.97 instrument provisions is to provide the data and information required by the operator. The data required are mandated by the operator's needs for effectively executing the EOP and ERF functions.

A complete set of criteria is developed in the R.G. 1.97 plan to form a basis for instrument selection. Utilizing the criteria, a plant specific list of accident monitoring instrumentation, qualification criteria and locations is developed.

The plant list also provides feedback to the control room survey and SPDS design basis. ERF design criteria may provide additional input to the plant list.

1.3.4 SPDS Element

The SPDS plan describes the tasks which will provide a method for developing, revising, assessing and implementing the safety parameter display system design bases, a method for documenting these efforts, and a method for implementing an SPDS. The objective of the SPDS is to provide a concise set of information to aid operating personnel in assessing plant safety status.

The plan will provide a description of each task involved and give the administrative guidance needed to perform the tasks, including defining source documents, determining manpower requirements, specifying vendor involvement, establishing a schedule, and specifying a method of configuration control. Interfaces with other Supplement 1 to NUREG-0737 elements should be clearly defined to ensure complete integration.

A list of human factors criteria pertaining to the SPDS should be developed as a basis for developing and assessing plant-specific SPDS designs. This list of criteria may be developed in conjunction with the human factors criteria required as input for the performance of a control room survey.

The EOPs, as a result of the efforts of the NSSS Owners Groups and plant-specific considerations, specify the critical safety functions for a plant. The SPDS design bases should incorporate this information so that the operator can use the SPDS, if available, in conjunction with the EOPs.

To ensure an effective SPDS, the design bases must specify hardware, inputs, and software, identify SPDS user(s), specify location, and define availability. The SPDS design is mandated by operator usability and compatibility with plant-specific EOPs.

Once the SPDS design bases have been determined, the adequacy of the design should be verified.

1.3.5 ERF Element

The ERF plan describes a method for designing, implementing, and utilizing the emergency response facilities. The purpose of an ERF is to support the operating personnel during degraded conditions.

The criteria that provide a basis for the design or upgrade of the Emergency Operating Facility (EOF), and Operational Support Center (OSC) need to be determined. The bases for these criteria should include consideration of 10 CRF 50.47, 10 CFR 50, Appendix E, NUREG-0696, Emergency Plans, and guidance provided by nuclear industry organizations.

1.4 THE ITERATION PROCESS

The EOP, CRDR, R.G. 1.97, SPDS and ERF elements should be involved in a iterative process that includes control room enhancements, plant-specific EOPs, specific R.G. 1.97 application, SPDS design, and ERF criteria. This iterative process should continue until all of the ERC criteria have been considered and satisfied.

SECTION 2

CONSIDERATIONS IN THE DEVELOPMENT OF AN INTEGRATED IMPLEMENTATION PROGRAM

OVERVIEW

This section identifies and discusses the factors that should be considered in the development of an integrated plant-specific implementation program. Each of these factors is discussed in detail in the following sections of this document.

2.1 DEFINITION OF ELEMENTS AND INTERFACES

An effective integrated implementation program requires a clear definition of each element involved and how these elements interface. The primary elements involved in this program include emergency operating procedures, control room design review, R.G. 1.97, safety parameter display system, and emergency response facilities. None of these elements are totally isolated from one another, as depicted in Figure 2-1. Because of their interrelationships, it is essential to clearly define the interfaces between the elements. Section 3 provides guidance in defining these elements and interfaces.

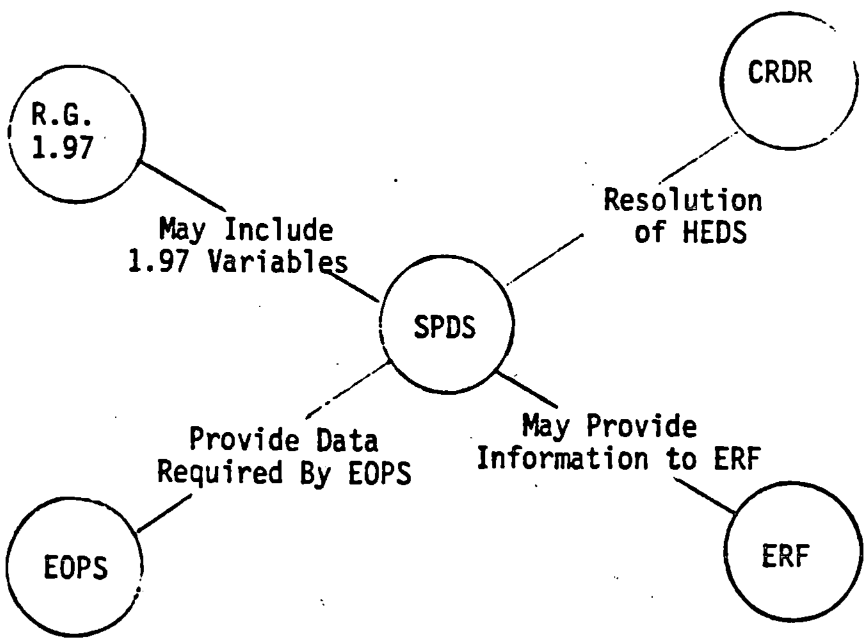


Fig. 2-1

SPDS Interfaces

SECTION 3

DEFINITION OF ELEMENTS AND INTERFACES

OVERVIEW

An effective integrated implementation program requires a clear definition of each element involved and how these elements interface. This section defines the elements and interfaces contained in Figure 1-1. In order to provide this information in a usable and logical format, this section has been divided into the following subsections:

- o Emergency Operating Procedures (EOPs)
- o Control Room Design Review (CRDR)
- o Reg. Guide 1.97 Provisions (R.G. 1.97)
- o Safety Parameter Display System (SPDS)
- o Emergency Response Facilities (ERF)
- o Implementation

Figure 3 illustrates which elements and interfaces will be included in each subsection.

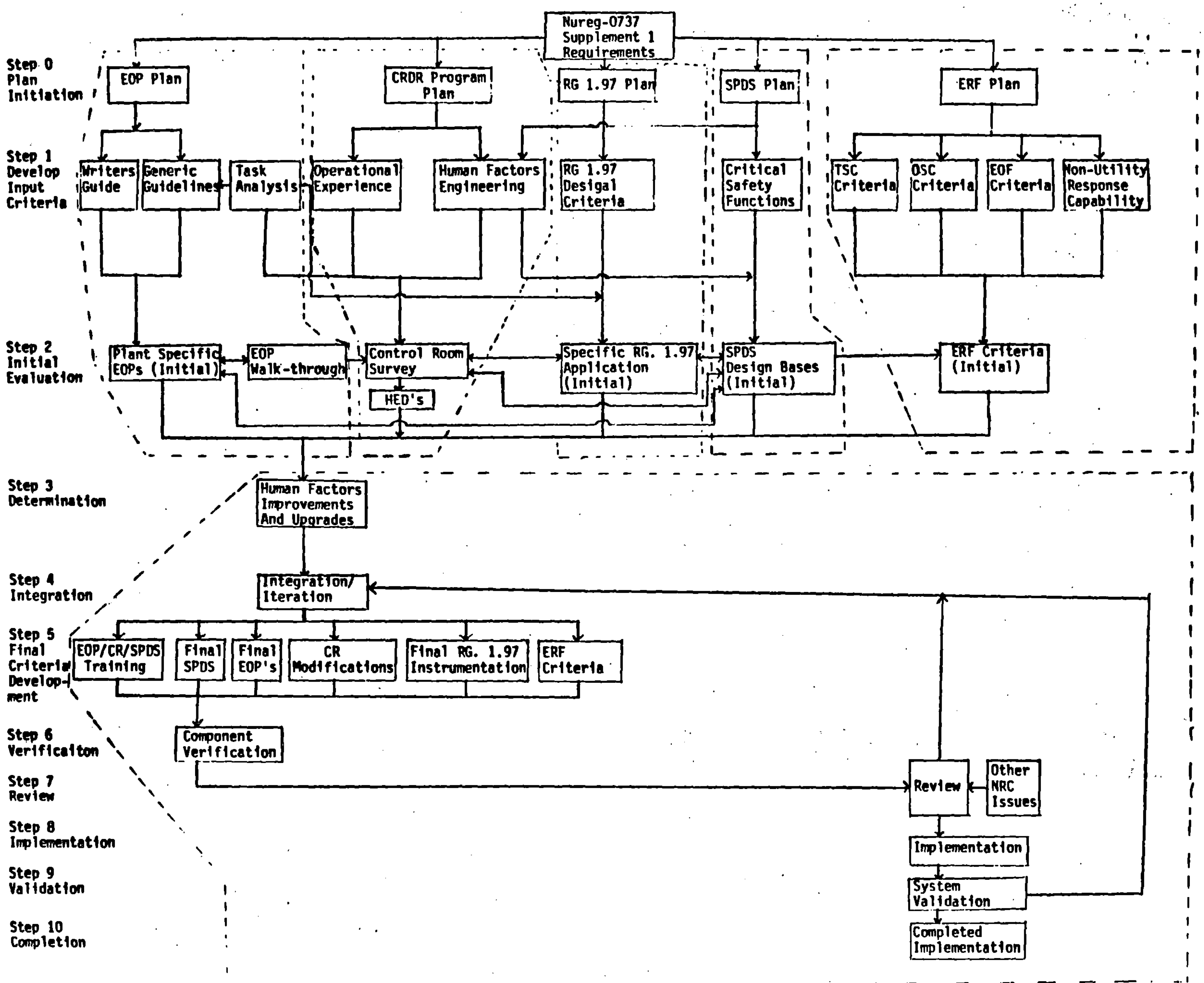

Nureg-0737 Supplement 1 Requirements
Step 0 Plan Initiation
EOP Plan
CRDR Program Plan
RG 1.97 Plan
SPDS Plan
ERF Plan
Step 1 Develop Input Criteria
Writers Guide
Generic Guidelines
Task Analysis
Operational Experience
Human Factors Engineering
RG 1.97 Desigal Criteria
Critical Safety Functions
TSC Criteria
OSC Criteria
EOF Criteria
Non-Utility Response Capability
Step 2 Initial Evaluation
Plant Specific EOPs (Initial)
EOP Walk-through
Control Room Survey
HED's
Specific RG. 1.97 Application (Initial)
SPDS Design Bases (Initial)
ERF Criteria (Initial)
Step 3 Determination
Human Factors Improvements And Upgrades
Step 4 Integration
Integration/ Iteration
Step 5 Final Criteria Development
EOP/CR/SPDS Training
Final SPDS
Final EOP's
CR Modifications
Final RG. 1.97 Instrumentation
ERF Criteria
Step 6 Verificaiton
Component Verification
Step 7 Review
Review
Other NRC Issues
Step 8 Implementation
Implementation
Step 9 Validation
System Validation
Step 10 Completion
Completed Implementation

Figure 3-

3.1 EMERGENCY OPERATING PROCEDURES (EOPs)

The elements and interfaces described in this section are detailed on Figure 3-1

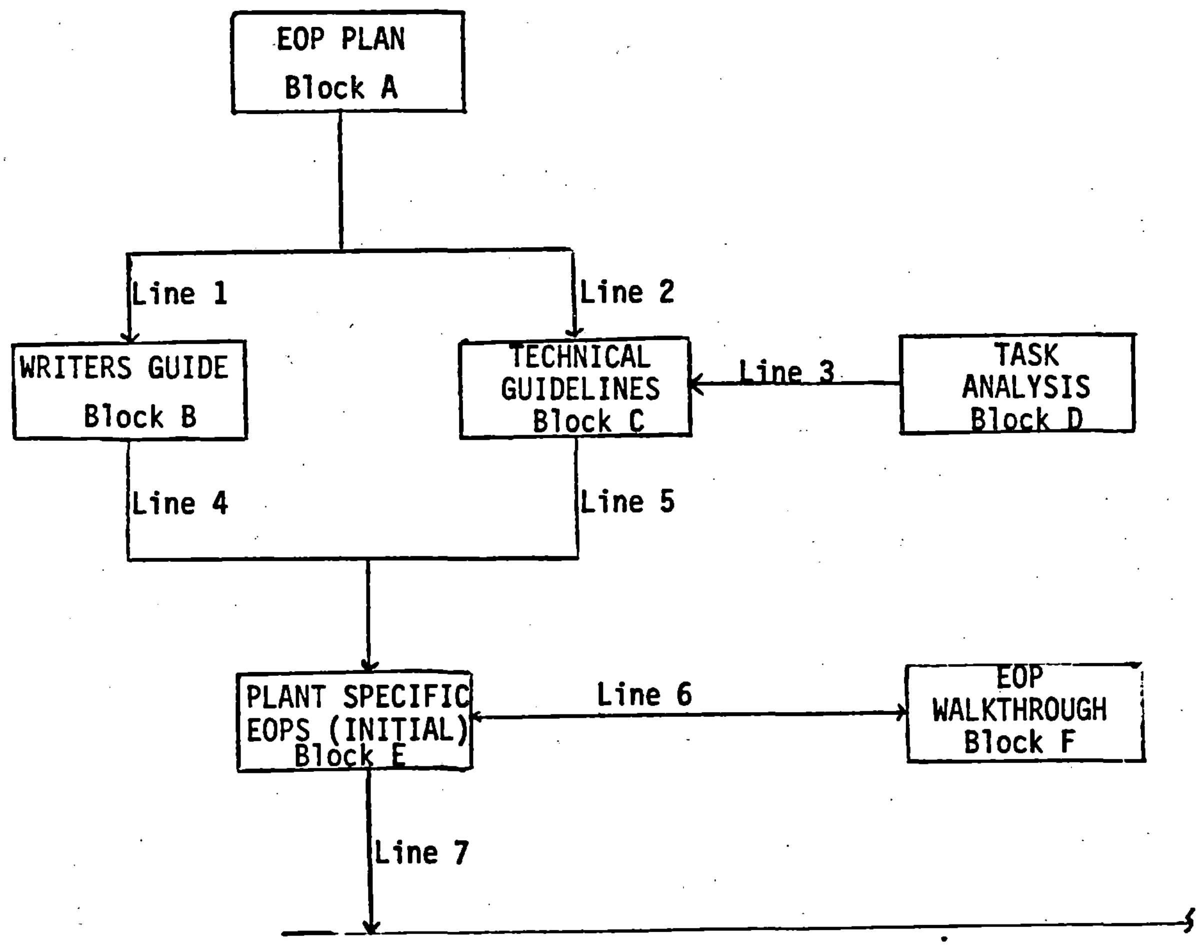


FIGURE 3-1
EOP ELEMENTS AND
INTERFACES

3.1.1 EOP Plan - Block A

NUTAC - Recommendations

A detailed EOP Plan is necessary for the development and implementation of Emergency Operating Procedures. The EOP Plan consists of those tasks which will provide a documented method for developing, utilizing, revising, and controlling Emergency Operating Procedures. The plan will be plant-specific, measurable in terms of deliverables, and consistent with current NRC guidance. The objective of the EOPs is to provide a framework to enhance the operator's ability to avoid degraded conditions.

References for Emergency Operating Procedures (Section 3.1)

"Emergency Operating Procedure Implementation Guideline" (INPO 82-016), EOPIA Review Group, June 1982

"Emergency Operating Procedures Generation Package Guideline" (INPO 83-007), EOPIA Review Group, February 1983

"Emergency Operating Procedures Writing Guideline" (INPO 82017), EOPIA Review Group, August 1982

"Emergency Operating Procedures Verification Guideline" (INPO 83-0XX), EOPIA Review Group, 1983

NUREG-0899, "Guidelines for the Preparation of Emergency Operating Procedures", USNRC, August 1982

NUREG-0660, "NRC Action Plan Developed as a Result of the TMI II Accident", NSNRC, May 1980

Plant-Specific Writer's Guide

Existing Plant-Specific Emergency Operating Procedures

NSSS Technical Guidelines

Generic Task Analysis

Supplement 1 to NUREG-0737, "Requirements for Emergency Response Capability", USNRC, December 1982

Vepco's EOP Plan will be described in the procedures generation package that will be submitted to the NRC.

3.1.2 Writer's Guide - Block B

The Writer's Guide should be developed to define specific methods for preparing plant-specific EOPs from technical guidelines. Vepco will use guidance documents developed by the INPO EOPIA NUTAC in combination with the WOG technical guidelines to develop EOPs.

3.1.3 Technical Guidelines - Block C

The purpose of these guidelines is to provide a technical foundation for the plant-specific EOPs. These guidelines identify operator tasks, information needs, and control functions required for emergency operating conditions. Vepco will use Technical Guidelines prepared by the WOG.

3.1.4 Task Analysis - Block D

The Task Analysis consists of an analysis of transients and accidents to identify operator tasks and information and control needs to avoid degraded conditions. This Task Analysis was performed and has been used by the Westinghouse Owners Group as a basis for the emergency operating procedure Technical Guidelines. Technical Guidelines can be used as input criteria for the Control Room Design Review and the R.G. 1.97 design.

3.1.5 Plant Specific EOPs (Initial) - Block E

Plant-specific EOPs are developed for the purpose of mitigating a broad range of initiating events and subsequent multiple failures or operator errors without the need to diagnose a specific event. These procedures are function-oriented and written with human factors considerations (provided by the Writer's Guide) to enhance human reliability.

3.1.6 EOP Walk-Through - Block F

EOPs should be checked for completeness, understandability, technical correctness, usability, and compatibility with the control room. EOP walk-through is a method used to provide assurance that the procedures are adequate.

The EOP walk-through may be performed as part of the EOP development process alone or with the CRDR. Due to the timing of the CRDR it may be necessary to perform a walk-through with both the CRDR and the EOP development process. The EOP walk-through serves as validation of the EOP. Initial validation has been completed by the W.O.G. This is documented in WCAP 10204, Summary Report for Emergency Response Guideline Validation Program.

3.1.7 EOP Plan/Writer's Guide Interface - Line 1

The EOP Plan should describe the development and use of a writer's guide in the preparation of the plant-specific EOPs. This is defined in the Procedures Generation Package.

3.1.8 EOP Plan/Technical Guidelines Interface - Line 2

The EOP Plan should describe the use of the WOG Technical Guidance in the preparation of the plant-specific EOPs.

3.1.9 Task Analysis/Technical Guidelines Interface - Line 3

The results of the Task Analysis performed by Westinghouse was used by the WOG as a basis for the Technical Guidelines. It should be noted that preparation of Technical Guidelines in this manner included a generic Task analysis and a plant specific task analysis is not required.

3.1.10 Writer's Guide/Plant-Specific EOPs (Initial) Interface - Line 4

The Writer's Guide is used in the development of the plant-specific EOPs.

3.1.11 Technical Guidelines/Plant-Specific EOPs (Initial) Interface - Line 5

The WOG Technical Guidelines are used as the basis for preparing the plant-specific EOPs.

3.1.12 Plant-Specific EOPs (Initial) and EOP Walk-Through Interface - Line 6

Plant-specific EOPs are used to perform an EOP walk-through. The results of the EOP walk-through may identify potential modifications as input to the CRDR.

3.1.13 EOP/Iteration Interface - Line 7

The initial generation of EOPs will be revised as ERC projects provide modifications to the control room i.e., SPDS, 1.97 and potential DCRDR modifications.

3.2 Control Room Design Review (CRDR)

The elements and interfaces described in this section are detailed on Figure 3-2.

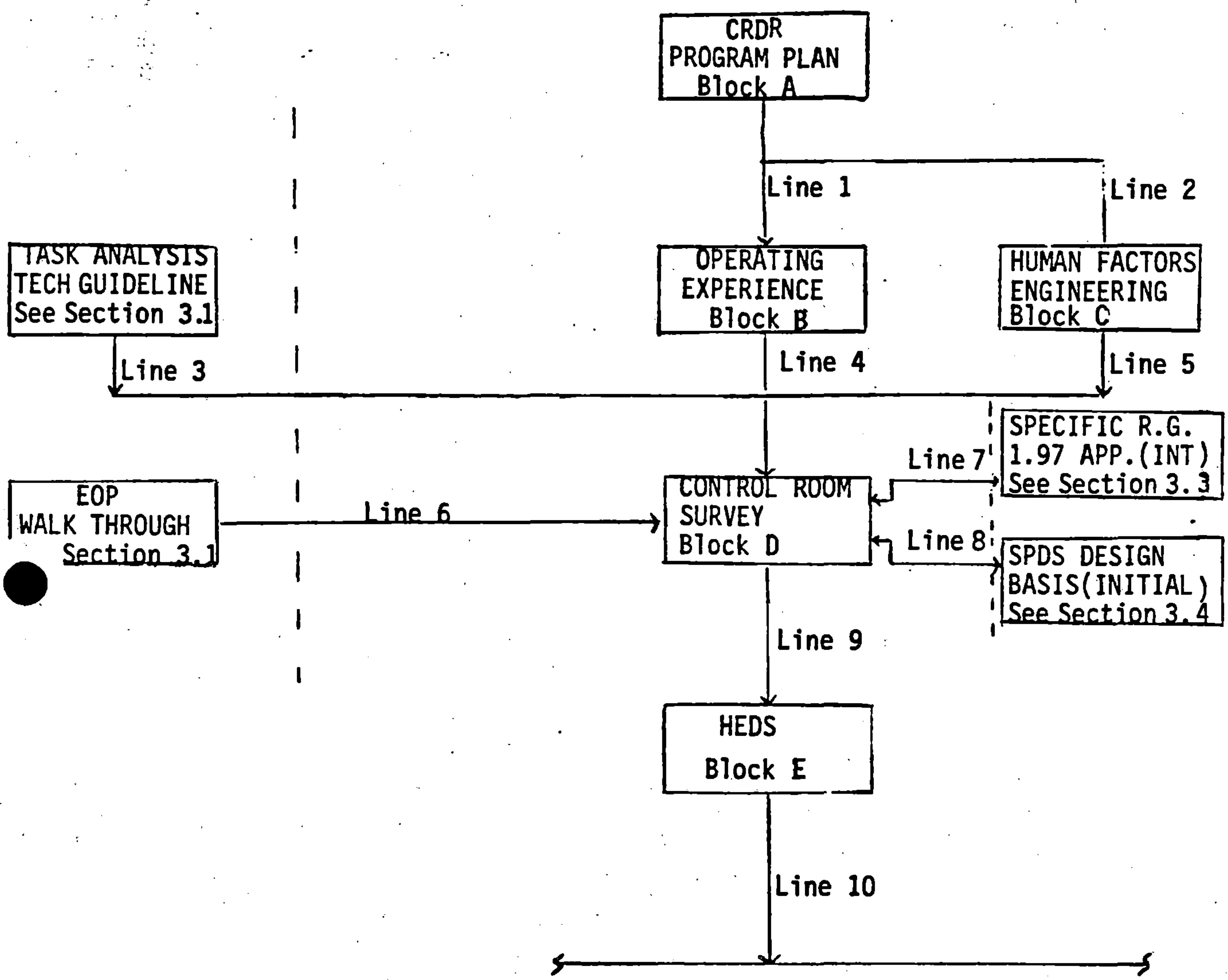


Figure 3-2

CRDR PROGRAM PLAN

3.2.1 CRDR Program Plan - Block A

The timing of the control room design review is critical to integration with other ERC projects (i.e., SPDS design, EOPs, and Reg. Guide 1.97). The overall control room design review should start only after it can take into account all other ERC design changes that are occurring in the control room. In addition the impact of these changes should be assessed against the initial Emergency Operating Procedures such that the generation of EOPs that will be used in the CRDR walk-through includes/account for planned modifications. Thus, in order for the CRDR to fully support integration with other efforts the actual review will include existing control room conditions along with planned modifications that have reached final design. The total assessment will be by design drawings (possible simulator mock-ups) and pre ERC-implementation control room conditions.

A CRDR Program Plan is required for NRC submittal and will define administrative guidance necessary to develop, perform, assess and document all aspects of the CRDR review. This includes establishing the review program and schedule, providing criteria for assessment of HEDs, and methods for correction. The detailed plan when developed will clearly identify all interfaces with other ERC efforts.

References for Control Room Design Review (Section 3.2)

Supplement 1 to NUREG-0737, "Requirements for Emergency Response Capabilities, USNRC, December, 1982

"Control Room Design Review Implementation Guideline", draft (INPO 83-OXX NUTAC), CRDR NUTAC

"Control Room Design Review Survey Development Guideline", draft (INPO 83-OXX NUTAC), CRDR NUTAC

"Human Factors Principles for Control Room Design Review", draft (INPO 83-OXX NUTAC), CRDR NUTAC

"Control Room Design Review Task Analysis Guideline", draft (INPO 83-OXX NUTAC), CRDR NUTAC

"INPO/TVA Pilot Systems Review" (INPO 82-014), INPO and TVA, June, 1982

NUREG-0700, "Guidelines for Control Room Design Review", USNRC, September, 1981

NUREG-0801, "Evaluation Criteria for Detailed Control Room Design Reviews", USNRC (draft for comment)

3.2.2 Operating Experience - Block B

Operating experience assists in identifying any operational problems resulting from design discrepancies. Available operating experience should be used to further ensure that the review is complete.

Various resources are available for the accumulation of operating experience data. The most important source of data is the operating personnel. This information may be obtained through operator interviews and reviews of industry experience, including LERs.

3.2.3 Human Factors Engineering - Block C

Human factors engineering principles should be used to determine if the design of the control room incorporates good human engineering practices. Various human engineering principles have been provided by the military, NRC, and other nuclear industry related organizations. As part of the detailed program plan, a composite list of those principles pertaining to nuclear power plant control rooms should be developed as a basis for the determination of HEDs.

3.2.4 Control Room Survey - Block D

The Control Room Survey is a static verification of the control room performed by comparing the existing instrumentation layout, lighting, and noise levels with accepted human engineering principles. This survey will utilize EOPs/task analysis, operating experience data, and human engineering principles to uncover any control room design problems. This survey should

include, among other things, an assessment of control room layout, the control room environment, and usefulness of audible and visual alarms, the readability of displays, the adequacy of instrumentation, and the information recording and recall capabilities.

3.2.5 Human Engineering Discrepancies (HEDs) - Block E

HEDs are characteristics of the existing control room that do not comply with accepted human engineering principles. HEDs must clearly be defined and enough information must be provided to allow some type of prioritization. This establishes a good basis for assessing the significance of a HED and determining whether some modification is required.

The criteria for prioritizing may include, among other things, impact on plant safety, impact on plant availability, involvement in previous operating events, economics, and impact of modification on plant operation.

3.2.6 CRDR Program Plan/Operating Experience Interface - Line 1

The CRDR program plan should require the accumulation and evaluation of operating experience data.

3.2.7 CRDR Program Plan/Human Factors Engineering Interface - Line 2

The CRDR program plan should describe the use of existing human factors engineering principles as a basis for determining HEDs. This will be accomplished in the Program Plan development.

3.2.8 Task Analysis (Technical Guidelines)/Control Room Survey - Line 3

The information and control needs identified in the Task Analysis (Technical Guidelines) (See Section 3.1.4), provide input criteria for the control room survey. The results of the Task Analysis are compared with the control room instrumentation during the survey process to determine if any enhancement of displays or controls is needed.

3.2.9 Operating Experience/Control Room Survey Interface - Line 4

The results of the operating experience review provide additional information for review in the control room survey.

3.2.10 Human Factors Engineering/Control Room Survey - Line 5

The human factors engineering principles developed for nuclear power plant control rooms and criteria developed in the plan provide the basis for identifying HEDs in the control room survey.

3.2.11 EOP Walk-Through/Control Room Survey Interface - Line 6

The operator's tasks and informational requirements validated by the EOP walk-through (see Section 3.1.6) provide input criteria to the control room survey process. This interface is classified as one-way. The results of an EOP walk-through may provide valuable information to the control room survey, but the survey has no effect on the walk-through.

If EOP production has been completed or is performed in parallel with the CRDR program, it may be possible to utilize the EOPs and information resulting from the walk-through instead of task analysis as input to the control room survey.

3.2.12 Specific R.G. 1.97 Application/Control Room Survey Interface - Line 7

Enhancements required by R.G. 1.97 provisions (see Section 3.3.3) should be considered during the control room survey. The instrumentation list prepared as part of the Reg. Guide 1.97 study may be used to supplement the control room survey. The timing of the control room survey and EOP walk-throughs should be such that 1.97 modifications planned in the control room are a part of the total CRDR process.

3.2.13 SPDS Design/Control Room Survey Interface - Line 8

Enchancements associated with the SPDS (see Section 3.4.4) may be included in the control room survey. This could minimize control board modifications.

3.2.14 Control Room Survey/HED Interface - Line 9

The evaluation of the results of the control room survey should provide a list of HEDs. It is important to incorporate final Reg. Guide 1.97 design, SPDS/PDS design and know modifications to EOPs associated with SPDS and Reg. Guide 1.97 in the CRDR to minimize the HEDs.

3.2.15 HED/Iteration Interface - Line 10

The control room enhancements should be coordinated with changes resulting from other programs, such as EOP, R.G. 1.97, SPDS, and ERF. The iteration with the CRDR program is an ongoing process as long as design changes are made to any of the other basic elements.

3.3 Regulatory Guide 1.97

The elements and interfaces described in this section are detailed on Figure 3-3

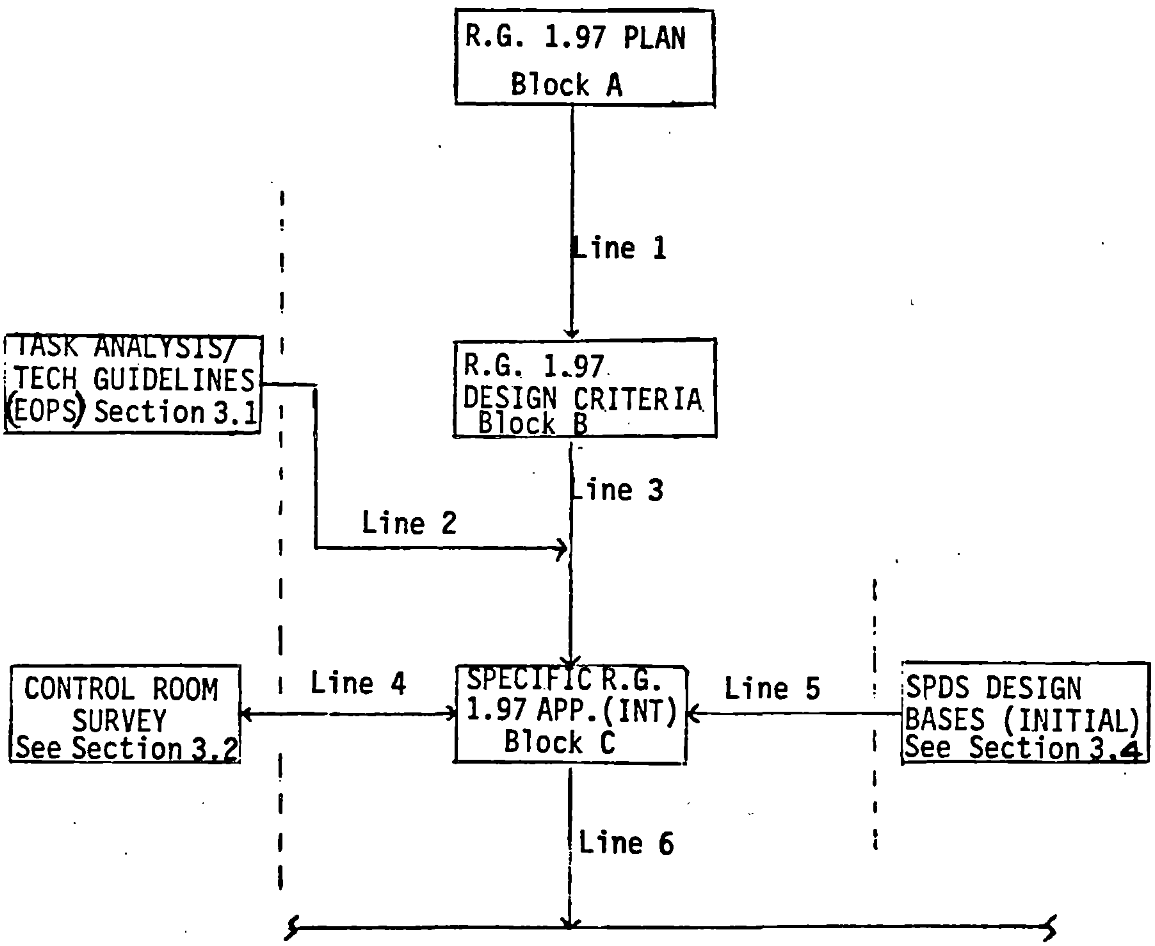


Figure 3-3
Reg. Guide 1.97
ELEMENT AND INTERFACES

3.3.1 Reg. Guide 1.97 Plan - Block A

Studies which provide a comparison of plant instrumentation with Reg. Guide 1.97 requirements are in progress. These studies will be completed. The results of these studies should be compared with the ongoing Equipment Qualification and Appendix R Work, the SPDS design, and the Task Analysis performed by Westinghouse as reflected in the Technical Guidelines. Appropriate Integration with the SPDS design may minimize the number of modifications required on the control board. When all design bases are established and modifications required for 1.97 are known, schedules can be defined.

References for Regulatory Guide 1.97 (Section 3.3)

"Accident Monitoring Instrumentation Implementation Guideline", draft (INPO 83-0XX NUTAC), ERC NUTAC

"Comments on Regulatory Guide 1.97", AIF Committee on Power Plant Design, Construction and Operation, February 1982

"Report on Emergency Response Facility Accident Monitoring", AIF Safety Parameter Integration Subcommittee to the Policy Committee on Nuclear Regulation, November 1980

"A PRA-Based Approach to Establishing Priorities for Equipment Qualification Needs", D.E. Leaver, W.A. Brinsfield, R.N. Kubik, presented at the International Meeting on Thermal Nuclear Reactor Safety, August 1982

ANS 4.5-1980, "American National Standard Criteria for Accident Monitoring Functions in Light-Water-Cooled-Reactors," December 1980.

IEEE Std 323-1974, "IEEE Standard for Qualifying Class 1E Equipment for Nuclear Power Generating Stations", IEEE, 1974 IEEE Std 497-1981, "IEEE Standard Criteria for Accident Monitoring Instrumentation for Nuclear Power Generation Stations", IEEE, 1981

IEEE Std 344-1975, "IEEE Recommended Practice for Seismic Qualification of Class 1E Equipment for Nuclear Power Generating Stations", IEEE, 1975

EPRI Report NP-2110, "On-Line Power Plant Signal Validation Technique Utilizing Parity-Space Representation and Analytic Redundancy", EPRI, November 1981 (an alternate to hardwired redundancy)

EGG-EE-6043, "Preliminary Recommendations for Changes to Regulatory Guide 1.97, Revision 2", September 1982

ANSI N13.1, "Guide to Sampling Airborne Radioactive Materials in Nuclear Facilities", 1969

Regulatory Guide 1.97, Revision 2, "Instrumentation for Light-Water-Cooled Nuclear Power Plants to Assess Plant and Environs Conditions During and Following an Accident", USNRC, December 1980

"Code of Federal Regulations", Title 10, Part 50, Appendix A, General Design Criteria 13, 19, and 64, published by the Office of the Federal Register, National Archives and Records Service, General Services Administration

Generic Letter 82-09, "Environmental Qualification of Safety-Related Electrical Equipment", USNRC, April 1980

Memorandum and Order CLI-80-21, USNRC, May 1980

NUREG-0588, Revision 1, "Interim Staff Position on Environmental Qualification of Safety-Related Electrical Equipment", USNRC, July 1981

Division of Operating Reactors, "Guidelines for Evaluating Environmental Qualification of Class 1E Electrical Equipment in Operating Reactors" (DOR Guidelines), USNRC, November 1979 IE Bulletin 79-01B, "Environmental Qualification of Class 1E Equipment", USNRC, October 1980

NUREG/CR-1440, "Light Water Reactor Status Monitoring During Accident Conditions", USNRC, June 1980

Supplement 1 to NUREG-0737, "Requirements for Emergency Response Capabilities", USNRC, December 1982

Owners Groups positions

3.3.2 R.G. 1.97 Design Criteria - Block B

An accurate assessment of plant instrumentation is needed to develop a set of explicit design and qualification criteria for necessary accident instrumentation.

3.3.3 Specific R.G. 1.97 Application (Initial) - Block C

Completion of the studies above will define those plant parameters and loops that require modification or new loops that may be needed.

3.3.4 R.G. 1.97 Plan/R.G. 1.97 Design Criteria Interface - Line 1

The R.G. 1.97 Plan will describe the development of criteria for the determination of required accident-monitoring instrumentation.

3.3.5 Task Analysis(Technical Guidelines (EOPs)/Specific R.G. 1.97 Application (Initial) Interface - Line 2

The information needed to mitigate degraded conditions is identified in the Task Analysis/Technical Guidelines (EOPs) (see Section 3.1.4) and should be in the specific R.G. 1.97 application.

3.3.6 R.G. 1.97 Design Criteria/Specific R.G. 1.97 Application (Initial) Interface - Line 3

The R.G. 1.97 provisions should provide a basis for developing a plant-specific instrument list.

3.3.7 Control Room Survey/Specific R.G. 1.97 Application (Initial) Interface - Line 4

Any modifications initiated as a result of specific R.G. 1.97 application should be input to the control room survey (see Section 3.2.4). The instrumentation identified as part of the Reg. Guide 1.97 listings can be used for Control Room survey.

3.3.8 SPDS Design Basis/Specific R.G. 1.97 Application (Initial) Interface - Line 5

This interface is classified as two-way. Credit may be taken for information displayed by the SPDS that meets R.G. 1.97 provisions on the plant-specific R.G. 1.97 instrument list.

3.3.10 Specific R.G. 1.97 Application (Initial)/Iteration Interface - Line 6

The specific R.G. 1.97 application should consider all changes associated with EOPs, control room improvements, SPDS design, and ERF design. This iteration is an ongoing process as long as R.G. 1.97 design changes are made to any of the other program deliverables. This interactive process provides essential coordination between R.G. 1.97 application and other NUREG-0737, Supplement 1, elements. Changes in any of these elements may impact the incorporation of R.G. 1.97 provisions.

3.4 Safety Parameter Display System (SPDS)

The elements and interfaces described in this section are detailed on Figure 3-4.

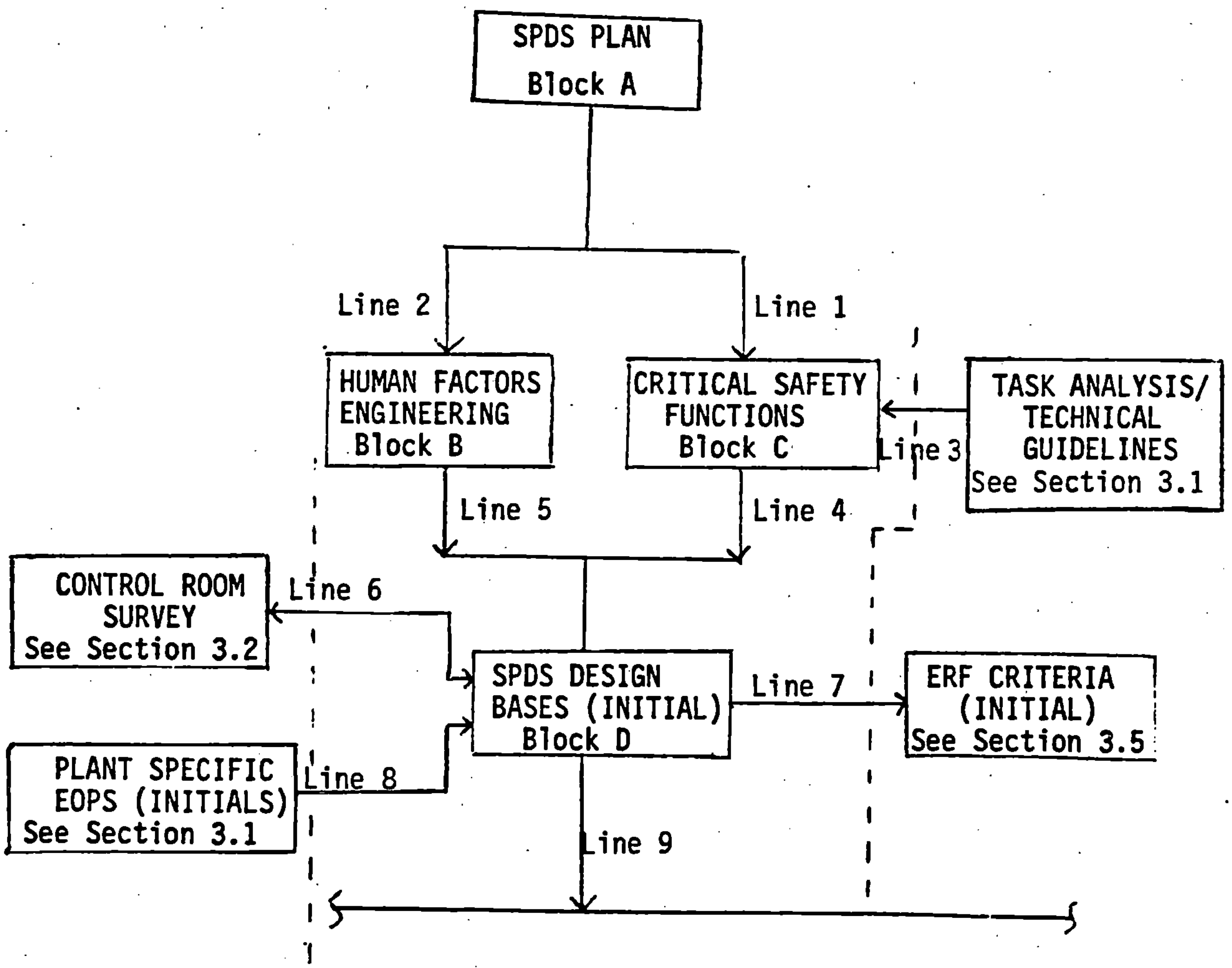


FIGURE 3-4
SPDS ELEMENTS
AND INTERFACES

3.4.1 SPDS Plan - Block A

A detailed SPDS plan is essential to the development and implementation of a Safety Parameter Display System. The SPDS plan should provide guidance for developing the design bases needed to provide a concise display of critical plant variables to the operating personnel to aid them in rapidly and reliably determining the safety status of the plant. This plan should provide a description of each task involved, including definition of source documents, establishing a schedule, and specifying a method of configuration.

References for Safety Parameter Display System (Section 3.4)

"Guidelines for an Effective SPDS Implementation Program", (INPO 83-003), SPDS NUTAC, January, 1983

"A Parameter Set for a Nuclear Plant Safety Console" (NSAC/10), NSAC

"Fundamental Safety Parameter Set for Boiling Water Reactors" (NSAC/21), NSAC

"Verification and Validation of Safety Parameter Display Systems" (NSAC/39), NSAC

NUREG-0835, "Human Factors Acceptance Criteria for Safety Parameter Display System", USNRC, October 1981 (draft for comment)

3.4.2 Human Factors Engineering - Block B

A description of human factors principles should be generated as a basis for developing and assessing SPDS designs.

3.4.3 Critical Safety Functions - Block C

Critical safety functions should be developed based on function-oriented EOPs if available. If not, the Technical Guidelines may be utilized.

3.4.4 SPDS Design Bases (Initial) - Block D

The design basis should define the minimum information required to be displayed on an SPDS, the users, the location, and the availability.

3.4.5 SPDS Plan/Critical Safety Function Interface - Line 1

The SPDS plan should describe the task of determining critical safety functions as part of the development of a SPDS. The method for safety analysis should be defined.

3.4.6 SPDS Plan/Human Factors Engineering - Line 2

The SPDS Plan should consider human factors principles.

3.4.7 Critical Safety Function/Task Analysis (Technical Guidelines (EOPs)

The Technical Guidelines are utilized to determine critical safely functions if validated EOPs are not available.

3.4.8 Critical Safety Functions/SPDS Design Bases (Initial) Interface - Line 4

The defined critical safety functions will be included in the initial SPDS design bases.

3.4.9 Human Factors Engineering/SPDS Design Bases - Line 5

Human factors principles will be utilized in giving consideration to the goal of SPDS usability.

3.4.10 Control Room Survey/SPDS Design Bases (Initial) Interface - Line 6

Enhancements associated with the SPDS may be included in the control room survey (see Section 3.2.4). This interface may be classified as one-way or two-way depending upon the intended use of the SPDS. If the SPDS is used only to fulfill minimum SPDS requirements, the interface is one-way. If it is intended to enhance existing control boards and reduce inefficiencies by incorporating additional information on the SPDS, the interface is two-way.

3.4.11 SPDS Design Bases (Initial)/ERF Criteria (Initial) Interface - Line 7

Consideration may be given to locating an SPDS in one or more of the ERFs and the resulting impact on the ERF design.

3.4.12 Plant Specific EOPs (Initial)/SPDS Design Bases (Initial) Interface - Line 8

Consideration should be given to how operating personnel utilize the SPDS in relation to EOPs (see Section 3.1.5). During the control room design review EOPs will be used to verify the SPDS.

3.4.13 SPDS Design Bases (Initial)/Iteration Interface - Line 9

The SPDS design bases should be coordinated with changes resulting from other programs, such as EOP, R.G.1.97, HED, and ERF. The iteration with the SPDS is an ongoing processas long as design changes are being made to any of the other basic elements.

3.5 Emergency Response Facilities (ERF)

The elements and interfaces described in this section are detailed on Figure 3-5.

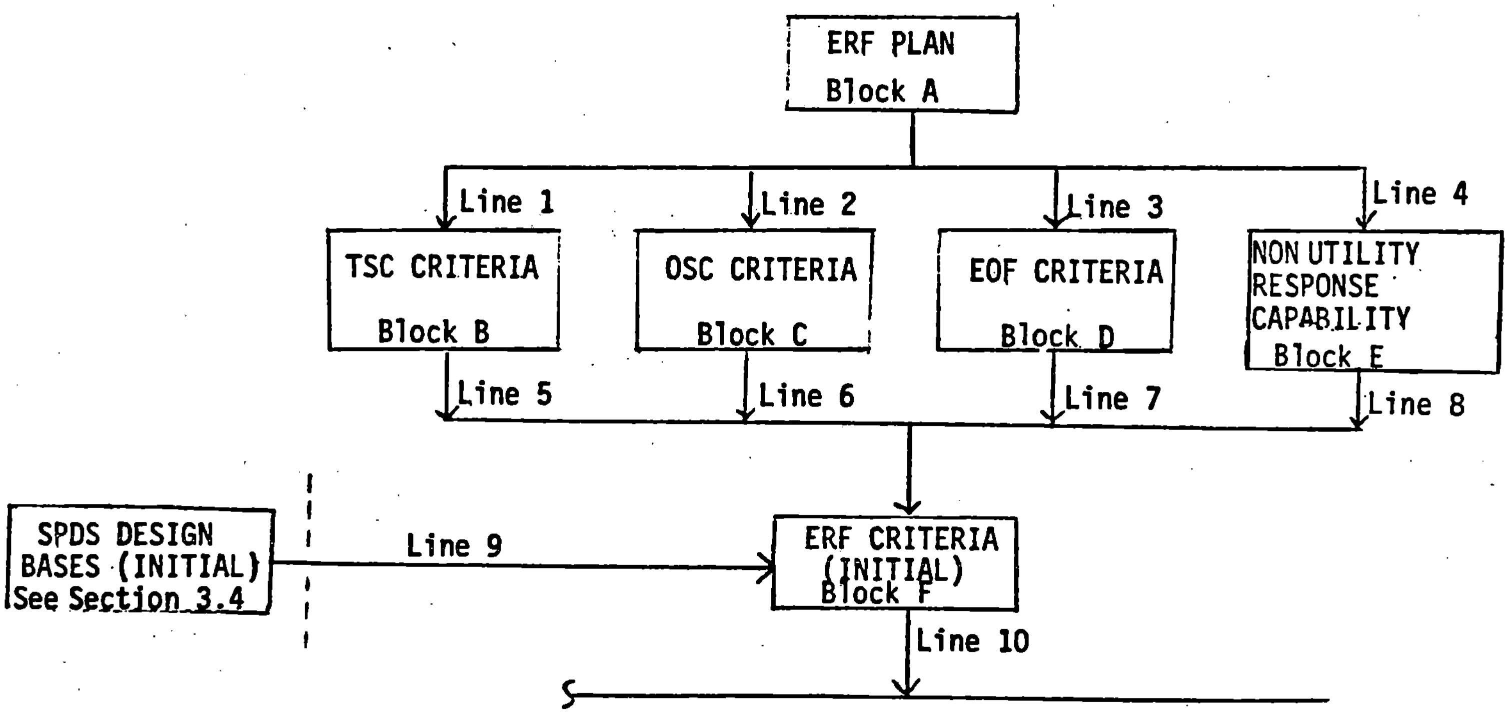


FIGURE 3-5
ERF ELEMENTS
AND INTERFACES

3.5.1 ERF Plan - Block A

A detailed ERF plan is important to the design and implementation of the emergency response facilities. This plan should provide a description of each task involved and the administrative guidance required to perform these tasks. This would include defining source documents, determining manpower requirements, specifying the vendors involved, establishing a schedule, and specifying a method of configuration control.

References for Emergency Response Facilities

"Guidelines for Implementation of Emergency Response Facilities", draft (INPO 83-0XX NUTAC), ERC NUTAC

"Code of Federal Regulations", Title 10, Part 50, Section 47 (Emergency Plans), op cit

"Code of Federal Regulations", Title 10, Part 50, Section 50 (Conditions of Licensees), op cit

"Code of Federal Regulations", Title 10, Appendix E (Emergency Planning and Preparedness for Production and Utilization Facilities), op cit

NUREG-0654, "Criteria for Preparation and Evaluation of Radiological Emergency Response Plans and Preparedness in Support of Nuclear Power Plants", USNRC, November 1980.

Supplement 1 to NUREG-0737, "Requirements for Emergency Response Capability", USNRC, December 1982

NUREG-0696, "Functional Criteria for Emergency Response Facilities", USNRC

NUREG-0814, "Methodology for Evaluation of Emergency Response Facilities" (Draft Report for Comment), USNRC, August 1981

NUREG-0818, "Emergency Action Levels for Light Water Reactors" (Draft Report for Comment), USNRC, October 1981

Eisenhut letter to licensees, 9/13/79, request for committment to meet requirements

Denton letter to licensees, 10/30/79, clarification of requirements

Eisenhut letter to licensees, 2/18/81, Commission-approved guidance on location, habitability, and staffing for ERFs. Request and deadline for submittal of conceptual designs.

COMJA-80-37, 1/21/81, Commission-approved guidance on EOF location and habitability

Secretary Memorandum 81-19, 2/19/83, Commission approval of NUREG-0696 as general guidance only

Utility Emergency Plans

Non-Utility Emergency Plans

3.5.2 Technical Support Center (TSC) Criteria - Block B

TSC criteria have been developed and incorporated into the design through review of the station emergency plan and the EPIPs. Data requirements are yet to be developed but will be developed in the same manner.

3.5.3 Operational Support Center (OSC) Criteria - Block C

OSC criteria are provided in the present Emergency Plan and no change in intended.

3.5.4 Emergency Operating Facility (EOF) Criteria - Block D

The Station and Corporate emergency plans will be utilized to develop the EOF criteria.

3.5.5 Non-utility Response Capability - Block E

Criteria must be developed to enable personnel to identify and assess interactions between utility and non-utility personnel during emergency conditions. These criteria may impact the design or manning requirements of the emergency response facilities. The bases for these criteria should include NRC regulatory requirements and utility and non-utility emergency plans.

3.5.6 Emergency Response Facility (ERF) Criteria (Initial) - Block F

The criteria developed for the TSC, OSC, EOF, and non-utility response capabilities should be evaluated to ensure the integration of all emergency response facilities.

3.5.7 ERF Plan/TSC Criteria Interface - Line 1

This activity has been completed for the facility but needs development for data displays.

3.5.8 ERF Plan/OSC Criteria Interface - Line 2

This is defined in the current emergency plan.

3.5.9 ERF Plan/EOF Criteria Interface - Line 3

This has been completed based on Corporate and Station Emergency Plans.

3.5.10 ERF Plan/Non-Utility Response Capability Interface - Line 4

This is defined in the Emergency Plans specifying how we interface with state and local governments and the NRC.

3.5.11 TSC Response Capability Criteria/ERF Criteria (Initial) Interface - Line 5

TSC criteria should be used to determine initial integrated ERF criteria, including resolution of deficient criteria, resources available, and planning objectives.

3.5.12 OSC Response Capability Criteria/ERF Criteria (Initial) Interface - Line 6

OSC criteria should be used to determine initial integrated ERF criteria, including resolution of deficient criteria, resources available, and planning objectives.

3.5.13 EOF Response Capability Criteria/ERF Criteria (Initial) Interface - Line 7

EOF criteria should be used to determine initial integrated ERF criteria, including resolution of deficient criteria, resources available, and planning objectives.

3.5.14 Non-Utility Response Capability Criteria/ERF Criteria (Initial) Interface - Line 8

Non-utility criteria should be used to determine initial integrated ERF criteria, including resolution of deficient criteria, resources available, and planning objectives.

3.5.15 SPDS Design Bases (Initial)/ERF Criteria (Initial) Interface - Line 9

Design bases for SPDS should be considered in the design and layout of the ERFs, as appropriate.

3.5.16 ERF Criteria (Initial)/Iteration Interface - Line 10

The ERF criteria should consider changes resulting from control room improvements, incorporation of upgraded EOPs, addition of accident monitoring instrumentation, and SPDS installation.

3.6 IMPLEMENTATION

The elements and interfaces described in this section are detailed on Figure 3-6. Although all of the lines illustrated in this figure are classified as interfaces, their descriptions are evident from the information provided for each element. As a result, this section provides separate guidance on only two interfaces that deal with an iteration process (see lines 1 & 2).

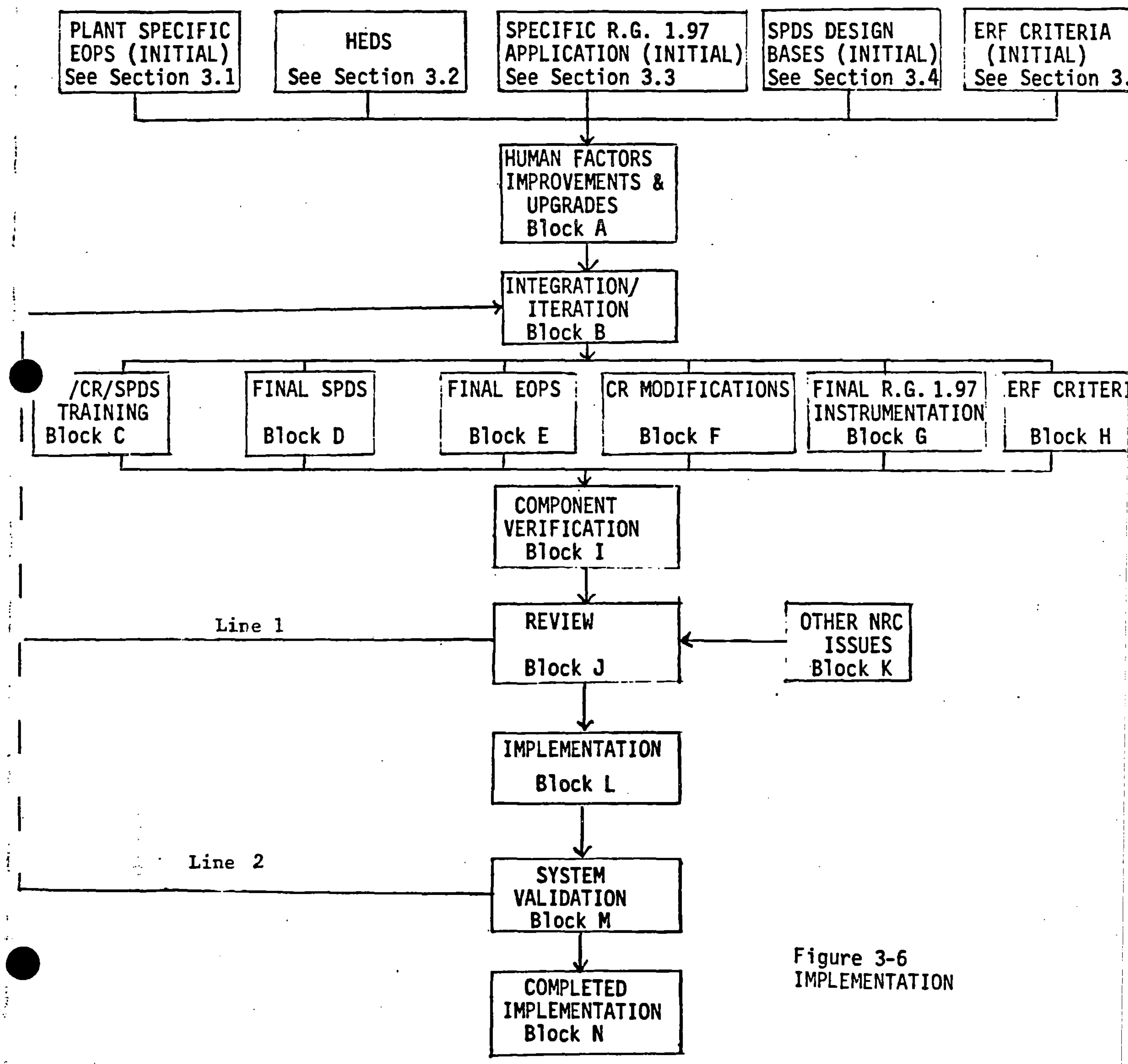


Figure 3-6
IMPLEMENTATION

3.6.1 Human Factors Improvements and Upgrades - Block A

After the HEDs have been identified as a result of the Control Room Survey and initial efforts are complete for the EOPs, R.G. 1.97, SPDS and ERFs, human factor improvements and upgrades should be identified to resolve HEDS. In addition to physical improvements in the Control Room, HEDs may be resolved through use of procedures, R.G. 1.97 instrumentation or the SPDS.

References for Human Factors Improvements and Upgrades

NUREG-0700, "Guidelines for Control Room Design Review", USNRC, September 1981

NUREG-0801, "Evaluation Criteria for Detailed Control Room Design Reviews", USNRC (draft for comments)

"Control Room Design Review Implementation Guideline", draft (INPO 83-0XX NUTAC), CRDR NUTAC

3.6.2 Integration/Iteration - Block B

All the initial criteria (i.e., plant specific EOPs, specific RG 1.97 application, SPDS design bases, and ERF criteria) and all identified human factor improvements and upgrades (see Section 3.6.1) should be reviewed in an integrated fashion. This review should further assure that all criteria are consistent with each other, all required human factor improvements and upgrades are being resolved, and all necessary interfaces between the EOP, CRDR, R.G. 1.97, SPDS, and ERF plans have occurred. This review should identify all necessary changes to the respective initial criteria in order to finalize these criteria.

This review may occur more than once as a result of problem areas identified by verification (see Section 3.6.9) or other NRC issues.

3.6.3 EOP/CR/ERF/SPDS Training - Block C

Training programs for operating personnel should be developed based on EOPs/CR/SPDS Criteria.

References for EOP/CR/SPDS Training

"Senior Control Room Operator and Shift Supervisor Qualification", (INPO 82-008), INPO, September 1982

"Nuclear Power Industry Training System Development", INPO, draft December 1982

"The Accreditation of Training in the Nuclear Power Industry", (INPO 82-011), INPO, May 1982

"Task Analysis Data Collection Procedure", TA-1, INPO, December 1982

"Emergency Operating Procedure Implementation Guideline", (INPO 82-016), EOPIA Review Group, June 1982

"Component Verification and System Validation Guideline", draft (INPO 83-0XX NUTAC), ERC NUTAC

"Emergency Operating Procedures Generation Package Guideline", (INPO 83-007), EOPIA Review Group, February, 1983

"Control Room Design Review Implementation Guideline", draft (INPO 83-0XX NUTAC), CRDR NUTAC

"Guideline for an Effective SPDS Implementation Program", (INPO 83-003), SPDS NUTAC, January, 1983

NUREG-0660, NRC Action Plan Developed as a Result of the TMIII Accident", USNRC, May 1980

NUREG-0899, "Guidelines for the Preparation of Emergency Operating Procedures", USNRC, August 1982

Supplement 1 to NUREG 0737, "Requirements for Emergency Response Capability", USNRC, December, 1982

3.6.4 Final SPDS - Block D

Design modification identified during the Integration/Iteration process should be incorporated to determine final SPDS design criteria.

3.6.4.2 References for Safety Parameter Display System

See Section 3.4

3.6.5 Final EOP - Block E

Procedure modifications identified during the Integration/Iteration Process should be incorporated to determine final EOPs.

References for Emergency Operating Procedures

See Section 3.1.

3.6.6 Control Room Modifications - Block F

Control room modifications as identified based on results from the Integration/Iteration process should be incorporated to determine final control room improvements.

References for Control Room Modifications

See Section 3.2.

3.6.7 Final R. G. 1.97 Instrumentation - Block G

Design modifications identified during the Integration/Iteration Process should be incorporated to determine final accident monitoring instrumentation.

References for R. G. 1.97 Instrumentation

See Section 3.3.

3.6.8 ERF Criteria - Block H

Emergency Response Facility modifications identified by the results from the Integration/Iteration process should be incorporated to determine final ERF design criteria.

References for Emergency Response Facility

See Section 3.5.

3.6.9 Component Verification - Block I

A verification should be included in the implementation program to ensure that each element achieves its design objective. All element (SPDS, EOPs, accident monitoring instrumentation, control room, training, and ERFs) should be involved in a component verification process.

References for Component Verification

Emergency Operating Procedure Verification Guideline, "(INPO 83-0XX)", EOPIA Review Group, 1983

"Verification and Validation of Safety Parameter Display Systems", (NSAC/39) NSAC

"Component Verification and System Validation Guideline", draft (INPO 83-oxx NUTAC), ERC NUTAC

3.6.10 Review - Block J

This review should compile verification findings and identify new NRC issuances and other regulatory guidance that affect the SPDS, control room, EOPs, ERF, Reg. Guide 1.97 instrumentation, and training. As a result of these findings and issuances, a methodology for revising the affected elements should be determined.

References for Review

Design Documentation

New Regulatory Guidance

3.6.11 Other NRC Issues - Block K

Future NRC issues that effect Supplement 1 to NUREG 0737 provisions should be considered in the review process.

3.6.12 Implementation - Block L

Implementation should include the purchase, installation, and testing of new equipment resulting from the Supplement 1 upgrades. Plant personnel should be fully trained on this equipment as well as EOPs during implementation.

Reference for Implementation

"Guidelines for an Effective SPDS Implementation Program", (INPO 83-003), SPDS NUTAC, January 1983.

3.6.13 System Validation - Block M

A system validation should be performed to determine that all elements together achieve the objective of mitigating the consequences of emergency conditions. All elements, SPDS, EOPs, control room, Reg. Guide 1.97 instrumentation, ERF, and training should be included in the validation.

References for System Validation

"Emergency Operating Procedure Verification Guideline", (INPO 83-oxx), EOPIA Review Group, 1983

"Verification and Validation of Safety Parameter Display Systems", (NSAC/39) NSAC

"Component Verification and System Validation Guideline", draft (INPO 83-OXX NUTAC), ERC NUTAC

3.6.14 Completed Implementation - Block N

The implementation plan for Supplement 1 to NUREG-0737 should be completed when all equipment has been installed in its permanent location, system validation has shown that the plant modifications achieve the objective of mitigating the consequences of emergency conditions, and plant personnel have been fully trained on EOPs and hardware.

3.6.15 Review/Integration Interface - Line 1

The revisions identified, if any, in the review process should be incorporated in the final design criteria for the SPDS, control room modifications, R.G. 1.97 instrumentation and ERF. Final EOPs and the EOP/CR/ERF/SPDS training program should also be upgraded to eliminate discrepancies determined during the review. Another component verification should be performed once the affected element has been revised.

References for Interface

See Section 3.6.2 through 3.6.14

3.6.16 System Validation/Integration/Iteration Interface - Line 2

The discrepancies identified during system validation, if any, should be classified according to their cause, EOPs, SPDS, control room, ERF, training, or a combination of these elements. Resolutions should be developed and appropriate modifications incorporated. Another component verification, review and validation should be performed if significant modifications are required to resolve the discrepancies.

References for Interface

See Sections 3.6.2 through 3.6.15