UNITED STATES
NUCLEAR REGULATORY COMMISSION
WASHINGTON, D. C. 20555

JUN 13 1985

Docket No: 50-414

APPLICANT: Duke Power Company

FACILITY: Catawba Nuclear Station, Unit 2

SUBJECT: SUMMARY OF MAY 14 AND 15, 1985, MEETING AND SITE VISIT TO AUDIT CATAWBA UNIT 2 SAFETY PARAMETER DISPLAY SYSTEM (SPDS)

General

On May 14 and 15, 1985, the NRC staff and its consultants met with Duke Power Company (DPC) representatives and visited the McGuire/Catawba simulator, the Catawba control room and the technical support center (TSC) for Units 1 and 2. A list of attendees is included in Enclosure 1.

Discussion

The meeting on May 14 and the site visit on May 15 followed the enclosed agenda (Enclosure 2). The viewgraphs of the presentations made by DPC representatives on May 14 are included as Enclosure 3. The staff's questions centered on the Verification and Validation (V&V) program review, data validation, critical safety functions, containment isolation, and radiation monitoring.

In the morning of May 15, 1985, the staff visited the McGuire/Catawba simulator. The staff witnessed a safety injection demonstration on high containment pressure due to a main steam line break. The event sequence is outlined in Enclosure 4. In the afternoon, the staff visited the Catawba control room and TSC. The viewgraphs of DPC's presentations are included as Enclosure 5. The staff's questions centered on the human factors engineering aspects. DPC representatives stated that the Catawba simulator delivery date is 1987 and operability date is 1988. The shift Technical Advisor (STA) is responsible for monitoring the SPDS.

Conclusion

The NRC representatives made the following statements at the conclusion of the visit.

1. DPC's V&V program appears to fulfill the intent of the SRP although the review is not performed by an independent consultant.

2. DPC's data validation program is adequate as an interim program. DPC committed to further improvements.

3. The human factors engineering aspects' review will be documented in the staff's audit report.

4. The staff noted an inconsistency between the SPDS status tree and the emergency operating procedures.

5. The staff noted that the containment isolation and radiation monitoring have to be reviewed by the NRC's Procedures and Systems Review Branch. The staff's target date for issuing the audit report is July 15, 1985, and for the SER is September 30, 1985.

Kahtan N. Jabbour

Kahtan N. Jabbour, Project Manager
Licensing Branch No. 4
Division of Licensing

Enclosures:
As stated

cc: See next page

DESIGNATED ORIGINAL

Certified By Angela Hatton

MEETING SUMMARY DISTRIBUTION

Docket No(s): 50-414
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Local PDR
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PRC System
LB #4 r/f
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E. Adensam
Project Manager K. Jabbour
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NRC PARTICIPANTS

G. Lapinsky
K. Jabbour

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Enclosure 1

ATTENDANCE LIST

May 14, 1985

NRC	Duke Power Company
G. Lapinsky	R. Sharpe
	R. White
NRC Consultants	J. Warren
	R. Morgan
J. DeBor	R. Collins
G. Bethke	H. Lee
C. Kain	R. Brown
	R. Dobson
	H. Davenport
	L. Frick
	A. Fairweather

May 15, 1985

NRC	Duke Power Company
K. Jabbour	R. Brown
G. Lapinsky	C. Majure
	G. Spurlin
NRC Consultants	R. Sharpe
	R. Morgan
G. Bethke	J. Ferguson
C. Kain	R. Collins
J. DeBor	

Enclosure 2

CATAWBA NUCLEAR STATION
SPDS AUDIT AGENDA

May 14-15, 1985
WC-1704

<u>TUESDAY, MAY 14, 1985</u>

8:30 AM	INTRODUCTIONS AND BRIEFING	NRC
8:45 AM	OVERVIEW OF SPDS IMPLEMENTATION	RL Brown
	STANDARDS	RL Brown
	HUMAN FACTORS ENGINEERING	RH White
	RELIABILITY	RC Collins
	VALIDATION AND VERIFICATION	RL Brown
	IMPLEMENTATION PLAN	RL Brown
	PROJECT MILESTONES	RL Brown
	IMPLEMENTATION STATUS	RL Brown
10:15 AM	BREAK	
10:30 AM	DESIGN BASIS	RL Brown
	WESTINGHOUSE EMERGENCY RESPONSE GUIDELINES AND CATAWBA EMERGENCY PROCEDURES GUIDELINES	HJ Lee
	LOGIC DESIGN	RL Brown
	DESIGN REVIEW	HJ Lee
	TASK ANALYSIS OF SPDS	RH White
	HUMAN FACTORS REVIEW	RH White
	SOFTWARE DEVELOPMENT	RC Collins
	VERIFICATION OF SOFTWARE	LR Frick
	VALIDATION TESTING OF SOFTWARE	LR Frick
12:00	LUNCH	
1:00 PM	SENSOR VALIDATION	LR Frick
	DYNAMIC TESTING	RL Brown
	SIMULATOR INSTALLATION	RL Brown
	DOCUMENTATION	RL Brown
2:00 PM	OPERATOR AID COMPUTER SYSTEM AND EMERGENCY RESPONSE FACILITIES	RL Brown
	DISCUSSION OF SPDS INPUTS AND INPUT ISOLATION	RM Meacham
	SPDS MAINTENANCE AND REVISION PROGRAM	RG Morgan
	OVERVIEW OF McGUIRE AND OCONEE SPDS AND IMPLEMENTATION STATUS	RL Brown
2:50 PM	BREAK	
3:00 PM	TOUR OF STAGING COMPUTER AND DEMONSTRATION OF OCONEE DISPLAYS	RC Collins
4:00 PM	NRC QUESTIONS AND REVIEW OF DOCUMENTATION	
4:30 PM	ADJOURN	

WEDNESDAY, MAY 15, 1985:

TECHNICAL TRAINING CENTER:

9:15 AM	TRAINING PROGRAM DESCRIPTION	GE Spurlin
9:45 AM	DESCRIPTION OF SIMULATOR AND SPDS	CA Majure
10:10 AM	BREAK	
10:30 AM	DISCUSSION OF SCENARIO	CA Majure
10:45 AM	SIMULATOR DEMONSTRATION	CA Majure
11:15 AM	QUESTIONS AND DISCUSSION	
11:30 AM	DEPART FOR CATAWBA AND LUNCH	

CATAWBA NUCLEAR STATION:

1:30 PM	TOUR OF CONTROL ROOM, EMERGENCY RESPONSE FACILITIES, AND DEMONSTRATION OF SPDS AND SUPPORTING DISPLAY SCREENS	RG Morgan, RC Collins
2:00 PM	TECHNICAL SUPPORT CENTER (TSC)	
2:15 PM	SCENARIO DISCUSSIONS	
2:45 PM	QUESTIONS AND DISCUSSIONS	
3:00 PM	NRC CAUCUS	
3:15 PM	EXIT BRIEFING	
3:30 PM	ADJOURN	

DUKE POWER'S OBJECTIVES IN DEVELOPING THE SAFETY PARAMETER DISPLAY SYSTEMS:

- INSURE THAT WE HAD AN SPDS WHICH AIDS THE OPERATOR IN DETERMINING THE SAFETY STATUS OF EACH OF THE CRITICAL SAFETY FUNCTIONS;
- COORDINATES WELL WITH THE NEW PLANT SYMPTOM ORIENTED EMERGENCY PROCEDURES;
- COMPLEMENTS HIS TRAINING; AND
- INTEGRATES FUNCTIONALLY INTO THE EXISTING CONTROL ROOM WITH DUE CONSIDERATION TO SOUND HUMAN FACTORS ENGINEERING PRINCIPLES.

FURTHER..... THE SPDS DEVELOPMENT WAS CONDUCTED IN AN INTEGRATED MANNER WITH THE CONTROL ROOM DESIGN REVIEW, EMERGENCY PROCEDURES DEVELOPMENT, REG GUIDE 1.97, AND EMERGENCY RESPONSE FACILITIES ACTIVITIES.

R. L. Brown
Production Support Department

PRIMARY STANDARDS USED
IN THE DEVELOPMENT
OF THE SPDS

NUREG-0696: "FUNCTIONAL CRITERIA FOR EMERGENCY RESPONSE FACILITIES"

NUTAC - "GUIDELINES FOR AN EFFECTIVE SAFETY PARAMETER DISPLAY SYSTEM IMPLEMENTATION PROGRAM"

NSAC/38: "A TRANSIENT LIBRARY FOR VALIDATING SAFETY PARAMETER DISPLAY SYSTEMS"

NSAC/39: "VERIFICATION AND VALIDATION FOR SAFETY PARAMETER DISPLAY SYSTEMS"

SECY-82-111: "REQUIREMENTS FOR EMERGENCY RESPONSE CAPABILITY"

NUREG-0737, SUPPLEMENT 1: "REQUIREMENTS FOR EMERGENCY RESPONSE CAPABILITY"

NUREG-0835: "HUMAN FACTORS REVIEW GUIDELINES FOR THE SAFETY PARAMETER DISPLAY SYSTEMS"

ADDITIONALLY:

HEAVY INVLOLVEMENT BY DUKE POWER EMPLOYEES IN THE FOLLOWING INDUSTRY ACTIVITIES RELATED TO THE SPDS:

- DISTURBANCE ANALYSIS AND SURVEILLANCE SYSTEM (DASS)
- DISTURBANCE ANALYSIS SYSTEM (DAS)
- ATOMIC INDUSTRY FORUM ACTIVITIES ON CONTROL ROOM DESIGN AND SPDS ISSUES
- EPRI, INPO, AND NSAC WORKING GROUPS

R. L. Brown
Production Support Department

CLARIFICATION: CATAWBA'S SPDS WAS DEVELOPED FROM McGUIRE

- SISTER PLANT, SAME NSSS
- BASICALLY SAME CONTROL ROOM
- UNITS AT McGUIRE WERE PLACED INTO OPERATION PRIOR TO CATAWBA

MUCH OF TODAY'S DISCUSSION WILL BE ABOUT THE DEVELOPMENT OF McGUIRE'S SPDS

McGUIRE'S COMPLETED SPDS WAS ADAPTED AND REVISED FOR APPLICATION AT CATAWBA WITH APPROPRIATE LOGIC, SPDS INPUT SELECTION REVIEWS AS WELL AS V & V OF COMPLETED SOFTWARE.

R. L. Brown
Production Support Department

CONTROL ROOM REVIEW PLAN

PROGRAM INTERFACES

- EMERGENCY PROCEDURES UPGRADE PROGRAM
- SAFETY PARAMETER DISPLAY SYSTEM (SPDS) PROGRAM
- POST ACCIDENT MONITORING ASSESSMENT PROGRAM
- SIMULATOR UPGRADE PROGRAM

DUKE POWER COMPANY
CONTROL ROOM REVIEW

MANAGEMENT APPROACH

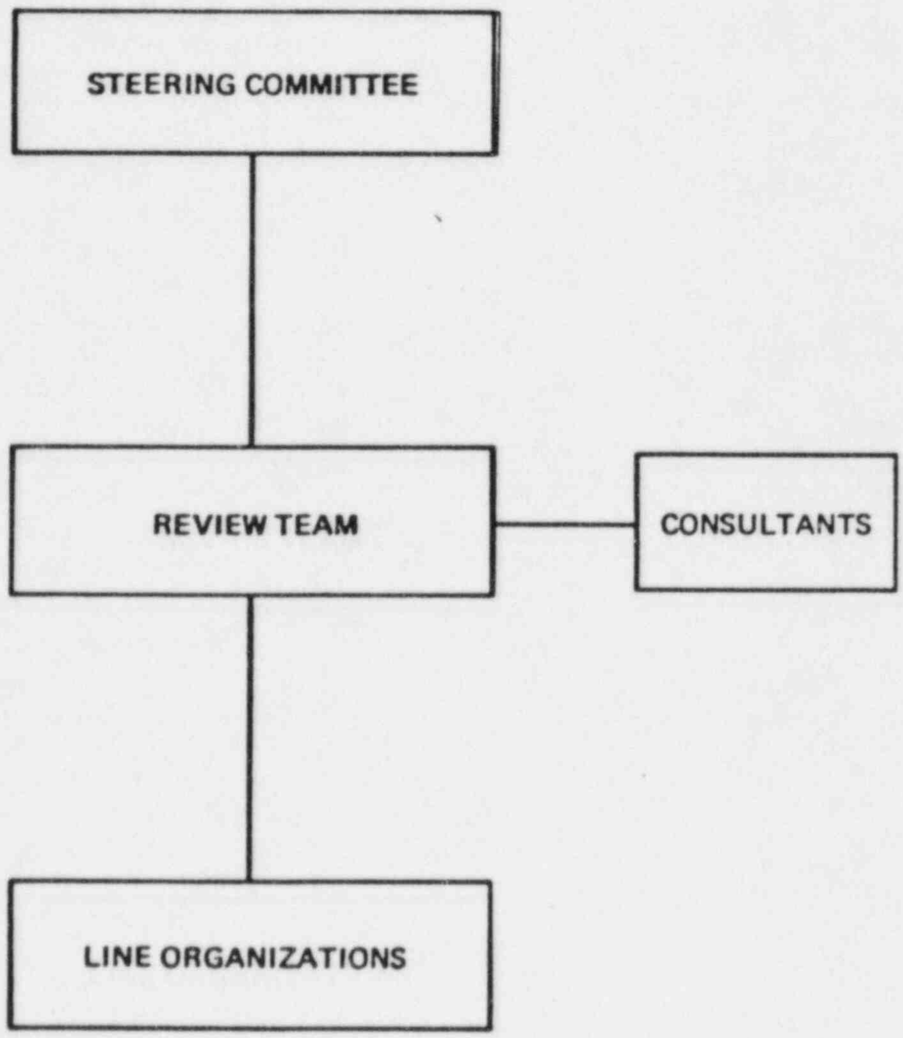


FIGURE 6

Steering Committee

- Design Engineering
- Nuclear Production — Stations
- Nuclear Production — General Office

Control Room Review Steering Committee

Name	Department
T. C. McMeekin	Design - Electrical Division
R. L. Brown	Nuclear - Engineering Services
H. R. Lowery	Nuclear - Oconee
G. D. Gilbert	Nuclear - McGuire
J. W. Hampton	Nuclear - Catawba
B. C. Moore	Nuclear - O & M
N. A. Rutherford	Nuclear - Licensing
R. S. Darke	Design - Electrical Division
W. H. Rasin	Design - SRAL
R. E. Hall	Design - Mechanical Division
C. A. Little	Nuclear - I & E Maintenance

CONTROL ROOM REVIEW STEERING COMMITTEE
MEMBER QUALIFICATION SUMMARY

STEERING COMMITTEE MEMBER	CURRENT			EDUCATION			LICENSES	EXPERIENCE							TECH SOC. MBR.	MISCELLANEOUS (Stds., Task Forces, Work Grps. and Related Activities)
	POS'N	LOC'N DIV'N or GRP	LOC'N DEPT	LEVEL	FIELD	OTHER		TOTAL YRS	PLANT OP	PLANT ENG	OTHER C/R	OTHER I/C	OTHER SYST	OTHER SUPP'T		
R. L. BROWN	System I&C Engr	Prod. Tech. Sup-Serv.	Prod. Sup-port	BS	ME		P.E. (NC-SC)	14	1	6		7			ISA	Paper "Enhanced FW Wtr Level Control" Mbr-EPRI/DOE - B&W DASS Project Mbr-AIF C.R. Consider'ns SubComm
C. A. LITTLE	System I&E Engr.	Nuc. Maint.	MPD	BS	EE		P.E. (NC)	10	1			5	4			Mbr. IEEE SC7, WG 7.1
W. S. PASEY	Consultant	Elec. Con. Sys.	D/E	BS	ME	AFTA-FERMI	P.E. (ILL)	43		5	10	18	10		IEEE NPS	Mbr IEEE SC6 - Safety Systems and WG 1.2 Control Room Std. Papers-Several on CR/CB's (IEEE, IAEA, JPJ
T. D. GILBERT	Operating Engr	MNS	Stm	BS	ME		SRO P.E. (NC)	11	11							
R. E. HALL	Superv's'g Eng. Engr	Mech	D/E	BS	ME		P.E. (NC)	13		2			11		ASME	CE Owners Grp -Systems/Oper'ns SubComm Chmn, DPC Metric Conversion Comm.
J. W. HAMPTON	Mgr. Catawba Nuc. Sta.	CNS	Stm	BS	ME	N Prototype Reactor	SRO CNS	34	16	8						EPRI - Eng'g & Operations Task Force INPO - Evaluation & Assistance Div'n, Industry Group
H. R. LOWERY	Operating Engr	CNS	Stm	H.S.	Gen		SRO	18	18							
T.C. McKEEVER	Principal Engr	Elect. Con. Sys.	D/E	BS	EE	USN-Nucl. Pwr & Sub. Schools	P.E. (SC-NC)	17	5			12			ANS IEEE ASME	Mbr - NSRB
W. R. BARIN	Senior Engr	Mech	D/E	BS	ME	USN-Nucl. Pwr Sch	P.E. (SC-NC) SRO (9 Yrs.)	31	14				7		ANS	EPRI - Safety and Analysis Task Force AIF-SubComm on ATWS-ANS paper Ind. Adv. Grp - TMI Site, 4/79
T. A. RUTHERFORD	Syst. Engr (Licensing)	Proj. & Lic.	Stm	BS M	ME BA			18	3	3			3	6	ANS	B&W Owners Grp, SubComm Dir NSRB '76-'79 Mbr-Equip Qual'n Grp. Mbr Nucl. Util Grp on Enforcement
P. C. MOORE	Syst. Prod. Engr	Op's Supp	Stm	BS	ME		P.E. (SC) SRO	10	10							
TOTALS								213 Man Yrs	78 Man Yrs	30 Man Yrs	10 Man Yrs	48 Man Yrs	31 Man Yrs	6 Man Yrs	6 Societies	

Review Team

- senior reactor operators
- mechanical/nuclear engineers
- electrical engineers

REVIEW TEAM

P. A. THOMPSON	NP - MCGUIRE
G. M. HAYNES	NP - OCONEE
S. H. BALLENGER	NP - CATAWBA
H. M. RYBCZYK	NP - CATAWBA
Y. G. TRUESDALE	NP - CATAWBA
M. D. RAINS	NP - PROJECTS
R. H. WHITE	DE - ELECTRICAL
L. T. HARBINSON	DE - ELECTRICAL
M. V. CARTER	DE - MECHANICAL/NUCLEAR
D. W. GWYNN	DE - MECHANICAL/NUCLEAR
M. R. CREWS	DE - ELECTRICAL

DUKE POWER COMPANY
CONTROL ROOM REVIEW

MANAGEMENT APPROACH

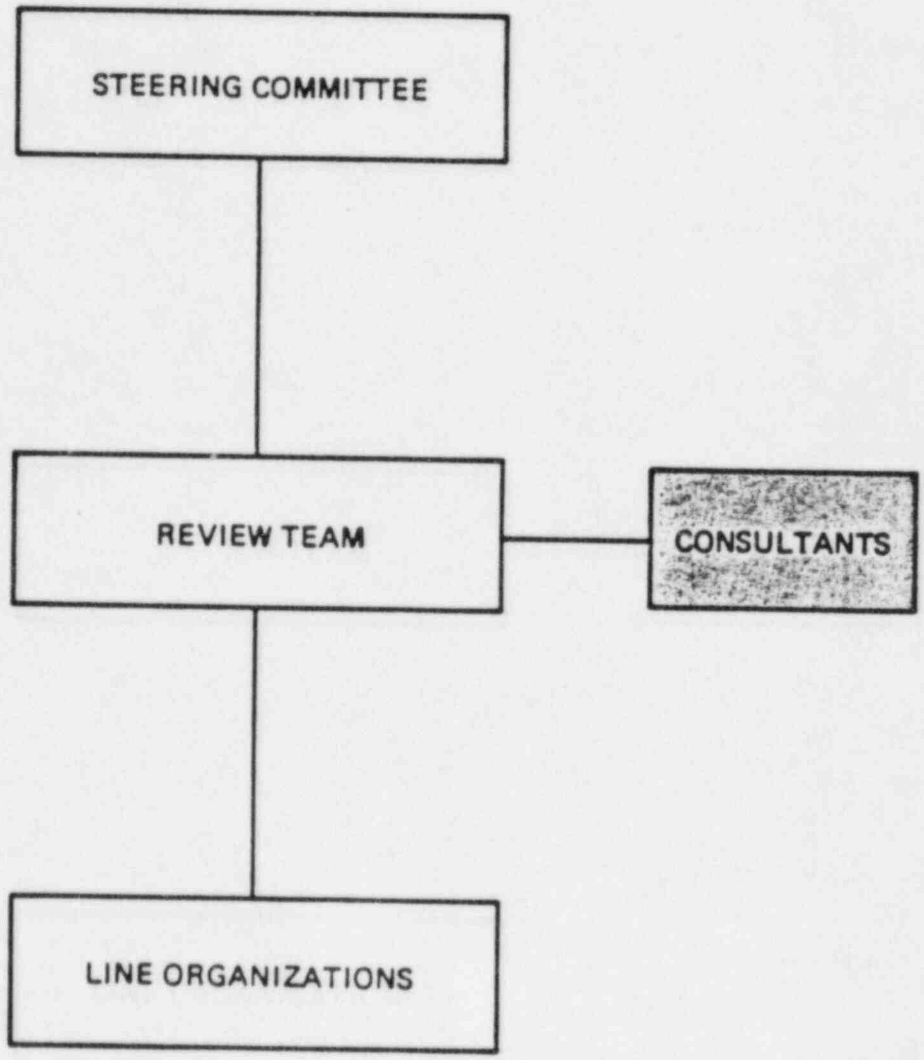


FIGURE 6

CONTROL ROOM REVIEW TEAM (Continued)

BIOTECHNOLOGY HUMAN FACTOR SUPPORT

Name	Department
Harold E. (Smoke) Price	Project Director; Leader OER Effort, and Assessment
Harold P. VanCott	Leader Task Analysis Effort, and Assessment
C. Richard Hatterick	Leader, Control Room Survey Effort
Barbara Paramore	Task Analysis & Assessment
Donald F. Taylor	OER, CRS & Assessment
James P. Bongarra	OER & Assessment
Joseph DeBor	Task Analysis
John H. Hill	Task Analysis
Mark Kirkpatrick	OER

CONTROL ROOM REVIEW PLAN

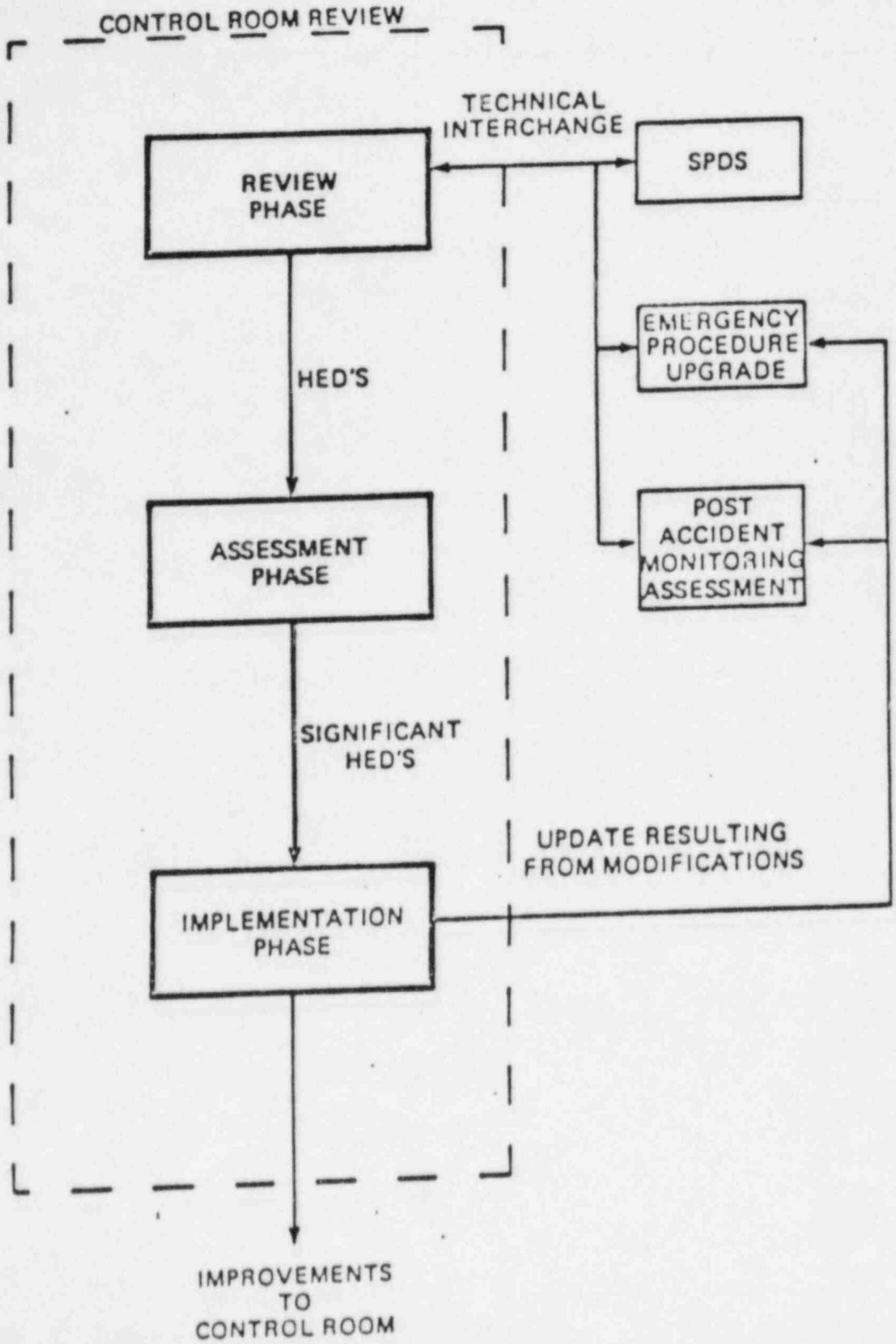
REVIEW TEAM TRAINING

- PRINCIPLES OF HUMAN FACTORS ENGINEERING
- SIMULATOR AND PROCEDURE FAMILIARIZATION
- SPECIFIC TRAINING FOR CONTROL ROOM REVIEW ACTIVITIES
    - TASK ANALYSIS
    - HUMAN FACTORS SURVEY
    - ETC.
- IMPORTANCE OF PROPER PREPARATION AND TRAINING RECOGNIZED

CRDR

• Review Team selected	2/82
• Final draft of Review Plan	5/82
• Plan presentation to NRC	5/82
• BioTechnology hired for Human Factors Assistance	6/82
• Mock-up space rented and construction begun	7/82
• Review efforts commenced	9/82
• Operating Experience Review	1/83
• Control Room Survey	1/83
• Task Analysis	2/83
• Assessment commences	3/83
• CRDR Summary Report	6/83

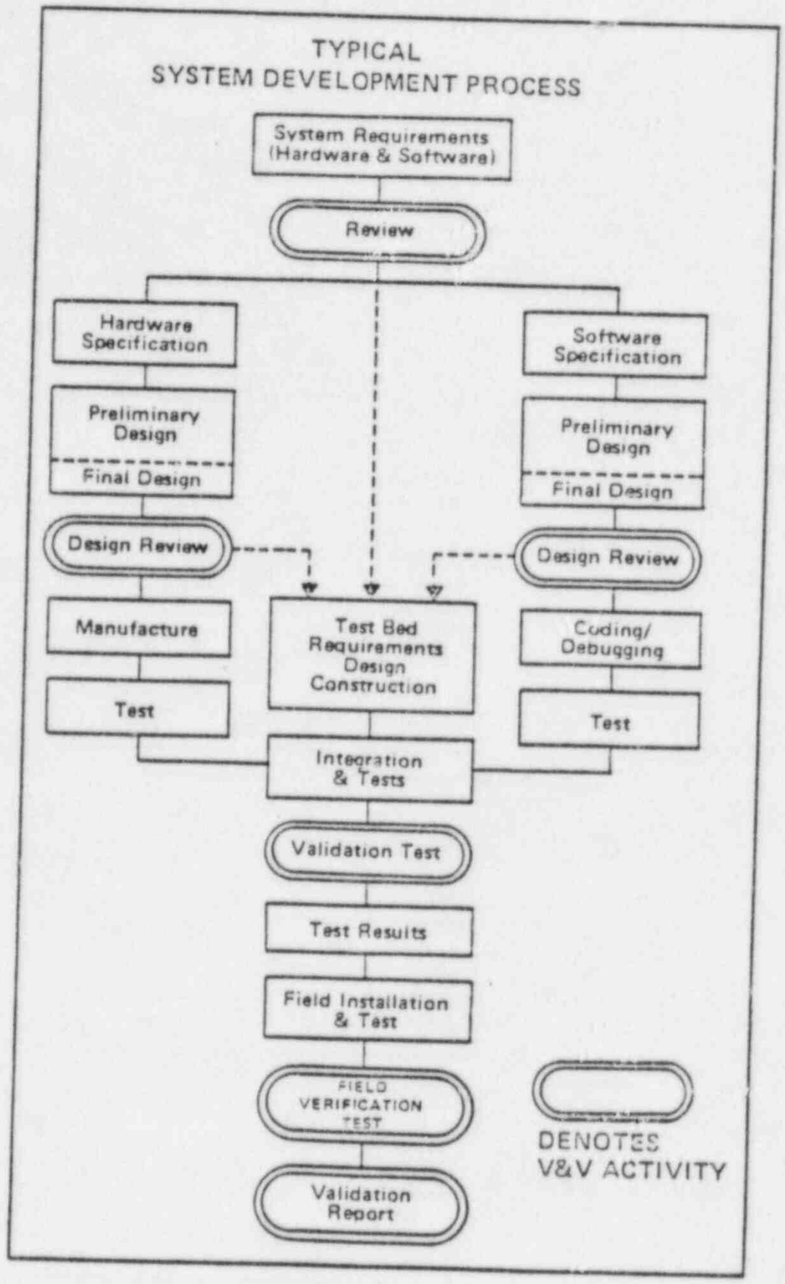
DUKE POWER COMPANY
CONTROL ROOM REVIEW PROCESS



VERIFICATION: A review to ensure that the identified problem is being solved properly and then the review of the resultant design to ensure that the SPDS meets functional requirements.

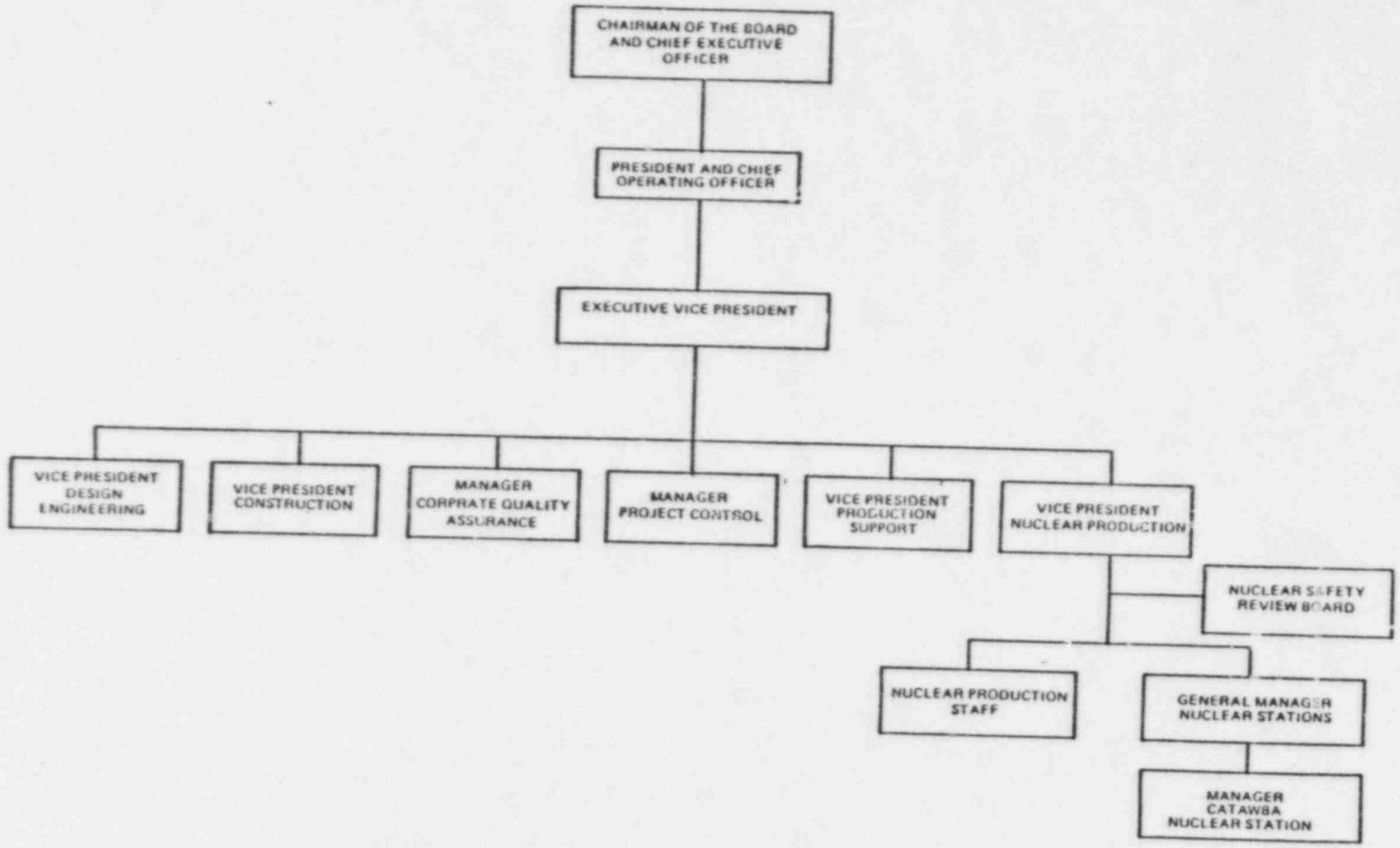
VALIDATION: A test and evaluation of the integrated hardware and software system to determine compliance with the functional, performance, and interface requirements.

R. L. Brown
Production Support Department



NSAC 39

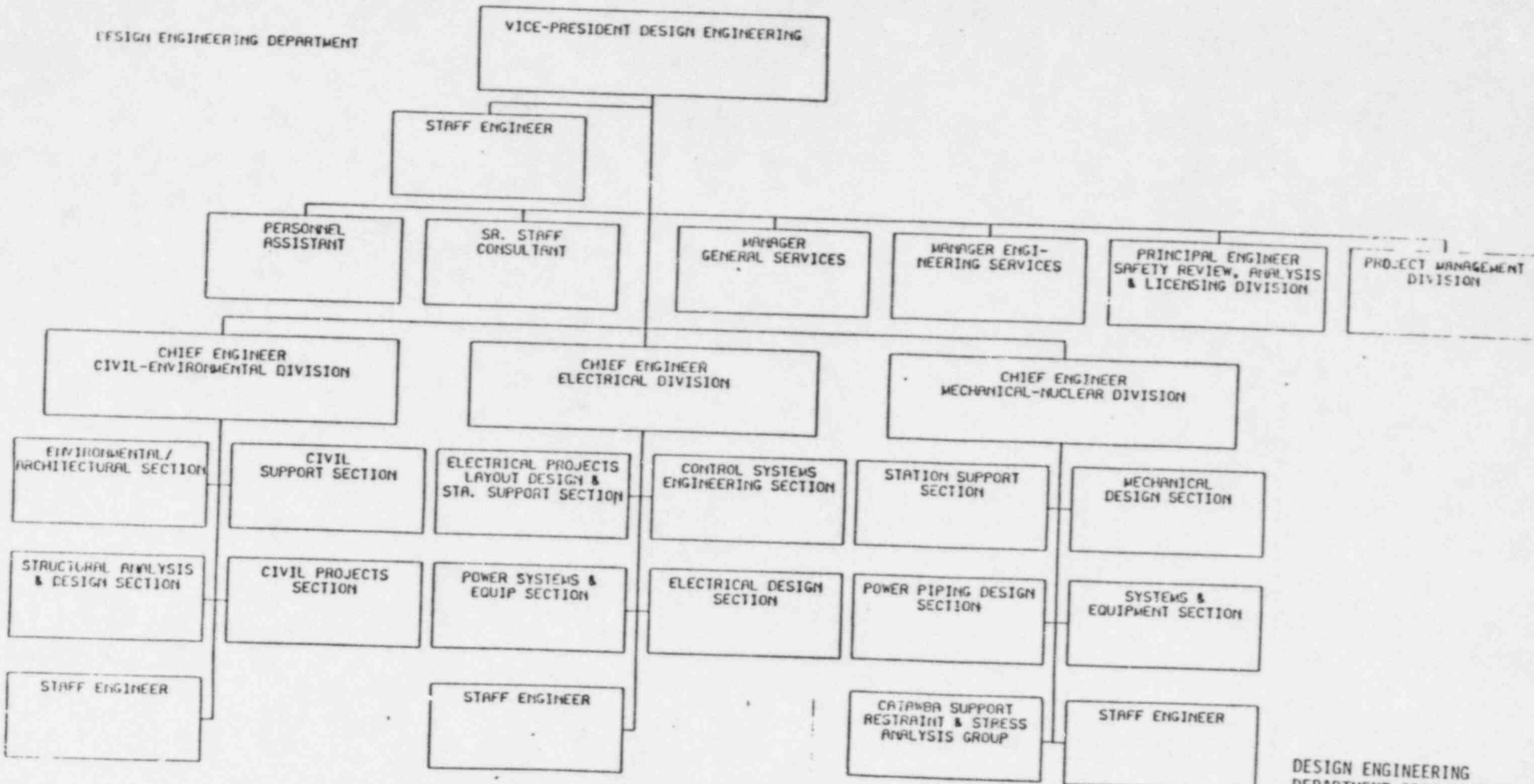
R. L. Brown
Production Support Department



R. L. Brown
Production Support

DUKE POWER CORPORATE ORGANIZATION

CATAWBA NUCLEAR STATION
Figure 13.1.1-1
Rev. 11

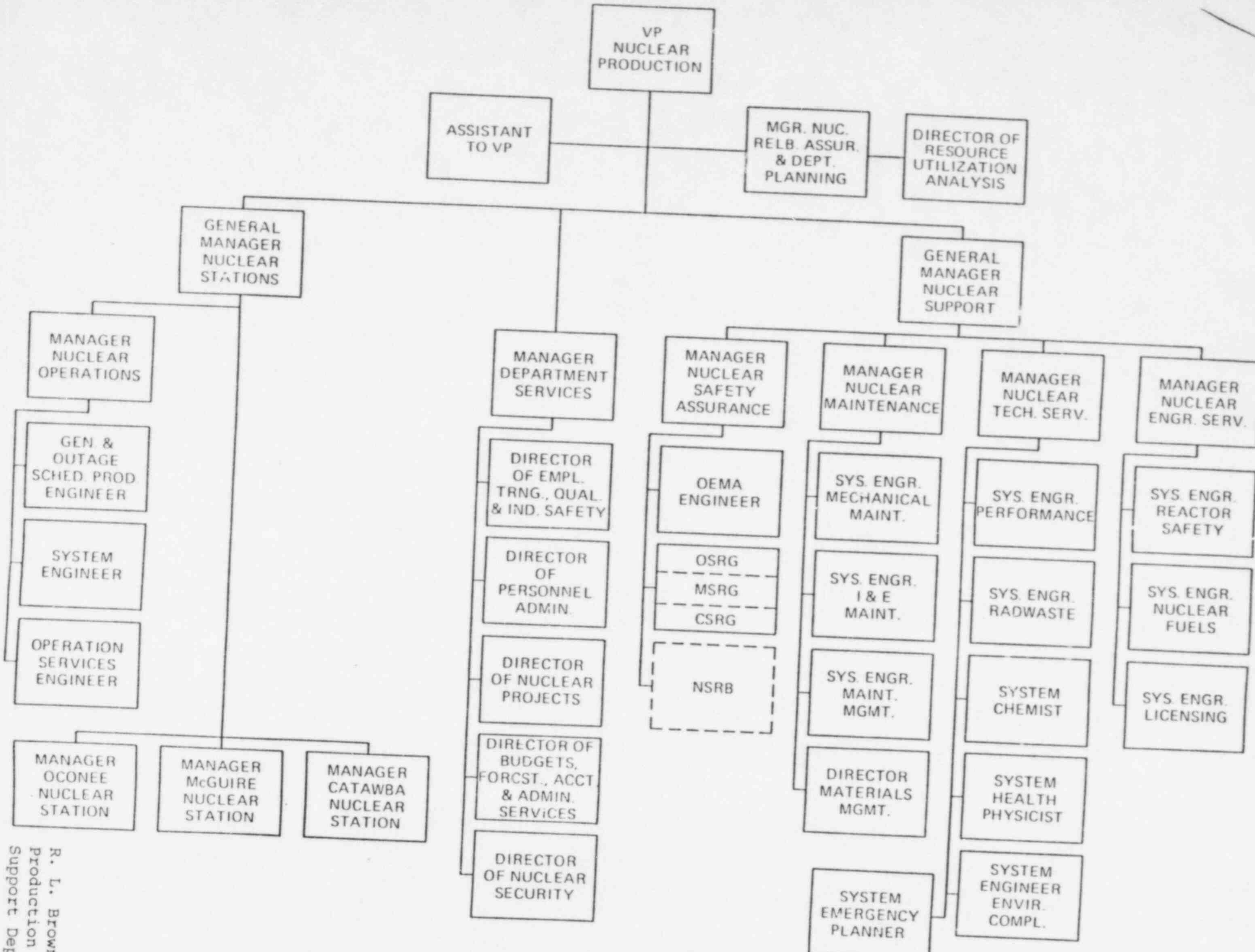


DESIGN ENGINEERING DEPARTMENT ORGANIZATION

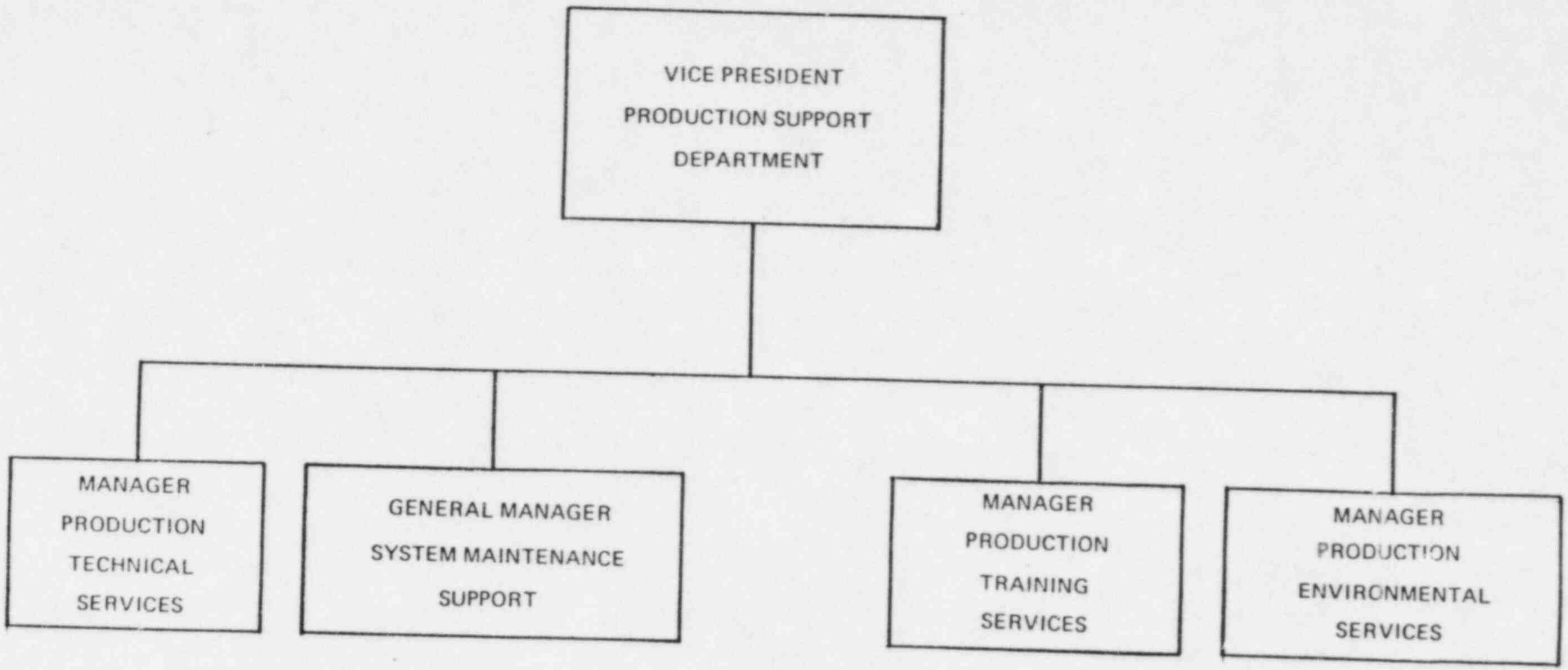
CATAWBA NUCLEAR STATION

Figure 13.1.1-2
Rev. 1

R. L. Brown
Production Support
Department



R. L. Brown
Production
Support Dept.



R. L. Brown
Production Support
Department

PRODUCTION SUPPORT
DEPARTMENT ORGANIZATION
CATAWBA NUCLEAR STATION
Figure 13.1.1-4
Rev. 7

VALIDATION AND VERIFICATION
OF
SAFETY PARAMETER DISPLAY SYSTEM

DESIGN BASIS:	CONTROL ROOM REVIEW STEERING COMMITTEE
REVIEW OF DESIGN BASIS:	CONTROL ROOM DESIGN REVIEW TEAM AND STATION PERSONNEL
	DAVE CAIN OF EPRI/NSAC
EMERGENCY RESPONSE GUIDELINES:	WESTINGHOUSE OWNERS GROUP
SELECTION OF CRITICAL SAFETY FUNCTIONS:	WESTINGHOUSE OWNERS GROUP
DEVELOPMENT OF CATAWBA EMERGENCY OPERATING PROCEDURES GUIDELINES:	NUCLEAR PRODUCTION'S REACTOR SAFETY UNIT
VALIDATION AND VERIFICATION PLAN:	DESIGN ENGINEERING ELECTRICAL
GENERATION OF SPDS LOGIC:	INSTRUMENTATION & ELECTRICAL OF NUCLEAR PRODUCTION
VERIFICATION OF SPDS LOGIC:	REACTOR SAFETY UNIT OF NUCLEAR PRODUCTION
GENERATION OF SPDS SOFTWARE:	PROCESS COMPUTER UNIT OF PRODUCTION SUPPORT
VERIFICATION OF SPDS SOFTWARE:	COMPUTER AND SECURITY ENGINEERING OF DESIGN ENGINEERING
VALIDATION OF INSTALLED SPDS SOFTWARE:	COMPUTER AND SECURITY ENGINEERING OF DESIGN ENGINEERING
TASK ANALYSIS OF SPDS:	CONTROL ROOM DESIGN REVIEW TEAM
DEVELOPMENT OF SUPPORTING DISPLAY SYSTEM:	PRODUCTION SUPPORT DEPARTMENT
HUMAN FACTORS REVIEW OF SPDS DISPLAY SYSTEM:	CONTROL ROOM DESIGN REVIEW TEAM
ONGOING VALIDATION OF SPDS SYSTEM:	PERFORMANCE SECTIONS AT EACH NUCLEAR STATION

R. L. Brown
Production Support Dept.

4 SAFETY PARAMETER DISPLAY SYSTEM

4.0 INTRODUCTION

This document describes the implementation plans for the installation of Safety Parameter Display Systems at Catawba. The approaches taken by Duke Power Company in providing these systems are consistent with the long-standing practice of utilizing in-house capabilities. This includes the use of technical and operations expertise in formulating the design of the SPDS as well as integrating the SPDS into existing highly reliable and well-developed plant data systems.

The SPDS systems described in the following meet the intent of the guidance documents NUREG-0737, Supplement 1 and was developed considering the guidance of the NUTAC Guidelines for an effective SPDS implementation program, NSAC/39, NUREG-0696, and other related documents.

4.1 IMPLEMENTATION PLAN

4.1.1 GENERAL SCHEDULE CONSIDERATIONS

The Safety Parameter Display System is being developed in an integrated manner with other activities associated with the overall emergency response capabilities being developed in response to NUREG-0737, Supplement 1.

As in the case with other emergency response capabilities activities, the SPDS system will be developed within Duke Power Company. By utilizing this in-house capability, many years of design, operating and maintenance experience will be incorporated into the SPDS design and implementation.

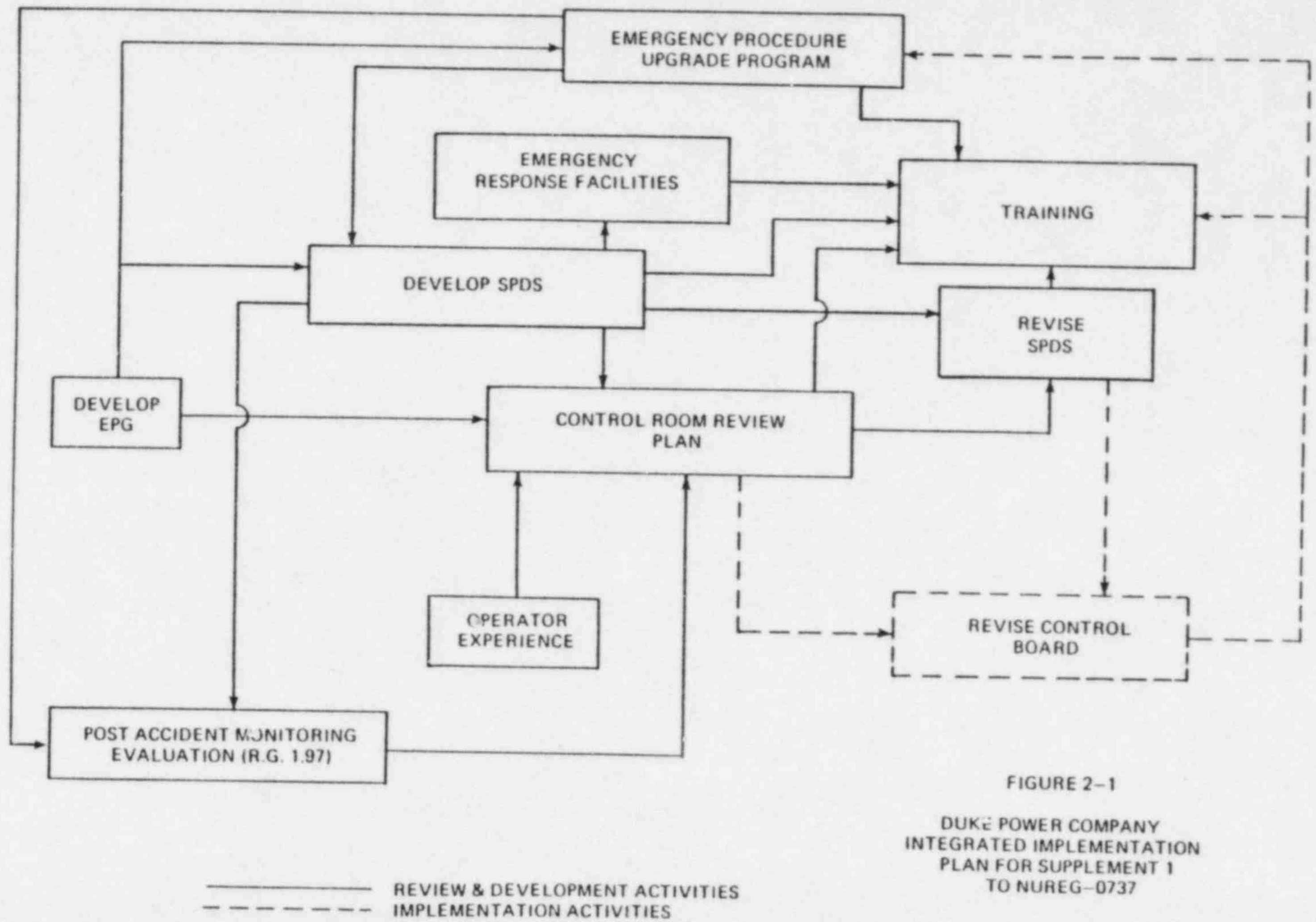


FIGURE 2–1

DUKE POWER COMPANY
INTEGRATED IMPLEMENTATION
PLAN FOR SUPPLEMENT 1
TO NUREG–0737

The SPDS design, development and implementation will be scheduled to take advantage of knowledge gained from the various elements of the Control Room Review and the development of the symptom oriented emergency procedures.

Further, the design of the SPDS will be an interactive process with input from various disciplines in both the development and testing phases. The validation and verification as well as on-line testing will be performed prior to the final installation of the SPDS to ensure an effective SPDS is provided to the operating crew. Results from V&V, on-line testing, CRR Task Analysis, Human Factors Review, and related activities will be evaluated and incorporated as needed prior to finalizing the SPDS design.

4.1.2 TRAINING

Control Room operators, shift supervisors, and shift technical advisors will receive training on the use of the SPDS. This training will be performed in conjunction with the operator training on the new Symptom Oriented Emergency Procedures.

This training will include the SPDS logic and its relationship to the emergency procedures. The panel functions and methods of calling up and interpreting supporting displays will be covered. The verification of SPDS indications using hardwired and other control room indications will be provided. Invalid or indeterminate SPDS indications (due to failed plant inputs) will be identified to the operator. Visual aids in the form of slides representing SPDS and supporting displays will be used.

In addition, appropriate instrument and electrical personnel at the station will receive training on the maintenance of the SPDS and field inputs. Training records will be maintained on those required to receive training.

4.1.2.1 Training Schedule Considerations

It is expected that the SPDS will be fully developed and tested prior to initiating training of operating crews and prior to final installation. Further, since the SPDS will provide the operator with information pertinent to the new symptom oriented emergency procedures, operator training for the SPDS will be performed in conjunction with training on the symptom oriented emergency procedures.

4.1.3 MANAGEMENT

The management of the SPDS project will be under the direction of the Control Room Review Steering Committee. Lead responsibility for the SPDS project was initially designated to be the Manager of Engineering Services of the Steam Production Department. After the initiation of this project, two major re-organizations occurred. This resulted in the Manager of Production Technical Services of the Production Support Department assuming lead responsibility for this project.

In this capacity, the SPDS Project Leader is responsible for the overall coordination and scheduling of the SPDS project. Under his direction, a number of other groups will design, review, and/or implement the SPDS systems.

A complete set of documentation related to the SPDS will be maintained by the SPDS Project Leader.

Design documents, software codes, system descriptions, V&V documents, and user documentation will be reviewed and approved and controlled consistent with established procedures for these classes of documentation. Revisions to these documents will also be controlled, reviewed and approved prior to use.

4.1.4 ROLE AND MISSION SPECIFICATION

The role and mission of the SPDS is to aid the Control Room Operating Crew in monitoring the status of the critical safety functions. The primary objective of the SPDS is to provide the operating crew with an overview of the safety status of the plant and how well the critical safety functions are being maintained.

The critical safety functions are defined by the generic Emergency Response Guidelines (ERG's) developed by the Westinghouse Owners Group. In the case of Catawba, the Emergency Procedure Guidelines identify the following critical safety functions:

- Subcriticality
- Core Cooling
- Heat Sink
- Integrity
- Containment
- Inventory

4.1.5 LOCATION OF THE SPDS

The SPDS will enable the operator to quickly assess the safety status without taking any manual actions from his normal operating positions. Further, the SPDS will be readily viewable from a wide area in the Control Room to enable shift technical advisors and shift supervisors to readily determine the safety status of each of the critical safety functions. The SPDS displays will also be available to the Technical Support Center personnel. The SPDS will be integrated into the plant control room without adding clutter and confusion. Further, a new and different man-machine interface will be avoided.

4.1.6 SPDS AVAILABILITY

The SPDS will be reliable and readily available to the operator during normal operation and during emergency operating conditions. It is not required during stable shutdown conditions nor during refueling outages.

It is not essential that the SPDS be operational for plant operations personnel to determine the safety status of the plant or to execute any of the symptom oriented emergency procedures since adequate instrumentation, instructions and training will be provided independent of the SPDS.

The plant operating crew will be trained and procedures will be in place to enable them to monitor the critical safety function status both with and without the SPDS. Further, this training and these procedures will require the operating crew to verify SPDS indications using reliable control board indicators prior to taking any corrective actions.

4.1.7 VALIDATION AND VERIFICATION

A component (SPDS) level Validation and Verification Program will be developed considering the guidance contained in the NUTAC Guidelines and NSAC/39. The V&V of the SPDS will be performed by the Design Engineering Department providing an independent review since the SPDS design will be developed by the Nuclear Production and Production Support Departments.

Further, a Human Factors Review and a Task Analysis will be performed of the SPDS and supporting displays by the Control Room Review Team to validate the SPDS as part of the total operating system.

4.2 SYSTEM DESCRIPTION

This section describes the design of the Safety Parameter Display System, Human Factors considerations and includes a description of the Operator Aid Computer (OAC) Systems.

4.2.1 HUMAN FACTORS CONSIDERATIONS

The Safety Parameter Display System will be designed with appropriate Human Engineering Factors incorporated.

4.2.1.1 Viewability

The SPDS will be implemented on the Operator Aid Computer System which has three color graphic CRT's located on each unit's main control board. In this location, these CRT's are readily viewable from all normal operating positions. The six color blocks, one for each critical safety function will be continuously displayed on the bottom of the alarm video. The alarm video is centrally located on the main control board. The dimensions of the color blocks are such that they are easily viewable from any position within the main control area of the Control Room. Since the color blocks will always be positioned in the same relative locations on the CRT, it will be easy for the operator, STA and shift supervisor to readily determine the status of any of the six critical safety functions.

The supporting displays for the SPDS will be available to the operator on demand on the other two videos located on the main control board. The man-machine interface used by the operators to call up the supporting displays is the same as he normally uses to call up system graphics, display menus, and other OAC programs. This man-machine interface is thoroughly familiar to the operators through their normal operation of the plant.

4.2.1.2 Information Hierarchy/Highlighting

4.2.1.3 SPDS

The Safety Parameter Display System will consist of logic based on the Westinghouse Owners Group decision trees which are part of the symptom oriented emergency procedures. This logic drives the six CSF color blocks.

4.2.1.4 Other Information

The SPDS is described above, other supporting information is provided through a variety of normally available control room tools. Supporting CRT displays will be provided which will allow the operator to call up displays that duplicate each of the decision trees. The alarmed path through the tree will be highlighted and will indicate the appropriate emergency procedure to implement.

Decision trees not in alarm will indicate that the critical safety function is satisfied.

Further, an additional level of detail will be available to the operator. He can determine the plant field inputs which have resulted in the logic generating an alarm, such as "NR level in SG A less than 5%", "Pressurizer level channel A less than 17%", etc.

In addition, the remaining OAC features such as system schematics, input display lists, trend recording, alarms, etc., will be available for the operator's use as needed.

4.2.1.5 Man-Machine Interfaces

The Operator Aid Computer System Man Machine interfaces have been developed over the past 20 years and takes advantage of the feedback from operators over this period of time. Control panels are conveniently placed on the lower control board below each CRT. Panel functions are designed to minimize the number of key strokes required of the operator consistent with the urgency of his needs.

Response to the operator's commands by the OAC's is nearly instantaneous with displays completed within two seconds.

4.2.2 DESIGN CONTROL

The SPDS logic design is the responsibility of the Nuclear Production Department's Instrument and Electrical Section. This logic was based upon the

Westinghouse Decision Trees. Inputs were selected to provide the information required to drive this logic.

The Design of the Catawba SPDS Systems will use the SPDS being developed for McGuire since both plants contain nearly identical Westinghouse NSSS Systems and Support Systems.

An independent validation and verification program will be performed by the Design Engineering Department (see Section 4.1.7).

4.2.3 RELIABILITY AND AVAILABILITY

As can be seen on the chart below, availability of the OAC systems at McGuire and Oconee has exceeded 99% when OAC downtime during unit outages is excluded. The reliability of the Catawba OAC's is expected to be similar to the McGuire OAC's since they are nearly identical.

Each OAC is fed by a dedicated static inverter which normally receives its power from DC batteries. Upon inverter, DC batteries or charger failure, a static transfer switch provides regulated AC power from two independent sources.

ANNUAL AVERAGE ADJUSTED SYSTEM AVAILABILITY*

	1980	1981	1982	Average Adjusted Availability*
McGuire 1	N/A	99.27%	99.38%	99.3%
Oconee 1	99.20%	99.75%	99.83%	99.6%
Oconee 2	99.82%	99.54%	99.80%	99.7%
Oconee 3	99.43%	99.74%	99.67%	99.6%

* = DOWNTIME DURING GENERATING UNIT OUTAGE NOT INCLUDED

4.2.4 OPERATOR AID COMPUTER SYSTEM

The Operator Aid Computer Systems at Catawba are Model 4400 Honeywell Corporation with 64K CPU memory and a one million word bulk core memory system. Rotating and tape bulk systems are not used due to their relatively slow memory access times and susceptibility to mechanical failures.

4.2.4.1 Color Videos

A compliment of five 19" color CRT's are driven by an AYDIN 5205-C color graphic video Display Generator. Three CRT's are located on the main control boards and have the following functions.

- Alarm Video - Dedicated to displaying plant alarms. Digital inputs are scanned every 400 milliseconds and are alarmed immediately upon detection. Analog values are scanned every 30 seconds and checked for high and/or low alarms as well as rate of change as appropriate. SPDS critical safety function status blocks will be permanently displayed on the bottom lines of the alarm video.

- Utility Video - The utility video is also located on the main control board. An alpha/numeric keyboard is provided to enable the operator to select any display or program available in the OAC. Twelve chart recorder pens are operator assignable from the utility and monitor videos. These 12 pens are located on the main control board in four-three pen recorders. Operator can select high and low ranges for any of the inputs available in the OAC.

- o Monitor Video - Same function as utility video above and includes its own keyboard similar to that described above. Panel buttons are also provided to allow the utility and monitor videos to display the contents of the alarm video in case of a failure of the alarm video.

Additional Videos

- o Performance Video - The performance video is located in the Computer Room and serves several different users to avoid interfering with plant operators in the Control Room. It is used for plant records and performance, reactor engineering, programmers console, field input calibrations, etc. All OAC displays, programs and functions are available at this console including the capability to display the contents of the alarm video.

- o Technical Support Center Video - This video is located in the Technical Support Center and has the same capabilities as the utility, monitor, and performance videos thereby making available the SPDS, supporting displays, alarm video information as well as access to all plant inputs to the OAC.

4.2.4.2 Typers and Printers

The following printers/typers are provided. An alarm typer is located in the Control Room which provides a hard copy of all alarms which appear on the alarm video. Also printed are status change messages such as pumps, motors, valves on/off open/closed, etc.

A utility typer is also provided in the Control Room. This typer allows the operator to print the output of a number of programs as well as any OAC input desired. Generation and plant logs are also typed out automatically each hour. The utility typer doubles as a backup to the alarm typer in case of failure.

A performance typer is located at the performance console in the Computer Room. Also, high speed line printers are available to type out large volumes of data from the OAC as needed.

4.2.4.3 Floppy Disc Drives

Magnetic floppy disc drives are also provided in the Computer Room for dumping copies of all OAC programs on a weekly basis in case of OAC program loss or damage. This allows the OAC to be restored to the latest version of system programs rapidly.

4.2.5 INSTALLATION AND TESTING

The SPDS will be thoroughly tested prior to being made available to the operator. This testing will include actual operation of the SPDS logic on the OAC for several weeks during startup, shutdown and normal operation. This testing will be transparent to the operators as the displays are inhibited from operating. However, an alarm summary table will be used to capture SPDS alarm changes as well as SPDS input parameter changes. This testing has been completed on the original version of McGuire's SPDS logic and was very useful in verifying the proper functioning of the SPDS as well as revealing some discrepancies primarily resulting to dynamic plant conditions. The results from these tests will be incorporated into a revision of the SPDS logic.

4.2.6 MAINTENANCE

Since the SPDS is being installed in the existing OAC Systems, the maintenance functions are already well defined and organized and demonstrated by the high availability of these computers.

4.2.6.1 Central Process Computer Group

Briefly, a Central Process Computer Group is responsible for generating and maintaining all application software. Further, this group provides hardware support and maintains a central set of spare parts for the OAC's. This group is responsible for the overall functioning of these systems including the implementation of factory recommended alterations and enhancements. Vendor support is also available when needed.

4.2.6.2 Station Maintenance Personnel

The Instrument and Electrical Section at the station is responsible for day to day hardware maintenance and preventative maintenance of the OAC Systems. Back up maintenance and spare parts support is generally available in less than four hours from the Central Process Computer Group located in Charlotte. The station also maintains a supply of normally required spare parts.

4.2.6.3 Availability Reports

Availability of the OAC's is monitored routinely. Additionally, procedures will be implemented to monitor SPDS logic and input performance to assure high availability of this function with periodic reviews of alarm summary tables.

4.3 SPDS SAFETY ANALYSIS

4.3.1 INTRODUCTION

A safety analysis review has been performed in order to verify the technical correctness and completeness of the McGuire and Catawba SPDS design. This independent review consisted of a series of detailed comment summaries which were provided to the SPDS designers at each phase of the design process. As a result of the review, the generic decision trees developed by the Westinghouse Owner's Group (WOG) have been slightly modified. These modifications have been incorporated within the SPDS logic and also in the emergency procedures referenced by the SPDS. The bases for the SPDS design, a comparison with the generic Owner's Group methodology, and the conclusions of the safety analysis review are discussed in the following sections.

4.3.2 OVERVIEW AND BASES

The SPDS is structured around the monitoring of the six critical safety functions and status trees specified in the Westinghouse Owner's Group Emergency Response Guidelines (ERGs) dated September 1, 1983. In order of decreasing severity these functions are: Subcriticality, Core Cooling, Heat Sink, Integrity, Containment, and Inventory. This set of critical safety functions and the corresponding status trees have undergone an exhaustive review within Westinghouse and the Owner's Group. The September 1, 1983, Revision 1 version, which is the bases of the McGuire/Catawba SPDS, also includes recommendations based on the NRC review of the ERGs. The NRC review culminated in the issuance of an SER dated June 1, 1983.

A Duke Power Company program was undertaken to convert the generic Emergency Response Guidelines into plant specific Emergency Procedure Guidelines. This program resulted in the identification of three modifications to the generic critical safety function status trees, each of which is included in the SPDS logic. Each modification can be considered as an enhancement of the generic version which remains consistent with the overall intent of the ERGs.

Modification #1: The generic status tree for Subcriticality is only valid following a reactor trip, since during normal operation the first branch point is "Power >5%", and a "yes" answer directs the operator to the "Response to Nuclear Power Generation/ATWS" procedure. In order for the SPDS to provide a meaningful unalarmed indication for all critical safety functions during normal power operation, a new first branch point in Subcriticality, "Reactor Trip Required", has been added. A "no" answer to this first decision point is appropriate during normal operation, and a valid unalarmed condition is indicated. A "yes" answer leads to the "Power >5%" branch point for continuation of the generic post-trip logic.

Modification #2: The generic status tree for Integrity has been revised to alarm on high Reactor Coolant System pressure. The alarm setpoint has been selected to indicate a high pressure condition that is well in excess of normal post-trip conditions. The alarm is useful with respect to alerting an overpressure condition and reduces the potential for challenging the pressurizer code safety valves.

Modification #3: The generic status tree for Containment has been revised to include monitoring of the containment hydrogen concentration.

With the exception of the above modifications, the McGuire/Catawba SPDS logic is designed to monitor plant computer field inputs so as to generate alarm conditions consistent with the Owner's Group critical safety function status trees.

4.3.3 SUMMARY OF SPDS LOGIC

The SPDS logic monitors the indications of pertinent plant instrumentation for comparison with setpoints that are characteristic of degraded plant conditions. The logic is designed to provide the best representation possible for each of the decision points in each of the status trees. Since the decision points have been uniquely specified in the Owner's Group documentation, development of the logic was a relatively straightforward task. Plant specific setpoints have been developed which include applicable instrument error and the effects of a degraded instrument environment as required.

Recognizing that the WOG critical safety functions and status trees have been subjected to a thorough NRC review, and due to the relative simplicity of converting the status trees into a logic scheme, a detailed summary of the logic utilized in the McGuire/Catawba SPDS is not warranted.

4.3.4 CONCLUSIONS

The McGuire/Catawba SPDS has been subjected to a thorough and independent review to ensure that the logic design accomplished the intended critical safety function monitoring task. The SPDS is based on the Westinghouse Owner's Group critical safety functions and status trees released with the Emergency Response Guidelines, Revision 1, dated September 1, 1983. The plant specific logic includes three minor enhancements consistent with the overall intent of the generic version. The utilization of the SPDS has been verified to be fully integrated with the upgraded emergency procedures. The logic has been verified to be technically correct. The SPDS will enhance operator response to transient events by alerting the operator to symptoms of degraded plant conditions, and by automating the prioritization of subsequent operator actions.

This review was completed pursuant to 10CFR 50.59 and has been determined not to result in an unreviewed safety question. The proposed SPDS meets or exceeds the existing design criteria as described in the Final Safety Analysis Report.

4.4 CATAWBA SPDS STATUS

Below is the status of the Catawba SPDS as of February 1984. The Catawba SPDS design is currently being developed using the McGuire SPDS system and will emerge as a separate design.

An analysis of the Westinghouse generic Emergency Response Guidelines (ERG's) resulted in the initial design basis and definition of the SPDS which was developed and approved in May, 1982.

In June, 1982, the Westinghouse Decision Trees which define the status of the six critical safety functions contained in the Westinghouse ERG's were defined in "and/or" logic arrays and OAC inputs were selected to drive this logic.

This logic was coded into software and installed on McGuire Unit 2 OAC in July, 1982 where the logic was tested to assure proper operation. At this time, the display presentations were reviewed and approved.

The Validation and Verification Plan was developed by the Design Engineering Department providing an independent analysis of the SPDS design and software developed by the Steam Production Department. This V&V process was then applied to the initial SPDS software. At this time, the SPDS software was installed on the McGuire Unit One OAC with the SPDS display inhibited. An alarm summary table was used to store all SPDS alarms and related SPDS field input alarms. This allowed a dynamic testing of the SPDS logic during normal unit maneuvers, start-ups and shutdowns.

In November, 1982, concepts for the supporting displays were developed which would provide methods for the operators to investigate the causes for SPDS alarms.

In January, 1983, revised Westinghouse Decision Trees (dated 9/1/82), results from Dynamic Testing, and information from the V&V Program were incorporated in a revised version of the SPDS logic.

This revised logic was installed on the OAC and V&V reinitiated in February, 1983. Alternative supporting displays were reviewed and design selected in March, 1983.

Additional revisions to the Westinghouse Emergency Response Guidelines were received in July 1983 which resulted in additional revisions to the SPDS logic and software. The Nuclear Production Department Reactor Safety Section performed numerous reviews and provided many suggestions on improving the SPDS design.

The supporting display system was developed and a Human Factors review was completed in October 1983 by the Control Room Review Team. Their comments were incorporated into the final SPDS design. A review of the Shift Technical Advisor function was also confirmed and a determination made to provide an additional CRT and keyboard for use by the STA. This CRT will be installed on the operators desk in the "horse shoe" area of the control room.

Training information was developed and provided to the Catawba operators in November 1983.

The software for the Catawba SPDS and supporting displays will be completed in March 1984. The final V&V is also in progress at this time and will be completed during March.

The SPDS should be installed and functional on schedule.

PROPOSED SCHEDULE FOR SUPPLEMENT 1 TO NUREG-0737 PROVISIONS - CATAWBA NUCLEAR STATION

Milestone Activity	Completion Date
CRR	
o Steering Committee Formed	October, 1981
o Program Concept	January, 1982
o Review Team Selected	February, 1982
o Final Draft of Review Plan	May, 1982
o Plan Presentation to NRC	May, 1982
o Bio Technology Hired for Human Factors Assistance	June, 1982
o Mockup Space Rented and Construction Began	July, 1982
o Review Phase Activities, Unit 1	March, 1983
o Assessment of Unit 1 HED's	June, 1983
o SUMMARY REPORT, Unit 1	JUNE, 1983*
o Review Phase Activities, Unit 2	August, 1983
o Assessment of Unit 2 HED's	March, 1984
o SUMMARY REPORT, Unit 2	MARCH, 1984*
SPDS	
o Initial Design Basis & SPDS Approved	May, 1982
o Initial Design V&V and Dynamic Testing	November, 1982
o Revised SPDS Design and V&V	January, 1984
o SPDS Training	April, 1984
o SPDS Safety Analysis	March, 1984
o SPDS OPERATIONAL, UNIT 1	MAY, 1984*
o SPDS OPERATIONAL, UNIT 2	OCTOBER, 1986*

* Indicates proposed commitment date

SPDS RELIABILITY

One of the most important requirements of a good Safety Parameter Display System is reliability.

Duke Power is a leader in the field of design, construction, and operation of Nuclear Power generation stations. With this experience and the guidelines established by Westinghouse and the NRC, Duke has developed a Safety Parameter Display System we know will perform the required task of reliable operator guidance.

The computer systems, on which the Safety Parameter System runs, maintains the high degree of reliability necessary for this function. Duke was one of the first power companies to install an Operator Aid Computer (OAC) in a power plant; this was at Marshall 1 and 2 in May, 1964 (See Attachment 1). Since that first OAC computer system at Marshall, Duke has developed, installed, and is maintaining 23 OAC and Security computers. All the OAC and Security systems installed are from the same family of computers, GE/Honeywell. This consistancy allowed Duke to obtain a high degree of expertise in both the areas of software and hardware.

Duke has a centrally located staff of highly trained and experienced personnel called the Process Computer Applications Unit (PCU) that develops, installs, and supports these systems. Duke is one of the few companies that develop and maintain their own OAC and Security software. The OAC and Security hardare is initially installed by the PCU technicians. Primary maintenance is supplied by the station I & E technicians with the PCU technicians supplying backup technical support as needed. The PCU's main core of 20 individuals each have an average of 16 years experience (Attachment 2). This staff of experts in the field of Operator Aid and Security Computers supports the reliability necessary for an SPD system.

Duke was one of the first power companies to develop graphics on its Operator Aid computers. This was accomplished in 1972 on the Belews Creek system. Graphics today play an important role in nuclear station operation and is a vital part in supporting the SPD system function. On the Duke nuclear station OAC's, graphics aid the operator in determining exact causes of SPDS alarms so that quick resolution can be obtained. This in turn establishes further reliability in the system.

Another item that reflects the reliability of the OAC's at Duke is the system availability. System availability is a measure of system uptime based on the generating unit being on line. The PCU has been able to maintain a total system adjusted availability for all its nuclear stations of greater than 99% (Attachment 3).

Nuclear Station design also supports the OAC reliability through its redundant power source. Regulated power to the OAC's is supplied primarily by station batteries through a static inverter and static transfer switch. If the battery chargers should fail, the batteries would maintain the load. If for some reason the inverter should fail, power would be transferred by the static transfer switch to a nonregulated power source. (See Attachment 4)

This was just a brief list of reasons why we at Duke are confident that the Safety Parameter Display System we have developed and implemented, will be a reliable aid to the Nuclear Station Operators.

ATTACHMENT 1

POWER GENERATION
PROCESS COMPUTERS

STATION & UNIT	COMMERCIAL DATE	FUEL	STEAM SUPPLY	T.G.	UNIT MW	COMPUTER MODEL	COMPUTER IN SERV.
Marshall 1&2	3-65/4-66	Fossil	CE	GE	385/380	GE412	5/64
Marshall 3	5-69	Fossil	CE	GE	630	GE4020	10/68
Marshall 4	5-70	Fossil	CE	GE	630	GE4020	10/69
Oconee 1	7-73	Nuclear	B & W	GE	860	GE4020	9/70
Oconee 1 Upgrade	--	--	--	--	--	H45000	1/85
Oconee 2	9-74	Nuclear	B & W	GE	860	GE4020	3/72
Oconee 2 Upgrade	--	--	--	--	--	H45000	5/85
Oconee 3	12-74	Nuclear	B & W	GE	860	GE4020	6/73
Oconee 3 Upgrade	--	--	--	--	--	H45000	11/84
Oconee Security	--	--	--	--	--	H4400	1/84
Cliffside 5	6-72	Fossil	CE	GE	572	GE4020	12/71
Keowee	4-71	Hydro	--	AC/W	165	GE4020	9/70
Jocassee 1,2, 3, and 4	12-73/ 5-75	Pumped Hydro	--	AC/W	680	GE4010	7/73
Belews Creek 1	8-74	Fossil	B & W	W	1140	GE4010	8/73
Belews Creek 2	12-75	Fossil	B & W	W	1140	GE4010	10/74
McGuire 1	12-81	Nuclear	W	W	1180	H4400	6/76
McGuire 2	3-84	Nuclear	W	W	1180	H4400	1/79
McGuire Security		--	--	--	--	H4400	8/78
Catawba 1		Nuclear	W	GE	1153	H4400	10/79
Catawba 2		Nuclear	W	GE	1153	H4400	10/81
Catawba Security		--	--	--	--	H4400	10/82
GO Staging - 4010		--	--	--	--	GE4010	3/76
GO Staging - 45000		--	--	--	--	H45000	1/84

23 Installations
5/9/85

ATTACHMENT 2

PROCESS COMPUTER UNIT
EMPLOYEE'S YEARS OF SERVICE

Employee	Years of Service
Boyte, F. T.	9
Burns, K. L.	20
Collins, R. C.	15
Cope, L. J.	19
Faulk, G. N.	9
Gabriel, G. B.	18
Goode, D. W.	15
Hartsell, T. L.	25
Holsonback, H. G.	15
Mason, E. B.	19
McCraw, J. A.	20
McCraw, J. S.	20
Miller, C. R.	15
Morgan, B. W.	15
Moser, III, M. I.	14
Moss, G. T.	9
Powell, M. L.	10
Rudisill, R. G.	24
Van Vynckt, M. E.	13
Walker, S. A.	17
	321 Man-years

ATTACHMENT 3

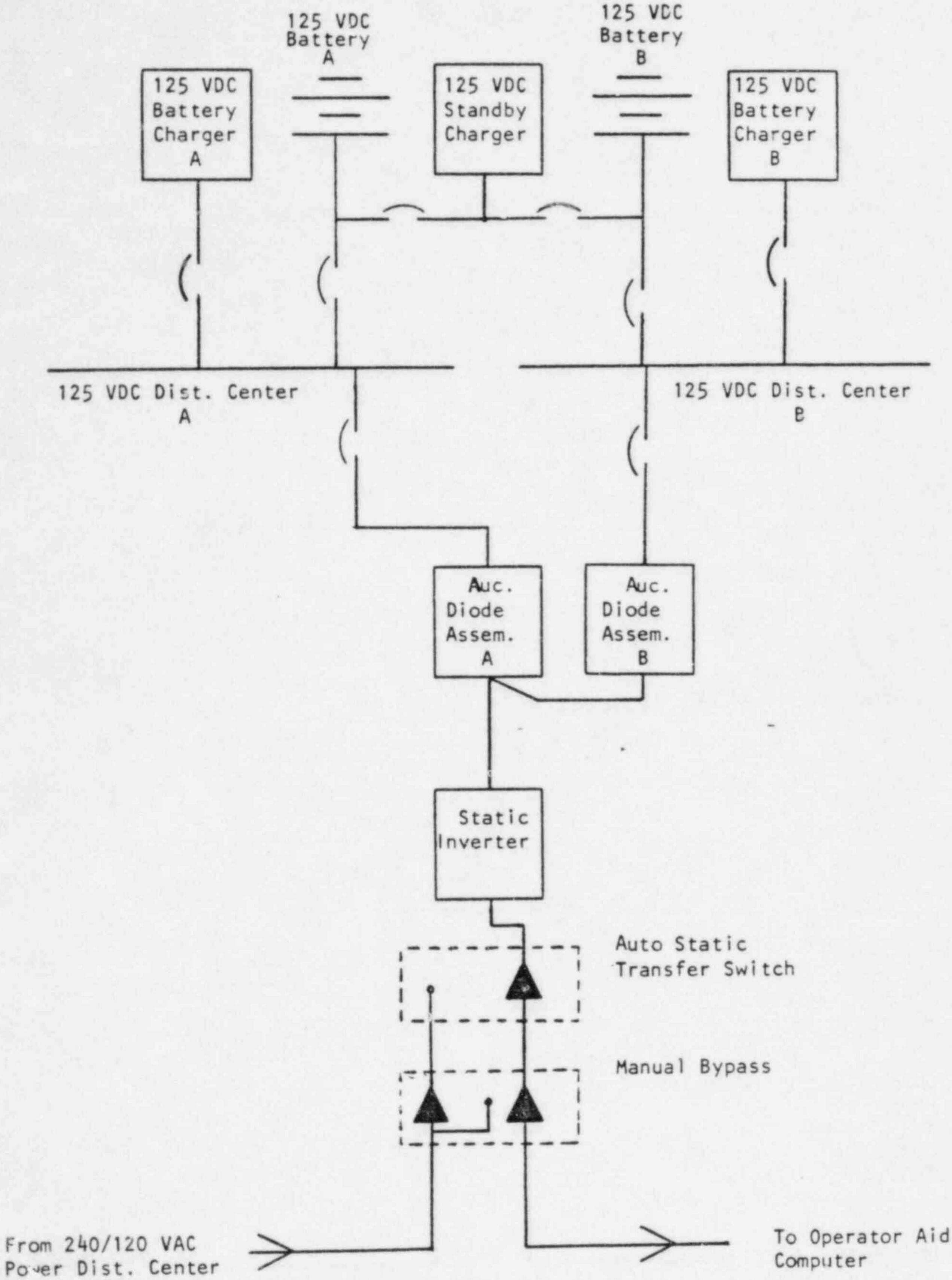
PROCESS COMPUTERS NUCLEAR STATION OAC AVERAGE AVAILABILITY

1984

STATION	SID	% HARDWARE	% SOFTWARE	% TOTAL SYSTEM UPTIME	% ADJUSTED SYSTEM UPTIME*
Oconee 1	UAX	99.92	99.91	99.71	99.83
Oconee 2	UAY	99.90	99.94	99.72	99.72
Oconee 3	AAT	99.95	99.76	98.73	99.83
McGuire 1	ATN	99.47	99.98	99.38	99.46
McGuire 2	ATO	99.81	99.99	99.49	99.81
Catawba 1	BGM	99.25	99.85	99.04	100.00
Catawba 2	BGN	97.54	99.81	95.44	100.00
TOTAL		99.41	99.89	98.93	99.81

* TOTAL SYSTEM UPTIME BASED ON GENERATING UNIT ON LINE

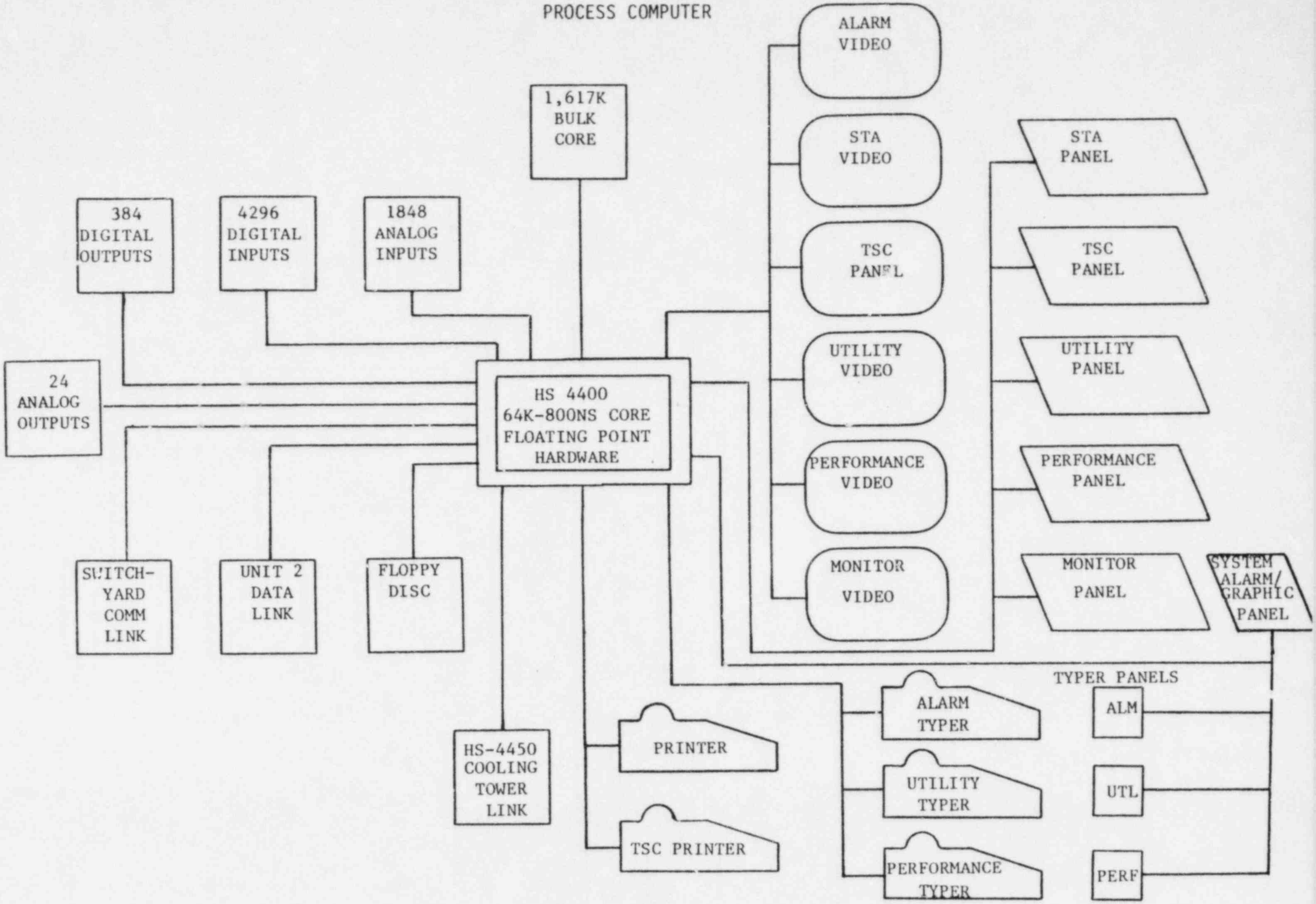
ATTACHMENT 4

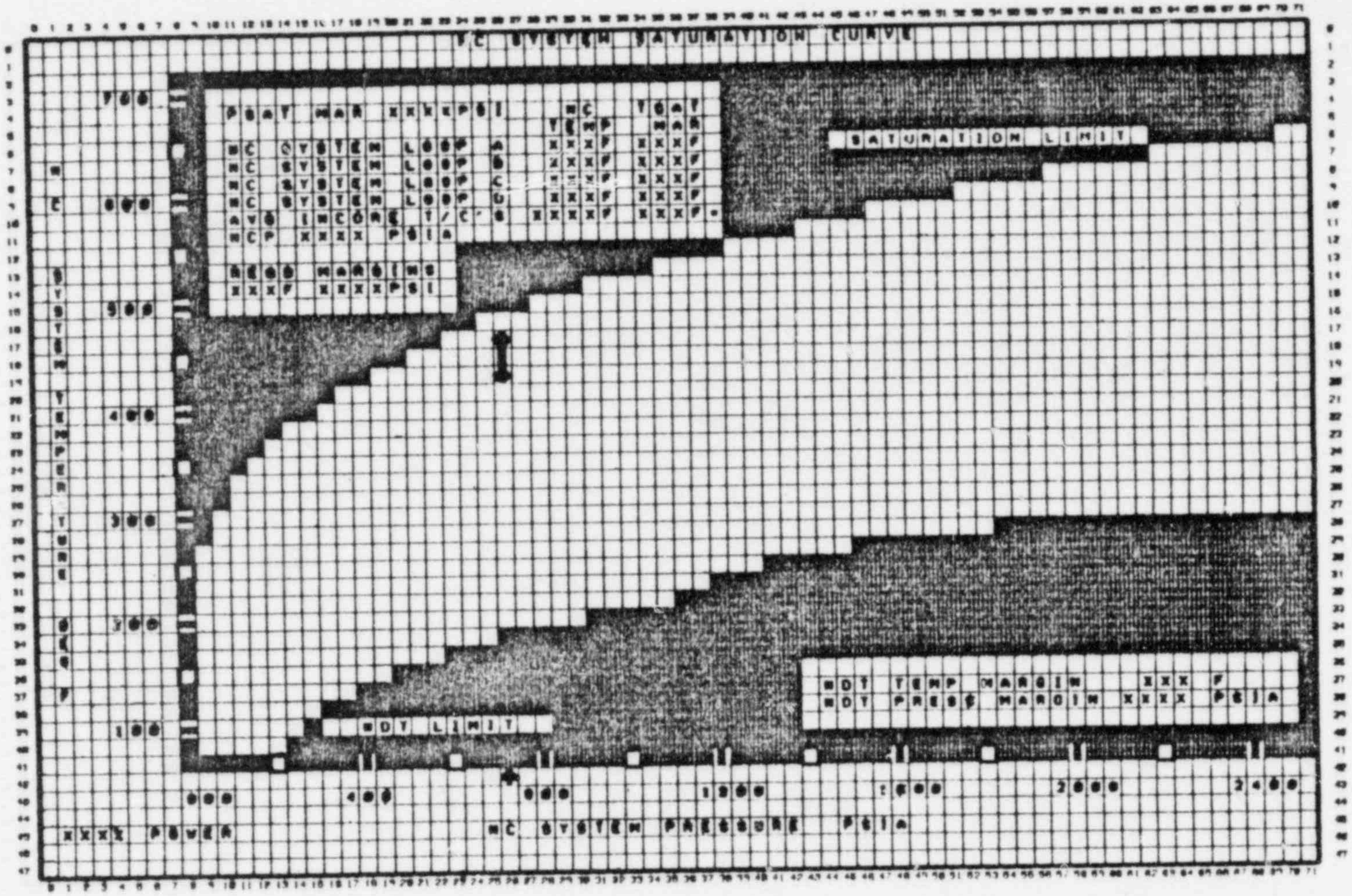


DUKE POWER COMPANY

CATAWBA NUCLEAR

PROCESS COMPUTER





DUKE POWER COMPANY

CATAWBA NUCLEAR STATION UNIT 1 & 2

1 SEGMENT 1 ADDED TO DATA BLOCK

CATAWBA SPDS
MILESTONE SCHEDULE

Design Basis Formulation	March, 1982
Review of Design Basis	Apr-June, 1982
Development of Design Documentation Standard	Mar-Apr, 1982
Initial Logic Design	Mar-Apr, 1982
Computer Program Request	April 30, 1982
Software Development	May-June, 1982
SPDS Installed (Displays Disabled) on McGuire 2	June 1, 1982
Software V&V Plan Development	July, 1982
Review of V&V Plan by Dave Cain	July, 1982
SPDS Alarm Summary Program (for Dynamic Testing at McGuire I)	Jul-Aug, 1982
V&V of Software Completed	August, 1982
SPDS Installed On McGuire #1 Decision - Not To Turn On SPDS	August, 1982
Westinghouse ERG'S Revised	September, 1982
Supporting Displays Development	Oct-Nov, 1982
Review of Supporting Displays	Nov-Dec, 1982
Revised SPDS Logic - (Westinghouse ERG Revision)	December, 1982 - January, 1983
Revised SPDS Software	December, 1982 - January, 1983
V&V Re-initiated	February, 1983
Review of Logic & Displays	Feb-March, 1983
0737, Supplement 1 Response	April, 1983
Shift Technical Advisor's Panel Proposed	May, 1983

Revised SPDS Logic Due to Westinghouse ERG Revisions	May-Oct, 1983
Set Points for Degraded and Non-degraded Containment	October, 1983
Human Factors Control Room Review of SPDS/Supp Displ	October, 1983
Operator Training Developed	October, 1983
Final Version of Logic For Catawba	December, 1983
Catawba Software Completed	March, 1984
V&V of Catawba Software	Mar-Apr, 1984
Catawba SPDS & EOP's Installed	May, 1984
Human Factors Survey Completed	June, 1984
Low Power License	December, 1984
Full Power License	March, 1985

spdsagen.513

Duke Power Company
Safety Parameter Display System Concept

Role:

The primary objective of the Safety Parameter Display System is to provide plant Operations personnel with an overview of the safety status of the plant. This objective is being met by defining the role of the Safety Parameter Display System as an operational aid which will provide plant Operations personnel with an overview of how well the plant critical safety functions are being maintained. The critical safety functions are defined by the Emergency Operating Procedures Guidelines developed by the individual NSSS Owners Groups. In the case of McGuire and Catawba, the Emergency Operating Procedures identify the following six critical safety functions:

- Subcriticality
- Reactor Coolant System Integrity
- Core Cooling
- Reactor Coolant Inventory
- Heat Sink
- Containment Integrity

In the ATOG Guidelines the B&W Owners Group has identified the following five critical safety functions:

- Reactivity
- RCS Integrity
- Core Cooling
- Secondary Heat Removal
- Containment Integrity

Design Considerations:

The Safety Parameter Display System is an operational aid for overview and execution of the new Emergency Operating Procedures. It is not essential that the SPDS be operational for plant personnel to determine the safety status of the plant or to execute any of the new Emergency Operating Procedures since adequate instrumentation, instructions, and training will exist or be provided independent of the SPDS. However, the Safety Parameter Display System can be an effective aid for facilitating

R. L. Brown
Production Support Department

this safety overview and executing the procedures. Consequently it is not necessary that the SPDS meet safety system criteria such as seismic qualification, single failure criteria, etc.

The SPDS will be implemented on the existing Operator Aid Computers for several reasons:

- The existing plant Operator Aid Computer Systems meet the equipment requirements based upon the defined role of the SPDS.
- The OAC Systems have been proven over many years to be highly reliable (approximately 99.9% measured availability on an annual basis).
- Duke has the inhouse expertise necessary for expeditious definition and implementation of the SPDS.
- Each plant OAC has several thousand implemented inputs which provide readily available parameters for developing the SPDS - a situation which further enhances the project schedule.
- The plant Operator Aid Computer represents a "normal" and familiar source of much operating information for the plant staff and installation of the SPDS function on the existing system will enhance the effectiveness of the system.
- The installation of a separate SPDS is undesirable from a human factors perspective since it introduces additional devices in the control room and competes for the operator's attention with other existing data systems.

Displays:

The SPDS Display will consist of five or six blocks arranged either vertically or horizontally on one of the CRT screens existing within the control boards. These blocks will represent the critical safety functions of the Emergency Operating Procedure Guidelines. In the case of McGuire and Catawba, the color of the Critical Safety Function blocks will change depending on departure of the CSF from the normal operating envelope. In the case of Oconee, color CRT's are not available and blinking or other mechanisms will be used to identify departure of a critical safety function from the normal envelope. The logic to identify the departure from normal envelope is developed directly from the Westinghouse "status trees" or the B&W ATOG guidelines as appropriate. The six critical safety function blocks will be continuously displayed on one of the CRT's on the control board and cannot be removed through operator action. Further, the CSF display will be large enough such that the shift supervisor or other Operations personnel can readily determine the status of each critical safety function from the back of the Control Room without requiring access into the immediate control board area.

SPDS Use:

In any mode of plant operation, the SPDS display will either confirm that the basic critical safety functions are being satisfactorily fulfilled or will identify to the operator the departure (and in some cases the degree of departure) of the critical safety function from the normal envelope. The fact that a critical safety function (or functions) has departed from the normal operating envelope will be readily apparent to the operator who can execute the appropriate EOP response to restore the unit within normal boundaries. Secondary displays (which are not a part of the SPDS) will be identified as the EOP's are developed to assist the operator in this task. Due to the simple nature of the Safety Parameter Display System and the existence of other aids, displays, indicators, and the most important element of operator training, the operator can effectively perform his surveillance and provide an appropriate response without the SPDS. However, the SPDS should be an effective aid in this overview/response mechanism.

R. L. Brown
Production Support Department

Duke Power Company
Safety Parameter Display System
Development Program Plan

The Duke Safety Parameter Display System will be developed through a planned systematic program. The plan centers on definition, implementation, and evaluation of safety parameter display system for McGuire Nuclear Station. This pilot program will be the basis for development of the Safety Parameter Display Systems for Oconee and Catawba.

McGuire has been established as a pilot location for development of the Safety Parameter Display System since the Westinghouse Owners Group has defined the six critical safety functions sufficiently to allow implementation at McGuire. Further, a more realistic evaluation of the SPDS is possible since McGuire Unit 1 is operational. It is anticipated that the SPDS Displays for Oconee can be developed expeditiously after the B&W Owners Group,as part of their ATOG guidelines, has fully defined the five critical safety functions and their normal envelopes.

The role of Safety Parameter Display System has been defined by Duke Power Company to provide the overview of the critical safety functions identified in the Emergency Operating Procedure Guidelines as an aid to the Operations staff in their overview of plant safety status and execution of the symptom oriented procedures. These procedures will be developed and reviewed during the course of the Control Room Review, at which time the SPDS will be reviewed and validated as an integral part of the EOP evaluation. Any secondary displays desired to support the basic Safety Parameter Display System will be identified in the course of this review.

The elements of the program plan for the SPDS development and their sequence are:

- Develop logic diagrams and Boolean equations from the Westinghouse status trees.
- Verify and select OAC inputs necessary to support the SPDS Displays.
- Develop a complete package documenting and defining one of the critical safety functions. This package will be reviewed and refined as the "standard" for documentation and definition of the critical safety functions. Concurrently, the Review Team will identify the display colors and orientation of the critical function CRT displays.

R. L. Brown
Production Support Department

- Develop the remaining five critical safety function definitions.

- Computer coding of the SPDS and evaluation on the McGuire 2 Operator Aid Computer.

- Installation of the SPDS on McGuire 1 and evaluation by the Review Team, Procedures Team, and McGuire Operations personnel.

- In the course of the Control Room Review, validate the use of the SPDS as an aid in providing the overview of plant safety status to support the Emergency Operation Procedure Guidelines and define any desired supporting displays.

R. L. Brown
Production Support Department

WESTINGHOUSE ERGs/CATAWBA EPGs

- WESTINGHOUSE OWNERS GROUP EMERGENCY RESPONSE GUIDELINES (ERGS), REVISION 1 CRITICAL SAFETY FUNCTION (CSF) STATUS TREES WERE THE STARTING POINT FOR DEVELOPMENT OF THE CATAWBA SPDS

- SAFETY ANALYSIS UNIT MODIFIED THE ERGS FOR PLANT SPECIFIC USE AT CATAWBA. RESULT WAS THE CATAWBA EMERGENCY PROCEDURE GUIDELINES (EPG) CRITICAL SAFETY FUNCTION STATUS TREES

- EPG STATUS TREES HAVE THE SAME CSF LIST, THE SAME HIERARCHY OF OPERATOR RESPONSE, AND THE SAME INTERFACE WITH SPECIFIC EMERGENCY PROCEDURES AS THE ERGS

- TWO TYPES OF MODIFICATIONS WERE MADE IN TRANSFORMING THE EPG TREES INTO THE EPG TREES:

  - ADDED LOGIC BRANCHES
  - REVISED SOME SETPOINTS

- ALL MODIFICATIONS HAVE BEEN PREVIOUSLY IDENTIFIED TO THE NRC IN NUREG-0737 SUPPLEMENT 1 RESPONSE

- ALL MODIFICATIONS HAVE BEEN SUBMITTED TO THE NRC AND APPROVED IN CONNECTION WITH THE CATAWBA EMERGENCY PROCEDURE REVIEW

  - LOGIC SUBMITTAL (JUNE 18, 1984)
  - SETPOINT SUBMITTAL (OCTOBER 17, 1984)
  - APPROVAL IN CATAWBA SER SUPPLEMENT 4 (DECEMBER, 1984)

SAFETY ANALYSIS UNIT
FUNCTIONS

- PART OF NUCLEAR ENGINEERING GROUP AND REACTOR SAFETY SECTION

- UNIT'S RESPONSIBILITIES INCLUDE . . FOLLOWING:

  - SAFETY ANALYSIS OF PLANT OPERATING TRANSIENTS AND POSTULATED ACCIDENTS

  - DEVELOPMENT OF PLANT SPECIFIC TECHNICAL GUIDELINES FOR EMERGENCY PROCEDURES

  - REVIEW AND APPROVAL OF TECHNICAL CONTENT OF PLANT EMERGENCY PROCEDURES

  - REVIEW OF VENDOR EMERGENCY RESPONSE GUIDELINES AND BACKGROUND MATERIAL

- UNIT IS THE INTERFACE FOR THE FOLLOWING GROUPS AS THEY AFFECT THE CATAWBA EMERGENCY PROCEDURES

  - NRC REVIEWERS

  - PLANT OPERATIONS STAFF

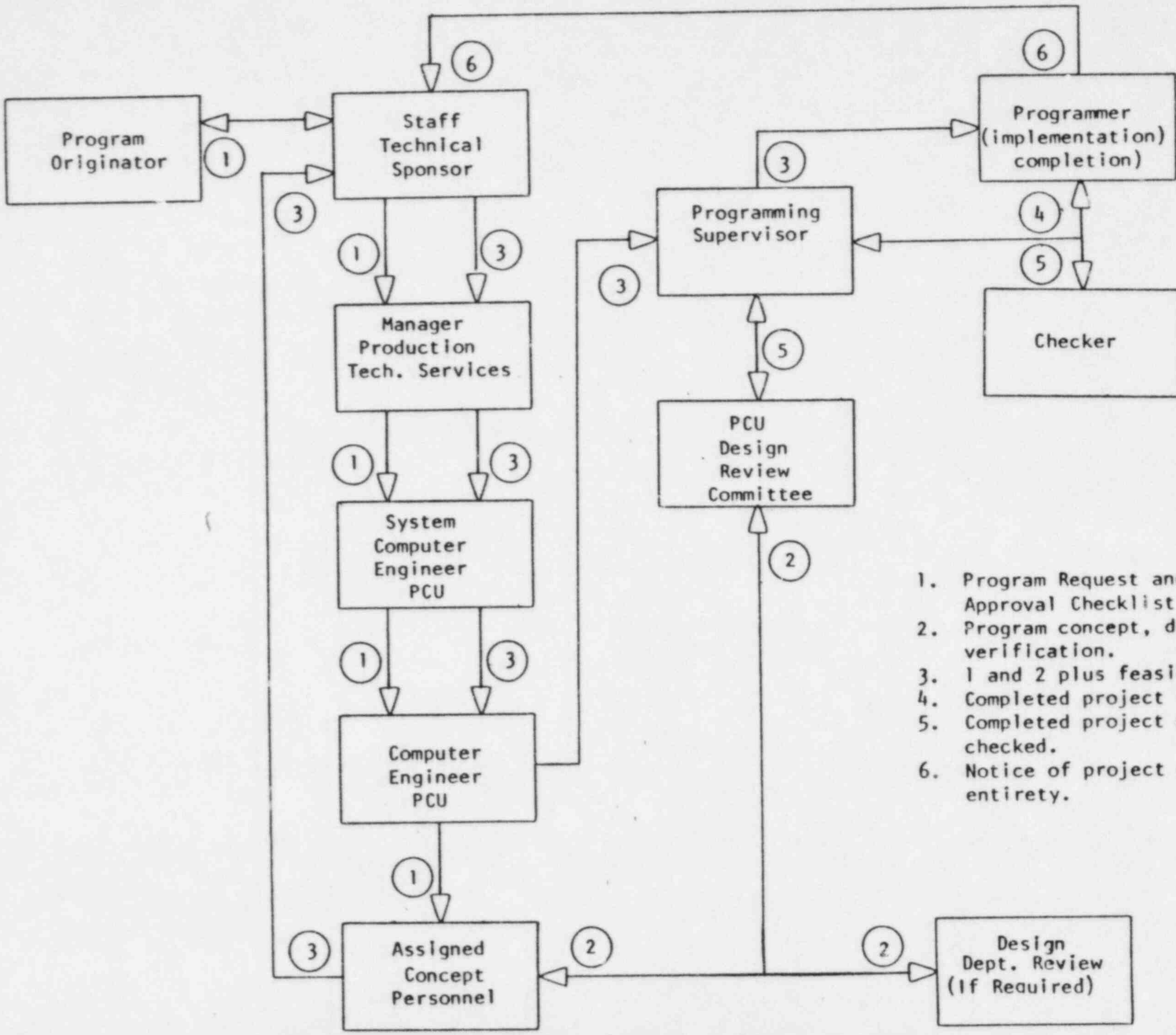
  - VENDOR TECHNICAL STAFF

  - PLANT TRAINING STAFF

SAFETY ANALYSIS UNIT
REVIEW OF SPDS LOGIC

- INITIAL REVIEW OF SPDS LOGIC WAS COINCIDENT WITH DEVELOPMENT OF CATAWBA EPG CSF STATUS TREES
- ELEMENTS OF SAFETY ANALYSIS UNIT REVIEW
  - ADDED LOGIC BRANCHES TO AGREE WITH EPG TREES
  - REVISED SPDS SETPOINTS TO AGREE WITH EPG SETPOINTS
  - REVISED SUBCRITICALITY LOGIC TO MAINTAIN GREEN INDICATION DURING NORMAL REACTOR TRIPS
  - REVISED CORE COOLING LOGIC TO PROPERLY MODEL UHI DYNAMIC HEAD RVLIS
  - CHECKED LOGIC FOR CORRECT MODELING OF MULTIPLE INSTRUMENTATION CHANNELS
  - CHECKED CORRECTNESS OF ALL LOGIC IN ALL CSF FUNCTIONS
- SAFETY ANALYSIS UNIT IS CONTINUALLY INVOLVED IN SPDS MAINTENANCE PROGRAM
- SAFETY ANALYSIS UNIT CONTINUALLY MONITORS CATAWBA AND MCGUIRE EMERGENCY PROCEDURE CHANGES FOR IMPACT ON SPDS

The Path of a Program Request
For the Process Computer Unit



1. Program Request and Routing and Approval Checklist.
2. Program concept, documentation, verification.
3. 1 and 2 plus feasibility study.
4. Completed project from programmer.
5. Completed project which has been checked.
6. Notice of project completion in entirety.

PROCESS COMPUTER PROGRAM RESPONSIBILITIES

I. Program Originator

A. This individual is the one charged with the responsibility of the initial program's written purpose, j: stification, functional definition, and other information required by the "Program Request Procedure." This individual should gather the ideas and comments of others regarding the intended program (including the Staff Technical Sponsor's). Very seldom should a program requested be the sole product of only one individual's thoughts.

B. After gathering the inputs of other knowledgeable persons and processing them into the program's definition, the Program Originator should complete the "Program Request Form" with all the required attachments and initiate the "Routing and Approval Checklist." The checklist should then be routed through the appropriate channels to the Station Superintendent or Manager or the Group Manager for approval/disapproval and signature.

II. Staff Technical Sponsor

A. This individual is a member of one of the Production Department's General Office Staff and has been assigned the responsibility of reviewing and coordinating certain program requests. The assignment of Staff Technical Sponsor is based on the group or unit jurisdictional policy and the station responsibility of indivduals within the group or unit. As programs are added to the existing base of Production Technical Services programs, they are assigned a Staff Technical Sponsor based on the function(s) desired and the station(s) involved.

B. Part of the Program Originator's responsibility is to plan with the Staff Technical Sponsor the program that is being requested. This should always take place for several reasons: the function being requested may already exist on another station's computer or in another program at the requesting site, changes to regulations, criteria, or operating procedure may be forthcoming and may impact the program requested or delete its requirement altogether, or the requested program could be combined with another similar request from that or another station and provide a better function of both. The Staff Technical Sponsor is responsible for the coordination of all related technical inputs into the original program request. Difference of opinion between technical areas should be resolved by the Staff Technical Sponsor before submitta, to his group Manager. If this is not possible, it becomes the esponsibility of his Group Manager and the Originator's Manager to resolve.

C. After the original program request or revision is passed on to the Production Technical Services from the Staff Technical Sponsor, it will return to the Sponsor after a feasibility study has been completed by the assigned unit. The Staff Technical Sponsor is responsible for determining if the assigned unit's feasibility study is acceptable in those areas affected by the requested program. He must also decide if the cost/benefit indicates whether the program is worthy of completion and how the tentative schedule for implementation will impact the areas involved with the program.

D. If the assigned unit must revise the functional definition, the Staff Technical Sponsor is responsible for determining if the revision causes the program's scope to deviate from the intended definition.

III. Manager Production Technical Services

This individual determines the responsible unit to handle the program request and whether a joint effort of units is necessary. The request is then sent to the Process Computer Unit for a feasibility study.

IV. System Computer Engineer and

V. Computer Engineer

These individuals receive the "Program Request" and evaluate the basic scope of the program and overall computer system and Process Computer Unit impact of the request. Appropriate unit supervision determine the individual to handle the software concepts and feasibility study. Comments from the System Computer Engineer and Com puter Engineer are incorporated into the technical remarks to be written by the assigned concept personnel.

VI. Assigned Concept Personnel

This individual is responsible for handling the software concepts required to initiate programming based on a program request. His responsibility includes: drawing an area diagram of the programs with the information and program flow (if more than one program is involved), gathering all of the technical information needed by the programmer from the program request and from the Staff Technical Sponsor and Program Originator, if required, expanding the high-level flowchart from which a programmer can begin coding, expanding the initial program documentation for the programmer, expanding the program verification procedure, and presenting all of this information to the Design Review Committee. After the committee's response, the assigned concept personnel develop the feasibility study in conjunction with the unit supervision and then send this study to the Staff Technical Sponsor.

VII. Design Review Committee

A. This is a group of individuals who are experienced and responsible for the computer system that the program request involves. The personnel involved are the Software Design Review Committee Coordinator, appropriate supervision, assigned concept person, programmer, and the system lead programmer. It is this group that insures that all software introduced into a computer system follows the guidelines for that system and that the introduction of the program requested does not impact the integrity of the whole computer. This committee also works to insure that the best possible methodology is followed in concept design and program execution.

B. The Design Review Committee also is responsible for the final review of a program before it is considered completed. This review includes the actual operation of the program on the computer system for which it is intended (if it is being staged in Charlotte), and an integrated test with the program running while the other system programs interact with this program. Any problems or changes required by the Design Review Committee must be incorporated into program by the programmer before completion.

C. final review of programs implemented on a computer system away from the staging area is accomplished by the computer system's responsible person instead of the full Design Review Committee.

VIII. Programming Supervisor

This individual is responsible for the assignment of personnel to a particular program request. This person also must set the schedule for program completions based on the unit's integrated schedule. The Programming Supervisor is the one responsible for the coordination of efforts between Programmer and Design Review Committee, Programmer and Checker, and Assigned Concept Personnel and Programmer.

IX. Programmer

This individual is responsible for preparing the code necessary for the computer to execute th prepared program. The Programmer must implement the code in an efficient manner to utilize existing space and timing limitations. He must use techniques to write the program exactly as specified by the definition, yet, take advantage of experience and perform his coding to avoid computer system problems or operator interface difficulties. He must empathize with the end-user and make the operation of the program to satisfy "Human Engineering Factors".

A. The Programmer must perform the program loading, debugging, and detailed verification plus upgrading the documentation to its final form. He must follow all of the programming guidelines while performing his duties. These guidelines are covered in the program information booklet available for each computer system.

B. After his initial program completion, the programmer's work is checked thoroughly by a checker. Any problems uncovered are resolved and then checked again.

C. After the checker's final review and approval, the program is tested further by the Design Review Committee which acts as an extension of the checker. Any problems uncovered by the Design Review Committee are to be handled the same as checker-found problems.

D. Upon final program completion(which includes implementation on the computer, checkouts all completed, and all required documents prepared and checked) the program is considered complete and the programmer informs the Staff Technical Sponsor of the completion as well as the Program Originator and Manager of Production Technical Services The Program Originator and Staff Technical Sponsor will be given the opportunity to review the program operation upon completion of implementation.

X. Checker

This individual is responsible for performing independent checks necessary to verify that every instruction of the program causes no erroneous or undefined action, that the documentation reflects a true definition of the program, and that a finished final product is ready to be released for use. This individual is an experienced programmer.

XI. Software Verificaion Procedure for Audit

A. Whenever a "Program Request" is drawn up, the Originator is responsible for including a Verification Procedure as part or the request. This procedure consists of basically two parts: Functional checks (a list of all cases that must be checked) and checkout procedure (a description of the procedure to be used to check each case, including test case parameters and all other information, material, and resources required to perform the test and the expected test results).

B. This procedure will usually be restated to include software checking techniques by the Assigned Concept Personnel. The Verification Procedure will also be expanded by the concept personnel to include all areas that are related and should be verified in the original program implementation and checkout.

C. The Verification Procedure is usually expanded again by the programmer as the details of the program are consummated and the software is implemented and debugged.

D. By the time the program is completed by the programmer, the Verification Procedure has been followed to the extent possible in the General Office. This must be accomplished before a checker can perform his responsibility of verifying the program's documentation and file folder. Each item of the Verification Procedure should be initialed and dated by the programmer performing the verification. Any additional notes should be attached to the Verification Procedure with an explanation as to to the meaning of the attachment. This will become a part of the permanent file for the program to be used for audit purposes.

E. All "Program Requests" are to include a Verification Procedure for the checkout and testing of the program after its initial completion. However, all programs which must be benchmarked because of various regulations or criticality of the program must have indicated in the program's functional definition that benchmarking or periodic reverification is required. This must be factored into the design of the program to afford the ability to perform the periodic program testing in a timely manner by the user of the program.

F. A method for benchmarking and periodic verification must be established along with a procedure and/or checklist to insure that the test has been performed satisfactorily for the organization which is accountable for the program's correct operation. All periodic testing will also be used as a part of the program's permanent record for audit reasons.

PRODUCTION SUPPORT DEPARTMENT
PRODUCTION TECHNICAL SERVICES
SERVICE REQUEST
—ROUTING AND APPROVAL FORM—

(PTS USE ONLY)
REQ. NO. ______

Form 34744 (R5-84)

ATTACH INDICATED FORM

- ☐ Computer Applications (34673)
- ☐ Process Computer (34977)
- ☐ Process Computer I/O (DP-100)
- ☐ Desk Top Computer (34978)
- ☐ Computer Services (16062)
- ☐ Qualification Test (34883)
- ☐ Project Request (34979)
- ☐ Standards Lab Support (34982)
- ☐ Info. Systems Resource Request (14264)

Name of Project ______

INITIAL REQUEST PHASE

1. Originator ______

Group ______

Location ______

Date ______

2. Local Mgmt. ☐ Approval ☐ Disapproval

By ______

Title ______

Date ______

3. Tech. Sponsor's ☐ Approval ☐ Disapproval

Staff Tech. Sponsor ______

Date Rec'd. ______

Tech. Sponsor's Group ______

Dept. ______

4. Staff Mgmt's. ☐ Approval ☐ Disapproval

By ______

Title ______ Date ______

FEASIBILITY AND COST STUDY PHASE

5. Received By Manager, Production Technical Services On ______ By ______

Assigned To ______ Unit Date ______

Estimated Cost $ ______ Estimated Mandays ______

Completed By ______ Date ______

FINAL REVIEW, JUSTIFICATION AND SCHEDULING PHASE

6. Staff Tech. Sponsor ______ Date Rec'd. ______

Tech. Sponsor's ☐ Approval ☐ Disapproval ______

Originator's ☐ Approval ☐ Disapproval ______

7. Staff Mgmt's. ☐ Approval ☐ Disapproval ______

By ______ Date ______

IMPLEMENTATION PHASE

8. Received By Manager, Production Technical Services On ______ By ______

Completed By ______ Date ______

NOTIFICATION OF COMPLETION

9. Staff Tech. Sponsor ______ Date ______

Originator ______ Date ______

Form 34977 (2-83)

PRODUCTION SUPPORT DEPARTMENT
PROCESS COMPUTER PROGRAM REQUEST

DATE ______________ (ATTACH TO A ROUTING AND APPROVAL CHECKLIST) P ______________

STATION: ______________ COMPUTER(S): ______________

PROGRAM: ______________

REQUESTOR: ______________ PHONE: ______________ DEPT.: ______________

CONTACTS: ______________ PHONE: ______________ DEPT.: ______________

______________ PHONE: ______________ DEPT.: ______________

PRIORITY (CHECK ONE):

REGULATORY/SAFETY_____, RELIABILITY/AVAILABILITY_____, PERFORMANCE/ECONOMIC/SECURITY_____, NEAR & LONG-TERM SUPPORT_____, LONG-TERM SUPPORT_____.

DESIRED COMPLETION DATE: ______________

I. **PURPOSE:** A statement of the need that the program would fill.

____

____

____

II. **JUSTIFICATION:** A concrete statement of benefits of the program. An attempt should be made to relate the value of the program in the form of dollars saved, man-hour savings, improved operation, increased safety, necessity to meet regulatory requirements, etc.

____

____

____

____

____

III. **FUNCTIONAL DEFINITION:** Detailed program requirements as outlined on the reverse side of this document should be attached to this sheet.

____

____

I. PROGRAM DESCRIPTION

A. PURPOSE
Brief statement of the purpose of the program.

B. FUNCTIONS
List of the specific functions performed by the program.

C. DEFINITION OF FUNCTIONS
This section comprises the bulk of the program description and consists of a detailed description of all functions, including details of run options and program operation.

D. INPUT DATA
For each input:
1. Method (cards, paper tape, floppy disc, typer, panel keyboard, analog or digital inputs, performance values, pseudo points, data link, tables, etc.)
2. Format

E. OUTPUT DATA
For each output:
1. Device (video, typer, paper tape punch, floppy disc, line printer, contact output, tables, etc.)
2. Format

F. CALCULATIONS
For each calculation:
1. Description of input variables (includes a description of inputs, range and units, validity checks and the action resulting from a detected error, as well as a description of any internal constants).
2. Equations
3. Accuracy requirements
4. Frequency performed
5. Results (includes a description of calculated values, range and units, and whether the value is an intermediate computational result or an end result).

Calculations should be described in logical sequence to show the program's mathematical analyses from input values to calculated output values.

G. VERIFICATION PROCEDURE
Description of the procedure used to prove that the program is properly performing each of its functions for all possible cases. The description includes:
1. Functional checks
   For each function includes a list of all cases that must be checked.
2. Checkout procedure
   Description of the procedure to be used to check each case, including set-up (including special hardware requirements), test case parameters, and expected results.

H. MISCELLANEOUS INFORMATION
Includes any additional information not included above, e.g., assumptions, tables, diagrams.

II. FUNCTIONAL FLOWCHART

A "do-it-block" diagram showing program functional organization and step-by-step treatment of all functions.

Form 34515 (1-81)

PROCESS COMPUTER UNIT
PROGRAM CHECKER QUESTION/PROBLEM LIST

Computer System ______________________ Page ______ of ______

Program Name ______________________ Mnem. ______________ No. ______________

Checker's Name ______________________ Resolver's Name ______________________

Date checking started ______________________ Date checking completed ______________________

Date copied ______________ Date copy returned to checker ______________________

*Resolver and then checker are to place date and initials under ITEM No. when resolved and rechecked.

*Item No.	Date	Question Or Problem – Leave Room For Response

Form 34515 (1-81)

PROCESS COMPUTER UNIT
PROGRAM CHECKING

The following guidelines are to be followed by an individual designated as a checker:

1. He is to become familiar with the programs to be checked in detail to the point that the checker knows basically the same information as the original programmer.

2. The checker should read the documentation and note errors and follow the flowchart to gain initial insight. If the checker is unfamiliar with the computer system, he should review the required areas as well.

3. The checker should operate the program if the computer system is available to familiarize himself with the program's operation and interface.

4. The checker should check every instruction and flow path for possible errors or interaction plus erroneous comments.

5. The checker is to list all questions and problems he has concerning the program or its interaction with other areas in the computer.

6. The checker is to verify the flowchart for accuracy and denote any problems. Whenever the flowchart is corrected, the checker should initial and date the title block "CHKD."

7. The checker is to verify the documentation and file folder and denote any inconsistencies or errors.

8. The checker works with his supervisor to ensure that the original programmer or a designated person other than the checker answers all questions and resolves all problems on the list. The original problem list is to be kept in a central file, but a copy is to be given to the person who is responsible for corrections.

9. The checker is to verify that any changes made to the program corrects the problem and does not cause another problem. As the problem or question is resolved, the programmer should initial the item number and include the date on his copy and return to his supervisor after completion of the list.

10. After the checker verifies the resolutions, he is to initial also the copied problem/question list indicating that the problem or question is resolved and designate the date. This copy is to be placed in the program's file folder. A copy of this copy will replace the original in the central file.

11. The checker, at times, will be responsible for making corrections, but his corrections are to be checked by another individual.

12. Any situations needing arbitration will be handled by the Design Review Committee for that system.

"Program Checker Question Problem List" forms are to be used in the above process. This form is also used for Design Review Committee comments on the final program review.

VERIFICATION OF SPDS SOFTWARE

VALIDATION OF SPDS SOFTWARE

SIGNAL VALIDATION

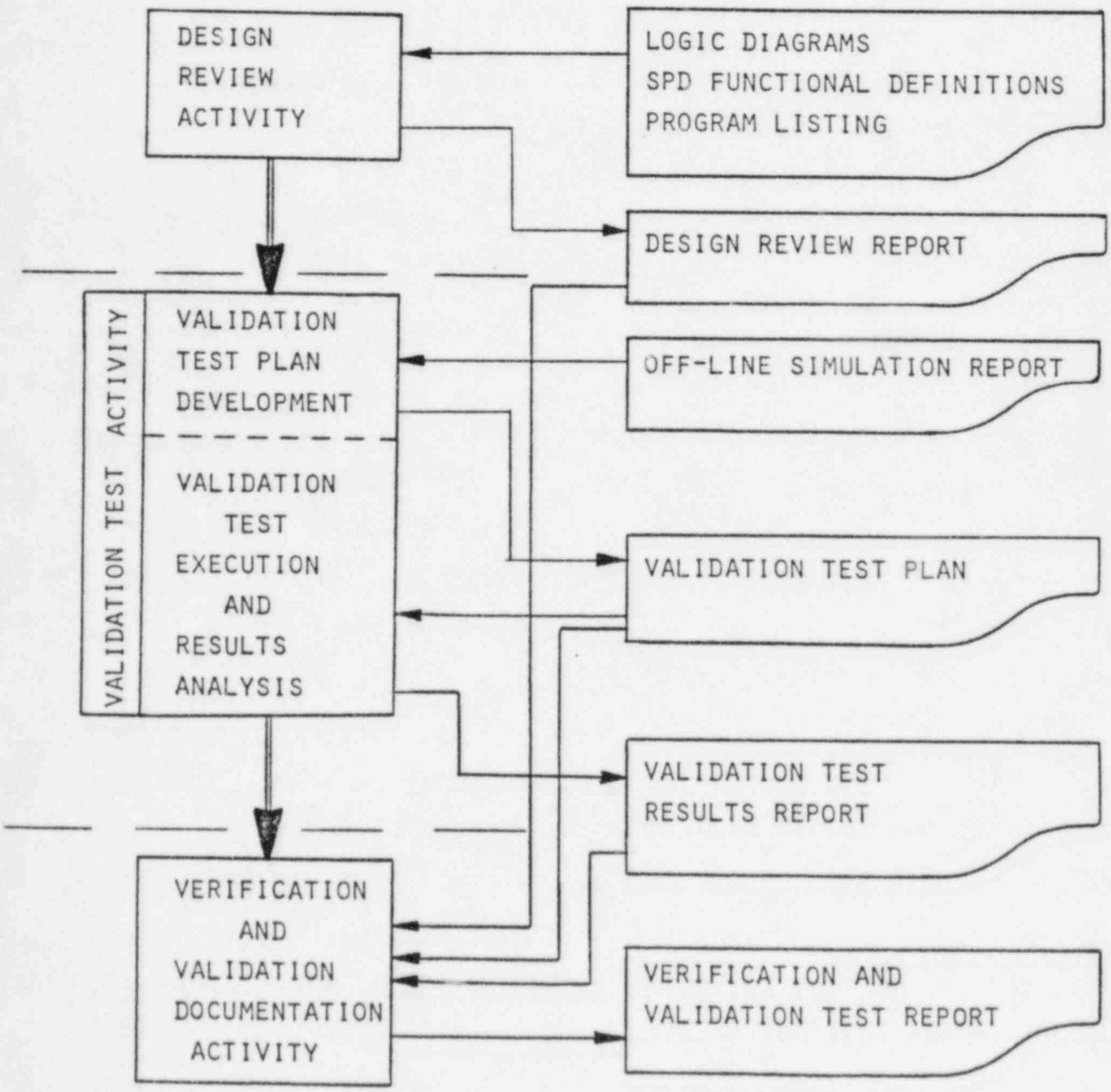
LARRY R. FRICK

PROCESS COMPUTER & SECURITY SYSTEMS GROUP

DESIGN ENGINEERING DEPARTMENT

SAFETY PARAMETER DISPLAY SYSTEM

SOFTWARE VERIFICATION AND VALIDATION PROGRAM



VERIFICATION OF SPDS SOFTWARE

DESIGN REVIEW ACTIVITY

PURPOSE:

VERIFY SPDS LOGIC IS CORRECTLY IMPLEMENTED IN OAC SOFTWARE

DETAILED REVIEW OF DESIGN DOCUMENTS

GENERATED DURING THE DESIGN PROCESS:

- o Logic Diagrams
- o Safety Parameter Display (SPD) Program Functional Definition
- o Program Listing (Assembly Language Code)

LOGICAL INTEGRITY

ABILITY TO SATISFY PERFORMANCE REQUIREMENTS

SOFTWARE ARCHITECTURE

DATA MANIPULATIONS

TESTABILITY

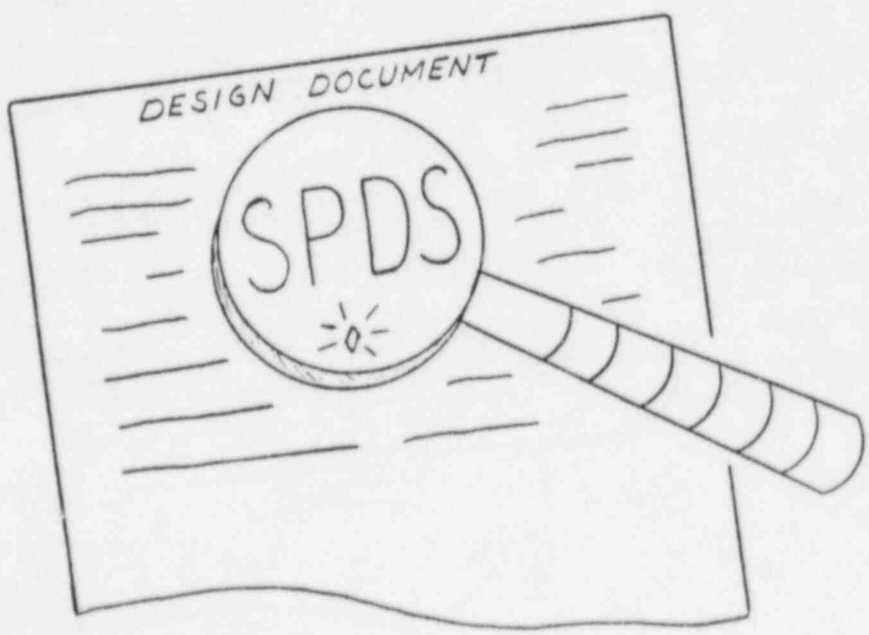
TIMING REQUIREMENTS

? ? ? ? ?

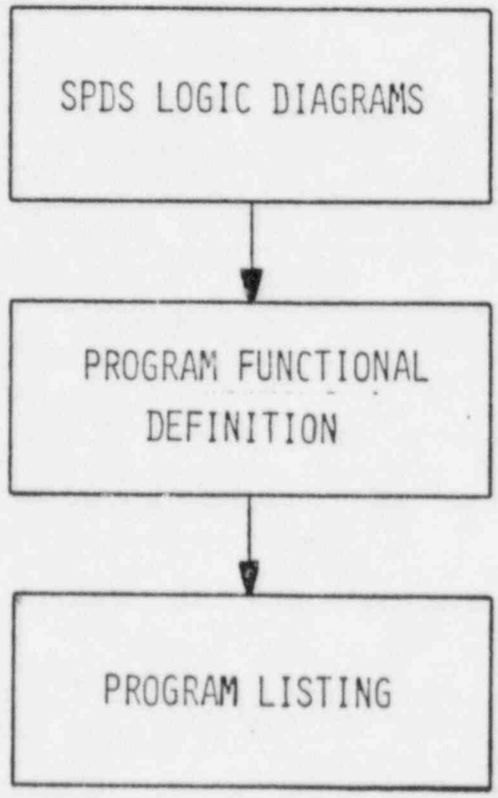
DIRECT EVALUATION AND

ANALYSIS OF DESIGN

DOCUMENTS (DESK CHECK)



VERIFY:



DATA BASE POINT ID'S

DIGITAL POINT STATUS

ANALOG POINT LIMITS

LOGIC SEGMENT GENERATION

SPECIAL CALCULATIONS

OUTPUTS - VIDEO FORMAT, ALARMS, PRINTER OUTPUT

ERROR DETECTION

- FORMALLY DOCUMENT DEFICIENCIES

- RESOLUTION OF DEFICIENCIES

- FOLLOW UP EVALUATION & ANALYSIS

- RESULTS OF DESIGN REVIEW ACTIVITIES

  DOCUMENTED IN DESIGN REVIEW REPORT

DISCREPANCY EXAMPLES RESULTING FROM DESIGN REVIEW ACTIVITY

CRITICAL SAFETY FUNCTION: HEAT SINK

FINDINGS: ANALOG INPUT A1107 (S/G A PRESS) WAS PROGRAMMED TO HAVE A LIMIT OF 1230 PSIG.

LIMIT AS ESTABLISHED BY FUNCTIONAL DESCRIPTION WAS 1225 PSIG.

CRITICAL SAFETY FUNCTION: CORE COOLING

FINDINGS: ANALOG INPUT A1306 (RVLIS NR TRA) WAS PROGRAMMED TO RECOGNIZE VALUE "LESS THAN" 45 PERCENT.

ANALOG POINT AS ESTABLISHED BY FUNCTIONAL DESCRIPTION SHOULD RECOGNIZE VALUE "GREATER THAN" 45 PERCENT.

VALIDATION TESTING OF SPDS SOFTWARE

VALIDATION TEST ACTIVITY

PURPOSE:

VALIDATE COMPLETED SPDS SOFTWARE:

- O MEETS ALL SYSTEM REQUIREMENTS
- O FUNCTIONS AS DESIGNED

SUBACTIVITIES

- VALIDATION TEST PLAN DEVELOPMENT

  Provides An Organized Test Procedure For

  Validation Testing

- VALIDATION TEST EXECUTION AND

  RESULTS ANALYSIS

  Demonstrates That The Implemented SPDS Functions

  As Designed

VALIDATION TEST PLAN DEVELOPMENT INCLUDES:

- Test Requirements
- Test Philosophy
- Test Environment
- Test Specifications
- Test Procedures
- Test Evaluation Approach
- Test Case Development

TEST CASE DEVELOPMENT

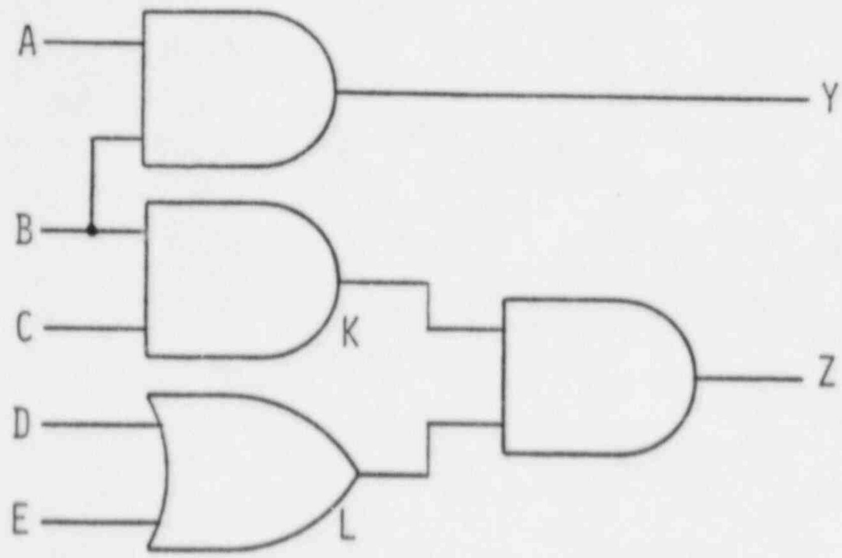
- ESTABLISH TEST CASE SELECTION CRITERIA
  - Test Each Input At Least Once
  - Test Each Gate At Least Once
  - Test Each Output Color Block For Each Critical Safety Function at Least Once
- USE OF OFF-LINE SIMULATION PROGRAM

OFF-LINE SIMULATION PROGRAM

- Simulation Report Is A Truth Table For Each Critical Safety Function Showing Colored Output Conditions For All Possible Combinations Of Parameter Inputs

- Simulation Report Developed By Transforming SPDS Logic Diagrams Into Boolean Expressions and Generating CSF Output Functions with Parameter Inputs

• SPDS LOGIC



• OFF-LINE SIMULATION REPORT

CASE	INPUTS	INTERMEDIATE	OUTPUT
#	A B C D E	K L	Y Z
0	0 0 0 0 0	0 0	0 0
1	0 0 0 0 1	0 1	0 0
2	0 0 0 1 0	0 1	0 0
⌇	⌇	⌇	⌇
31	1 1 1 1 1	1 1	1 1

• APPLY TEST CASE CRITERIA

• TEST CASE SELECTION

VALIDATION TEST EXECUTION AND

RESULTS ANALYSIS

INCLUDES:

- o VALIDATION TEST EXECUTION
- o TEST RESULTS DOCUMENTATION
- o TEST RESULTS ANALYSIS
- o DOCUMENTATION OF DEFICIENCIES
- o FOLLOWUP TESTING OF RESOLVED DEFICIENCIES
- o VALIDATION TEST RESULTS REPORT

DISCREPANCY EXAMPLES RESULTING FROM VALIDATION TEST ACTIVITY

CRITICAL SAFETY FUNCTION: CORE COOLING

FINDINGS: DURING STATIC TEST, DIGITAL INPUT D4148 (SATURATION MARGIN IN ALARM), YIELDED A "YELLOW" OUTPUT CONDITION INSTEAD OF THE EXPECTED "GREEN" OUTPUT CONDITION.

CAUSE: INPUT D4148 WAS INVERTED.

CRITICAL SAFETY FUNCTION: REACTOR COOLANT INVENTORY

FINDINGS: DURING STATIC TEST ALIGNMENT, ANALOG INPUT A1300 (RVLIS UP "A" @ 100%) YIELDED A "YELLOW" OUTPUT CONDITION INSTEAD OF THE EXPECTED "GREEN" OUTPUT CONDITION.

CAUSE: THE CSF OUTPUT WAS USING THE WRONG INPUT SEGMENT FOR COLOR GENERATION.

VERIFICATION AND VALIDATION

DOCUMENTATION ACTIVITY

PURPOSE:

- o Summarize V&V Activities Performed on The SPDS

ACTIVITIES:

- o Prepare A Summary Document Describing:

  SCOPE

  ACTIVITIES PERFORMED

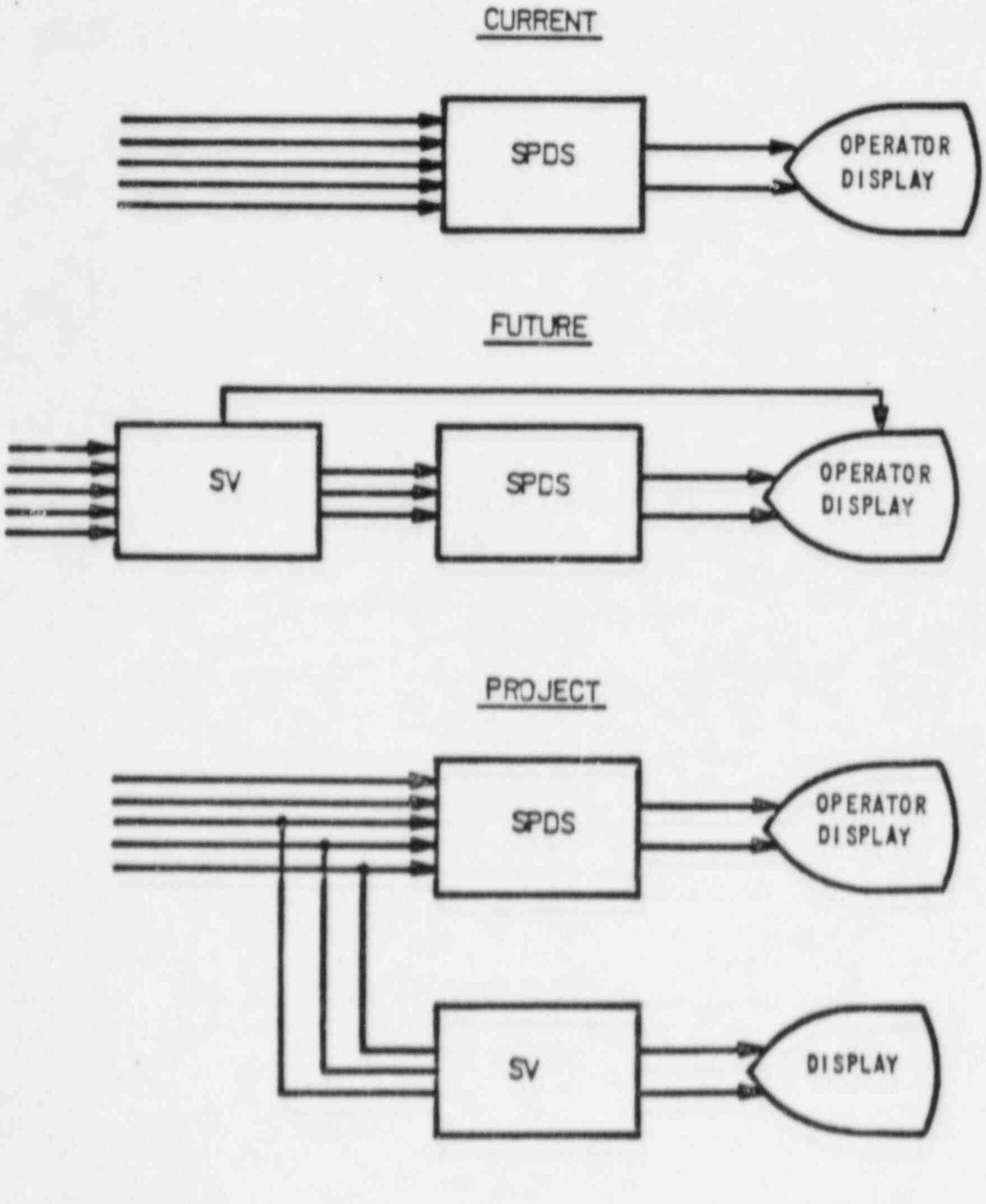
  FINDINGS

  CORRECTIVE ACTIONS

Signal Validation

- What is signal validation ?
- How are signals validated ?

Relationship Between Signal Validation and SPDS



EPRI Project 2292-1
Validation and Integration of PWR Signals

- Project objective
- Project scope and organization
- Design methodology
  - — Physical redundancy
  - — Analytic redundancy

PROJECT OBJECTIVES:

- Develop system to validate plant signals
- Build on advanced techniques
- Develop generic methodology and software modules
- Demonstrate the concepts
- Provide EPRI members with commercially usable system design

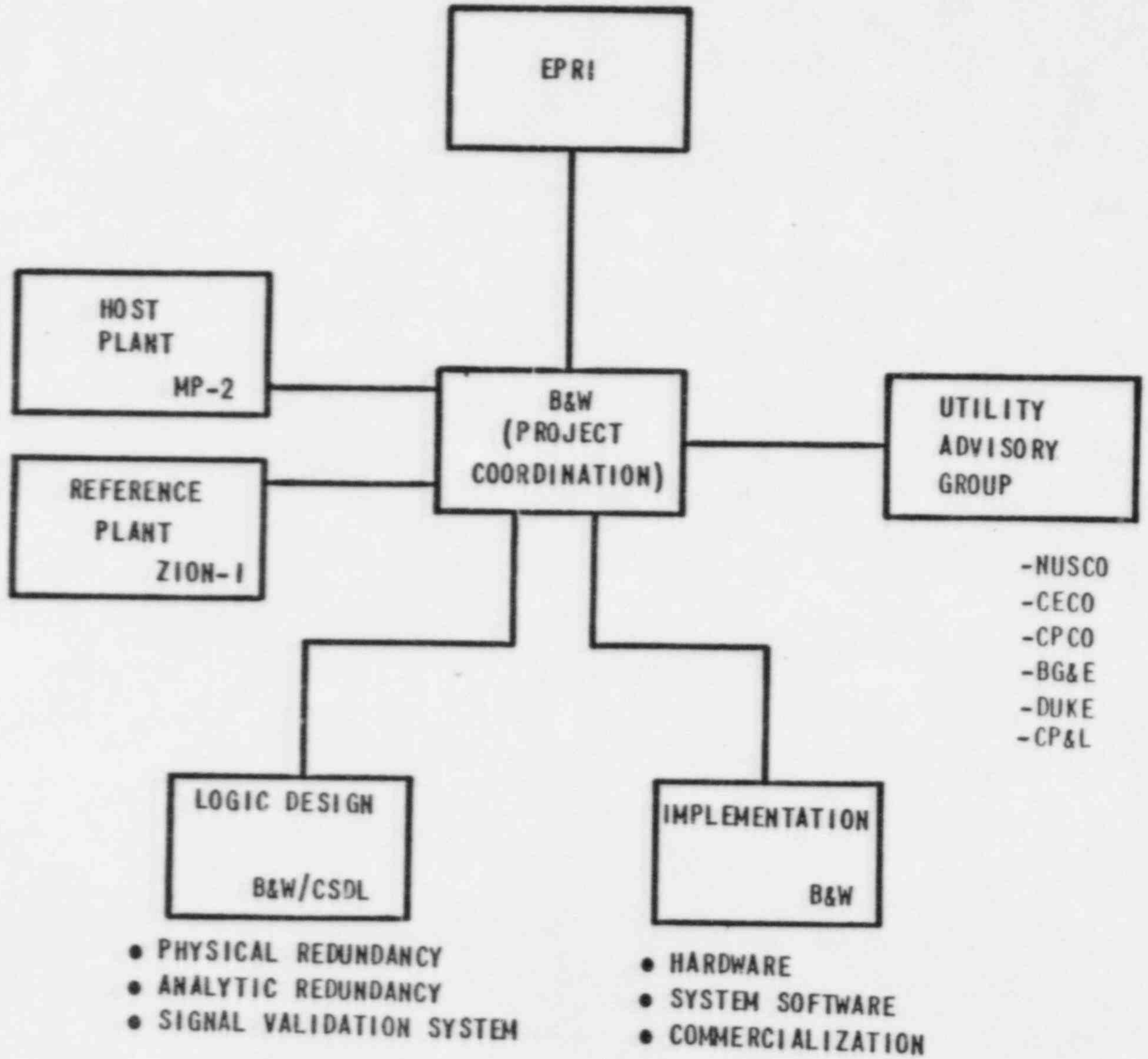
PROJECT SCOPE:

- PWR signals used to monitor Critical Safety Functions

Critical Safety Functions

Variables	Core Cooling	RCS Integrity	RCS Inventory	Containment Integrity	Reactivity Control	RCS Heat Removal
P_{RC}	4	6			1	4
T_{HOT}	5				1	5
T_{COLD}	5	1			5	5
N_{CORE}	2				7	
F_{RHR}						4
P_{SG}			1			5
P_{CNT}		4	2	7		2
L_{SUMP}		4	4	3		1
F_{RC}	5	5				4
F_{FW}						4
F_{LTD}			4			1
L_{SG}			1			6
L_{PZR}			6			
T_{CORE}	7	1				
Z_{CR}				1	6	
T_{SUB}	6					
F_{SI}	1		1			

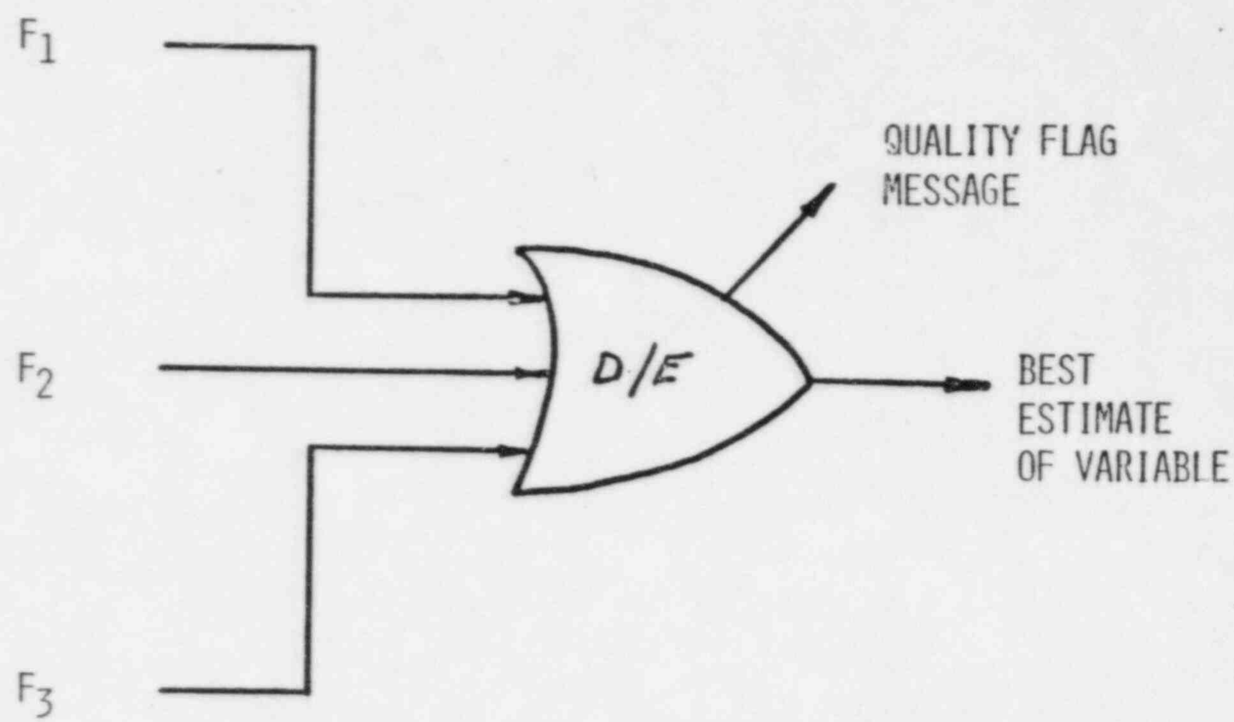
Project Organization



Signal Validation Techniques

- Physical redundancy
- Analytical redundancy

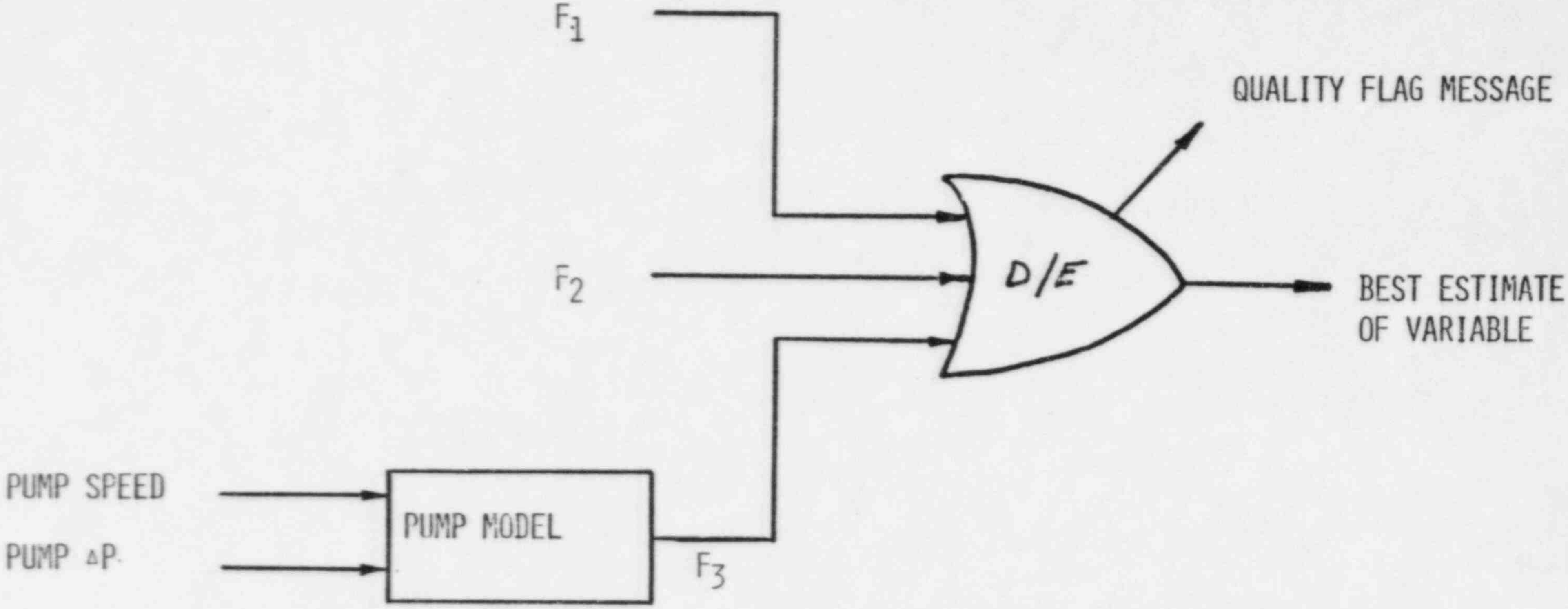
PHYSICAL REDUNDANCY



- DIRECT SENSOR INPUTS
- SIMPLEST D/E LOGIC
- CONCEPTUALLY SIMPLE
- MINIMUM ENGINEERING

- REQUIRES MULTIPLE COLOCATED SENSORS
- DOES NOT DETECT COMMONMODE FAILURES

ANALYTIC REDUNDANCY



- o SUPPLEMENTS SENSOR INPUTS
- o DETECTS SEVERAL COMMONMODE FAILURES
- o DEPENDENT ON MODEL ACCURACY
- o DOES NOT DETECT ALL COMMONMODE FAILURES
- o REQUIRES MEASURED INPUTS FOR MODELS

1984 SV PROJECT ACCOMPLISHMENTS

- DATA ACQUISITION SYSTEM OPERATIONAL
- SUCCESSFUL DEMONSTRATIONS OF SV TECHNIQUES
- FINAL DRAFT OF INTERIM REPORT
- FOUR PHYSICAL REDUNDANCY MODULES FOR MILLSTONE-2 COMPLETE
- TACIT APPROVAL OF THESE SIGNAL VALIDATION TECHNIQUES BY NRC

SV PROJECT STATUS

- DATA ACQUISITION SYSTEM OPERATIONAL
- I/O LIST COMPLETED
- READ MODEM MODULE COMPLETED
- TEST BED EXECUTIVE OPERATIONAL
- MAN/MACHINE INTERFACE OPERATIONAL
- SV EXECUTIVE OPERATIONAL
- FOUR SV MODULES COMPLETE
- FINAL DRAFT OF INTERIM REPORT COMPLETE
- INITIAL DRAFT OF DETAILED DESIGN DOCUMENT COMPLETE

DYNAMIC TESTING

NSAC/38 "ACCIDENT SEQUENCES FOR DESIGN, VALIDATION, & TRAINING, SAFETY PARAMETER DISPLAY SYSTEMS"

- o ABANDONED DUE TO LARGE SOFTWARE DESIGN EFFORT REQUIRED TO MATCH NSAC/38 DATABASE WITH OAC COMPUTER SPDS PROGRAMS (ASSEMBLY LANGUAGE)

TRANSPARENT IMPLEMENTATION OF SPDS AT McGUIRE

- o DISPLAYS TURNED OFF
- o ALARM TABLE DEVELOPED
- o EFFECTIVE IN SPOTTING PROBLEMS AND VERIFYING PROPER OPERATION

R. L. Brown
Production Support Dept.

INSTALLATION OF SPDS
ON THE
OPERATOR TRAINING SIMULATOR

EARLY IMPLEMENTATION ON SIMULATOR EVALUATED TO ALLOW OPERATOR FEEDBACK; DYNAMIC HUMAN FACTORS REVIEW; AND TEST BED OF PROCEDURE/ CONTROL ROOM/OPERATOR......

NOT FEASIBLE:

- BACKLOG OF REVISIONS IN PART DUE TO CONTROL ROOM REVIEW ACTIVITIES......
- LIMITED PLANT MODEL DID NOT CONTAIN SUFFICIENT INPUTS NEEDED BY THE SPDS TO ALLOW SATISFACTORY EVALUATION.

SPDS HAS NOW BEEN INSTALLED (MAY 2, 1985) AND IS UNDERGOING DEBUGGING AND TESTING. IT IS A LIMITED SIMULATION BUT WILL PROVIDE SUFFICIENT FIDELITY FOR OPERATOR TRAINING PURPOSES.

A TRAINING SIMULATOR HAS BEEN PURCHASED FOR THE CATAWBA UNITS AND IS EXPECTED TO BE INSTALLED BY 1988.

SAFETY PARAMETER DISPLAY SYSTEMS

DATES PLACED IN SERVICE

CATAWBA UNIT 1	OPERATIONAL	JUNE 5, 1984
McGUIRE UNIT 1	OPERATIONAL	AUGUST 2, 1984
McGUIRE UNIT 2	OPERATIONAL	AUGUST 30, 1984
NEW EMERGENCY OPERATING PROCEDURES IMPLEMENTED (McGUIRE)		NOVEMBER 30, 1984
CATAWBA UNIT 2 FUEL LOAD (SPDS ALREADY INSTALLED)		JANUARY, 1986

SPDS DOCUMENTATION

CURRENT WORKING FILES:

o	SPDS PROJECT LEADER:	PROJECT FILES
o	DESIGN ENGINEERING:	VALIDATION AND VERIFICATION TASK ANALYSIS HUMAN FACTORS REVIEW
o	NUCLEAR PRODUCTION:	LOGIC DESIGN LOGIC REVIEW ON-GOING SUPPORT
o	PRODUCTION SUPPORT	SOFTWARE FILES

PROPOSED SCHEDULE FOR FINALIZATION OF DOCUMENTATION PACKAGES:

o	INDEX OF ALL FILES	JULY 1, 1985
o	REVIEW AND CLOSURE OF OPEN ITEMS:	OCTOBER 1, 1985
o	DOCUMENTATION SUMMARY PACKAGES:	JANUARY 1, 1986

SPDS ALARM SUMMARY
03/13/85

09:08:40P D4354 SPDS CONT INTEGRITY IN ORANGE STATUS

1 8 10 13 20 24 25 26 29 30 32 34 35 41

09:08:55P D4356 SPDS CONT INTEGRITY IN GREEN STATUS

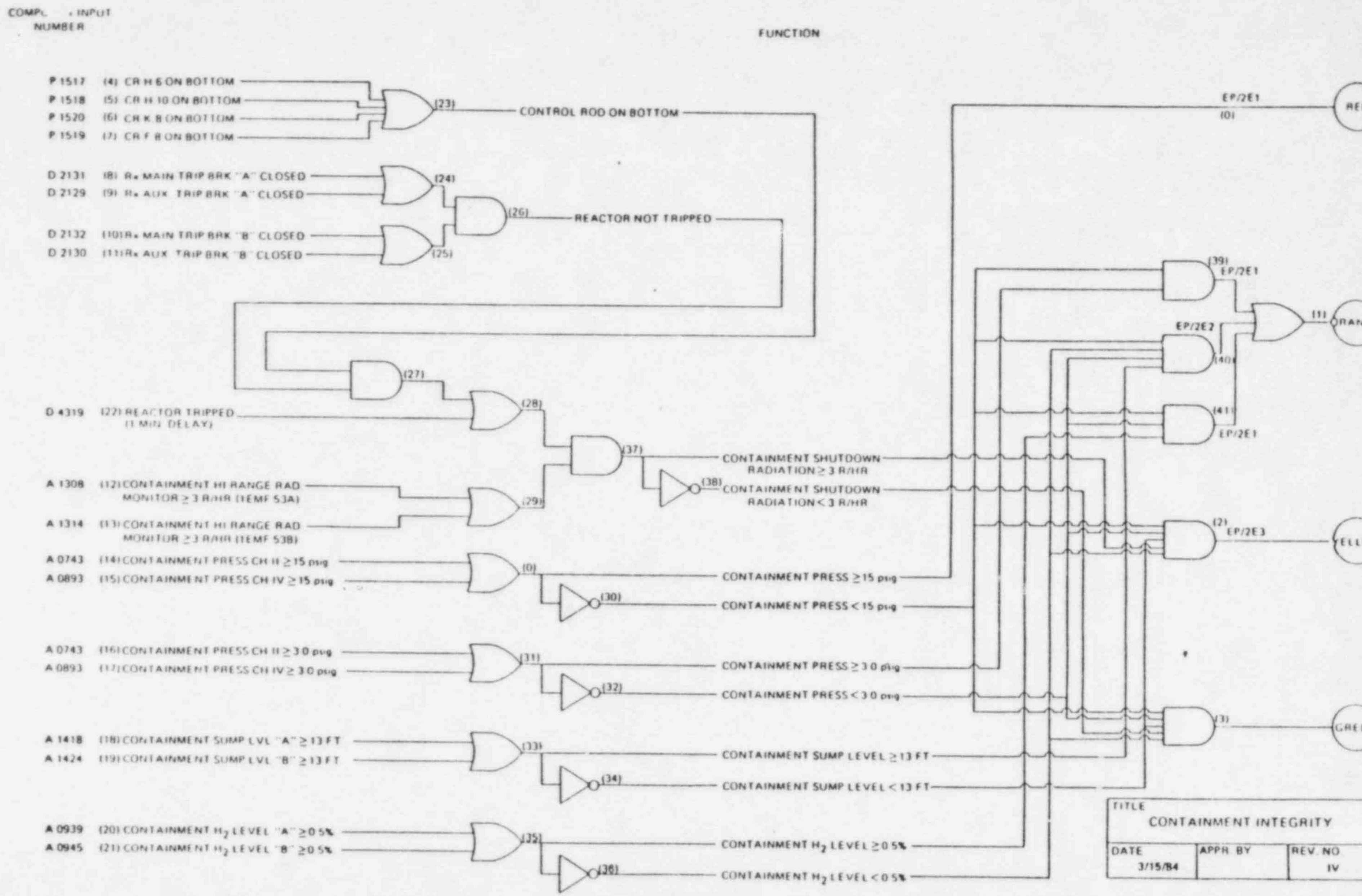
3 8 10 13 24 25 26 29 30 32 34 36

1, 41, 35, 20 CLEARED

3, 36 NORMAL

CATAWBA
SAFETY PARAMETER DISPLAY SYSTEM (SPDS)
LOGIC DIAGRAM FOR "CONTAINMENT INTEGRITY"

New 6/29/84



TITLE CONTAINMENT INTEGRITY		
DATE 3/15/84	APPR. BY	REV. NO. IV

ATTACHMENT 6

FOR INFORMATION ONLY

SPDS ALARM SUMMARY
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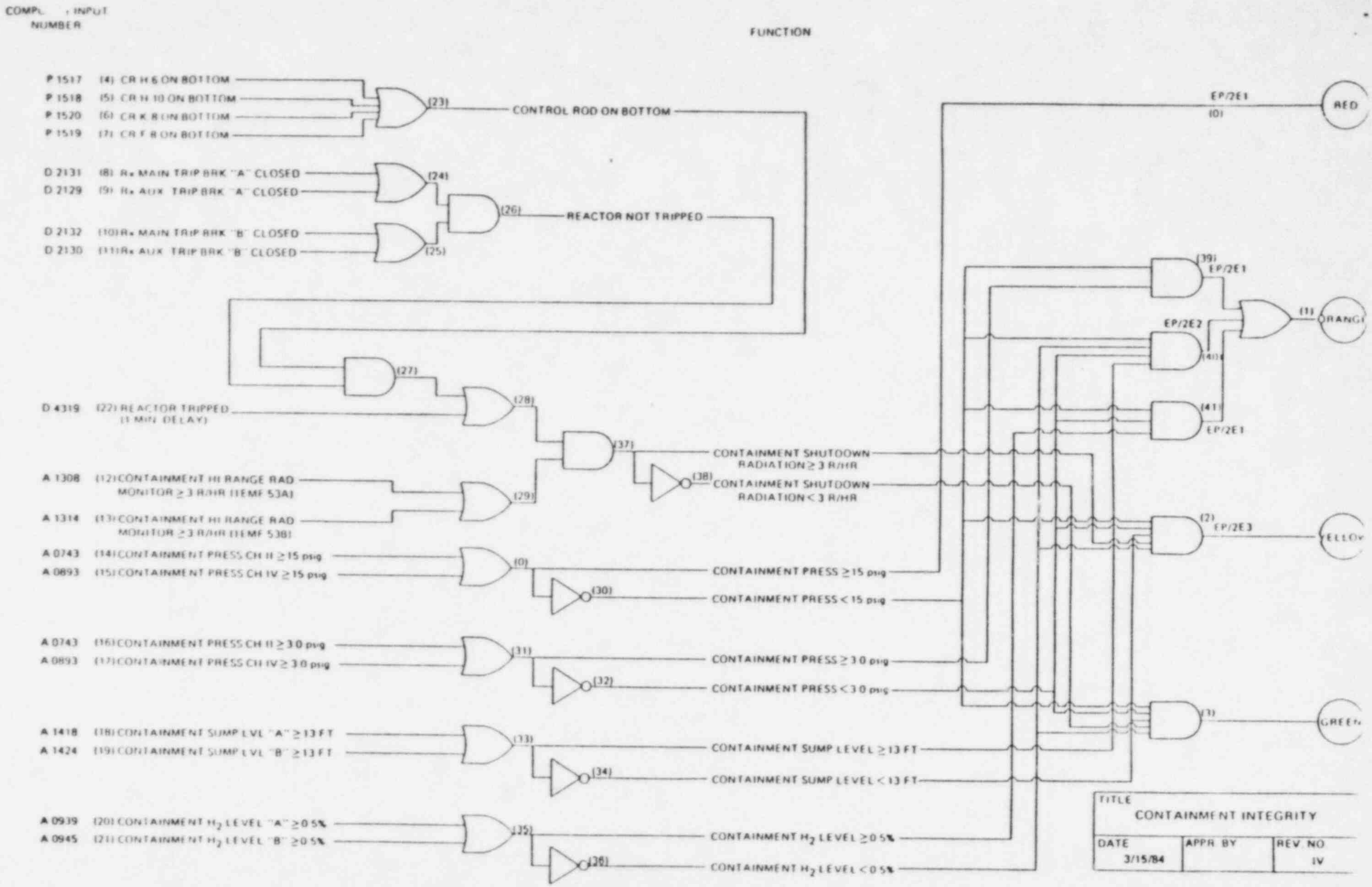


ATTACHMENT 6

FOR INFORMATION ONLY

ROBERT G. MORGAN

NUCLEAR OPERATIONS

NUCLEAR PRODUCTION DEPARTMENT

TECH SPONSOR FOR SPDS

SPDS MAINTENANCE & REVISION PROGRAM

- Ensures SPDS performs properly during plant dynamic conditions

- Facilitates prompt detection of equipment or software problems

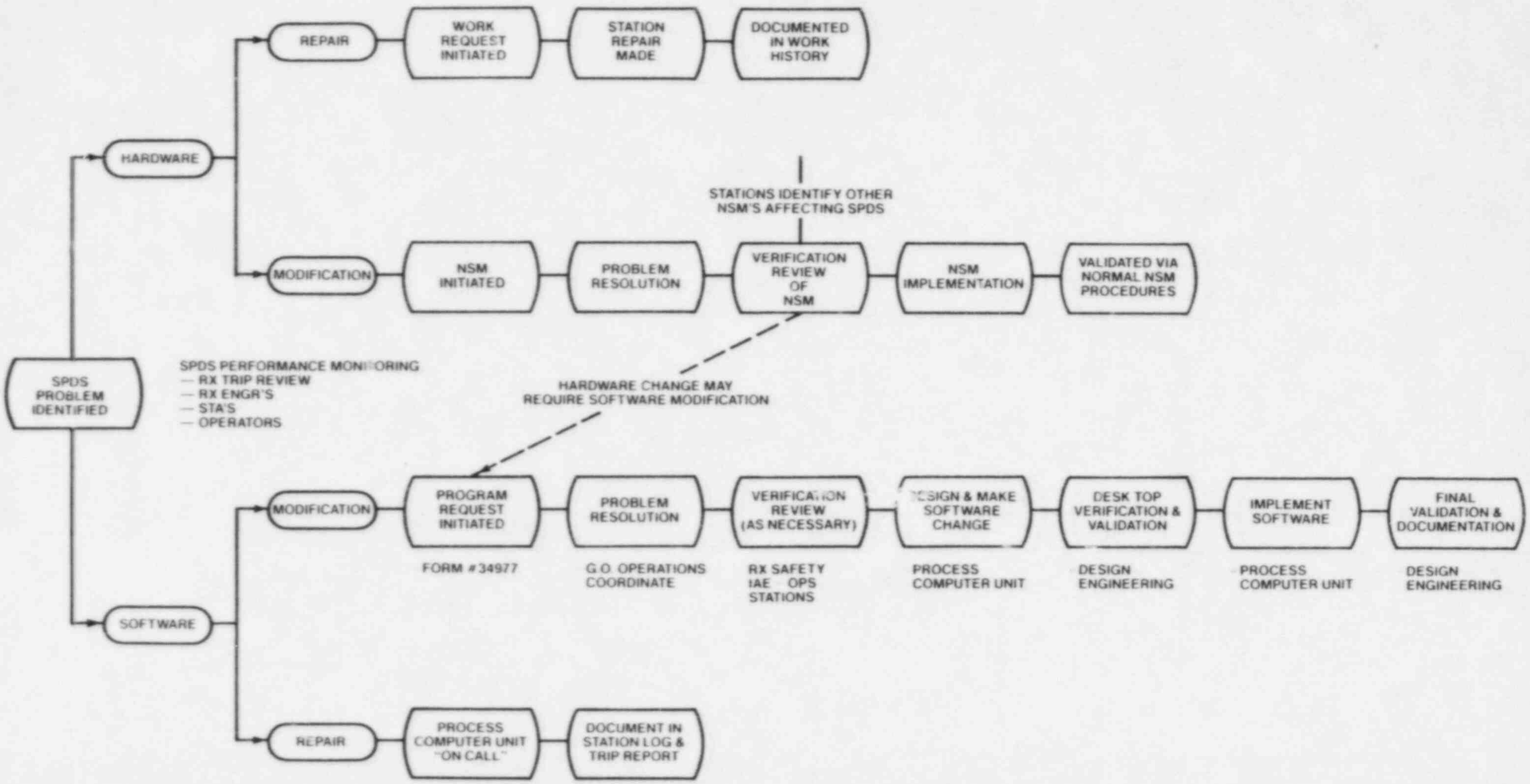
- Ensures future modifications to SPDS receive the same degree of V&V as initial development

PERFORMANCE MONITORING

- Shift Technical Advisors
- Control Room Operators
- Reactor Post Trip Review
- Reactor Engineer/Station OAC Software Coordinator

SPDS MAINTENANCE

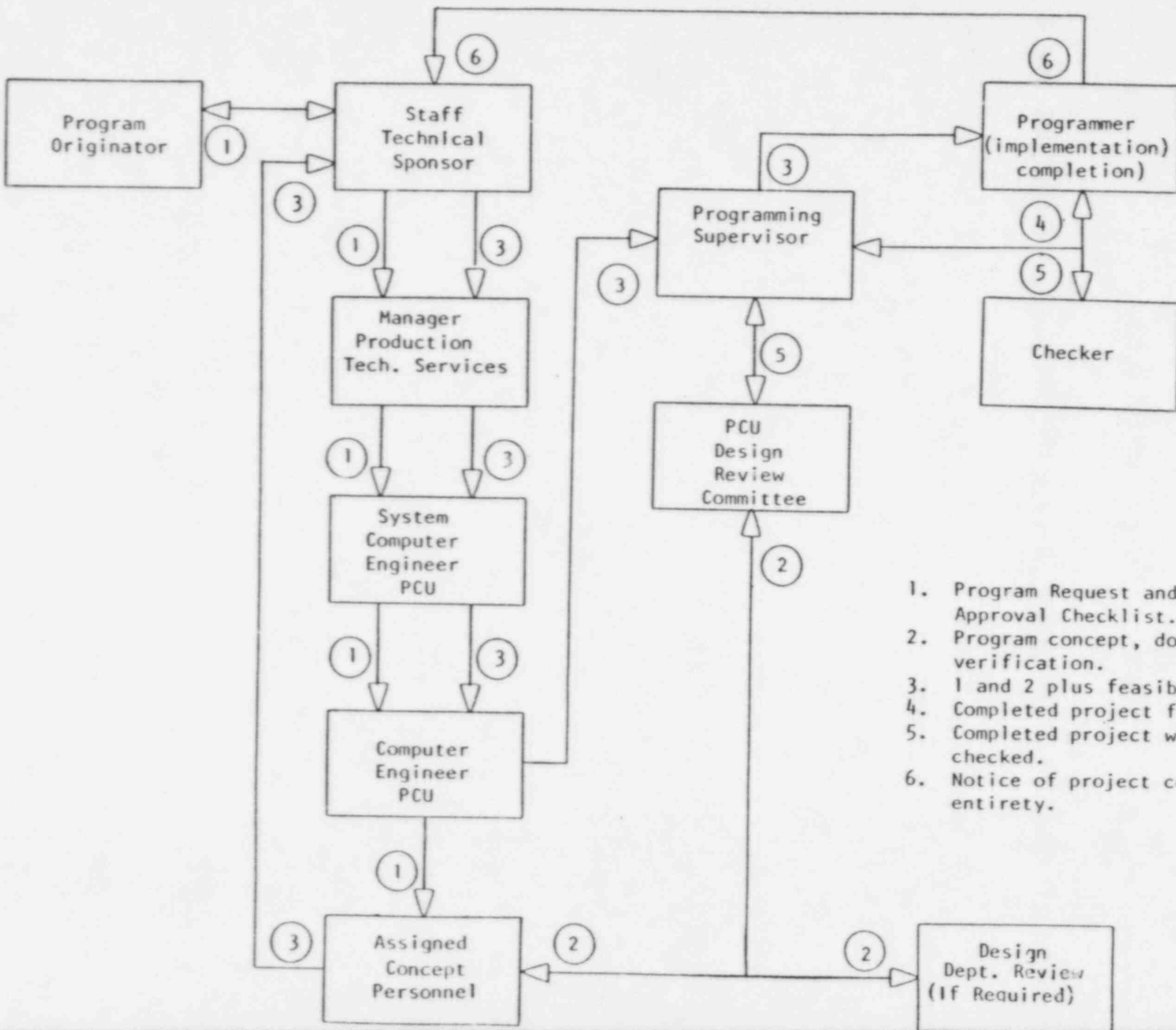
RGM/1-3-85



VERIFICATION: A review to ensure that the identified problem is being solved properly and then the review of the resultant design to ensure that the modified SPDS continues to meet functional requirements.

VALIDATION: A test and evaluation of the integrated hardware and software system to determine compliance with the functional, performance, and interface requirements.

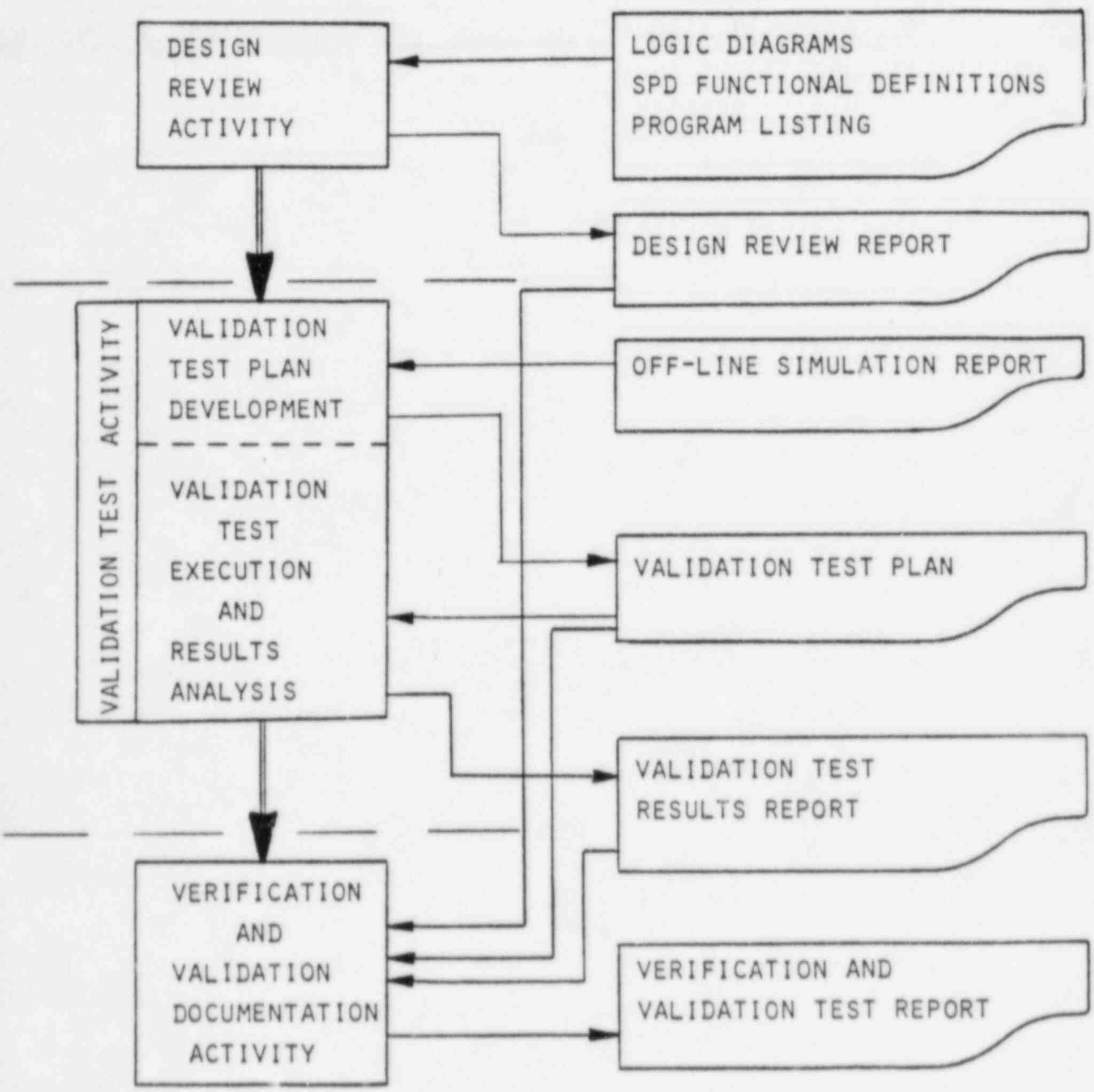
The Path of a Program Request
For the Process Computer Unit



1. Program Request and Routing and Approval Checklist.
2. Program concept, documentation, verification.
3. 1 and 2 plus feasibility study.
4. Completed project from programmer.
5. Completed project which has been checked.
6. Notice of project completion in entirety.

SAFETY PARAMETER DISPLAY SYSTEM

SOFTWARE VERIFICATION AND VALIDATION PROGRAM



SPDS SOFTWARE MODIFICATIONS

CATAWBA NUCLEAR STATION

Modify SPDS Supporting Displays based upon recommendations of Human Factors Survey Program Request P840104-0	Completed October 1984
Modifications to SPDS Software for Rx Safety's recommendations (setpoint revisions, etc.) Program Request P840125-0	Completed January 1985
Addition of Software to Support STA CRT in Control Room Program Request P840038-0	Completed January 1985
Modification of SPDS Alarm Summary to enhance performance monitoring Program Request P840129-0	Completed February 1985
Develop program to validate RVLIS signals to prevent spurious alarms (to be consistent with MNS) Program Request P850011-0	Completed May 1985

OCONEE NUCLEAR STATION
SPDS IMPLEMENTATION

OPERATOR AID COMPUTER UPGRADES:

- GE 4020 OAC's INSTALLED DURING LATE 1960 AND EARLY 1970
- ORIGINAL OAC'S HAD BLACK AND WHITE CRT's
- ALPHA/NUMERIC CHARACTER GENERATOR
- INPUT/OUTPUT SPARE CAPACITY USED UP
- LIMITED PROCESSOR CAPABILITY
- BY TMI TIMEFRAME, NEED TO UPGRADE OAC's HAD BEEN IDENTIFIED
- DELAYED DUE TO HIGH ACTIVITY ASSOCIATED WITH TMI ISSUES:
  - EMERGENCY PROCEDURES
  - CONTROL ROOM DESIGN REVIEW
  - EMERGENCY RESPONSE FACILITIES
  - PLANT MODIFICATIONS

OPERATOR AID COMPUTER UPGRADE PROGRAM:

- o REQUIREMENTS SETTLED
- o THREE PHASE UPGRADE PROGRAM IDENTIFIED

  PHASE I: ADD NEW CPU, COLOR GRAPHIC DOT ADDRESSABLE CRT's, MULTIPLEXED INPUT I/O NEEDED FOR SPDS AND PRESSURE TEMPERATURE DISPLAY IMPLEMENTATION

  PHASE II: DEVELOP INPUT/OUTPUT MIGRATION PLAN, CONVERT 4020 SOFTWARE TO NEW HONEYWELL 45000

  PHASE III: CHANGE OVER INPUT/OUTPUT SYSTEM AND REMOVE GE 4020 COMPUTER SYSTEM

OCONEE NUCLEAR STATION

OPERATOR AID COMPUTER SYSTEM

AND

SPDS IMPLEMENTATION MILESTONES

WORK ORDER INITIATED	MAY, 1983
DETAILED DESIGN OF SPDS ITITIATED	MAY, 1983
WORK ORDER APPROVED	JULY, 1983
ORDERS PLACED WITH HONEYWELL	AUGUST, 1983
SPDS LOGIC DESIGN COMPLETED	DECEMBER, 1983
NEW COMPUTERS RECEIVED	FEBRUARY - JUNE 1984
OAC'S INSTALLED:	
UNIT 3	NOVEMBER, 1984
UNIT 1	JANUARY, 1985
UNIT 2	JULY, 1985

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UNIT 1	JANUARY, 1985
UNIT 2	JULY, 1985

SPDS INPUTS

AND

INPUT ISOLATION

MAY 14, 1985

R.M. MEACHAM

SPDS INPUTS

- APPROXIMATELY 50% DERIVED FROM ANALOG SOURCES AND 50% DERIVED FROM DIGITAL SOURCES

- ALL SPDS INPUTS WHICH ARE DERIVED FROM CLASS 1E SIGNALS ARE FULLY ISOLATED BEFORE BEING CONNECTED TO OPERATOR AID COMPUTER

- MOST SPDS INPUTS CONSIST OF CIRCUIT DESIGNS WHICH WERE REVIEWED AT THE TIME OF CATAWBA FSAR APPROVAL -- THE SMALL NUMBER ADDED SINCE THAT TIME (FOR EXAMPLE, THE HI-RANGE CONTAINMENT RAD. MONITOR) UTILIZE FULLY PROVEN AND NRC-LICENSED OPTICAL ISOLATORS

- LARGE NUMBER OF SPDS INPUTS ARE DERIVED FROM WESTINGHOUSE PROCESS CONTROL CABINETS

WESTINGHOUSE PROCESS CONTROL CABINETS AND ISOLATION

- SPDS INPUTS DERIVED FROM WESTINGHOUSE CABINETS UTILIZE ISOLATION DEVICES WHICH HAVE BEEN ACCEPTED BY THE NRC AND WHICH MEET REGULATORY GUIDE 1.75

- TEST DATA IN WCAP-8892A, "WESTINGHOUSE 7300 SERIES PROCESS CONTROL SYSTEM NOISE TESTS"

- NRC APPROVAL LETTER APRIL 20, 1977 TO WESTINGHOUSE

DUKE ISOLATOR DESIGN

- DUKE ISOLATORS FOR ANALOG AND DIGITAL SIGNALS ARE MANUFACTURED BY E-MAX INSTRUMENTS, INC.

- OPTICAL DESIGN WITH HIGH-DIELECTRIC STRENGTH CASE AND PHYSICALLY - SEPARATED INPUT AND OUTPUT TERMINALS

- REVIEWED IN CATAWBA FSAR, CHAPTER 7, AND APPROVED FOR USE IN NUCLEAR SAFETY RELATED APPLICATIONS

- E-MAX ISOLATORS ARE UTILIZED EXTENSIVELY IN CIRCUITS AT CATAWBA NUCLEAR STATION FOR MEETING SEPARATION CRITERIA -- CLASS 1E TO NON-CLASS 1E AND NON-CLASS 1E TO CLASS 1E

DUKE ISOLATOR TESTING

- E-MAX CONDUCTED EXTENSIVE TESTING OF ALL MODELS USED BY DUKE POWER

- OTHER DUKE TESTING HAS BEEN PROPOSED AND PROCEDURES ARE BEING WRITTEN FOR A TEST WHICH PLACES A FAULT DIRECTLY ACROSS THE NON-1E (SPDS) OUTPUT TERMINALS TO DEMONSTRATE THE LACK OF DAMAGE TO THE CLASS 1E INPUT CIRCUIT

- PRELIMINARY TESTS HAVE BEEN PERFORMED IN WHICH A 30 AMPERE/120 VOLT AC SOURCE WAS APPLIED TO THE NON-1E TERMINALS, WITH THE RESULT THAT ONLY SLIGHT DAMAGE WAS NOTED (BURNED RESISTOR, FOR EXAMPLE) ON THE NON-1E SIDE OF THE ISOLATOR CIRCUIT BOARD. NO DAMAGE TO THE CLASS 1E INPUT AND ONLY A SLIGHT, SHORT DURATION PERTURBATION ($<$ 10 MILLIVOLT) IN THE CLASS 1E SIGNAL.

SPDS DEMONSTRATION

Initial Conditions:

100% Power. All control stations in automatic steady state operation.

Malfunctions:

1. Failure of the bypass valves to open on high ΔP across the polishing demineralizers.

2. Failure of automatic Reactor trip on a reactor protection trip signal.

3. Steamline break inside containment on S/G B.

BRIEF DESCRIPTION

The polishing demineralizers in the condensate system have a high ΔP and the bypass valves around the polishers are not opening. The feedwater pumps trip on low suction pressure which trips the turbine generator. The turbine generator sends a trip signal to the Reactor but the Reactor fails to trip. SPDS display for subcriticality will go Red after 5 seconds. The operator will trip the Reactor and subcriticality will go Green.

The steam line break occurs causing containment press. to increase resulting a safety injection signal at 1.0 psig and SPDS display for containment will go Orange at 3.0 psig.

The steam break cooldown causes Pzr level to decrease less than 17% causing a yellow path on inventory. Pzr level drops out of indicating range and Reactor coolant system eventually becomes saturated. The saturated condition produces a yellow path on core subcooling. With safety injection flow the Pzr level returns, increasing pressure above saturation causing core cooling to return to Green.

EVENT SEQUENCE

T = 00:00	Start Scenario
00:44	Polisher High ΔP Alarm
1:06	Rx Fails to Trip on Turbine Trip (Red)
1:14	Manual Rx Trip (Green)
1:28	Safety Injection on High Cont. Pressure (Steam Break)
1:43	Containment Pressure > 3.0 psig (Orange)
1:50	Pzr Lvl < 32% or 17% (Yellow)
2:42	NCS Saturated (Yellow)
3:35	NCS Subcooled (Green)
4:52	Pzr > 17% but because of ACC still (Yellow)

USE OF SPDS BY OPERATORS

* SPDS BASED ON WOG SIX CRITICAL SAFETY FUNCTIONS OF THE PLANT.

* SPDS STATUS TREES ARE USED TO DIRECT OPERATOR INTO FUNCTIONAL RECOVERY GUIDELINES WHEN NEEDED TO PRESERVE CRITICAL SAFETY FUNCTIONS.

* SPDS IS ALWAYS AVAILABLE FOR OPERATOR USE, BUT MONITORING IS REQUIRED UPON KICK-OUT FROM EP/01 REACTOR TRIP OR SAFETY INJECTION PROCEDURE. (COMMON ENTRY POINT FOR ALL EVENT RELATED EP'S EXCEPT FOR LOSS OF ALL AC POWER)

* SPDS IS MONITORED IN PARALLEL WITH PERFORMANCE OF EVENT RELATED EP'S.

    * RED OR ORANGE TERMINUS TAKES PRIORITY OVER EVENT RELATED EP FOR OPERATOR ACTIONS.

    * YELLOW TERMINUS CAN BE PERFORMED IN PARALLEL WITH EVENT RELATED EP.

    * GREEN TERMINUS INDICATES SAFETY FUNCTION SATISFIED.

    * TERMINUS COLOR DETERMINES PRIORITY FOR ACTION FOR SIMULTANEOUS SPDS ALARMS:

      EX. RED ON ANY FUNCTION TAKES PRIORITY OVER ANY ORANGE.

        * SIMULTANEOUS RED TERMINI - PRIORITY DETERMINED BY SAFETY FUNCTION.

          EX. SUBCRITICALITY IN HIGHEST PRIORITY, INVENTORY IS LOWEST.

CATAWBA NUCLEAR STATION June 30, 1984

EP INDEX

EP/1/A/5000/01	Reactor Trip or Safety Injection
/1A	Reactor Trip Response
/1A1	Natural Circulation Cooldown
/1B	S/I Termination Following Spurious S/I
/1C	High-Energy Line Break Inside Containment
/1C1	S/I Termination Following High-Energy Line Break Inside Containment
/1C2	Post LOCA Cooldown and Depressurization
/1C3	Transfer to Cold Leg Recirculation
/1C4	Transfer to Hot Leg Recirculation
/1C5	Loss of Emergency Coolant Recirculation
/1C6	LOCA Outside Containment
/1D	Steam Line Break Outside Containment
/1D1	S/I Termination Following Steam Line Break
/1E	Steam Generator Tube Rupture
/1E1	Post-SGTR Cooldown and Depressurization
/1E2	SGTR Alternate Cooldown Using Backfilling
/1E3	SGTR With Continuous NC System Leakage - Subcooled Recovery
/1E4	SGTR With Continuous NC System Leakage - Saturated Recovery
/1E5	SGTR Without Pressurizer Pressure Control
/1E6	SGTR Cooldown Using ND
EP/1/A/5000/02	Critical Safety Function Status Trees
/2A	Subcriticality
/2A1	Nuclear Power Generation/ATWS
/2A2	Loss of Core Shutdown
/2B	Core Cooling
/2B1	Inadequate Core Cooling
/2B2	Degraded Core Cooling
/2B3	Saturated Core Cooling Conditions
/2C	Heat Sink
/2C1	Loss of Secondary Heat Sink
/2C2	S/G Overpressure
/2C3	S/G High Level
/2C4	Loss of Normal Steam Release Capabilities
/2C5	S/G Low Level
/2D	Reactor Coolant Integrity
/2D1	Imminent Pressurized Thermal Shock Condition
/2D2	Anticipated Pressurized Thermal Shock Condition
/2D3	High Pressurizer Pressure
/2E	Containment
/2E1	High Containment Pressure
/2E2	High Containment Sump Level
/2E3	High Containment Radiation Level
/2F	Inventory
/2F1	Pressurizer Flooding
/2F2	Low NC System Inventory
/2F3	Void in Reactor Vessel
EP/1/A/5000/03	Loss of All AC Power
/3A	Loss of All AC Power Recovery Without S/I Required
/3B	Loss of All AC Power Recovery With S/I Required

STEAMLINE BREAK INSIDE CONTAINMENT SCENARIO

INITIAL CONDITIONS - REACTOR OPERATING AT 100% POWER
END OF LIFE CONDITIONS

MAIN STEAMLINE LEAVING A STEAM GENERATOR RUPTURES INSIDE CONTAINMENT DUE TO A SEISMIC EVENT.

SYMPTOMS: (PRIOR TO TRIP)

* CONTAINMENT PRESSURE INCREASING
* A S/G PRESSURE AND LEVEL DECREASING
* PRESSURIZER PRESSURE AND LEVEL DECREASING
* FEEDWATER FLOW INCREASE TO A S/G

SAFETY INJECTION OCCURS DUE TO

* LOW STEAMLINE PRESSURE (725 PSIG)
* HIGH CONTAINMENT PRESSURE (1.2 PSIG)
* LOW PRESSURIZER PRESSURE (1845 PSIG)

S/I RESULTS IN CONTAINMENT PHASE A ISOLATION FEEDWATER ISOLATION

MAIN STEAMLINE ISOLATION OCCURS DUE TO LOW STEAMLINE PRESSURE (725 PSIG)

CONTAINMENT ISOLATION PHASE B OCCURS AT 3.0 PSIG CONTAINMENT PRESSURE

OPERATOR ENTERS EMERGENCY PROCEDURES UPON REACTOR TRIP OR SAFETY INJECTION

STEAM LINE BREAK INSIDE CONTAINMENT PROCEDURE FLOW PATH

EVENT RELATED	FUNCTION RELATED
ENTER EP/01 REACTOR TRIP OR SAFETY INJECTION	
PERFORM IMMEDIATE ACTIONS. STEP 3 SUBSEQUENT ACTION. VERIFY S/I VALVE ALIGNMENT (INCLUDES CONTAINMENT ISOLATION VALVES)	
KICKOUT ON CONTAINMENT PRESSURE AND ICE CONDENSER DOORS OPEN	
ENTER EP/1C HIGH ENERGY LINE BREAK INSIDE CONTAINMENT	
AND	
CONCURRENTLY IMPLEMENT EP/02 CRITICAL SAFETY FUNCTION STATUS TREES.	ORANGE CONTAINMENT - REFER TO EP/2E1
ISOLATE FAULTED S/G REFERRING TO EP/1D STEAM LINE BREAK OUTSIDE CONTAINMENT STEPS 1 THROUGH 4	YELLOW ON HEAT SINK - REFER TO EP/2C5
KICKOUT OF EP/1C TO EP/1D1 S/I TERMINATION FOLLOWING STEAM LINE BREAK	YELLOW ON REACTOR COOLANT INTEGRITY - REFER TO EP/2D2

ACTION/EXPECTED RESPONSE	RESPONSE NOT OBTAINED

****
CAUTION
****

IF D/G supplying bus, THEN loads placed on the 4160V Bus should NOT exceed 5750 KW to limit engine crankshaft and piston stresses.

___ 2. Verify E/S Sequencers ACTUATED:

- o Following status lights: LIT (1SI-14)
  - o Manually initiate S/I.
  - o "E/S Load Seq Actuated Train A"
    AND
  - o "E/S Load Seq Actuated Train B".

___ 3. Ensure Proper Safety System S/I Valve Alignment:

****
CAUTION
****

The following monitor light configuration is valid only in Injection mode.

- o Monitor light configuration on 1MC-14:
  - a. Check Group 1:
    - 1) Ss equipment: LIT
      - 1) Manually operate equipment.
    - 2) Containment pressure has remained below: 3.0 PSIG

      AND

      Sp Equipment: DARK.
      - 2) IF Containment pressure has remained below 3.0 PSIG AND Sp equipment LIT, THEN:
        - o Manually operate equipment.

        IF Containment pressure has gone above 3.0 PSIG, THEN:
        - o Ensure Sp equipment: LIT

          AND

          NS Flow: INDICATED.

ACTION/EXPECTED RESPONSE	RESPONSE NOT OBTAINED
	o IF NOT initiated, THEN: a) Manually initiate "PHASE B NS-VX INIT CONT VENT ISOL" o TRN A o TRN B OR b) Manually operate equipment. o Ensure all NC Pumps: STOPPED. o Ensure VX operation after 9 minute time delay: o Following fans: RUNNING o ARF-1A, -1B (Cont Air Return Fan) o HSF-1A, -1B (H2 Skimmer Fan) o Following dampers: OPEN o ARF-D-2 (ARF-1A Ret Fan Damper) o ARF-D-4 (ARF-1B Ret Fan Damper).
b. Check Group 2: DARK	b. Manually operate equipment.
c. Check Group 3: DARK	c. Manually operate equipment.
d. Check Group 4: LIT	d. Manually operate equipment.

ACTION/EXPECTED RESPONSE	RESPONSE NOT OBTAINED
e. Check Group 5:	
1) St equipment: LIT	1) Manually INITIATE "Phase A Cont Vent Isol": o TRN A o TRN B OR Manually operate equipment.
2) Containment pressure has remained below: 3.0 PSIG AND SM pressure: > 725 PSIG AND Sp Equipment: DARK	2) IF Containment pressure has remained below 3.0 PSIG AND SM pressure > 725 PSIG AND Sp LIT, THEN: o Manually operate equipment. IF Containment pressure has gone above 3.0 PSIG, THEN o Ensure Sp equipment: LIT AND SM PORV's: CLOSED IF SM pressure < 725 PSIG THEN ensure SM isolation status lights LIT. o M -2, -3, -10, -11
f. Check Group 6: DARK	f. Manually operate equipment.

ACTION/EXPECTED RESPONSE	RESPONSE NOT OBTAINED
g. IF NC pressure > 1240 PSIG THEN: Check Group 7 DARK	g. IF NC pressure < 1240 PSIG, THEN ensure UHI injecting: o UHI Surge Tank Level A (B): < 14% AND/OR DECREASING rapidly. o UHI Surge Tank Press A (B): < 1240 PSIG AND/OR DECREASING rapidly. Continue with procedure AND WHEN NC System Press < 450 PSIG, ensure UHI Isolation: Group 7 LIT
h. Check Group 8: DARK	h. Manually operate equipment.
____ 4. Verify ECCS Operation AND Flow:	
a. NV S/I Flow: INDICATING FLOW.	a. Ensure correct valve alignment AND NV Pump operation.
b. NI Pump 1A AND 1B Disch Flow: INDICATING FLOW.	b. Check NC pressure: 1) IF > 1520 PSIG, THEN go to Step D.5. 2) IF < 1520 PSIG, THEN: a) Ensure correct valve alignment AND NI Pump operation.
c. ND HX 1A AND 1B Outlet Flow: INDICATING FLOW.	c. Check NC pressure: 1) IF > 195 PSIG, THEN go to Step D.5. 2) IF < 195 PSIG, THEN: a) Ensure correct valve alignment AND ND Pump operation.

ACTION/EXPECTED RESPONSE	RESPONSE NOT OBTAINED
	a) Close its block valve OR b) IF block valve CANNOT be CLOSED, THEN go to EP/1/A/5000/1C, HIGH ENERGY LINE BREAK INSIDE CONTAINMENT. AND Concurrently implement EP/1/A/5000/02, CRITICAL SAFETY FUNCTION STATUS TREES.
b. PZR Spray Valves: CLOSED	b. IF PZR press < 2260 PSIG, THEN: 1) Manually CLOSE PZR Spray Valves 2) IF any PZR Spray Valve CANNOT be CLOSED, THEN stop NC Pump supplying that valve.
___ 10. Check Containment Conditions:	
o Check following containment conditions: SATISFIED o Containment pressure: < .3 PSIG o "Ice Cond Lower Inlet Doors Open" alarm: DARK (1AD-13, A-7) o 1EMF-53A(B) "Containment High Range Monitor": < 3 R/HR. o Containment Sump Lvl: ZERO o TRN A o TRN B.	o Go to EP/1/A/5000/1C, HIGH ENERGY LINE BREAK INSIDE CONTAINMENT. AND o Concurrently implement EP/1/A/5000/02, CRITICAL SAFETY FUNCTION STATUS TREES.

ACTION/EXPECTED RESPONSE	RESPONSE NOT OBTAINED
___ 11. **Check For Unisolated Steam Line Break Outside Containment:**	
o Check all S/G pressures: o All S/G's Press: STABLE (NO S/G Pressure: DECREASING IN UNCONTROLLED MANNER) o All S/G's: PRESSURIZED (NO S/G DEPRESSURIZED)	o Go to EP/1/A/5000/1D, STEAM LINE BREAK OUTSIDE CONTAINMENT. <u>AND</u> o Concurrently implement EP/1/A/5000/02, CRITICAL SAFETY FUNCTION STATUS TREES.
___ 12. **Check For SGTR:**	
o Check following Secondary Conditions SATISFIED: o EMF-26 (27,28,29) "Stmline Hi Rad" (1RAD-3, E-5) alarm: DARK o 1EMF-33 "Cond AE Exh Hi Gas Rad" (1RAD-1, B-1) alarm: DARK o 1EMF-34 "S/G Sample Hi Rad" (1RAD-1, C-3) alarm: DARK o S/G levels: Responding Normally. (<u>NOT</u> increasing out of control).	o Go to EP/1/A/5000/1E, STEAM GENERATOR TUBE RUPTURE <u>AND</u> o Concurrently implement EP/1/A/5000/02, CRITICAL SAFETY FUNCTION STATUS TREES.

ACTION/EXPECTED RESPONSE	RESPONSE NOT OBTAINED
c. ND HX 1A AND 1B Outlet Flow: INDICATING FLOW.	c. Check NC pressure: 1) IF > 195 PSIG, THEN go to Step 3. 2) IF < 195 PSIG, THEN: a) Ensure correct valve alignment AND ND Pump operation.

NOTE During subsequent recovery actions, reference leg flashing may cause temporarily erroneous S/G Level indication on a faulted S/G.

ACTION/EXPECTED RESPONSE	RESPONSE NOT OBTAINED
___ 3. Check For Loss Of Secondary Integrity:	
o Check all S/G pressures: o ALL S/G Pressures: STABLE (NO S/G Pressure DECREASING in Uncontrolled Manner). o ALL S/G's: PRESSURIZED (NO S/G DEPRESSURIZED).	o Refer to EP/1/A/5000/1D, STEAM LINE BREAK OUTSIDE CONTAINMENT, Steps 1 through 4 AND isolate faulted S/G.

ACTION/EXPECTED RESPONSE	RESPONSE NOT OBTAINED
___ 11. Check NC Cold Leg Temperatures:	
o All NC T-COLD: > 400°F.	o Go to EP/1/A/5000/1D1, S/I TERMINATION FOLLOWING STEAM LINE BREAK.

NOTE: The following step is only to help diagnose an event AND not to provide additional guidance.

___ 12. Check For Feedwater Line Break Upstream Of CF Check Valve(s).

- o Check following conditions for indication of CF break:
  - o S/G pressure(s): STABLE OR INCREASING
  - o NC pressure: STABLE OR INCREASING
  - o Containment EMFs: NOT IN ALARM.

___ 13. To Terminate S/I:

- o Go to EP/1/A/5000/1C1, S/I TERMINATION FOLLOWING HIGH ENERGY LINE BREAK INSIDE CONTAINMENT.

___ 14. Initiate NC Cooldown AND Depressurization:

NOTE Steps 16 thru 18 below should be performed concurrently with the performance of EP/1/A/5000/1C2, POST LOCA COOLDOWN AND DEPRESSURIZATION.

- o Go to EP/1/A/5000/1C2, POST LOCA COOLDOWN AND DEPRESSURIZATION.

Form 34731 (10-81)
(Formerly SPD-1002-1)

FOR INFORMATION ONLY

DUKE POWER COMPANY
PROCEDURE PREPARATION
PROCESS RECORD

(1) ID No: EP/1/A/5000/02
Change(s) 0 to
0 Incorporated
Retype #2

(2) STATION: Catawba Nuclear Station

(3) PROCEDURE TITLE: Critical Safety Function Status Trees

(4) PREPARED BY: Ron M. DATE: 3/12/85

(5) REVIEWED BY: J R Ferguson DATE: 3/13/85

Cross-Disciplinary Review By: ______ N/R: JRF

(6) TEMPORARY APPROVAL (IF NECESSARY):

By: ______ (SRO) Date: ______

By: ______ Date: ______

(7) APPROVED BY: CW Graves, Jr. Date: 3/13/85

(8) MISCELLANEOUS:

Reviewed/Approved By: ______ Date: ______

Reviewed/Approved By: ______ Date: ______

This copy has been compared with the control copy and is verified correct.

Initial ______ Date ______ Time ______

Form 34912 (8-82)

TABLE OF CONTENTS

A. PURPOSE

The purpose of this procedure is to provide a convenient and effective means of monitoring the Critical Safety Functions (CSFs) through the use of status trees. The status trees are intended to serve two purposes:

1. GENERAL SURVEILLANCE under all sets of unusual or abnormal conditions that can lead to, or result from, initiation of safety injection.

2. DIRECT OPERATOR GUIDANCE in those rare events that go beyond the design basis of the Engineered Safeguards Systems and normal emergency procedures.

B. SYMPTOMS

- o Safety Injection Actuated

- o Operations Emergency Procedure Implemented

ACTION/EXPECTED RESPONSE	RESPONSE NOT OBTAINED

C. IMMEDIATE ACTIONS

None

D. SUBSEQUENT ACTIONS

NOTE The condition colors used in CASES A to F are to aid in identifying priorities of action.

- o Condition GREEN ( ) - Safety function satisfied: no operator action required.
- o Condition YELLOW (●●●●●●●●) - Safety function not fully satisfied: operator action may be needed.
- o Condition ORANGE ( ) - Safety function under severe challenge: prompt operator action required after other CSFs are verified acceptable.
- o Condition RED ( ) - Safety function in jeopardy: immediate operator action required.

NOTE Cases are numbered in order of priority. CASE A has top priority, CASE F the lowest priority. IF two cases have red conditions, THEN the higher priority case should be handled first.

ACTION/EXPECTED RESPONSE	RESPONSE NOT OBTAINED

___ 1. **Monitor Critical Safety Functions:**

a. Use attached Status Trees OR TECH SPEC Computer Program to monitor Critical Safety Functions (CSFs).

Critical Safety Function	Status Tree	TECH SPEC Computer Program
Subcriticality	EP/1/A/5000/2A	21
Core Cooling	EP/1/A/5000/2B	22
Heat Sink	EP/1/A/5000/2C	23
Reactor Coolant Integrity	EP/1/A/5000/2D	24
Containment	EP/1/A/5000/2E	25
Reactor Coolant Inventory	EP/1/A/5000/2F	26

b. IF all conditions are GREEN, THEN scan Status Trees OR TECH SPEC Computer Programs every 10 to 20 minutes.

b. IF any condition other than GREEN is encountered, THEN scan Status Trees OR TECH SPEC Computer Programs continuously.

c. Log CSF status on Enclosure 1.

___ 2. Monitor RP/0/A/5000/01, CLASSIFICATION OF EMERGENCY, For As Long As EP/1/A/5000/02, CRITICAL SAFETY FUNCTION STATUS TREES, Is In Effect.

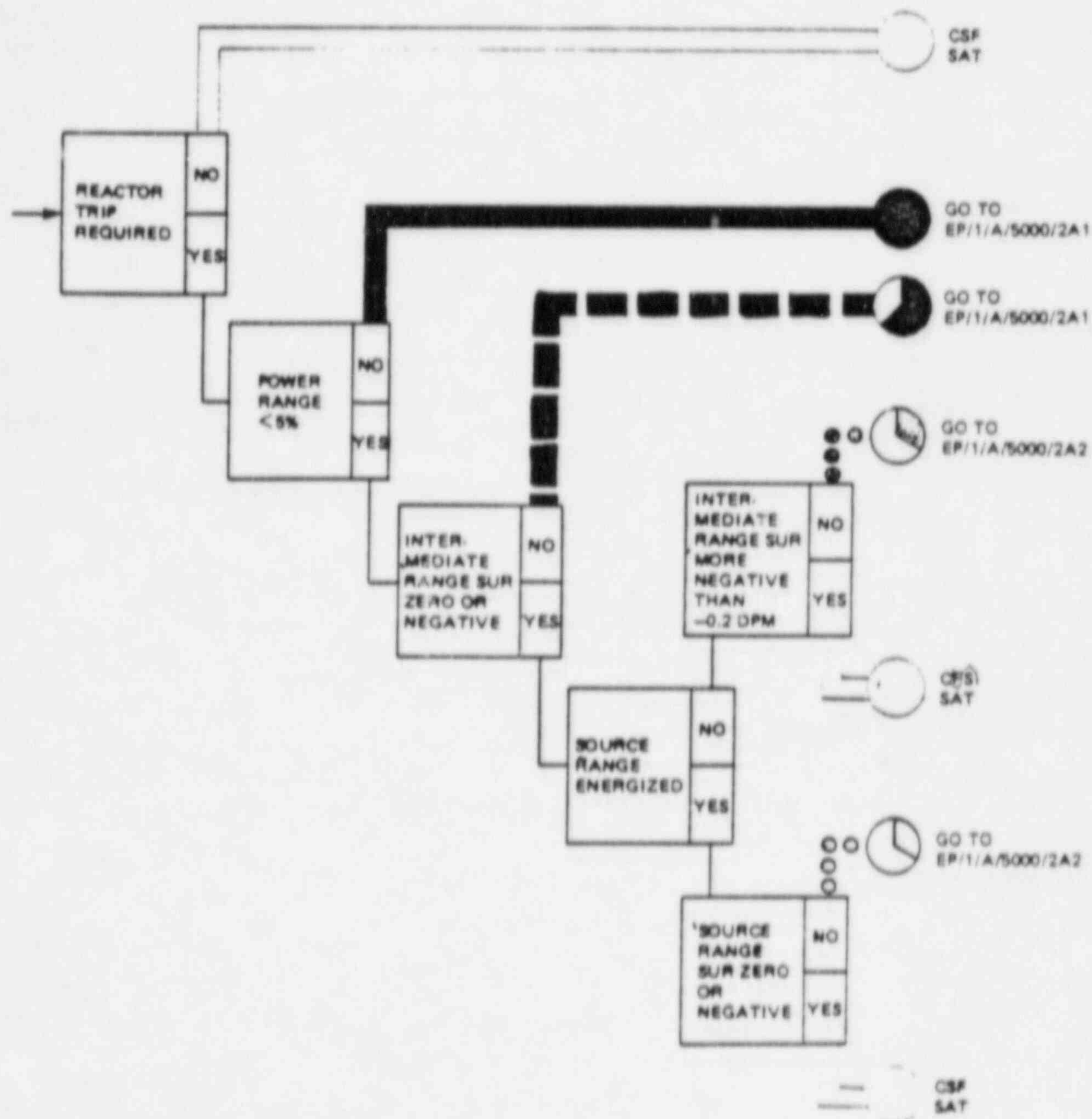
___ 3. Continue To Monitor CSFs Until The Emergency Procedure(s) Are Completed OR Until No Longer Necessary As Determined By The Shift Supervisor.

END

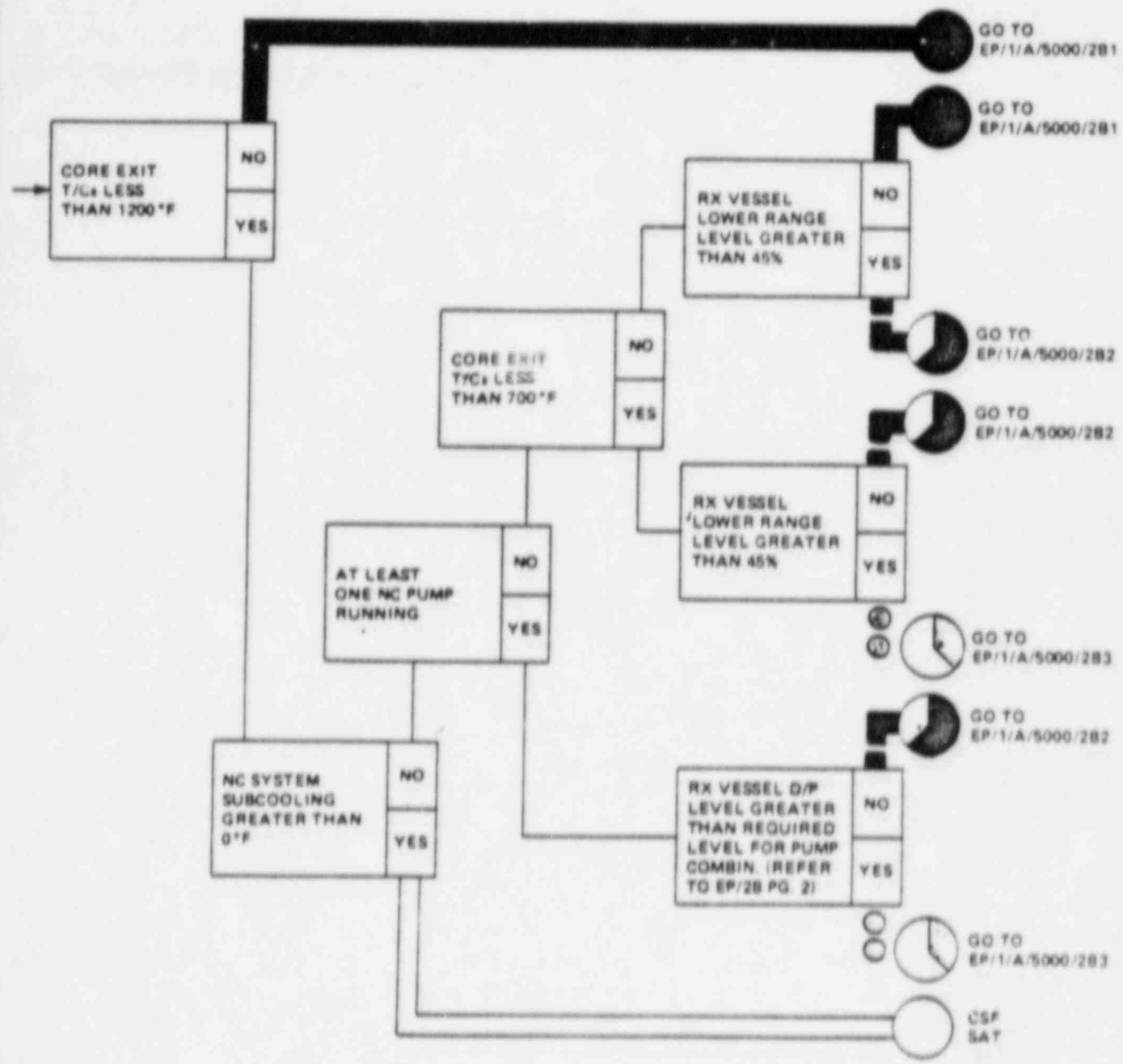
CSF STATUS LOG

NOTE Record conditions Green, Yellow, Orange, Red for each function as encountered.

DATE/TIME	SUB-CRITICALITY	CORE COOLING	INTEGRITY	HEAT SINK	CON-TAINMENT	INVENTORY	INITIAL



REACTOR TRIP REQUIRED
NO
YES
CSF SAT
POWER RANGE < 5%
NO
YES
GO TO EP/1/A/5000/2A1
INTER-MEDIATE RANGE SUR ZERO OR NEGATIVE
NO
YES
GO TO EP/1/A/5000/2A1
INTER-MEDIATE RANGE SUR MORE NEGATIVE THAN −0.2 DPM
NO
YES
GO TO EP/1/A/5000/2A2
SOURCE RANGE ENERGIZED
NO
YES
CSF SAT
SOURCE RANGE SUR ZERO OR NEGATIVE
NO
YES
GO TO EP/1/A/5000/2A2
CSF SAT

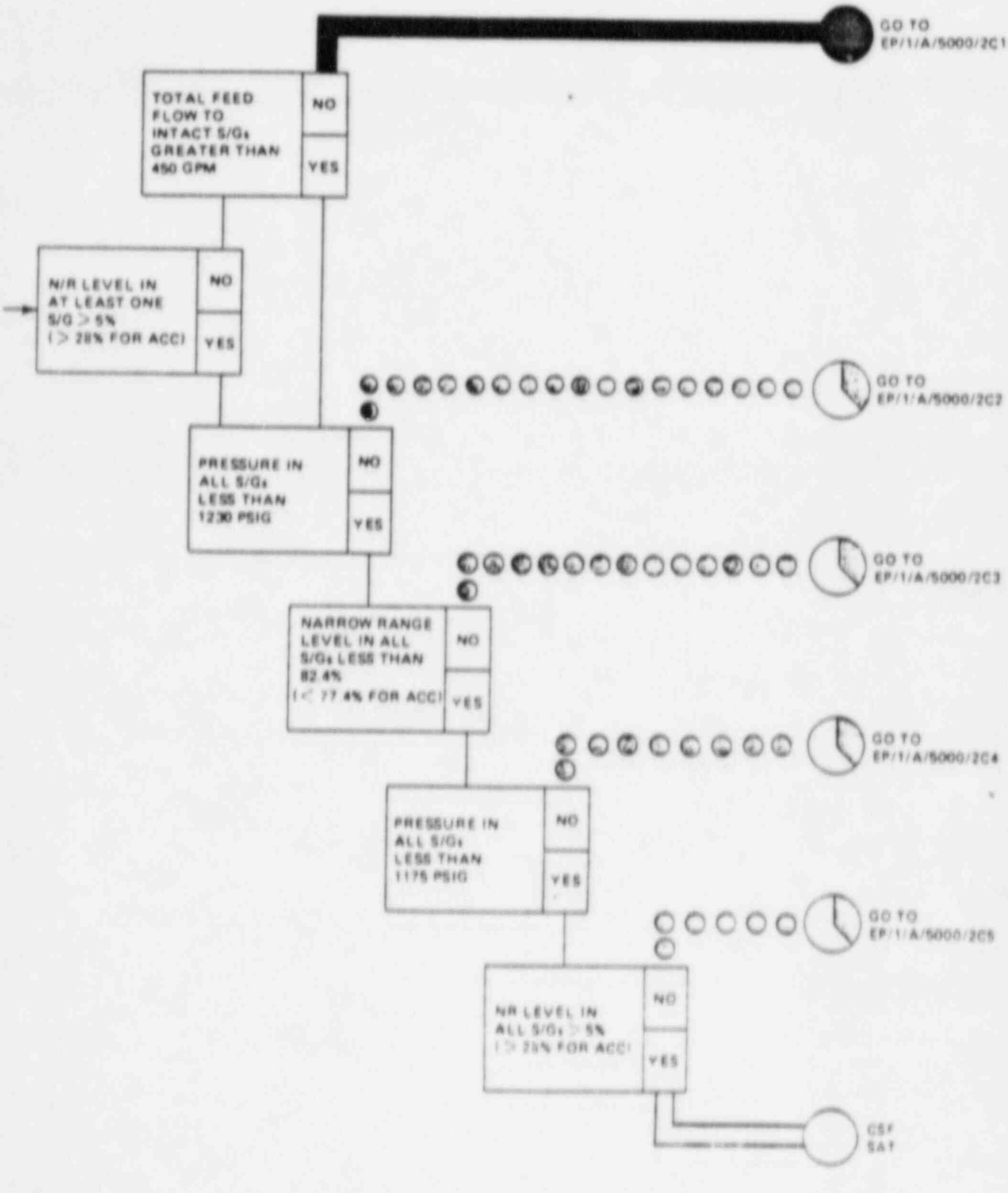

CORE EXIT T/Cs LESS THAN 1200°F
NO
YES
GO TO EP/1/A/5000/2B1
GO TO EP/1/A/5000/2B1
RX VESSEL LOWER RANGE LEVEL GREATER THAN 45%
NO
YES
CORE EXIT T/Cs LESS THAN 700°F
NO
YES
GO TO EP/1/A/5000/2B2
GO TO EP/1/A/5000/2B2
RX VESSEL LOWER RANGE LEVEL GREATER THAN 45%
NO
YES
AT LEAST ONE NC PUMP RUNNING
NO
YES
GO TO EP/1/A/5000/2B3
GO TO EP/1/A/5000/2B2
NC SYSTEM SUBCOOLING GREATER THAN 0°F
NO
YES
RX VESSEL D/P LEVEL GREATER THAN REQUIRED LEVEL FOR PUMP COMBIN. (REFER TO EP/2B PG. 2)
NO
YES
GO TO EP/1/A/5000/2B3
CSF SAT

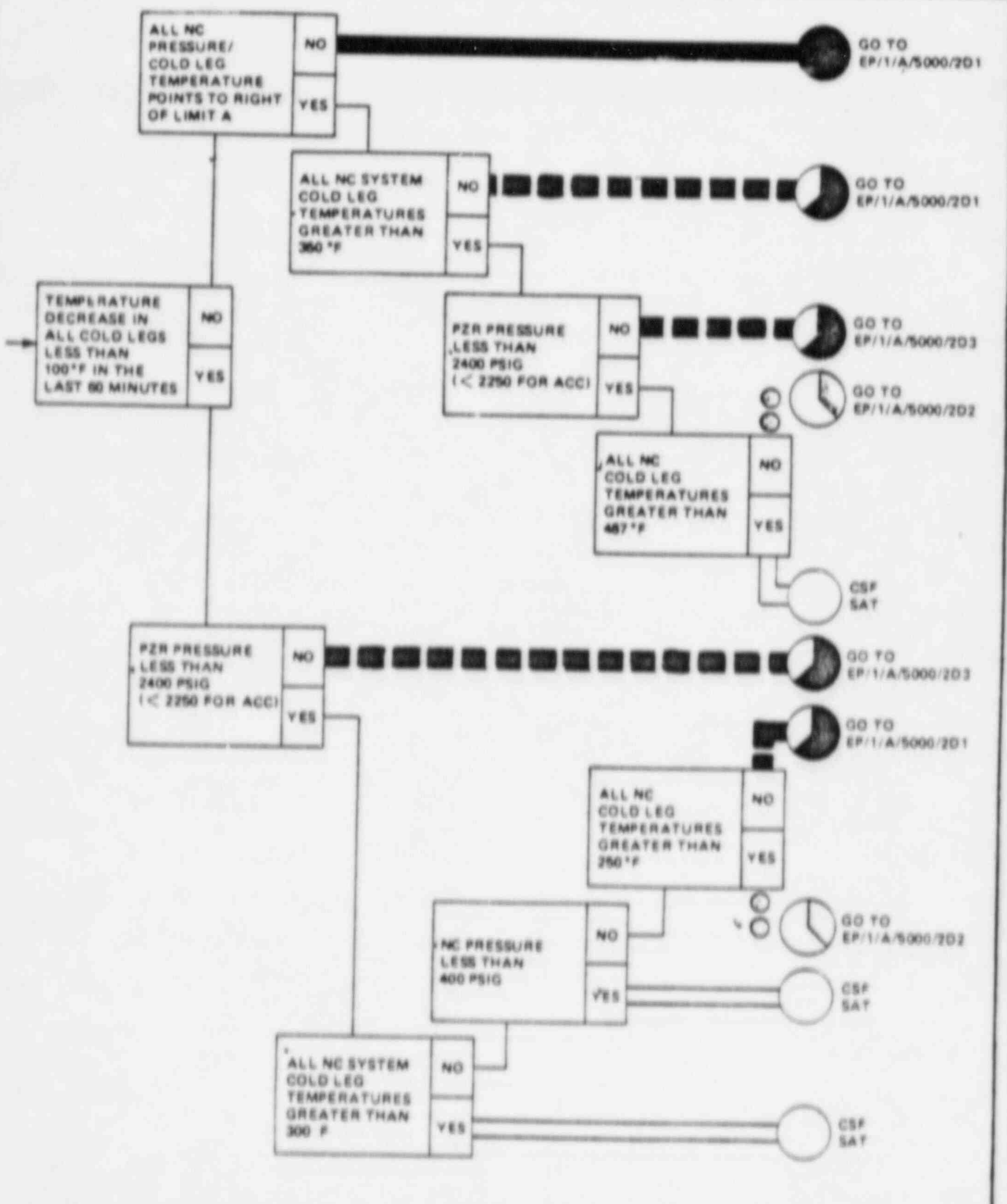
Form 34912 (8-82)

DYNAMIC HEAD RANGE RVLIS
SETPOINTS FOR
DEGRADED CORE COOLING

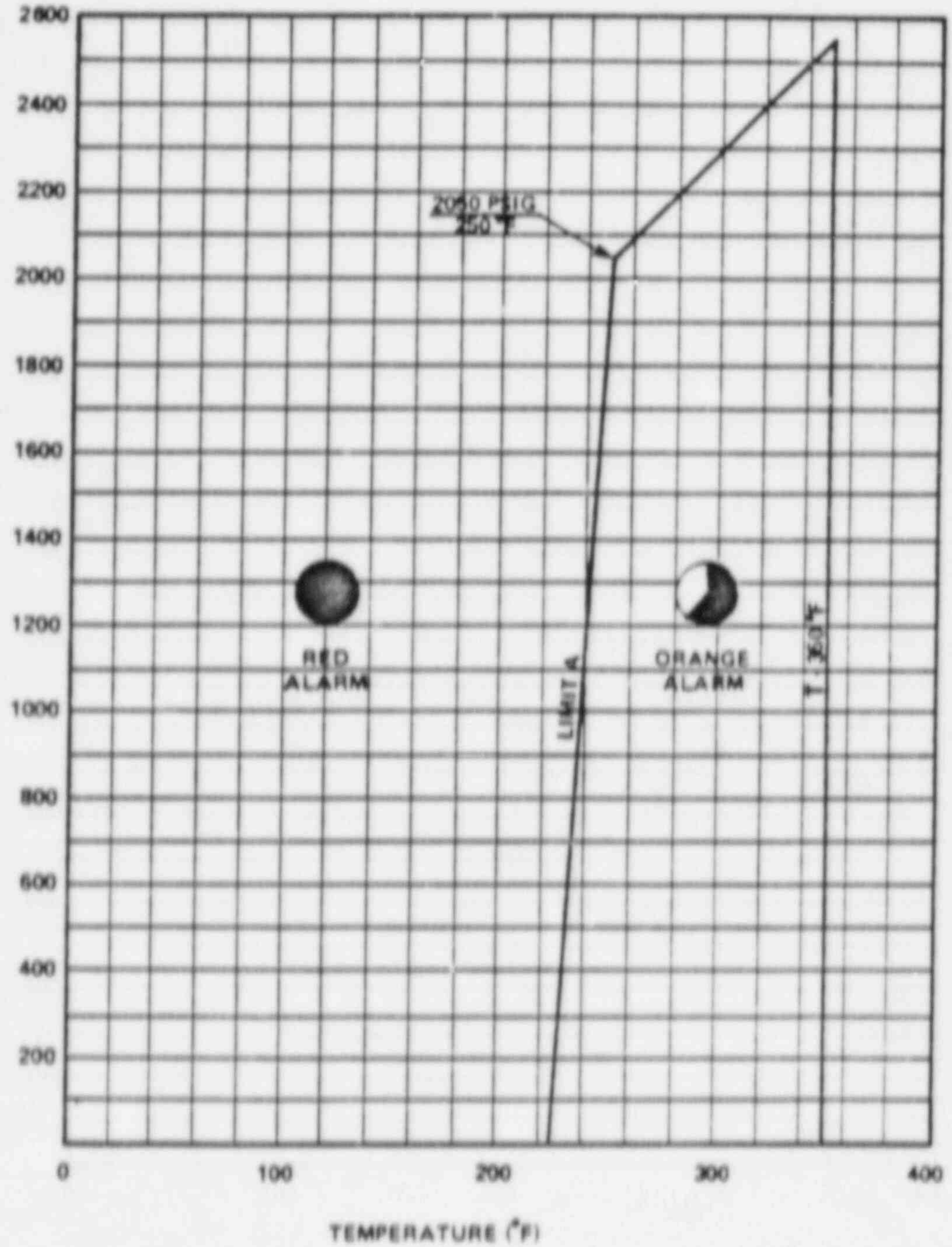
Number of NC Pumps Running	Channel A with NC Pump A		Channel B with NC Pump C	
	Running	Not Running	Running	Not Running
4	80%	--	80%	--
3	60%	35%	60%	35%
2	45%	23%	45%	23%
1	35%	15%	35%	15%

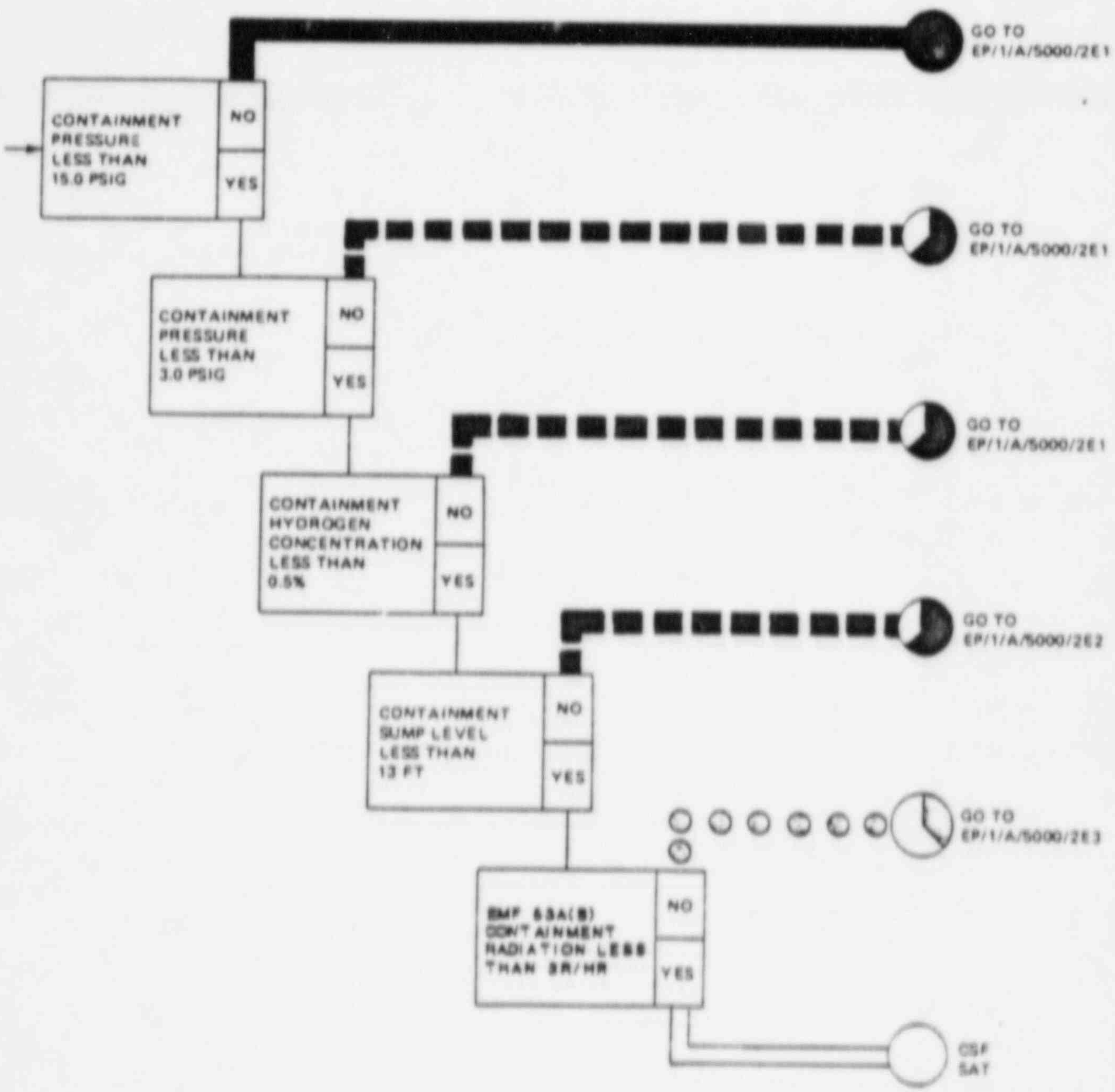
CNS EP/1/A/5000/2C	HEAT SINK	PAGE NO. 1 RETYPE #2



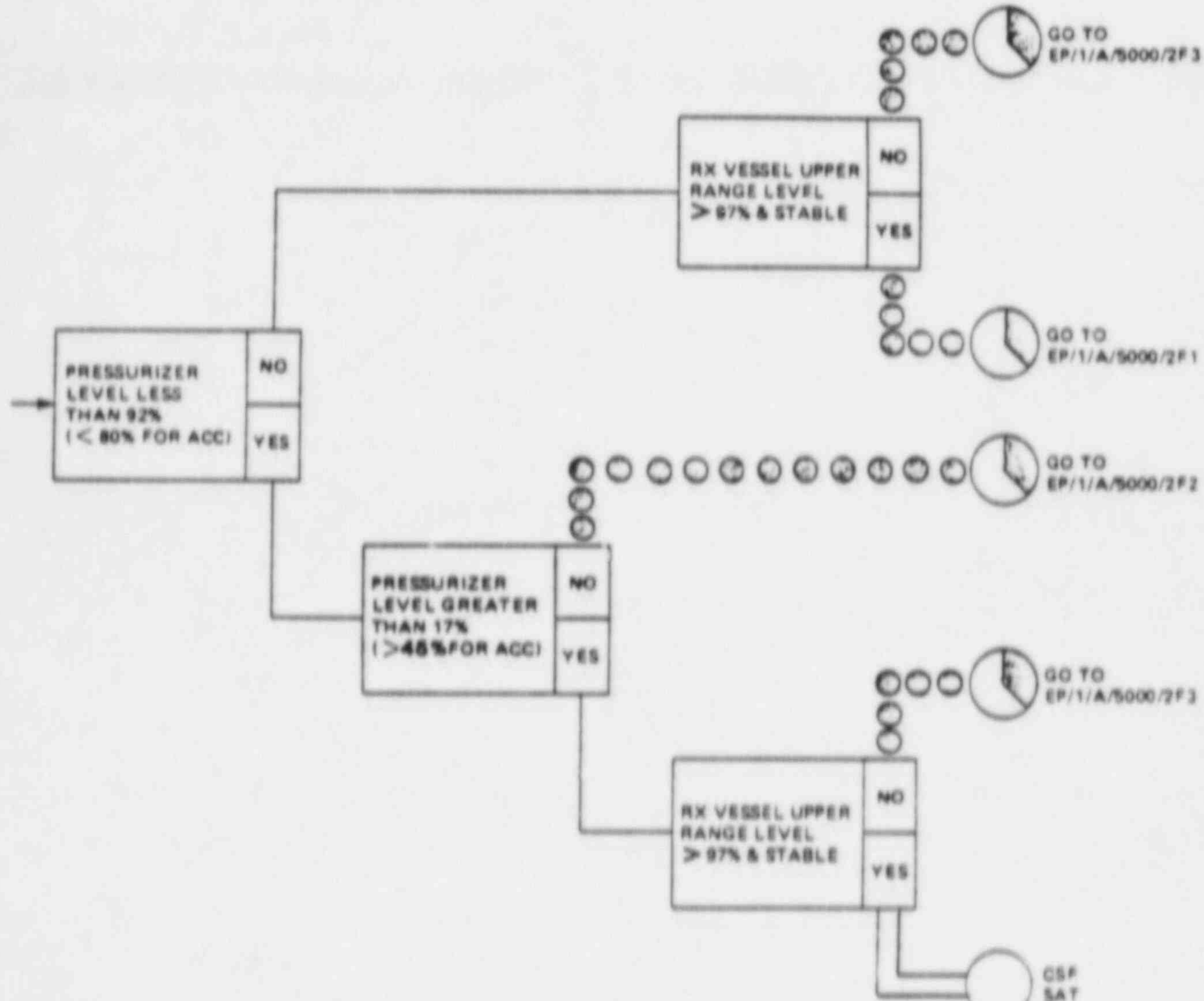

ALL NC PRESSURE/ COLD LEG TEMPERATURE POINTS TO RIGHT OF LIMIT A
NO
YES
GO TO EP/1/A/5000/2D1
ALL NC SYSTEM COLD LEG TEMPERATURES GREATER THAN 350 °F
NO
YES
GO TO EP/1/A/5000/2D1
TEMPERATURE DECREASE IN ALL COLD LEGS LESS THAN 100°F IN THE LAST 60 MINUTES
NO
YES
PZR PRESSURE LESS THAN 2400 PSIG (< 2250 FOR ACC)
NO
YES
GO TO EP/1/A/5000/2D3
GO TO EP/1/A/5000/2D2
ALL NC COLD LEG TEMPERATURES GREATER THAN 487 °F
NO
YES
CSF SAT
PZR PRESSURE LESS THAN 2400 PSIG (< 2250 FOR ACC)
NO
YES
GO TO EP/1/A/5000/2D3
GO TO EP/1/A/5000/2D1
ALL NC COLD LEG TEMPERATURES GREATER THAN 250 °F
NO
YES
GO TO EP/1/A/5000/2D2
NC PRESSURE LESS THAN 400 PSIG
NO
YES
CSF SAT
ALL NC SYSTEM COLD LEG TEMPERATURES GREATER THAN 300 F
NO
YES
CSF SAT

PTS LIMIT CURVE




CONTAINMENT PRESSURE LESS THAN 15.0 PSIG
NO
YES
GO TO EP/1/A/5000/2E1
CONTAINMENT PRESSURE LESS THAN 3.0 PSIG
NO
YES
GO TO EP/1/A/5000/2E1
CONTAINMENT HYDROGEN CONCENTRATION LESS THAN 0.5%
NO
YES
GO TO EP/1/A/5000/2E1
CONTAINMENT SUMP LEVEL LESS THAN 12 FT
NO
YES
GO TO EP/1/A/5000/2E2
EMF 53A(B) CONTAINMENT RADIATION LESS THAN 3R/HR
NO
YES
GO TO EP/1/A/5000/2E3
CSF SAT

CNS EP/1/A/5000/2F	REACTOR COOLANT INVENTORY	PAGE NO. 1 RETYPE #



Form 34913 (8-82)

CNS EP/1/A/5000/2E1	HIGH CONTAINMENT PRESSURE	PAGE NO. 1 Retype #2

ACTION/EXPECTED RESPONSE	RESPONSE NOT OBTAINED

NOTE Actions performed in other procedures may require Containment Isolation Valves to be opened. An evaluation of any Containment Isolation Valve found to be open should be made before it is closed in Step 1.

___ 1. Verify Containment Isolation:

a. All Containment Isolation Phase A AND Containment Ventilation Isolation Valves CLOSED

- o Monitor Light Panel, Group 5 ST Valves: LIT.

a. IF any valve is OPEN that is NOT required to be open, THEN Manually OR Locally CLOSE it.

b. All Containment Isolation Phase B Valves: CLOSED

- o Monitor Light Panel, Group 5, Sp Valves: LIT.
- o Monitor Light Panel Group 1, Sp Valves: LIT.

b. IF any valve is OPEN that is NOT required to be open, THEN Manually OR Locally CLOSE it.

****
CAUTION
****

- o IF FWST level reaches Lo-Lo Level (11%) setpoint, NS system should be switched to sump recirculation referring to EP/1/A/5000/1C3, TRANSFER TO COLD LEG RECIRCULATION.
- o IF EP/1/A/5000/1C5, LOSS OF EMERGENCY COOLANT RECIRCULATION has been implemented, THEN NS should be operated in accordance with that procedure rather than in Step 2 below.

___ 2. Verify Containment Spray Actuation:

a. Ensure NS Pumps: ON.

- o NS PMP 1A
- o NS PMP 1B.

ACTION/EXPECTED RESPONSE	RESPONSE NOT OBTAINED
b. Ensure required NS valve alignment established (Refer to Monitor Light Panels):	
o During Injection Mode, Group 3: DARK.	
OR	
During Recirculation Mode, Group 3: LIT	
o Group 1 (G-1,2,11,12): LIT.	
___ 3. Ensure MSIVs <u>AND</u> MSIV Bypass Valves: CLOSED.	

****
<u>CAUTION</u> At least one S/G should be maintained available for NC cooldown.
****

ACTION/EXPECTED RESPONSE	RESPONSE NOT OBTAINED
___ 4. Check <u>IF</u> Feed Should Be Isolated To Any S/G:	
a. Check S/G pressures:	a. <u>IF</u> NEITHER condition exists, <u>THEN</u> go to Step 5.
o Any S/G pressure: DECREASING in an UNCONTROLLED manner	
OR	
o Any S/G: DEPRESSURIZED.	
b. Verify CF Flow To Faulted S/G Isolated:	
o Verify required monitor light configuration on 1MC-14.	

ACTION/EXPECTED RESPONSE	RESPONSE NOT OBTAINED
o Group 1 Lights for faulted S/G: LIT o S/G 1A: B-3, B-4; F-3 o S/G 1B: C-3, C-4; F-4 o S/G 1C: C-9, C-10; F-9 o S/G 1D: B-9, B-10; F-10	o Manually CLOSE valves.
o Group 2 Lights for faulted S/G: DARK o S/G 1A: C-1, C-3; D-1 o S/G 1B: C-2, C-4; D-2 o S/G 1C: C-9, C-11; D-11 o S/G 1D: C-10, C-12; D-12	o Manually CLOSE valves.
c. Isolate CA Flow To Faulted S/G:	
o <u>IF</u> 1A S/G FAULTED, <u>THEN</u>: CLOSE: 1CA-62A (CA Pmp A Disch To S/G 1A Isol) <u>AND</u> Ensure 1CA-66B (CA Pmp 1 Disch To S/G 1A Isol): CLOSED.	
o <u>IF</u> 1B S/G FAULTED, <u>THEN</u> CLOSE: 1CA-58A (CA Pmp A Disch To S/G 1B Isol) <u>AND</u> 1CA-54B (CA Pmp 1 Disch To S/G 1B Isol).	
o <u>IF</u> 1C S/G FAULTED, <u>THEN</u> CLOSE: 1CA-46B (CA Pmp B Disch To S/G 1C Isol) <u>AND</u>	

ACTION/EXPECTED RESPONSE	RESPONSE NOT OBTAINED
1CA-50A (CA Pmp 1 Disch To S/G 1C Isol).	
o <u>IF</u> 1D S/G FAULTED, <u>THEN</u>:	
CLOSE: 1CA-42B (CA Pmp B Disch To S/G 1D Isol)	
<u>AND</u>	
Ensure 1CA-38A (CA Pmp 1 Disch To S/G 1D Isol): **CLOSED.**	
___ 5. **Verify VE Operation:**	
o At rear of 1MC5, verify:	
a. VE Fan A <u>AND</u> B: **RUNNING.**	a. Manually start Fans.
b. Annulus Press Trending to <u>OR</u> ≥ -1.0 IN. W.C. o TRN A o TRN B	b. <u>IF</u> < -1.0 IN. W.C. <u>AND</u> increasing, <u>THEN</u>: 1) Ensure following VE Fan Isol Dampers: OPEN o 1AVS-D-5 o 1AVS-D-10 2) Check VE Flow to Stack INDICATED: o 1A o 1B. <u>IF</u> flow <u>NOT</u> indicated, <u>THEN</u> dispatch operator to locally verify status of following dampers based on position of their operating pistons: o Recirculation Dampers CLOSED: o 1AVS-D-2 (AB-600, JJ-51) o 1AVS-D-7 (AB-603, HH-52)

ACTION/EXPECTED RESPONSE	RESPONSE NOT OBTAINED
	o Exhaust Dampers OPEN: o 1AVS-D-3 (AB-603, JJ-52) o 1AVS-D-8 (AB-600, HH-52) 3) Consult plant engineering staff AND request I&E/ Maintenance troubleshoot AND repair as necessary.

NOTE The NS Pumps AND VX Air Return Fans should stop automatically WHEN Containment pressure < 0.3 PSIG AND start automatically IF pressure > 0.4 PSIG. Once reset, NS must be manually initiated to operate the system with pressure < 3 PSIG.

ACTION/EXPECTED RESPONSE	RESPONSE NOT OBTAINED
___ 6. Ensure VX Operation:	
a. 10 minute time delay expired.	a. Go to Step 7 AND WHEN > 10 minutes after Sp, THEN do Step 6.b.
b. At rear of 1MC-5, ensure VX Fans: RUNNING. o ARF A AND B o HSF A AND B.	
___ 7. Check IF ND Auxiliary Containment Spray Can Be Provided:	
a. ND Train 1A OR 1B aligned for Cold Leg Recirculation (to satisfy interlocks.) o Monitor Light Panel Group 3: LIT.	a. Continue with Step 9.

ACTION/EXPECTED RESPONSE	RESPONSE NOT OBTAINED
NOTE The 50 minute time limit is based on the time required for all the ice condenser contents to melt during a Design Basis LOCA, after which containment pressure will increase.	
b. Time since Sp signal: > 50 MIN.	b. IF containment pressure > 10 psig AND STABLE OR INCREASING, THEN go to Step 7.c, "Response Not Obtained". Otherwise, go to Step 9.
c. Verify Cold Leg Injection Flow available: o NV S/I Flow: INDICATED AND o NI Pump 1A OR 1B Disch Flow: INDICATED.	c. IF both ND Trains operable AND neither is aligned to provide ND Auxiliary Containment Spray, THEN go to Step 8. IF only one ND Train operable OR one currently aligned to provide ND Auxiliary Containment Spray, THEN go to Step 9.
___ 8. Align ND For Auxiliary Containment Spray:	
a. Align 1 available ND Train to provide ND Auxiliary Containment Spray. o Train A: 1) Place PWR DISCON FOR 1NI-173A to "ENABL". 2) CLOSE 1NI-173A (ND Hdr 1A To Cold Legs C & D) 3) OPEN 1NS-43A (ND Pmp 1A To Cont Spray Hdr) o Train B: 1) Place PWR DISCON FOR 1NI-178B to "ENABL".	a. IF NO ND Train available, THEN go to Step 9.

ACTION/EXPECTED RESPONSE	RESPONSE NOT OBTAINED
2) CLOSE 1NI-178B (ND Hdr 1B To Cold Legs A & B).	
3) OPEN 1NS-38B (ND Pmp 1B To Cont Spray Hdr).	
b. IF ONE OR MORE of the following conditions exist, THEN continue with Step 9:	b. IF NONE of the conditions are met in Step 8.b, THEN return to Step 7.c. to determine if a second ND Train should be aligned for Containment Auxiliary Spray.
o Time since Sp signal: < 50 MIN.	
OR	
o Both ND Trains aligned to provide ND Auxiliary Containment Spray	
OR	
o Containment Pressure: < 10 psig.	

NOTE
- o Appropriate steps in any other procedure in effect may be performed while waiting for results of H2 measurement.
- o IF 10 minute time delay has expired since Sp signal, THEN Step 6.b may be performed.

____ 9. Check Containment H2 Concentration:

a. Dispatch an operator to place Containment H2 Analyzers in service referring to OP/1/A/6450/10, CONTAINMENT HYDROGEN CONTROL SYSTEMS.

- o H2 Analyzer Locations:
  - o TRN A: AB-579 (DD-53)
  - o TRN B: AB-559 (DD-53)

****
CAUTION
****
The Containment Hydrogen Purge System should NOT be operated without TSC AND Plant Engineering Staff concurrence due to the potential for radioactive release.

b. H2 concentration: < 3.5%.	b. IF EP/1/A/5000/2B2, DEGRADED CORE COOLING previously OR currently in effect, THEN go to Step 10. IF EP/1/A/5000/2B2 NOT previously OR currently in effect, THEN place Hydrogen Purge System in service referring to OP/1/A/6450/10, CONTAINMENT HYDROGEN CONTROL SYSTEMS AND go to Step 11.
c. H2 concentration: < 0.5%.	c. Initiate H2 Recombiner/EHM System operation referring to OP/1/A/6450/10 CONTAINMENT HYDROGEN CONTROL SYSTEMS.
d. Return to procedure in effect.	

___ 10. In Conjunction With TSC Evaluate Options:

a. Start Recombiners based on further analysis of flammability limits.

b. Reset Phase A Containment Isolation to establish H2 Purge System operation referring to OP/1/A/6450/10, CONTAINMENT HYDROGEN CONTROL SYSTEMS.

___ 11. Monitor H2 Concentration:

a. Hydrogen concentration: STABLE OR DECREASING.

b. Ensure TSC informed of H2 concentration at least once per 30 minutes OR WHEN any significant change in H2 concentration occurs.

NOTE H2 Recombiners should NOT be started with H2 concentration > 6% since they have not been tested to operate above this limit AND could become a H2 ignition source if started.

___ 12. Check IF H2 Recombiners Should Be Started:

a. H2 concentration < 6%.	a. Return to procedure in effect.
b. Start recombiners referring to OP/1/A/6450/10, CONTAINMENT HYDROGEN CONTROL SYSTEMS.	

___ 13. Return To Procedure In Effect.

END

ACTION/EXPECTED RESPONSE	RESPONSE NOT OBTAINED

<u>NOTE</u> Throughout this procedure, "AFFECTED" refers to a S/G in which N/R level < 5% (< 28% for adverse containment conditions (ACC)).

****
<u>CAUTION</u> Steam releases from affected S/G(s) should be minimized.
****

___ 1. Identify Affected S/G(s):

- o N/R level: < 5% (< 28% ACC).

  RESPONSE NOT OBTAINED: o <u>IF</u> N/R Level <u>NOT</u> Low, <u>THEN</u> return to procedure in effect.

___ 2. Ensure Affected S/G(s) Blowdown Valves: CLOSED.

- o S/G 1A:
  - o 1BB-56A (S/G 1A Bldwn Cont Isol Insd)
  - o 1BB-57B (S/G 1A Bldwn Cont Isol Otsd)
  - o 1BB-148B (S/G 1A Bldwn Cont Isol Byp)
  - o 1BB-69 (S/G 1A Bldwn Flow Ctrl)
- o S/G 1B:
  - o 1BB-19A (S/G 1B Bldwn Cont Isol Insd)
  - o 1BB-21B (S/G 1B Bldwn Cont Isol Otsd)
  - o 1BB-150B (S/G 1B Bldwn Cont Isol Byp)
  - o 1BB-73 (S/G 1B Bldwn Flow Ctrl)

ACTION/EXPECTED RESPONSE	RESPONSE NOT OBTAINED

- o S/G 1C:
  - o 1BB-60A (S/G 1C Bldwn Cont Isol Insd)
  - o 1BB-61B (S/G 1C Bldwn Cont Isol Otsd)
  - o 1BB-149B (S/G 1C Bldwn Cont Isol Byp)
  - o 1BB-24 (S/G 1C Bldwn Flow Ctrl)
- o S/G 1D:
  - o 1BB-8A (S/G 1D Bldwn Cont Isol Insd)
  - o 1BB-10B (S/G 1D Bldwn Cont Isol Otsd)
  - o 1BB-147B (S/G 1D Bldwn Cont Isol Byp)
  - o 1BB-65 (S/G 1D Bldwn Flow Ctrl)

****
CAUTION CA flow should not be established to a faulted S/G that has been isolated.
****

ACTION/EXPECTED RESPONSE	RESPONSE NOT OBTAINED
___ 3. **Verify Affected S/G(s) Intact:**	o IF affected S/G(s) has been previously isolated, THEN return to procedure in effect.
o S/G pressure: NOT DECREASING IN AN UNCONTROLLED MANNER	OR
OR	o IF NOT isolated, THEN:
o S/G pressure: NOT COMPLETELY DEPRESSURIZED.	a. CLOSE affected S/G(s) MSIV AND MSIV Bypass Valve.
	b. Refer to Enclosure 1 AND stop all feed flow to affected S/G.

ACTION/EXPECTED RESPONSE	RESPONSE NOT OBTAINED
	c. IF either S/G 1B OR 1C affected, THEN dispatch an operator to unlock AND close CA Pump 1 Steam Supply from affected S/G. o S/G 1B: 1SA-3 (Main Steam 1B to CA Pump No. 1 Stop Check) AB-551 (DD-53). o S/G 1C: 1SA-6 (Main Steam 1C to CA Pump No. 1 Stop Check) AB-551 (DD-53). d. Ensure affected S/G(s) PORV: CLOSED. e. Return to procedure in effect.
___ 4. Check CA Flow To Affected S/G(s): o CA flow: > 25 GPM.	o Check affected S/G W/R level: o IF > 3% (> 31% ACC), THEN establish CA flow rate required to refill affected S/G(s) OR o IF < 3% (< 31% ACC) THEN establish CA flow to affected S/G(s) at a rate NOT to exceed 100 GPM per S/G to prevent thermal shock to S/G.

ACTION/EXPECTED RESPONSE	RESPONSE NOT OBTAINED
___ 5. Continue Filling Affected S/G(s) To Restore N/R Level: > 5% (> 28% ACC).	
___ 6. Return To Procedure In Effect.	

END

___ A. Ensure Affected S/G Feedwater Isolation Valves: CLOSED

- o S/G 1A:
  - o 1CF-33 (S/G 1A CF Cont Isol)
  - o 1CF-90 (S/G 1A CF Cont Isol Byp)
  - o 1CA-149 (S/G 1A CF Byp To CA Nozzle Isol)
  - o 1CA-185 (S/G 1A CA Nozz Tempering Isol).
- o S/G 1B:
  - o 1CF-42 (S/G 1B CF Cont Isol)
  - o 1CF-89 (S/G 1B CF Cont Isol Byp)
  - o 1CA-150 (S/G 1B CF Byp To CA Nozzle Isol)
  - o 1CA-186 (S/G 1B CA Nozz Tempering Isol).
- o S/G 1C:
  - o 1CF-51 (S/G 1C CF Cont Isol)
  - o 1CF-88 (S/G 1C CF Cont Isol Byp)
  - o 1CA-151 (S/G 1C CF Byp To CA Nozzle Isol)

<u>IF</u> required valves <u>CANNOT</u> be closed, <u>THEN</u> ensure following valves for affected S/G: CLOSED

- o S/G 1A:
  - o 1CF-28 (S/G 1A CF Ctrl)
  - o 1CF-30 (S/G 1A CF Byp Ctrl).
- o S/G 1B:
  - o 1CF-37 (S/G 1B CF Ctrl)
  - o 1CF-39 (S/G 1B CF Byp Ctrl).
- o S/G 1C:
  - o 1CF-46 (S/G 1C CF Ctrl)
  - o 1CF-48 (S/G 1C CF Byp Ctrl).

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- o 1CA-187 (S/G 1C CA Nozz Tempering Isol.

- o S/G 1D:
  - o 1CF-60 (S/G 1D CF Cont Isol)
  - o 1CF-87 (S/G 1D CF Cont Isol Byp)
  - o 1CA-152 (S/G 1D CF Byp To CA Nozzle Isol)
  - o 1CA-188 (S/G 1D CA Nozz Tempering Isol).

- o S/G 1D:
  - o 1CF-55 (S/G 1D CF Ctrl)
  - o 1CF-57 (S/G 1D CF Byp Ctrl).

____ B. Isolate CA Flow To Affected S/G:

- o Close the following valves for affected S/G:
  - o S/G 1A:

    1CA-62A (CA Pmp A Disch To S/G 1A Isol)
    1CA-66B (CA Pmp 1 Disch To S/G 1A Isol).
  - o S/G 1B:

    1CA-58A (CA Pmp A Disch To S/G 1B Isol)
    1CA-54B (CA Pmp 1 Disch To S/G 1B Isol).
  - o S/G 1C:

    1CA-50A (CA Pmp 1 Disch To S/G 1C Isol)
    1CA-46B (CA Pmp B Disch To S/G 1C Isol).

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- S/G 1D:

  1CA-38A (CA Pmp 1 Disch To S/G 1D Isol)
  1CA-42B (CA Pmp B Disch To S/G 1D Isol).

ACTION/EXPECTED RESPONSE	RESPONSE NOT OBTAINED

___ 1. **Check NC Cold Leg Temperature:**

o T-COLD W/R:

STABLE OR INCREASING.

o IF decreasing, THEN try to stop NC System cooldown

a. Ensure S/G PORV(s): CLOSED.

b. Ensure Steam Dumps: CLOSED.

c. Check for faulted S/G(s):

o Any S/G Pressure: DECREASING IN UNCONTROLLED MANNER

OR

o Any S/G Pressure: DEPRESSURIZED.

d. Isolate any faulted S/G(s):

1) Refer to Enclosure 1 AND stop all feed flow to faulted S/G(s) UNLESS needed for NC temperature control.

2) Close S/G MSIV AND MSIV Bypass Valves.

3) IF S/G 1B AND/OR 1C FAULTED, THEN dispatch operator to unlock AND locally close steam supply valve(s) to CA Pump No. 1:

o S/G 1B: 1SA-3 (Main Steam 1B To CA Pump No. 1 Stop Check), AB-551 (DD-53)

ACTION/EXPECTED RESPONSE	RESPONSE NOT OBTAINED

o S/G 1C: 1SA-6 (Main Steam 1C To CA Pump No. 1 Stop Check), AB-551 (DD-53).

4) Close "S/G OTLT HDR BLDWN CV" for faulted S/G:

- o S/G 1A: 1SM-77A
- o S/G 1B: 1SM-76B
- o S/G 1C: 1SM-75A
- o S/G 1D: 1SM-74B.

e. Regulate feed flow to non-faulted S/G(s) to STOP NC cooldown.

1) Maintain total feed flow > 450 GPM until N/R level in at least one (1) non-faulted S/G: > 5% (> 28% for adverse containment conditions (ACC)).

2) IF CA supplying S/G feed, THEN re-establish CA flow control:

a) "RESET" ECCS TRN A AND TRN B.

b) CA Pmp #1 - Select "OFF".

c) Ensure NC System pressure < 1955 PSIG AND P-11 permissive LIT.

d) Depress "CA PUMP AUTO-START DEFEAT":

- o 1A DEFEAT
- o 1B DEFEAT.

ACTION/EXPECTED RESPONSE	RESPONSE NOT OBTAINED
	e) "RESET" "CA SYS VLV CTRL" switches o TRN A o TRN B.
	3) Throttle feed as necessary to stop NC cooldown.
	f. IF all S/G's faulted OR faulted S/G required for NC temperature control, THEN decrease feed flow to faulted S/G(s) to ~ 25 GPM.
	g. IF ND in service, THEN stop any cooldown from ND.

****
CAUTION
****
Preference should be given to running an NC pump that will provide PZR spray (NC Pump B first, NC Pump A second).

___ 2. Check NC Pump Status:

ACTION/EXPECTED RESPONSE	RESPONSE NOT OBTAINED
a. Less than two (2) NC Pumps: RUNNING.	a. Stop all but one (1) NC Pump. Go to Step 3.
b. PZR Level: < 95% (<80% for adverse containment conditions (ACC)).	b. IF one NC Pump running, THEN go to Step 3. IF NO NC Pump running, THEN: o Monitor PZR Level AND continue with Step 3 until PZR Level < 95% (<80% ACC), THEN perform Step 2.c.
c. Verify one NC Pump: RUNNING.	c. IF NO NC Pump running, THEN: 1) Check if NC Pump can be started: a) NC Subcooling: > 0°F AND

ACTION/EXPECTED RESPONSE	RESPONSE NOT OBTAINED
	b) Following conditions satisfied: - PZR level response: NORMAL AND RX VESSEL UPPER RANGE LEVEL TRN A (TRN B): > 97% AND STABLE o IF NOT satisfied, THEN refer to EP/1/A/5000/2F3, VOID IN REACTOR VESSEL, AND continue.
	2) IF NC Pump CANNOT be started, THEN go to Step 3.
	3) IF NC Pump can be started, THEN establish conditions for starting NC Pump referring to OP/1/A/6150/02A, REACTOR COOLANT PUMP OPERATION, AND start one (1) NC Pump.
___ 3. Check If S/I Terminated:	
o NV S/I Flow: NOT INDICATED AND NI Pumps 1A AND 1B: NOT RUNNING.	o IF S/I NOT terminated, THEN go to Step 6.
___ 4. Check NC Pressure:	
o NC pressure: WITHIN cooldown curve limits of Enclosure 2.	o IF NC pressure greater than limits, THEN decrease NC pressure to within curve limits using normal PZR Spray.

ACTION/EXPECTED RESPONSE	RESPONSE NOT OBTAINED
	o IF normal PZR spray NOT available, THEN: o IF Charging AND Letdown in service, THEN use NV Auxiliary Spray OR o IF Charging AND Letdown NOT in service, THEN use one (1) PZR PORV.
___ 5. Determine If Additional NC Cooldown Restrictions Are Required:	
o IF NC temperature decrease has exceeded 100°F in any 60 minutes period during transient, THEN NC cooldown is permitted subject to the following restrictions: 1) Maintain NC pressure AND T-COLD W/R within limits of Enclosure 2. 2) Maintain NC cooldown rate < 50°F in any 60 minute period.	a. IF NC temperature decrease limit has NOT been exceeded, THEN additional cooldown restrictions are NOT required. Return to procedure in effect.
___ 6. Return To Procedure In Effect.	

END

___ A. **Ensure Affected S/G Feedwater Isolation Valves: CLOSED.**

- o S/G 1A:
  - o 1CF-33 (S/G 1A CF Cont Isol)
  - o 1CF-90 (S/G 1A CF Cont Isol Byp)
  - o 1CA-149 (S/G 1A CF Byp To CA Nozzle Isol)
  - o 1CA-185 (S/G 1A CA Nozz Tempering Isol)
- o S/G 1B:
  - o 1CF-42 (S/G 1B CF Cont Isol)
  - o 1CF-89 (S/G 1B CF Cont Isol Byp)
  - o 1CA-150 (S/G 1B CF Byp To CA Nozzle Isol)
  - o 1CA-186 (S/G 1B CA Nozz Tempering Isol)
- o S/G 1C:
  - o 1CF-51 (S/G 1C CF Cont Isol)
  - o 1CF-88 (S/G 1C CF Cont Isol Byp)
  - o 1CA-151 (S/G 1C CF Byp To CA Nozzle Isol)
  - o 1CA-187 (S/G 1C CA Nozz Tempering Isol)

<u>IF</u> required valves <u>CANNOT</u> be closed, <u>THEN</u> ensure following valves for affected S/C: CLOSED

- o S/G 1A:
  - o 1CF-28 (S/G 1A CF Ctrl)
  - o 1CF-30 (S/G 1A CF Byp Ctrl)
- o S/G 1B:
  - o 1CF-37 (S/G 1B CF Ctrl)
  - o 1CF-39 (S/G 1B CF Byp Ctrl)
- o S/G 1C:
  - o 1CF-46 (S/G 1C CF Ctrl)
  - o 1CF-48 (S/G 1C CF Byp Ctrl)

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- o S/G 1D:
  - o 1CF-60 (S/G 1D CF Cont Isol)
  - o 1CF-87 (S/G 1D CF Cont Isol Byp)
  - o 1CA-152 (S/G 1D CF Byp To CA Nozzle Isol)
  - o 1CA-188 (S/G 1D CA Nozz Tempering Isol)

___ B. Isolate CA Flow To Affected S/G:

- o Close the following valves for affected S/G:
  - o S/G 1A:

    1CA-62A (CA Pmp A Disch To S/G 1A Isol)

    1CA-66B (CA Pmp 1 Disch To S/G 1A Isol).
  - o S/G 1B:

    1CA-58A (CA Pmp A Disch To S/G 1B Isol)

    1CA-54B (CA Pmp 1 Disch To S/G 1B Isol).
  - o S/G 1C:

    1CA-50A (CA Pmp 1 Disch To S/G 1C Isol)

    1CA-46B (CA Pmp B Disch To S/G 1C Isol).

- o S/G 1D:
  - o 1CF-55 (S/G 1D CF Ctrl)
  - o 1CF-57 (S/G 1D CF Byp Ctrl)

- o S/G 1D:

  1CA-38A (CA Pmp 1 Disch
  To S/G 1D Isol)

  1CA-42B (CA Pmp B Disch
  To S/G 1D Isol).

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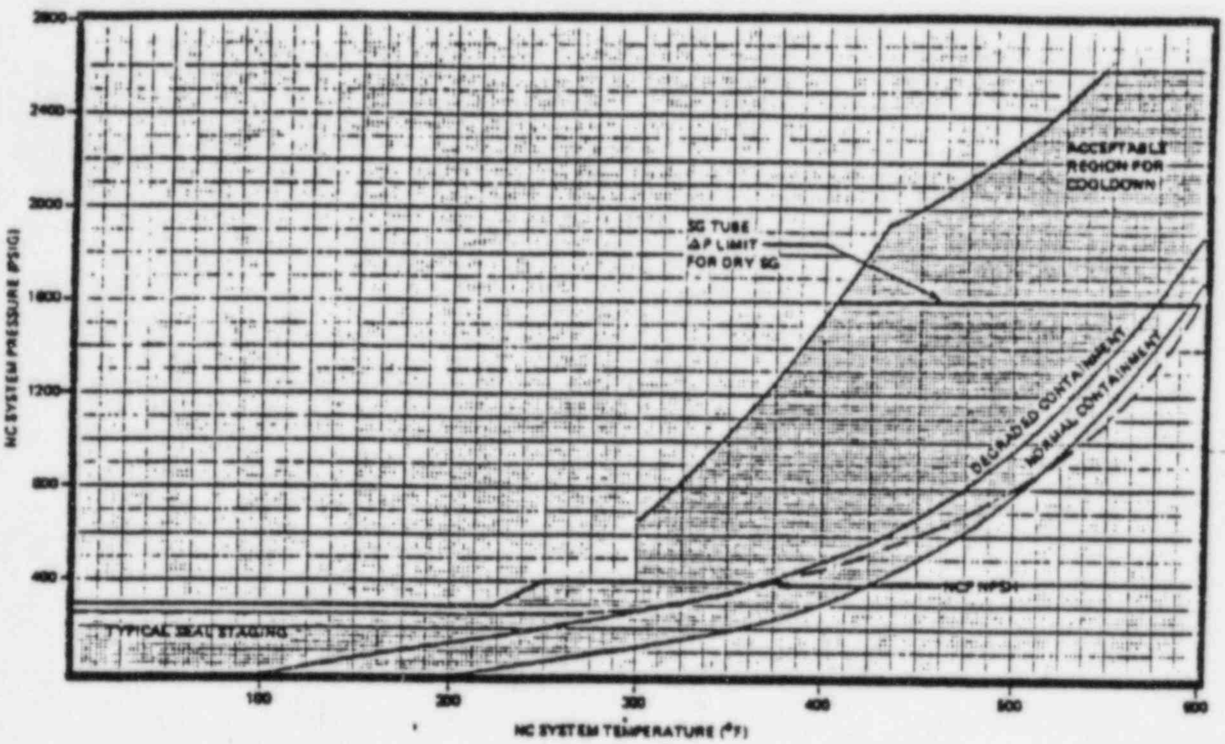

Form 34731 (10-81)
(Formerly SPD-1002-1)

DUKE POWER COMPANY
PROCEDURE PREPARATION
PROCESS RECORD

(1) ID No: RP/0/A/5000/11
Change(s) 0 to
1 Incorporated

(2) STATION: CATAWBA

(3) PROCEDURE TITLE: PROTECTIVE ACTION RECOMMENDATIONS WITHOUT THE OAC

(4) PREPARED BY: Mike Bolch DATE: Aug. 16, 1984

(5) REVIEWED BY: Peter G. LeRoy DATE: 8/20/84

Cross-Disciplinary Review By: [signature] N/R: 8-21-84

(6) TEMPORARY APPROVAL (IF NECESSARY):

By: ______________________ (SRO) Date: ______________

By: ______________________ Date: ______________

(7) APPROVED BY: [signature] Date: 8/21/84

(8) MISCELLANEOUS:

Reviewed/Approved By: ______________ Date: ______________

Reviewed/Approved By: ______________ Date: ______________

2.3 Recommendations

2.3.1 Determine Protective Action Recommendations from Step 1 of Enclosure 4.4.

2.3.2 Determine the affected zone(s) from Step 2 of Enclosure 4.4.

2.3.3 Always include Zone A-0 in Recommendations.

2.3.4 See RP/0/A/5000/05 (General Emergency) for Recommendation Format.

3.0 SUBSEQUENT ACTIONS

3.1 Determine the need for protective actions once every hour if:

3.1.1 The Reactor Building radiation level is > 35 R/hr for > 1 hour, or

3.1.2 EMF-36(L) is > 30,000 cpm for > 1 hour.

4.0 ENCLOSURES

4.1 Clock and Meteorological Data Sheet

4.2 Reactor Building Data - Calculation Sheet

4.3 Unit Vent Data - Calculation Sheet

4.4 Protective Action Recommendation Worksheet

4.5 Limits and Precautions

DUKE POWER COMPANY
CATAWBA NUCLEAR STATION
RP/0/A/5000/11
ENCLOSURE 4.1

CLOCK AND METEOROLOGICAL DATA SHEET

Unit ____________________

Protective Actions Determined By __________________________

1. Clock Data

Time Now ______________________ Date Now ____________________

Time of Reactor Trip ____________ Date of Reactor Trip __________

Hours Since Reactor Trip ________

2. Meteorological Data (from station EEB system or National Weather Service [NWS] at 704-399-6000 if EEB is not available)

Wind Direction - Upper Tower __________ degrees

- Lower Tower __________ degrees

- NWS __________ degrees

NOTE: If wind direction is indicated to be > 360° then subtract 360° and proceed.

Wind Speed - Lower Tower __________ mph

- Upper Tower __________ mph

- NWS __________ mph

Actual ΔT - Upper minus Lower Tower __________ °C

Assumed ΔT - Time now of 1000 to 1600 -0.4°C

- Time now of 1600 to 1000
with wind speed > 15 mph -0.1°C
with wind speed ≤ 15 mph +1.3°C

NOTE: Assumed ΔT is for use when EEB system is inoperable. ΔT is not available from NWS.

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CATAWBA NUCLEAR STATION
RP/0/A/5000/11
ENCLOSURE 4.2

REACTOR BUILDING DATA - CALCULATION SHEET

1. Based upon hours since reactor trip, determine the Reactor Trip Time Factor (RTTF) from the table below and record. ______________

Hours Since Reactor Trip	RTTF
0.0 - 1.0	12
1.1 - 2.0	17
2.1 - 5.0	27
5.1 - 10.0	42
> 10.0	N/A*

* After 10 hrs. TSC will perform dose calculations.

2. Reactor Building Dose Rate (RBDR).

   a) EMF-53A ____________ R/hr.
      EMF-53B ____________ R/hr.

      NOTE: Use the highest EMF reading in calculations.

   b) HP/0/B/1009/06 ____________ R/hr.

3. Calculate Time Determined Dose (TDT).

   TDT ____________ = RBDR ____________ x RTTF ____________

4. Calculate Wind Determined Dose (WDD) based on Wind Speed (WS).

   WDD ____________ = TDT __________ ÷ WS ____________

   NOTE 1: Lower WS is preferred. If not available, use upper WS, then WS from National Weather Service.

   NOTE 2: If WS $\leq$ 1 mph then use the value of <u>1</u>.

5. Go to Enclosure 4.4.

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CATAWBA NUCLEAR STATION
RP/0/A/5000/11
ENCLOSURE 4.3

UNIT VENT DATA - CALCULATION SHEET

1. Unit Vent EMF Readings

   EMF-36(L) = ______________ cpm

   EMF-36(H) = ______________ cpm

   Unit Vent Flow Rate = ______________ cfm

2. Calculate Time Determined Dose (TDT). If EMF-36(H) is < 100 cpm, calculate DT with Section 2.1. If EMF-36(H) is > 100 cpm, calculate DT with Section 2.2.

   2.1 TDT ______________ = EMF-36(L) ________ cpm x ________ cfm x 6.4E-7

   2.2 TDT ______________ = EMF-36(H) ________ cpm x ________ cfm x 4.3E-3

3. Calculate Wind Determined Dose (WDD) based on Wind Speed.

   WDD ___________ = TDT _____________ ÷ WS ___________

   NOTE 1: Lower WS is preferred. If not available, use upper WS, the WS from National Weather Service.

   NOTE 2: If WS $\leq$ 1 mph then use the value of 1.

4. Go to Enclosure 4.4.

DUKE POWER COMPANY
CATAWBA NUCLEAR STATION
RP/0/A/5000/11
ENCLOSURE 4.4

PROTECTIVE ACTION RECOMMENDATION WORK SHEET

1. Based on WDD and ΔT, determine distances and level of protective action from Tables 1.1 and 1.2 below. Circle ΔT, WDD and Protective Action Recommendation.

Table 1.1

0-5 Mile Radius Protective Action Recommendations

	WDD Values		
ΔT: ≤ -0.6	≤ 4.10E6	4.10E6 to 2.00E7	> 2.00E7
-0.6 to -0.5	≤ 1.10E5	1.10E5 to 5.50E5	> 5.50E5
-0.4 to -0.2	≤ 3.50E4	3.50E4 to 1.70E5	> 1.70E5
-0.1 to +0.4	≤ 2.00E4	2.00E4 to 1.00E5	> 1.00E5
+0.5 to +1.2	≤ 9.80E3	9.80E3 to 4.90E4	> 4.90E4
≥ +1.2	≤ 4.50E3	4.50E3 to 2.20E4	> 2.20E4
Protective Action Recommendations	NO ACTION	Consider EVACUATION PARTICULARLY FOR CHILDREN AND PREGNANT WOMEN	EVACUATE EVERYONE

Table 1.2

5-10 Mile Radius Protective Action Recommendations

	WDD Values		
ΔT: ≤ -0.6	≤ 2.00E7	2.00E7 to 1.00E8	> 1.00E8
-0.5 to -0.4	≤ 1.80E6	1.80E6 to 9.20E6	> 9.20E6
-0.4 to -0.2	≤ 4.10E5	4.10E5 to 2.00E6	> 2.00E6
-0.1 to +0.4	≤ 2.00E5	2.00E5 to 1.00E6	> 1.00E6
+0.5 to +1.2	≤ 7.90E4	7.90E4 to 3.90E5	> 3.90E5
≥ +1.2	≤ 2.90E4	2.90E4 to 1.40E5	> 1.40E5
Protective Action Recommendations	NO ACTION	Consider EVACUATION PARTICULARLY FOR CHILDREN AND PREGNANT WOMEN	EVACUATE EVERYONE

DUKE POWER COMPANY
CATAWBA NUCLEAR STATION
RP/0/A/5000/11
ENCLOSURE 4.4

PROTECTIVE ACTION RECOMMENDATION WORK SHEET

2. Based on wind direction (WD), determine the affected zones from the tables below. Circle the wind direction and affected zones.

NOTE: Upper tower wind direction is preferred. If not available, use lower WD, then use WD from National Weather Service.

A. IF WIND SPEED IS < 5 MPH, THE AFFECTED ZONES ARE A-0, A-1, B-1, C-1, D-1, E-1 and F-1.

B. IF WIND SPEED IS > 5 MPH, SELECT THE AFFECTED ZONES FROM THE TABLES BELOW AS APPLICABLE.

Table 2.1

0-5 Mile Radius Wind Direction	Affected Zones
0.1° - 360° →	A-0
PLUS	
0.1° - 22° →	C-1, D-1
22° - 73° →	C-1, D-1, E-1
73° - 108° →	C-1, D-1, E-1, F-1
108° - 120° →	D-1, E-1, F-1
120° - 159° →	E-1, F-1
159° - 207° →	E-1, F-1, A-1
207° - 247° →	F-1, A-1, B-1
247° - 265° →	A-1, B-1
265° - 298° →	A-1, B-1, C-1
298° - 338° →	B-1, C-1
338° - 360° →	B-1, C-1, D-1

Table 2.2

5-10 Mile Radius Wind Direction	Affected Zones
0.1 - 27° →	C-2, D-2
27° - 69° →	C-2, D-2, E-2
69° - 95° →	D-2, E-2, F-2
95° - 132° →	D-2, E-2, F-2, F-3
132° - 144° →	E-2, F-2, F-3
144° - 160° →	E-2, F-2, F-3, A-2
160° - 201° →	F-2, F-3, A-2
201° - 229° →	F-2, F-3, A-2, B-2
229° - 249° →	F-3, A-2, B-2
249° - 259° →	A-2, A-3, B-2
259° - 290° →	A-2, B-2, C-2, A-3
290° - 304° →	A-3, B-2, C-2
304° - 333° →	B-2, C-2
333° - 360° →	B-2, C-2, D-2

DUKE POWER COMPANY
CATAWBA NUCLEAR STATION
RP/0/A/5000/11
ENCLOSURE 4.5

LIMITS AND PRECAUTIONS

1. This procedure is to be used by Control Room Operations personnel only in the event the Operator Aid Computer is not available to perform the calculation of protective action recommendation and the Technical Support Center is not activated.

   NOTE: This procedure is applicable only in the first 10 hours after the Reactor Trip.

2. This procedure is conservative in its ability to protect the public in that:

   a. A 45° wide plume is assumed with an additional 22½° on each side of the plume.

   b. Wind determined dose (WDD) has a built in margin of safety.

   c. There are three sources of meteorological data:

      1) EEB System upper and lower towers
      2) National Weather Service at Charlotte Office of National Weather Service
      3) Established data from CNS FSAR

3. All protective action recommendations relate to child thyroid dose protective action guides.

4. The ratio of I-131 eq. to Xe-133 eq. in the unit vent is assumed to be 9.74E-3.

5. The basis for the unit vent method is HP/0/B/1009/13, Offsite Dose Projection - Uncontrolled Release of Radioactive Material Through the Unit Vent.

6. 6.4E-7 and 4.3E-3 are unitless constants which relate unit vent data to the WDD value tables used to determine protective action recommendations.

Figure 2

DOSE ASSESSMENT

CURRENT TIME	02:01:18P		
CURRENT DATE	01/25/84		
UNIT VENT STACK FLOW		X#	55000 CFM
EMF36L UNIT VENT GAS MONITOR		X#	95000E-03 CPM
EMF36H UNIT VENT GAS MONITOR		X#	90000E-03 CPM
EMF53A CONT HIGH RANGE MONITOR TRAIN A		X#	0 R/HR
AMBIENT AIR D/T ELEV 662 & ELEV 762		X#	-0.1 DEG C
AVERAGE WIND SPEED (MPH)	5.00		
AVERAGE WIND DIRECTION (DEG)	30.00		
SHUTDOWN TIME (HHMM)	01:00:00P		
SHUTDOWN DATE (MMDDYY)	01/25/84		
R B DOSE RATE (R/HR)	6.300E 04		
DELTA T (DEG C)	-0.10		
TIME FACTOR	17.00		
DOSE FACTOR	2.14195E 05		

RANGE (MILES)	ACTION TO TAKE	AFFECTED AREAS
0-5	EVACUATE EVERYONE	A0,A1,B1,C1,D1,E1,F1
5-10	EVACUATE PW & C	C2,D2,E2