NORTHEAST UTILITIES

THE CONNECTICUT LIGHT AND POWER COMPANY
WESTERN MASSACHUSETTS ELECTRIC COMPANY
HOLYOKE WATER POWER COMPANY
NORTHEAST UTILITIES SERVICE COMPANY
NORTHEAST NUCLEAR ENERGY COMPANY

General Offices • Selden Street, Berlin, Connecticut

P.O. BOX 270
HARTFORD, CONNECTICUT 06141-0270
(203) 665-5000

May 24, 1985

Docket No. 50-423
A02959
A04508
A04615
A04752

Director of Nuclear Reactor Regulation
Attn: Mr. B. J. Youngblood, Chief
Licensing Branch No. 1
Division of Licensing
U. S. Nuclear Regulatory Commission
Washington, D. C. 20555

References:

(1) W. G. Counsil letter to B. J. Youngblood, dated April 5, 1984.

(2) B. J. Youngblood letter to W. G. Counsil, dated November 28, 1984.

(3) W. G. Counsil letter to B. J. Youngblood, dated December 7, 1984.

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(7) W. G. Counsil letter to B. J. Youngblood, dated June 14, 1984.

Dear Mr. Youngblood:

Millstone Nuclear Power Station, Unit No. 3
Supplement 1 to NUREG-0737
Safety Parameter Display System

In Reference (1), Northeast Nuclear Energy Company (NNECO) submitted the Safety Analysis Report (SAR) for the Millstone Unit No. 3 Safety Parameter Display System (SPDS). This SAR provided information to the NRC Staff

A003
1/40

demonstrating that our SPDS was being designed to meet the provisions of Supplement 1 to NUREG-0737. Based upon its review of our SPDS SAR, the NRC Staff requested additional information in Reference (2). Our response to the items contained in Reference (2) can be found in Attachment No. 1.

We provided the NRC Staff in Reference (3) with a schedule for our phased implementation approach. This two-phase approach (i.e., Phase I completed by fuel load and Phase II completed by the first refueling outage) represented a revision to the implementation schedule submitted to the NRC Staff in Reference (4). NRC Staff concurrence with our two-phase approach was requested. The NRC Staff responded to our request in a very timely fashion in Reference (5) and concurred with our two-phase approach with the exception of three (3) items. Although no response was requested by the NRC Staff, we have chosen to address these three (3) items in Attachment No. 2.

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To incorporate our responses to Reference (2) and to reflect some minor changes, the SAR submitted in Reference (1) has been revised. The revised SAR is included in Attachment No. 4. Revised sections are identified by change bars. This submittal also fulfills the commitment made in Reference (7) to provide the NRC Staff with the ICC core map display as part of the revised SPDS submittal.

We trust that this submittal adequately addresses the NRC Staff's concerns related to the Millstone Unit No. 3 SPDS identified in References (2), (5), and (6).

Very truly yours,

NORTHEAST NUCLEAR ENERGY COMPANY
et. al.

BY NORTHEAST NUCLEAR ENERGY COMPANY
Their Agent

J. F. Opeka

J. F. Opeka
Senior Vice President

cc: Mr. G. W. Lapinsky, Jr.
NRC Human Factors Engineering Branch

Mr. F. Orr
NRC Procedures & Systems Review Branch

STATE OF CONNECTICUT)
) ss. Berlin
COUNTY OF HARTFORD)

Then personally appeared before me J. F. Opeka, who being duly sworn, did state that he is Senior Vice President of Northeast Nuclear Energy Company, an Applicant herein, that he is authorized to execute and file the foregoing information in the name and on behalf of the Applicants herein and that the statements contained in said information are true and correct to the best of his knowledge and belief.

Lorraine J. D'Amico
Notary Public

My Commission Expires March 31, 1988

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Supplement 1 to NUREG-0737
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Docket No. 50-423

Attachment No. 1

Millstone Nuclear Power Station, Unit No. 3

Response to NRC Staff's Letter Dated November 28, 1984

May, 1985

Response to NRC Staff's Letter Dated November 28, 1984

Instrumentation and Control Systems Information

General:

The Millstone Unit No. 3 SPDS is part of the Plant Process Computer (PPC), which is a control grade system. The PPC is isolated from the Reactor Protection Systems primarily by isolation devices that are classified as protection components. This isolation conforms to General Design Criterion 24 of Appendix A to 10 CFR 50, as discussed in Section 3.1.2.24 of the FSAR, and also complies with the provisions of Regulatory Guide 1.75 and IEEE-384-1974, as discussed in Sections 1.8 and 7.1.2.2.1 of the FSAR. A detailed description of our compliance with Regulatory Guide 1.75 is discussed in FSAR Section 8.3.1.4.

Two types of isolation devices are generally utilized, namely:

1. Analog Isolators

2. Digital Isolators

These isolators are discussed in the FSAR Sections 7.2.1.1.8 and 7.7.2.1. Most of the analog isolators are part of the Westinghouse 7300 Series Process Control System. Other Class 1E instruments that input to the PPC are isolated by Class 1E isolation devices such as the Foxboro N-2A0-VAI. The digital isolators are the Struthers-Dunn CX3917NE Reed Relay Isolation Devices located in various isolation cabinets.

Test results for these isolators are documented in the following Reference Test Reports:

1. Westinghouse WCAP - 8892A - 7300 Series Process Control System Noise Test.

2. Westinghouse WCAP - 8587 Report No. ESE-13 - Environmental and Seismic Test Report for the 7300 Series Process Control System.

3. Foxboro QOAAB44 - Type Test Report for 2A0-VAI Custom (ECEP 9206) Style A CS-N/SRC Voltage to Current Converter.

4. Foxboro QOAAA20-1 Test Report T7-6082 - Seismic Testing of Spec 200 Current Production Model Rack Mounted Modules on N-2ES Style B Rack.

5. Struthers-Dunn Test Report No. 6379 - Final Test Report for Qualification of CS3917NE Reed Relay Isolation Device to IEEE-323-1974 and IEEE-344-1975.

The following responses to the NRC Staff questions pertain to the analog and digital isolators described above. Information regarding isolation between the PPC/SPDS and the Inadequate Core Cooling (ICC) system follows our response to Question 420.12. In addition, we will be prepared to discuss during the forthcoming SPDS audit any other isolation devices.

Question 420.7:

For each type of device used to accomplish electrical isolation, describe the specific testing performed to demonstrate that the device is acceptable for its application(s). This description should include elementary diagrams when necessary to indicate the test configuration and how the maximum credible faults were applied to the devices.

Response:

The analog isolators of the Westinghouse 7300 Series Process Control System were tested with the applications of 580VAC and 250VDC on the output of the isolators. The Foxboro isolators were tested with 600 VAC. Results of the tests show that the voltages applied to the output of the isolator did not affect the input side. Details of the tests are documented in Reference Test Reports #1 and #3.

The digital isolators were tested with the applications of 532VAC, 60HZ, 2000 Amp. short circuit capability and 500 Amp., 134VDC on both the output and input of the isolators to show that burnout of one side of the isolator does not affect the other side. Results of the tests show that the input is adequately isolated from the output. This is documented in Reference Test Report #5.

Question 420.8:

Data to verify that the maximum credible faults applied during the test were the maximum voltage/current to which the device could be exposed, and define how the maximum voltage/current was determined.

Response:

The voltages used in the tests are conservative. The separation criteria at Millstone Unit No. 3 assures that the instrumentation cables are separated from high voltage cables. This is discussed in the FSAR Section 8.3.1.4.

Question 420.9:

Data to verify that the maximum credible fault was applied to the output of the device in the transverse mode (between signal and return) and other faults were considered (i.e., open and short circuits).

Response:

Details of the tests are fully documented in Reference Test Reports #1, #3, and #5.

Question 420.10:

Define the pass/fail acceptance criteria for each type of device.

Response:

The pass/fail acceptance criteria for each type of isolation device require that the input side of the isolation device (safety system side) is not affected by the application of electrical disturbances in the output side.

Question 420.11:

A commitment that the isolation devices comply with the environmental qualifications (10 CFR 50.49) and with seismic qualifications which are the basis for plant licensing.

Response:

The isolation devices are environmentally qualified in accordance with Regulatory Guide 1.89 (IEEE-323-1974) and 10 CFR 50.49, and seismically qualified in accordance with Regulatory Guide 1.100 (IEEE-344-1975). These are documented in Reference Test Reports #2, #4, and #5.

Question 420.12:

A description of the measures taken to protect the safety systems from electrical interference (i.e., Electrostatic Coupling, EMI, Common Mode and Crosstalk) that may be generated by the SPDS.

Response:

To protect the safety systems from electrical interference (i.e., Electrostatic Coupling, EMI, Common Mode and Crosstalk) that may be generated by the SPDS, all instrumentation cables are twisted and shielded. Also, separation criteria are established as described in FSAR Section 8.3.1.4.

Information Regarding Isolation Between the PPC/SPDS and the Inadequate Core Cooling System

The entire Inadequate Core Cooling (ICC) system is Class 1E except for four (4) serial data links to non-Class 1E terminals and to the non-Class 1E PPC. These data links are designed to provide the isolation between Class 1E and non-Class 1E devices. The data links consist of Honeywell Model HFM 5000-003 Fiber Optic RS232C Link Module Kits interconnected with a minimum of two feet of fiber optic cable. The modules connected to the Class 1E side of the link are powered from Class 1E power. The modules connected on the non-Class 1E side are powered from non-Class 1E power. There are no electrical conducting paths between Class 1E and non-Class 1E. The two foot (minimum) length of fiber optic cable provides electrical isolation to well over 3000 volts, as verified by the physical characteristics of the cable material. Approximately seventy-five (75) feet of cable exists between Class 1E and non-Class 1E components at Millstone Unit No. 3.

Seismic qualification was performed along with the entire system, as documented in Energy Incorporated Report No. EI-84-15. These data links are environmentally qualified in accordance with 10 CFR 50.49.

Human Factors Engineering Information

Question 620.1 (Human Factors Program):

Provide a description of the display system, with emphasis on its human factored design, and the methods and results of a human factors program to ensure that the displayed information can be readily perceived and comprehended so as not to mislead the operator. Color photographs or reproductions of display pages and interface devices may be helpful in supporting the discussion.

Response:

Section 7.0, entitled Human Factors Engineering, of our SAR has been revised and is included in Attachment No. 4. A representation of the process computer function keyboard and copies of the SPDS displays can be found at the end of this attachment. Some changes may be made to these displays, as well as the keyboard configuration, prior to fuel load. Three sets of colored displays are being submitted to the NRC Human Factors Engineering Branch. More detailed information can be provided to the NRC Staff during the SPDS audit.

Question 620.2 (Data Validation):

Describe the method used to validate data displayed by the SPDS. Also describe how invalid data is identified to the operator on the displays.

Response:

Section 5.0, entitled Signal Validation, of our SAR has been revised and is included in Attachment No. 4. The quality tags discussed in this section can be seen on the colored displays being submitted to the NRC Human Factors Engineering Branch. The quality tags for each SPDS signal will be shown on the displays as follows:

Quality of Signal	Quality Tag
validated	none
unvalidated	"U" inside a magenta colored box
invalid	"X" inside a magenta colored box
substituted	"S" inside a magenta colored box

Question 620.3 (Verification and Validation Program):

Define and discuss the Verification and Validation Program which was used or will be used in the development of the SPDS. Also, describe results to date from the Verification and Validation Program, and the corrective actions taken to address identified design deficiencies.

Response:

Section 6.0, entitled Verification and Validation, of our SAR has been revised and is included in Attachment No. 4. Many positive benefits have resulted from our verification and validation (V&V) efforts to date. In performing V&V, much more detailed design was done earlier and issues were resolved sooner, resulting in a more stable design at the end of development. In tandem with earlier resolution of issues, problems were identified and corrected earlier. Because of a more stable design towards the end, integration of all the software went much more smoothly than could otherwise have been expected. Also, corrections of many conceptual errors were accomplished before any code was developed.

Errors encountered at the level of the requirements and design documents are identified and corrected and thus are not propagated into the software coding stage. Errors encountered in the test procedure document are corrected to ensure that the test is in compliance with the functional specification.

Our experience has shown that the independent review process has aided in the discovery of potential errors at an early stage, has forced the various sub-system designs and interfaces to be consistent with each other and overall objectives, and has kept the documentation and audit trail in order.

The V&V procedures forced the production of an extensive set of documentation with audit trails. Overall, this has been an excellent tool in defining the total software package and has been an aid in problem solving. Because of the types of documentation produced, the milestones associated with the various documents were excellent project management tools.

Question 620.4 (Unreviewed Safety Questions):

Provide conclusions regarding unreviewed safety questions or changes to technical specifications.

Response:

Since an operating license for Millstone Unit No. 3 has not yet been received, this item is not directly applicable. Unreviewed safety question determinations made pursuant to 10 CFR 50.59 and changes to technical specifications only relate to operating plants. The SAR submitted in Reference (1) represents our safety evaluation for the Millstone Unit No. 3 SPDS and at this time we foresee no technical specifications pertaining to the SPDS.

Question 620.5 (Implementation Plan):

Provide a schedule for full implementation of the SPDS including hardware, software, operator training, procedures and users manuals.

Response:

Our SPDS implementation plan was initially submitted to the NRC Staff in Reference (4). This implementation plan was subsequently revised in Reference (3). The two-phase approach proposed in Reference (3) was approved by the NRC Staff in Reference (5).

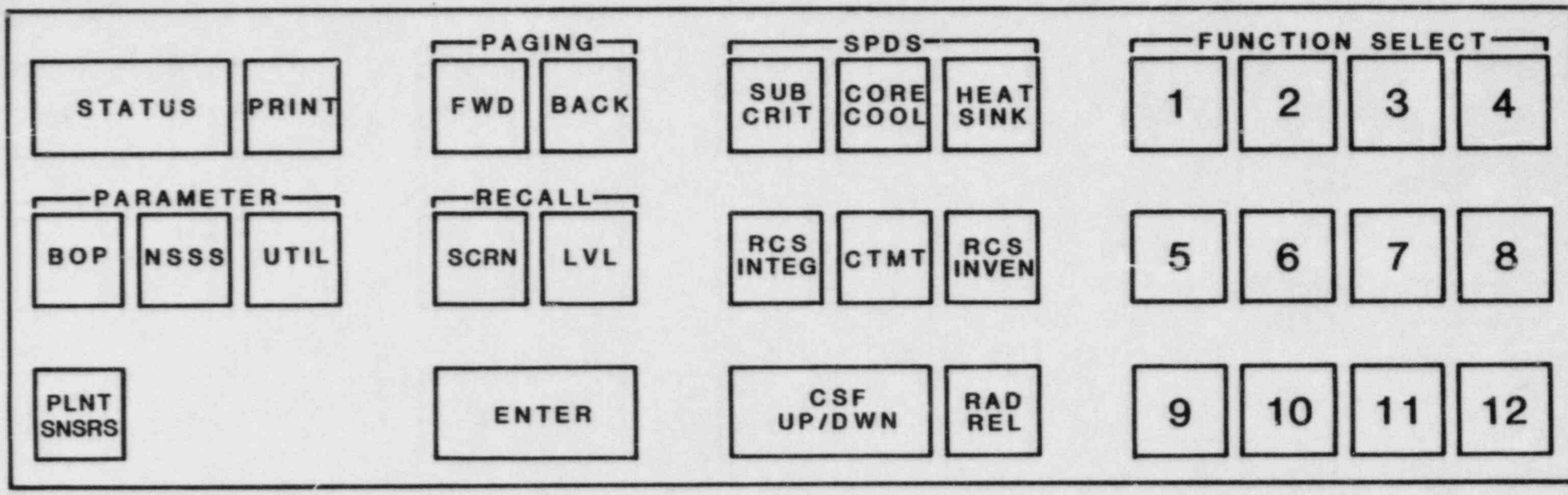


FIGURE 1 PROCESS COMPUTER FUNCTION KEYBOARD

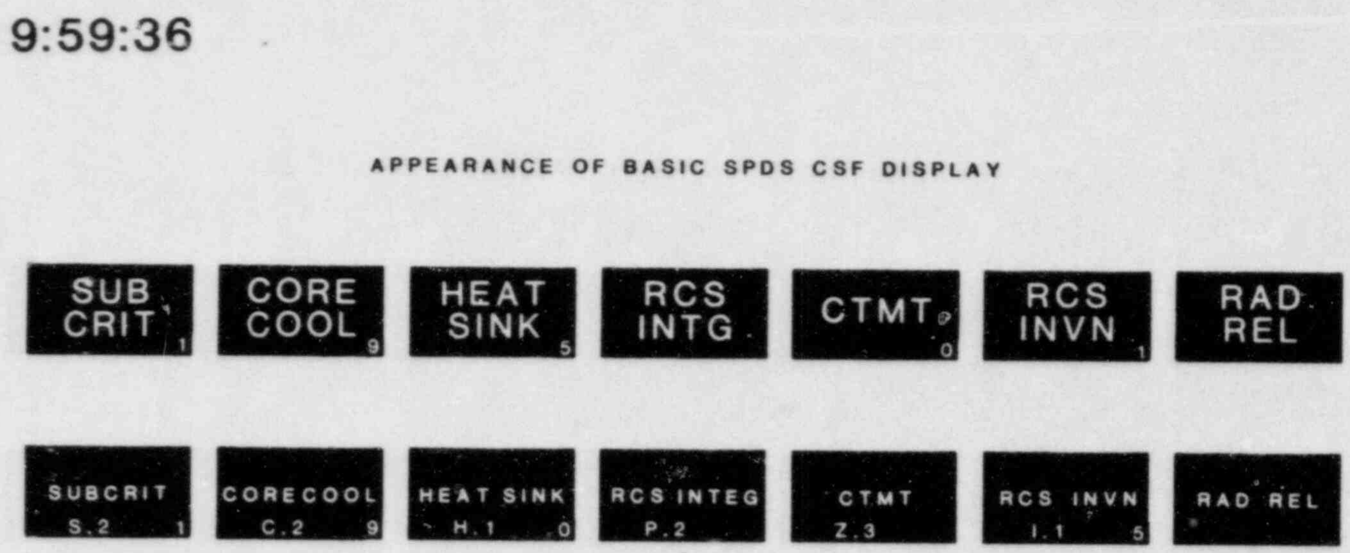

9:59:36
APPEARANCE OF BASIC SPDS CSF DISPLAY
SUB CRIT 1
CORE COOL 9
HEAT SINK 5
RCS INTG
CTMT 0
RCS INVN 1
RAD REL
SUBCRIT S.2 1
CORECOOL C.2 9
HEAT SINK H.1 0
RCS INTEG P.2
CTMT Z.3
RCS INVN I.1 5
RAD REL

FIGURE 2

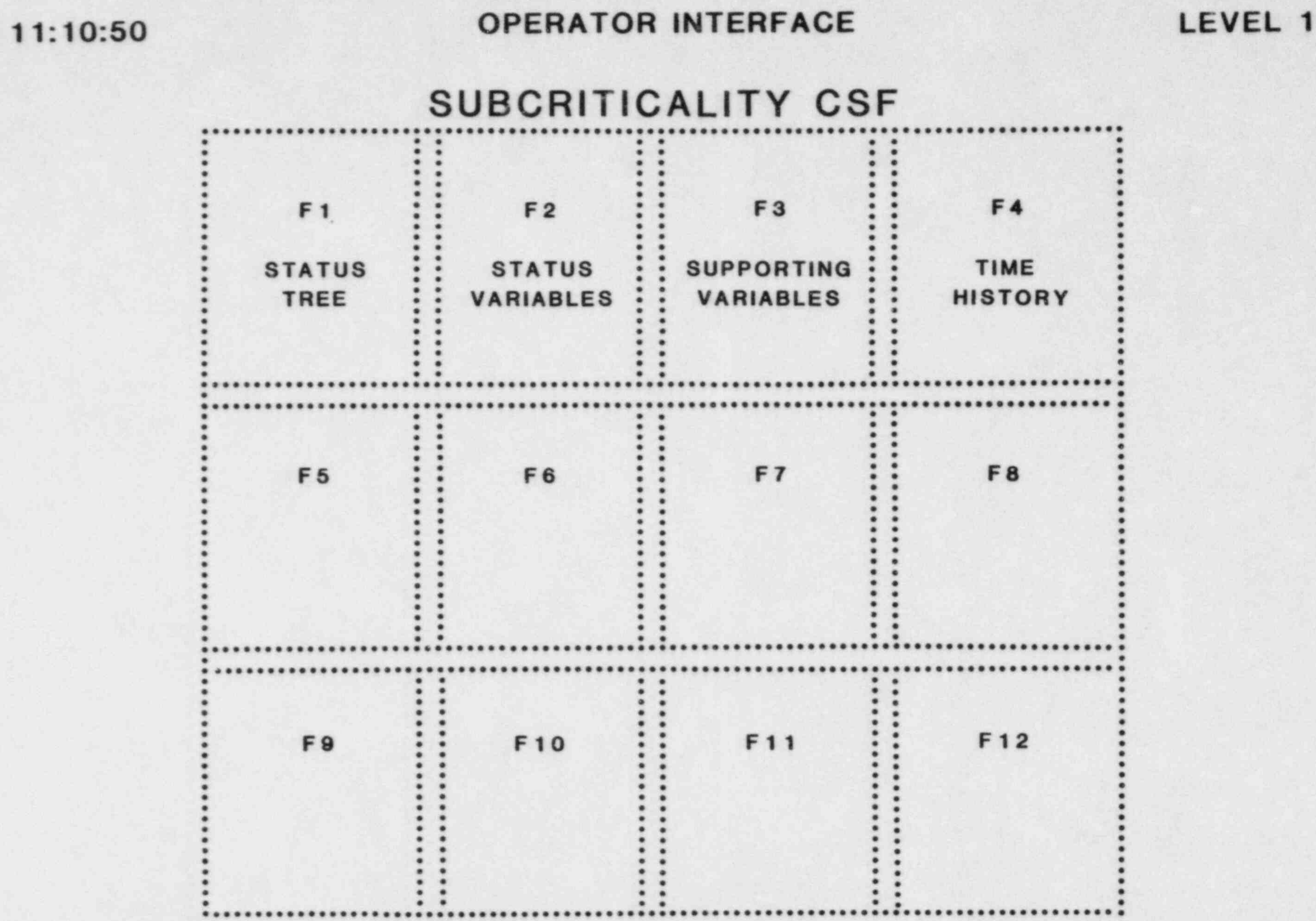

11:10:50
OPERATOR INTERFACE
LEVEL 1
SUBCRITICALITY CSF
F1
STATUS TREE
F2
STATUS VARIABLES
F3
SUPPORTING VARIABLES
F4
TIME HISTORY
F5
F6
F7
F8
F9
F10
F11
F12

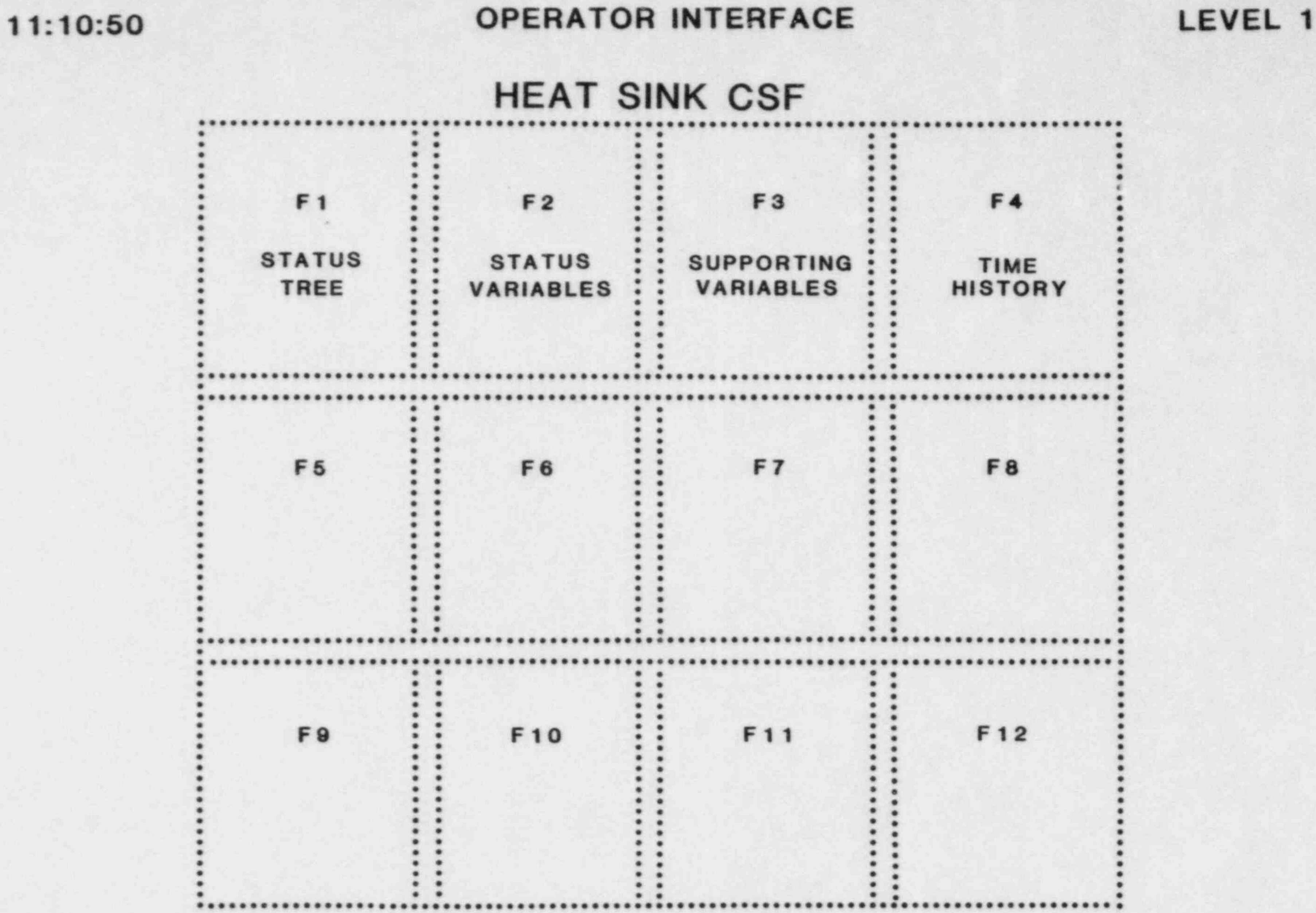
FIGURE 3.0

11:10:50 OPERATOR INTERFACE LEVEL 1

CORE COOLING CSF

F1 STATUS TREE	F2 STATUS VARIABLES	F3 SUPPORTING VARIABLES	F4 TIME HISTORY
F5 PRESSURE-TEMPERATURE PLOT	F6 SUBCOOLING SUPERHEAT	F7 REACTOR VESSEL LEVEL	F8
F9	F10	F11	F12

FIGURE 3.1



11:10:50
OPERATOR INTERFACE
LEVEL 1
HEAT SINK CSF
F1
STATUS TREE
F2
STATUS VARIABLES
F3
SUPPORTING VARIABLES
F4
TIME HISTORY
F5
F6
F7
F8
F9
F10
F11
F12

FIGURE 3.2

11:10:50 OPERATOR INTERFACE LEVEL 1

RCS INTEGRITY CSF

F1 STATUS TREE	F2 STATUS VARIABLES	F3 SUPPORTING VARIABLES	F4 TIME HISTORY
F5	F6	F7	F8
F9	F10	F11	F12

FIGURE 3.3

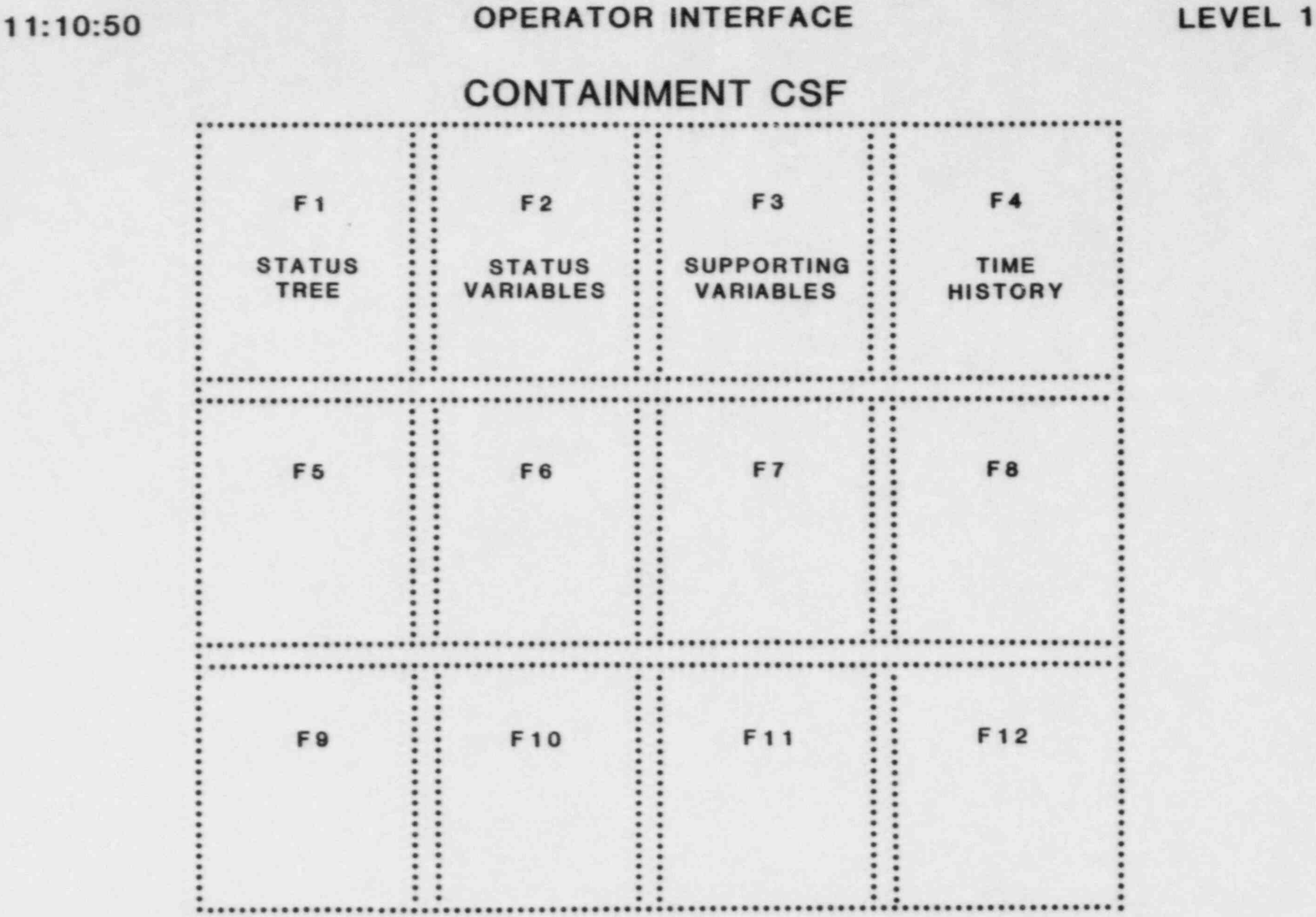

11:10:50
OPERATOR INTERFACE
LEVEL 1
CONTAINMENT CSF
F1
STATUS TREE
F2
STATUS VARIABLES
F3
SUPPORTING VARIABLES
F4
TIME HISTORY
F5
F6
F7
F8
F9
F10
F11
F12

FIGURE 3.4

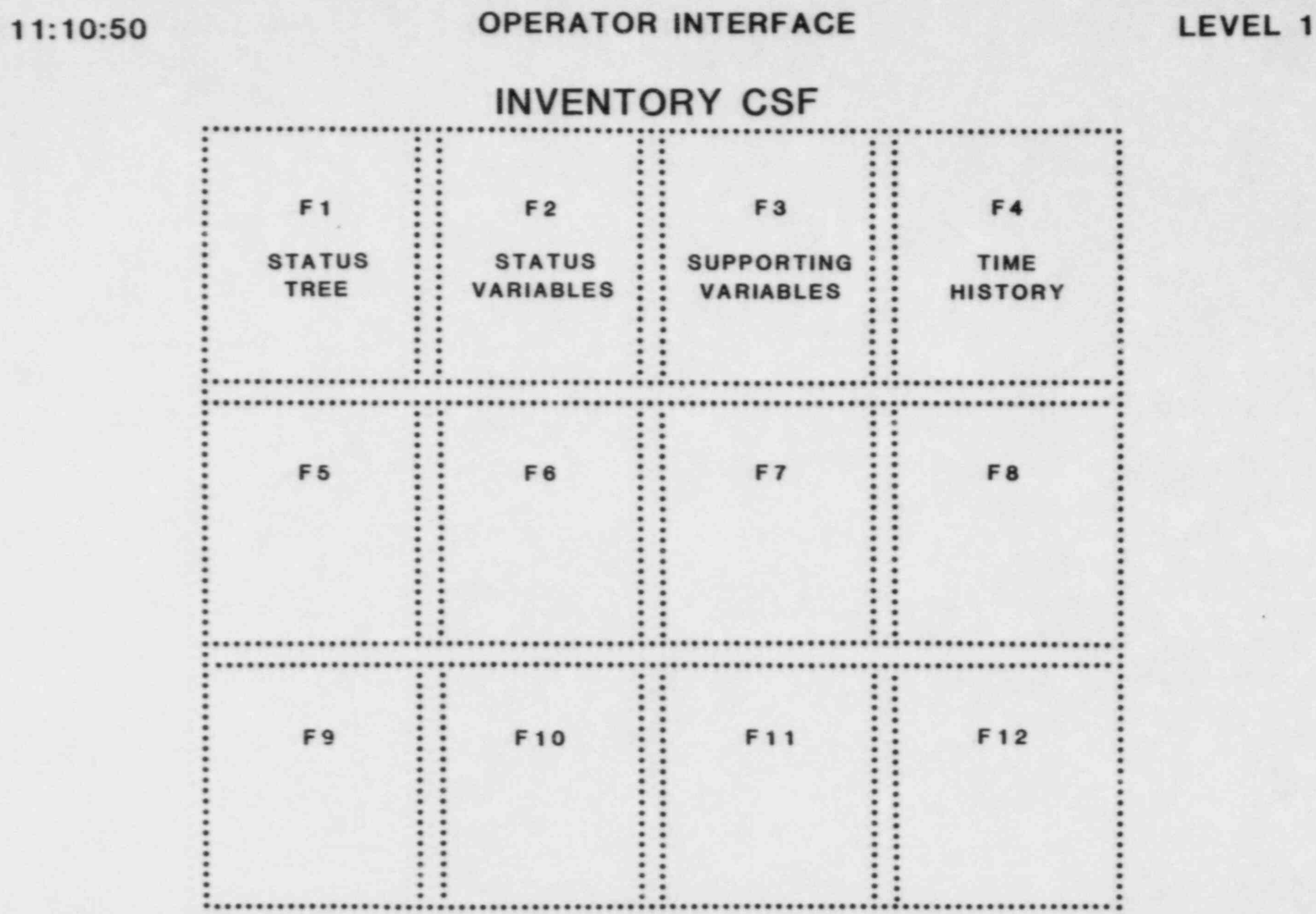

11:10:50
OPERATOR INTERFACE
LEVEL 1
INVENTORY CSF
F1
STATUS TREE
F2
STATUS VARIABLES
F3
SUPPORTING VARIABLES
F4
TIME HISTORY
F5
F6
F7
F8
F9
F10
F11
F12

FIGURE 3.5

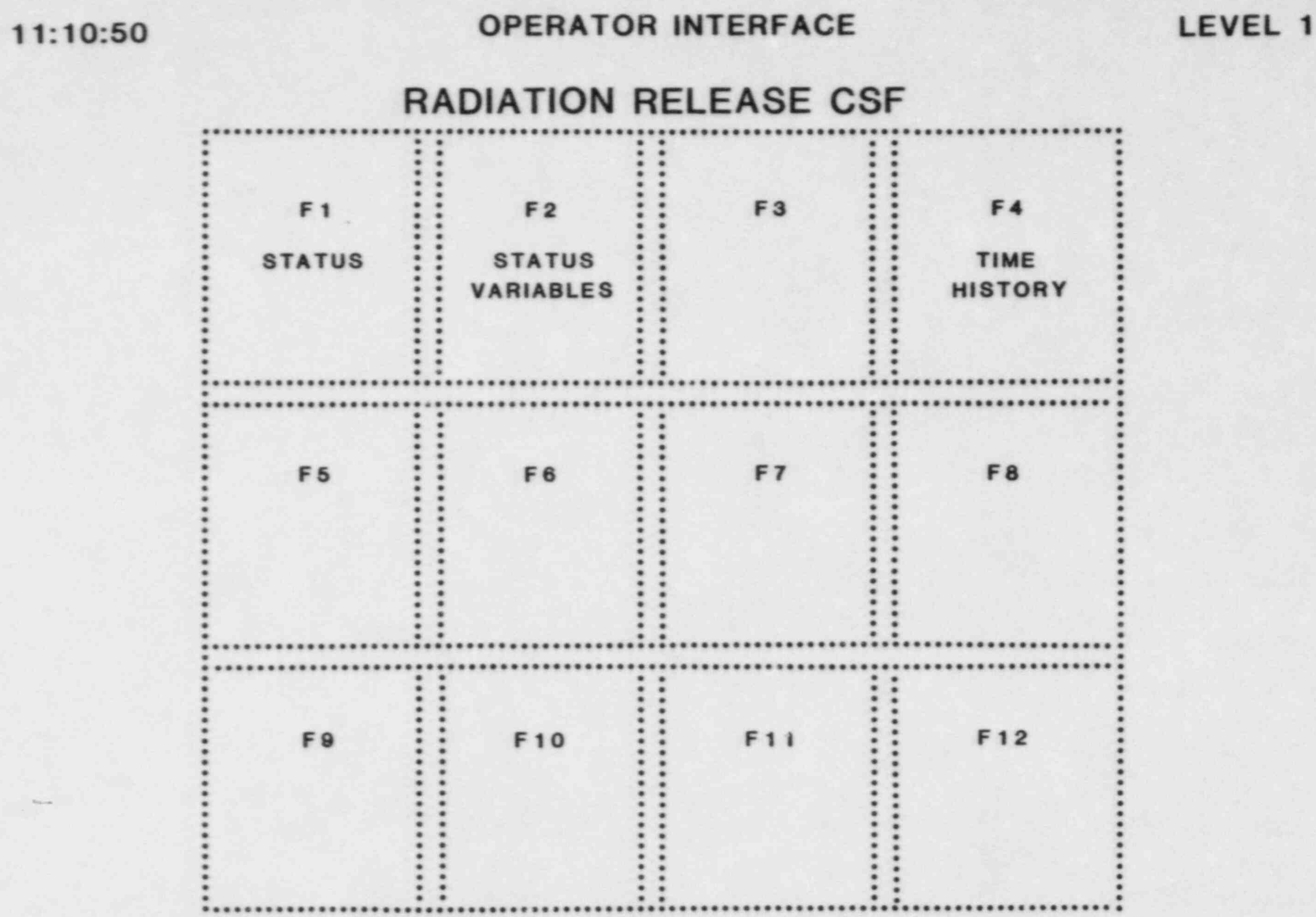

11:10:50
OPERATOR INTERFACE
LEVEL 1
RADIATION RELEASE CSF
F1
STATUS
F2
STATUS
VARIABLES
F3
F4
TIME
HISTORY
F5
F6
F7
F8
F9
F10
F11
F12

FIGURE 3.6

10: 1:36

SUB CRIT

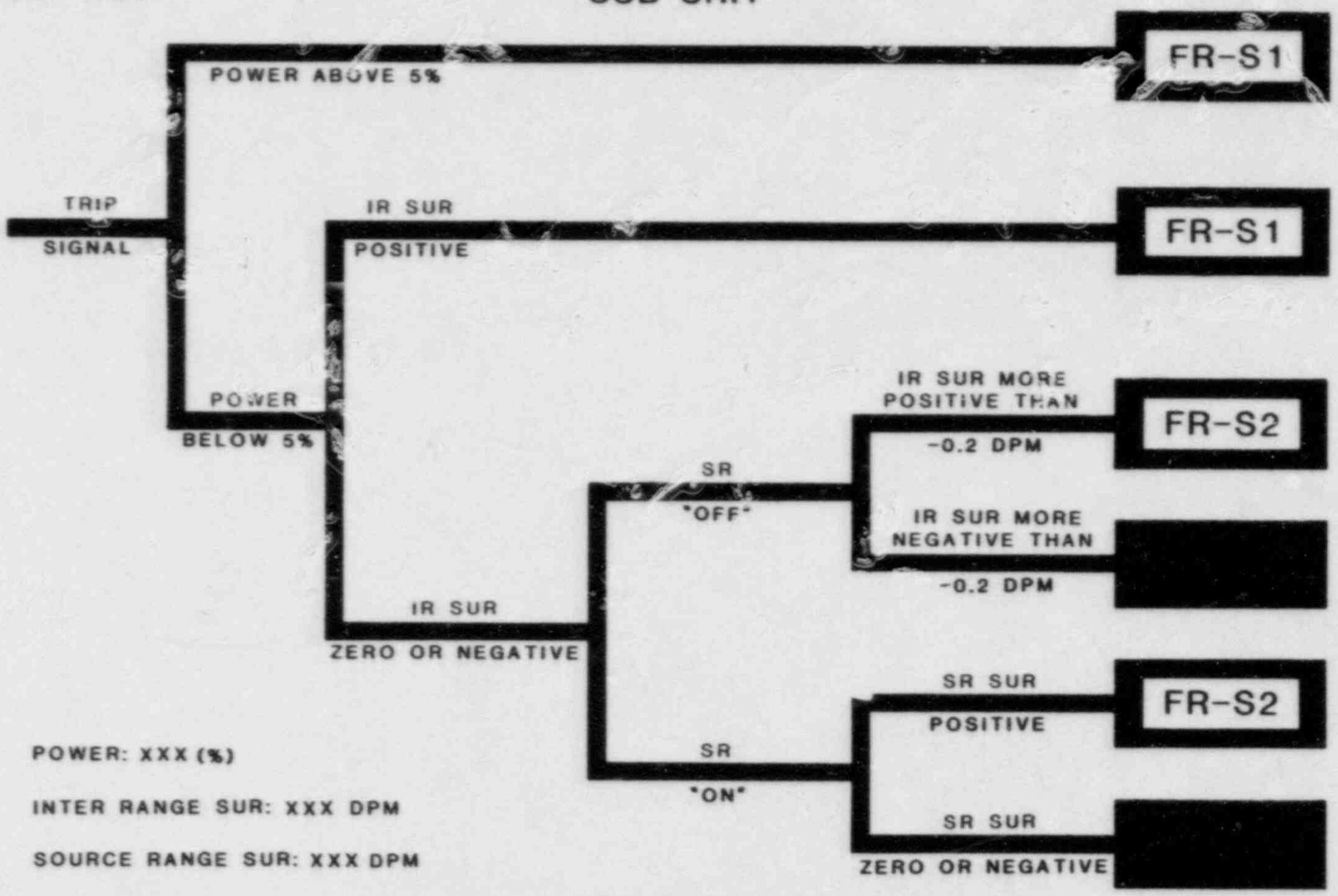


FIGURE 4.0

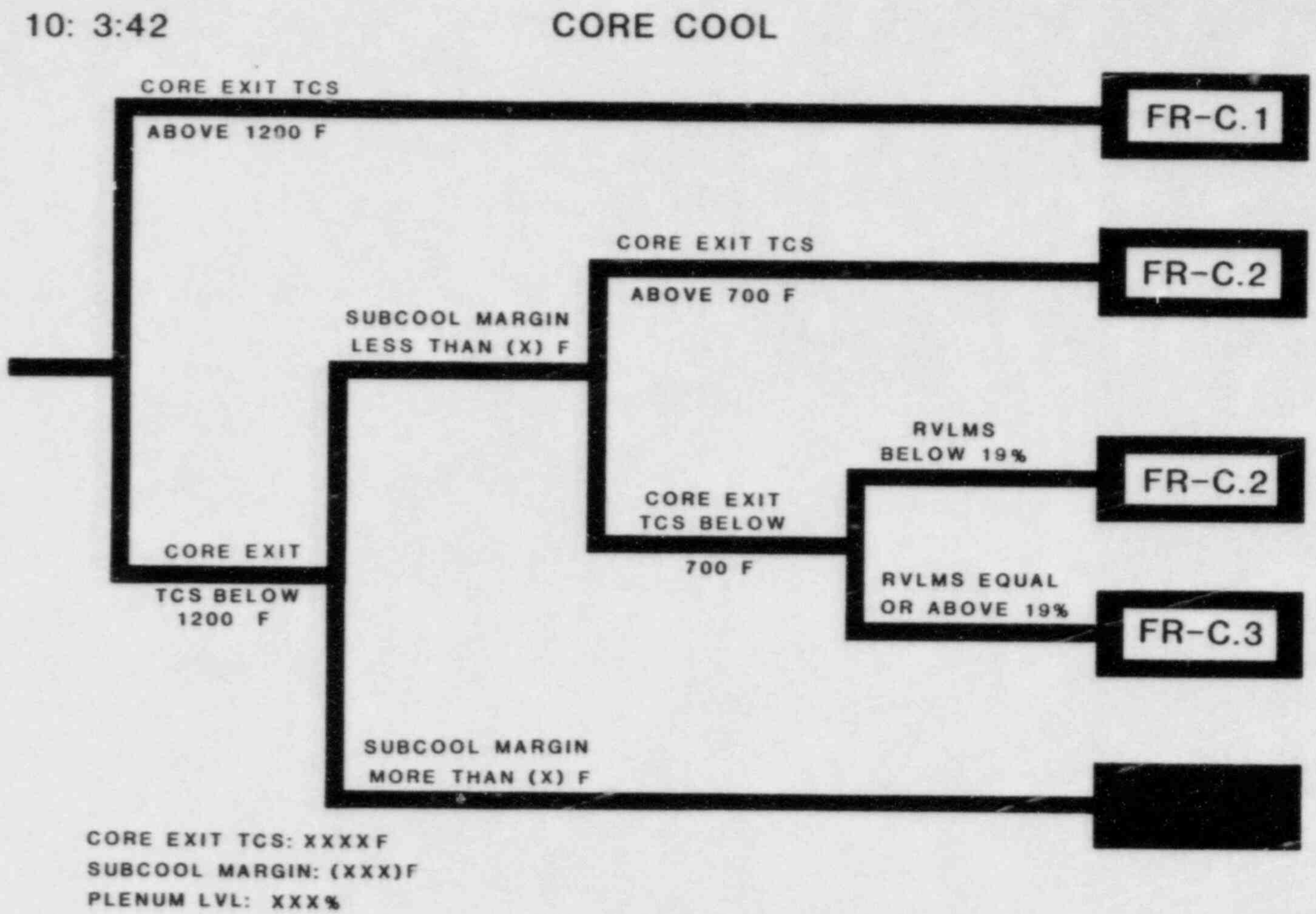

10: 3:42
CORE COOL
CORE EXIT TCS
ABOVE 1200 F
FR-C.1
CORE EXIT TCS
ABOVE 700 F
FR-C.2
SUBCOOL MARGIN
LESS THAN (X) F
RVLMS
BELOW 19%
FR-C.2
CORE EXIT
TCS BELOW
700 F
CORE EXIT
TCS BELOW
1200 F
RVLMS EQUAL
OR ABOVE 19%
FR-C.3
SUBCOOL MARGIN
MORE THAN (X) F
CORE EXIT TCS: XXXXF
SUBCOOL MARGIN: (XXX)F
PLENUM LVL: XXX%

FIGURE 4.1

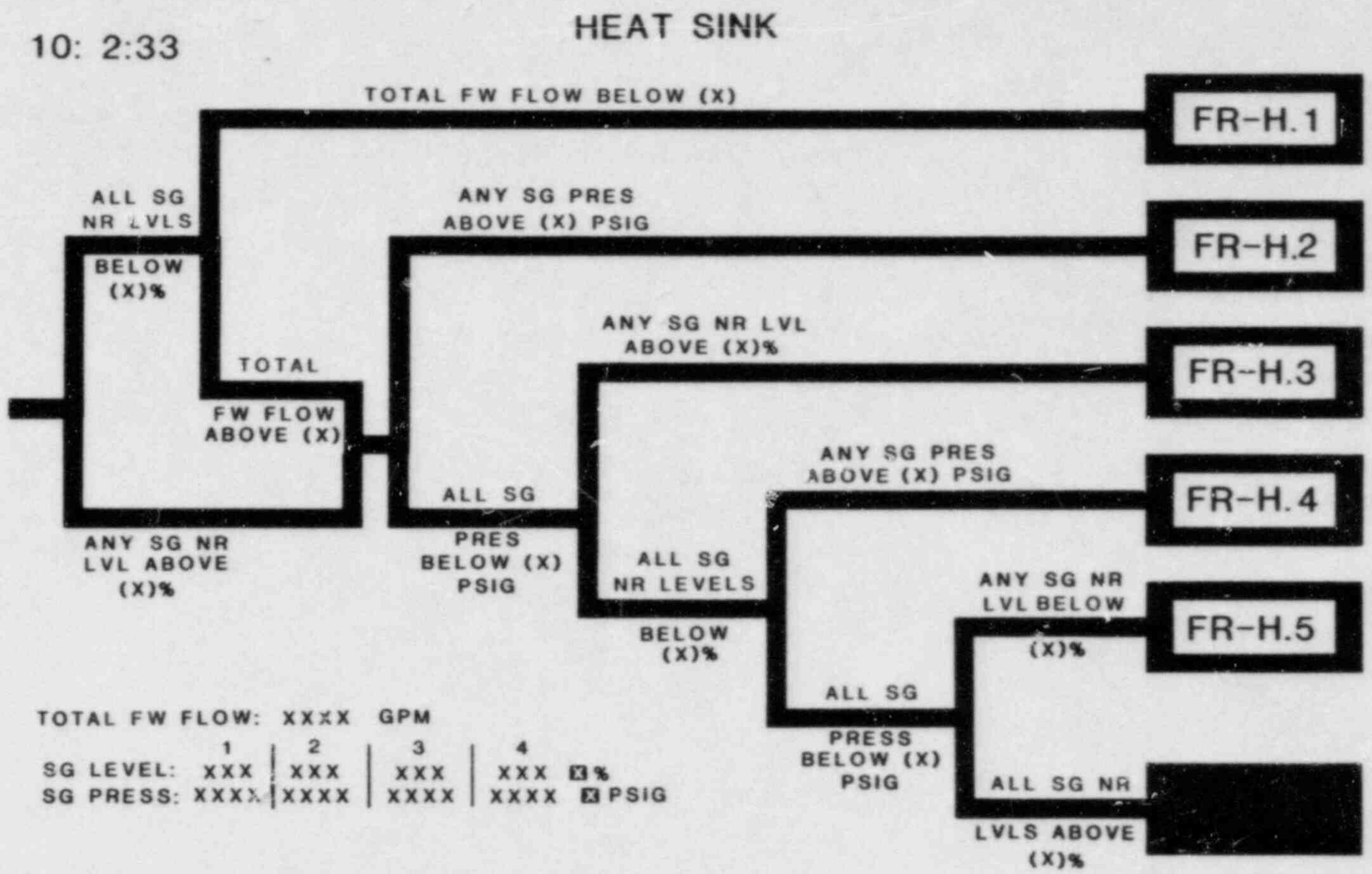

10: 2:33
HEAT SINK
TOTAL FW FLOW BELOW (X)
FR-H.1
ALL SG NR LVLS BELOW (X)%
ANY SG PRES ABOVE (X) PSIG
FR-H.2
ANY SG NR LVL ABOVE (X)%
FR-H.3
TOTAL FW FLOW ABOVE (X)
ANY SG PRES ABOVE (X) PSIG
FR-H.4
ALL SG PRES BELOW (X) PSIG
ANY SG NR LVL ABOVE (X)%
ALL SG NR LEVELS BELOW (X)%
ANY SG NR LVL BELOW (X)%
FR-H.5
ALL SG PRESS BELOW (X) PSIG
ALL SG NR LVLS ABOVE (X)%
TOTAL FW FLOW: XXXX GPM
1 2 3 4
SG LEVEL: XXX XXX XXX XXX %
SG PRESS: XXXX XXXX XXXX XXXX PSIG

FIGURE 4.2

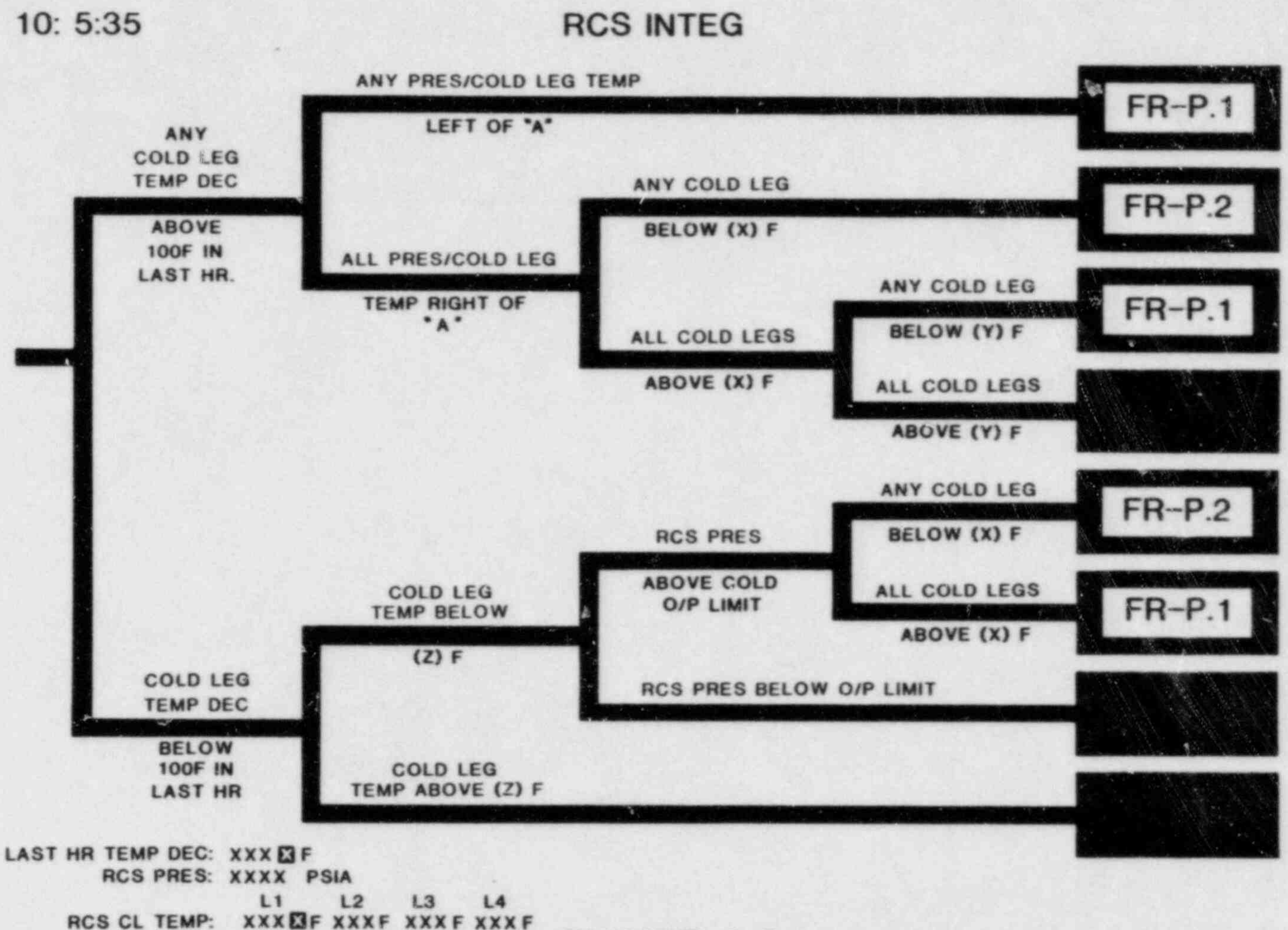

10: 5:35
RCS INTEG
ANY PRES/COLD LEG TEMP
LEFT OF "A"
FR-P.1
ANY COLD LEG TEMP DEC
ABOVE 100F IN LAST HR.
ANY COLD LEG
BELOW (X) F
FR-P.2
ALL PRES/COLD LEG
TEMP RIGHT OF "A"
ANY COLD LEG
BELOW (Y) F
FR-P.1
ALL COLD LEGS
ABOVE (X) F
ALL COLD LEGS
ABOVE (Y) F
ANY COLD LEG
BELOW (X) F
FR-P.2
RCS PRES
ABOVE COLD O/P LIMIT
ALL COLD LEGS
ABOVE (X) F
FR-P.1
COLD LEG TEMP BELOW
(Z) F
RCS PRES BELOW O/P LIMIT
COLD LEG TEMP DEC
BELOW 100F IN LAST HR
COLD LEG TEMP ABOVE (Z) F
LAST HR TEMP DEC: XXX F
RCS PRES: XXXX PSIA
L1 L2 L3 L4
RCS CL TEMP: XXX F XXX F XXX F XXX F

FIGURE 4.3

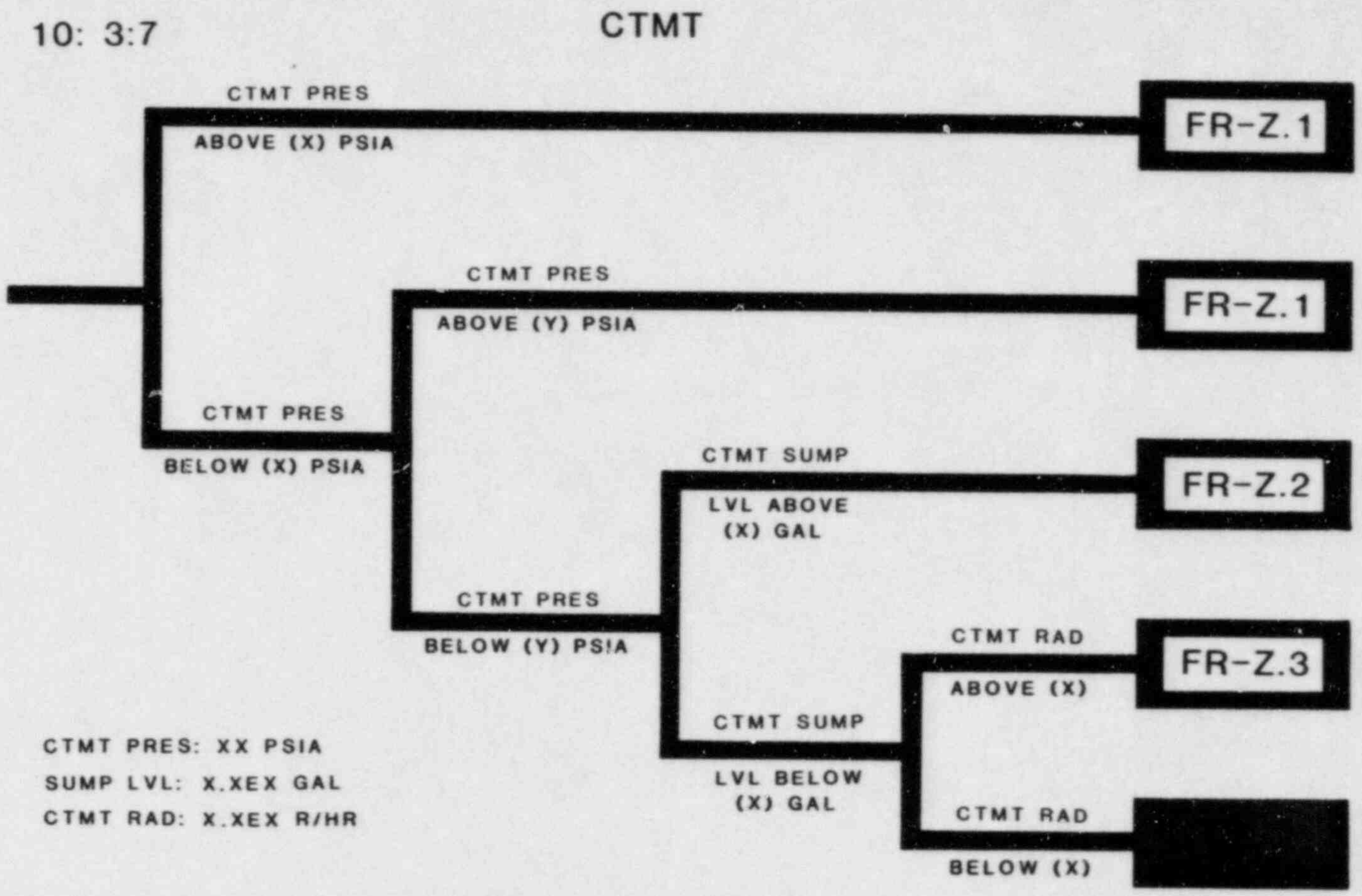

10: 3:7
CTMT
CTMT PRES
ABOVE (X) PSIA
FR-Z.1
CTMT PRES
ABOVE (Y) PSIA
FR-Z.1
CTMT PRES
BELOW (X) PSIA
CTMT SUMP
LVL ABOVE
(X) GAL
FR-Z.2
CTMT PRES
BELOW (Y) PSIA
CTMT RAD
ABOVE (X)
FR-Z.3
CTMT SUMP
LVL BELOW
(X) GAL
CTMT RAD
BELOW (X)
CTMT PRES: XX PSIA
SUMP LVL: X.XEX GAL
CTMT RAD: X.XEX R/HR

FIGURE 4.4

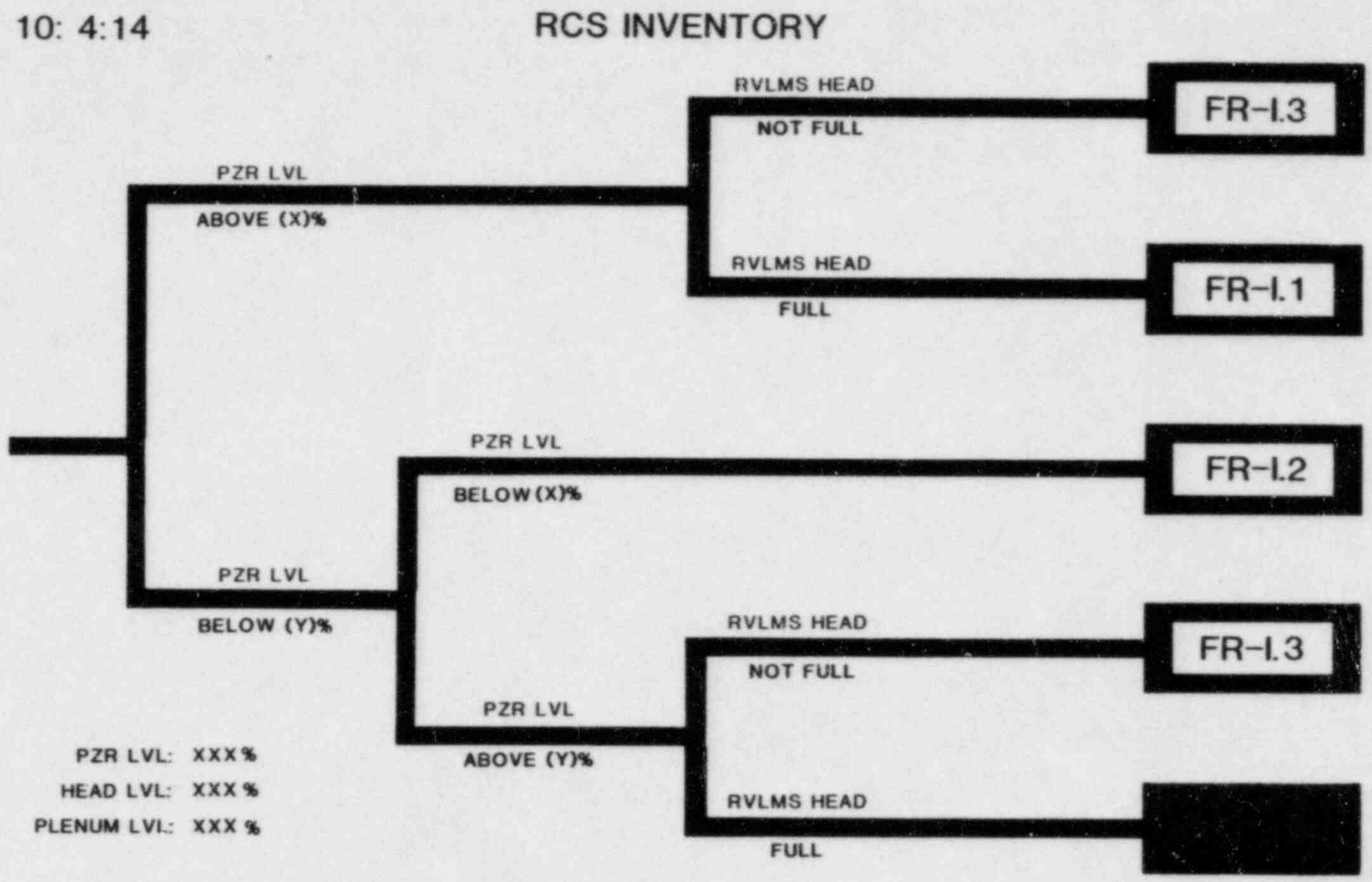

10: 4:14
RCS INVENTORY
RVLMS HEAD
NOT FULL
FR-I.3
PZR LVL
ABOVE (X)%
RVLMS HEAD
FULL
FR-I.1
PZR LVL
BELOW (X)%
FR-I.2
PZR LVL
BELOW (Y)%
RVLMS HEAD
NOT FULL
FR-I.3
PZR LVL
ABOVE (Y)%
RVLMS HEAD
FULL
PZR LVL: XXX %
HEAD LVL: XXX %
PLENUM LVL: XXX %

FIGURE 4.5

10: 4:51 RAD RELEASE STATUS

SENSOR LOCATION	CONDITION	DATA	
VENTILATION VENT NOBLE GAS MONITOR	OVER 0.6 BRAVO	X.XE-X	UC/CC
SLORS NOBLE GAS MONITOR	OVER 5.0 CHARLIE 2	X.XE+X	UC/CC
MAIN STEAMLINE RAD MONITOR	BKND NO. RELEASE	X.XE-X	UC/CC
AUX FEEDPUMP RAD MONITOR	OVER BKGND MINOR	X.XE-X	UC/CC

FIGURE 4.6

10: 6:39

SUBCRIT SENSOR DATA

POWER

DATA	SENSOR
XXX	NMP NM41F
XXX	42F
XXX ☒	43F
XXX	44F

INTERMEDIATE (AMPS)

DATA	SENSOR
X.XE+X	NMI NM35B
XX.XE-X	36B

SOURCE RANGE (CPS)

DATA	SENSOR
X.XE+X	NMS NM 31F
X.XE-X	32F

POWER	SENSOR
OFF	NMS NC 31H
ON	32H

BREAKERS

1st OUT TRIP SIGNAL
PRESENT (ABSENT)

BREAKERS

OPEN (CLOSED)	TBM RX

FISSION COUNTERS

WIDE RANGE (%)

DATA	SENSOR
X.XE+X	(LATER)
X.XE-X	(LATER)

SOURCE RANGE (CPS)

DATA	SENSOR
(LATER)	(LATER)
(LATER)	(LATER)

FIGURE 5.0

10:10: 2

CORE COOLING SENSOR DATA

CORE EXIT TC MAP

LOOP 2 | H15 XXXX | LOOP 3

	C14 XXXX	E14 XXXX	G14 XXXX	J14 XXXX	L14 XXXX	N14 XXXX	
	C12 XXXX	E12 XXXX	G12 XXXX	J12 XXXX	L12 XXXX	N12 XXXX	
A10 XXXX	C10 XXXX	E10 XXXX	G10 XXXX	J10 XXXX	L10 XXXX	N10 XXXX	R10 XXXX
A8 XXXX	C8 XXXX	E8 XXXX	G8 XXXX	J8 XXXX	L8 XXXX	N8 XXXX	R8 XXXX
A6 XXXX	C6 XXXX	E6 XXXX	G6 XXXX	J6 XXXX	L6 XXXX	N6 XXXX	R6 XXXX
	C4 XXXX	E4 XXXX	G4 XXXX	J4 XXXX	L4 XXXX	N4 XXXX	
	C2 XXXX	E2 XXXX	G2 XXXX	J2 XXXX	L2 XXXX	N2 XXXX	

LOOP 1 | H1 XXXX | LOOP 4

	DATA
HIGH TEMP:	XXXX
LOW TEMP:	XXXX

RPV LVL (%)

	TRAIN A	TRAIN B
HEAD	XXX	XXX
PLENUM	XXX	XXX

RCS PRES PSIA

WIDE RANGE

DATA	SENSOR
XXXX	RCS P403
XXXX	P405

NARROW RANGE

DATA	SENSOR
XXXX	RCS P455A
XXXX	P456
XXXX	P457
XXXX	P458

- ■ TEMP BELOW 700F
- ■ TEMP BETWEEN 700 & 1200F
- ■ TEMP ABOVE 1200F

FIGURE 5.1

10: 7:20

HEAT SINK SENSOR DATA

	SG 1		SG 2		SG 3		SG 4	
MAIN FW		FWS						
FLOW (LB/HR)	X.XEX	F510	X.XEX	F520	X.XEX ☒	F530	X.XEX	F540
	X.XEX	511	X.XEX	521	X.XEX	531	X.XEX	541
AUX FW		FWA						
FLOW (GPM)	XXX	F51A	XXX	F33B	XXX	F33C	XXX	F51D
		MSS						
SG PRES	XXXX	P514	XXXX	P524	XXXX	P534	XXXX	P544
(PSIG)	XXXX	515	XXXX	525	XXXX	535	XXXX	545
	XXXX ☒	516	XXXX	526	XXXX	536	XXXX	546
SG LVL		FWS						
	XXX	L517	XXX	L527	XXX	L537	XXX	L547
NR (%)	XXX	518	XXX	528	XXX	538	XXX	548
	XXX	519	XXX	529	XXX	539	XXX	549

FIGURE 5.2

10: 7:65 RCS INTEG SENSOR DATA

COLD LEG

LAST HR TEMP DECREASE: XXX F

TEMPERATURE: (WR, F)

LEG	DATA	SENSOR
1	XXX	RCS T413B
2	XXX	423B
3	XXX	433B
4	XXX	443B

RCS PRES (PSIA)

NARROW RANGE

DATA	SENSOR
XXXX	RCS P455A
XXXX ☒	P456
XXXX	P457
XXXX	P458

WIDE RANGE

DATA	SENSOR
XXXX ☒	RCS P403
XXXX	P405

FIGURE 5.3

10:8:36

RCS INVENTORY SENSOR DATA

PZR LEVEL (%)

DATA	SENSOR
XXX	RCS L459
XXX	460
XXX ☒	461

HEAD LEVEL (%)

DATA	SENSOR
XXX	TRAIN A
XXX	TRAIN B

PLENUM LEVEL (%)

DATA	SENSOR
XXX	TRAIN A
XXX	TRAIN B

FIGURE 5.4

10:12: 3

RAD RELEASE SENSOR DATA

NOBLE GAS:

	DATA	SENSOR
VENTILATION VENT MONITOR:	X.XE-X	HVR 10B
SLORS MONITOR:	X.XE+X	HVR 19B
AUX FEEDPUMP RAD MONITOR:	X.XE-X	MSS 79

MAIN STEAM

	SG1	SVV	SG2	SVV	SG3	SVV	SG4	SVV
SAFETY & DUMP VALVE POSITION	CLOSED	F28A	CLOSED	F28B	OPEN	F28C	CLOSED	F28D
	CLOSED	F29	CLOSED	F29	CLOSED	F29	CLOSED	F29
	OPEN	F30	OPEN	F30	OPEN	F30	CLOSED	F30
	CLOSED	F31	CLOSED	F31	CLOSED	F31	OPEN	F31
	CLOSED	F32	OPEN	F32	OPEN	F32	CLOSED	F32
		MSS						
STEAM LINE RAD MONITORS	X.XE-X	75	X.XE+X	76	X.XE-X	77	X.XE-X	78

FIGURE 5.5

10: 9:20

PRESSURE TEMPERATURE PLOT

ADVERSE CTMT

TIME BASE
5 MIN

TIME BASE

TO GET	PUSH
5 MIN	F9
30 MIN	F10
60 MIN	F11

SENSOR DATA
0-TC XXX F
■-TH XXX F
*-CETC XXX F

TO GET	PUSH
TC	F5
TH	F6
CETC	F7

PRES: XXXX PSIA

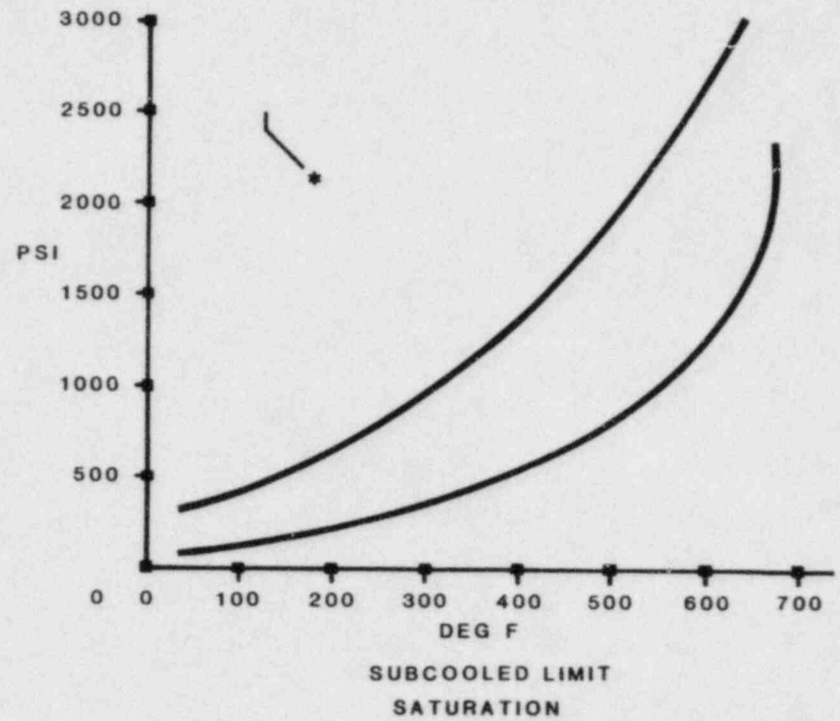


FIGURE 6.0

10:11:21

SUBCOOL–SUPERHEAT

CORE MARGIN CONDITIONS

LOOP 2 | H15 XXX | LOOP 3

	C14 XXX	E14 XXX	G14 XXX	J14 XXX	L14 XXX	N14 XXX	
	C12 XXX	E12 XXX	G12 XXX	J12 XXX	L12 XXX	N12 XXX	
A10 XXX	C10 XXX	E10 XXX	G10 XXX	J10 XXX	L10 XXX	N10 XXX	R10 XXX
A8 XXX	C8 XXX	E8 XXX	G8 XXX	J8 XXX	L8 XXX	N8 XXX	R8 XXX
A6 XXX	C6 XXX	E6 XXX	G6 XXX	J6 XXX	L6 XXX	N6 XXX	R6 XXX
	C4 XXX	E4 XXX	G4 XXX	J4 XXX	L4 XXX	N4 XXX	
	C2 XXX	E2 XXX	G2 XXX	J2 XXX	L2 XXX	N2 XXX	

LOOP 1 | H1 XXX | LOOP 4

ADVERSE CTMT

	MARGIN	SENSOR
BEST CETC:	XXX F	XXX
WORST CETC:	XXX F	XXX
RCS PRES:	XXXX PSIA	

■ ABOVE SATURATION
■ ABOVE SUBCL LIMIT
■ SUBCOOLED

FIGURE 6.1

10:10:43

RPV LEVEL-ICC

LEVEL

	TRAIN A	TRAIN B
HEAD REGION	100 %	100 X %
UPPER PLENUM	XXX %	82 %

TC TEMPERATURE: (F)

		TRAIN A		TRAIN B	
	%	UHJTC TEMP	COV/ UNCOV	UHJTC TEMP	COV/ UNCOV
HD RGN	100	XXXX	C	XXXX	U
	63	XXXX	C	XXXX	C
UPR PLM	100	XXXX U	U	XXXX	C
	82	XXXX	C	XXXX	C
	64	XXXX	C	XXXX	C
	47	XXXX	C	XXXX	C
	32	XXXX	C	XXXX	C
	19	XXXX	C	XXXX	C

FIGURE 6.2

Docket No. 50-423

Attachment No. 2

Millstone Nuclear Power Station, Unit No. 3

Response to NRC Staff's Letter Dated January 18, 1985

May, 1985

Response to NRC Staff's Letter Dated January 18, 1985

Item No. 1:

Phase 1 must include implementation of electrical and electronic isolation suitable to prevent interference with equipment and sensors that are used in safety systems.

Response:

Phase I will include implementation of adequate electrical and electronic isolation. A discussion of such isolation is contained in Attachment No. 1.

Item No. 2:

Phase 1 must include implementation of a parameter set that is acceptable to the staff. The inclusion of necessary parameters cannot be deferred beyond fuel load.

Response:

Specific NRC Staff concerns regarding our parameter selection were provided in Reference (6). Our response to these concerns is included in Attachment No. 3.

Item No. 3

The staff is unsure of the meaning of the Phase II feature described as, "Plant variable information to aid critical safety function assessment and execution of emergency operating procedures..." Based on the teleconference between the staff and Northeast Utilities on December 7, 1984, the staff assumes that this phrase means "information that is helpful to the operator for non-SPDS functions, such as event diagnosis." On that basis the staff concurs that this feature may be deferred to first refueling. If the staff's assumption about the meaning of this phrase is incorrect, the applicant should provide further information to clarify the meaning.

Response:

The NRC Staff's assumption is correct. This Phase II feature is not necessary to provide the SPDS users with adequate information to determine the safety status of the plant. However, some displays may be identified that would be useful primarily to aid the operators in the execution of the emergency operating procedures. As indicated in Reference (3), it may not be necessary to develop any additional displays. As such, the SAR has been appropriately revised.

Docket No. 50-423

Attachment No. 3

Millstone Nuclear Power Station, Unit No. 3

Response to NRC Staff's Letter Dated March 18, 1985

May, 1985

Response to NRC Staff's Letter Dated March 18, 1985

The NRC Staff identified seven (7) variables in Reference (6) that may not have already been included in our SPDS. We were requested to address these variables and their functions by:

(1) adding the recommended variable to the Millstone Unit No. 3 SPDS,

(2) providing alternate added variables along with justifications that these alternates accomplish the same safety functions for all scenarios,

(3) providing justification that variables currently on the Millstone Unit No. 3 SPDS do in fact accomplish the same safety functions for all scenarios, or

(4) identifying that these variables are in fact available from the SPDS console.

It is important to note that the Millstone Unit No. 3 SPDS is part of the plant process computer. As such, SPDS users have direct access to the plant process computer from all SPDS consoles. In addition, our SPDS has been designed with sufficient flexibility to allow for necessary expansion of our SPDS in the future due to revisions to the Westinghouse Owners' Group ERGs or for other reasons.

Each of the seven (7) variables identified by the NRC Staff are discussed below:

1. <u>RHR Flow</u>

We have considered RHR flow for inclusion as a SPDS variable and we have concluded that it is not necessary for critical safety function monitoring. However, RHR flow is available as an input to the plant process computer. Thus, it can be displayed, as required, as part of the plant process computer displays. Since plant process computer displays are accessible from all SPDS consoles, RHR flow is available from any SPDS console. Availability of RHR flow from any SPDS console is consistent with the fourth option identified by the NRC Staff.

The SPDS design philosophy is to provide the minimum, yet complete, set of variables necessary to monitor the critical safety functions. As such, only those parameters that directly monitor the critical safety functions have been included. Secondary variables and variables verifying system operation have not been included.

The variables monitored for core cooling are: reactor coolant system subcooling, core exit temperature and reactor vessel level. These are sufficient to monitor challenges to core cooling. The RHR system is one of a number of systems providing core cooling. The set of core cooling variables, monitored by the SPDS, cover failures of core cooling systems, including RHR. RHR flow is a system verification variable rather than a direct measure of core cooling and as such, need not be included as a SPDS variable.

In addition, the RHR system is primarily used in the later stages of transients and normal operation, corresponding to Modes 5 and 6. As stated in the Safety Analysis Report, Modes 5 and 6 are beyond the scope of the SPDS design.

Since critical safety function status is based upon the trees developed by the Westinghouse Owners' Group, adding RHR flow as a SPDS variable would not alter the status determination scheme. It could be added as part of the SPDS sensor displays. However, if implemented in this manner, it would add very little to the usefulness of the SPDS. It would not provide any more information to the operators by its inclusion in SPDS than by its current availability through the plant process computer.

2. Containment Isolation

We have considered monitoring containment isolation and we have concluded that it is not necessary for critical safety function monitoring. As discussed previously, variables verifying system operation have not been included in SPDS. Containment isolation verification falls into this category.

Upon receipt of a containment isolation signal, the operators will confirm that proper isolation has occurred. This can be easily performed by examining the ESF status panel. When a containment isolation signal occurs, the containment isolation section of the panel will show a lit condition. Any non-lit portions of this section will indicate an improper isolation condition to be corrected by operator action. This indication is backed up by valve position indication on the main control board.

The ESF status panel provides a concise, highly visible indication of the containment isolation status. It is located on one of the main control boards directly across from the SPDS console station, which consists of a plant process computer/SPDS CRT and two keyboards. The ESF status panel is clearly visible and easily monitored at this SPDS console station by the Senior Reactor Operator(s) who are responsible for the overview function.

Because of the number of valves involved in containment isolation, status indication for all the valves is not provided as input for the plant process computer. Even if they were, we believe that the ESF status panel provides as good indication of containment isolation status as would be displayed on the plant process computer. Thus, we have concluded that containment isolation status need not be monitored by the plant process computer or included in our SPDS.

3. Containment Hydrogen Concentration

We have considered containment hydrogen concentration and we have concluded that it need not be included in our SPDS. However, Millstone Unit No. 3 is equipped with hydrogen monitors that are included as input to the plant process computer. Thus, containment hydrogen concentration

can be displayed, as required, from any SPDS console. Availability of containment hydrogen concentration from any SPDS console is consistent with the fourth option identified by the NRC Staff.

Containment hydrogen concentration is an important variable for long-term containment integrity management. However, it is a slowly varying parameter for which sampling is one of the primary means of its determination. Because the analysis from the Post Accident Sampling System (PASS) is one of the primary means of determining hydrogen concentration when required by the EOPs, it is not amenable to monitoring by the SPDS.

Since critical safety function status is based upon the trees developed by the Westinghouse Owners' Group, adding the output from the hydrogen monitors would not alter the status determination scheme. It could be added as part of the SPDS sensor displays. However, it would be unvalidated by the PASS analysis and, if implemented in this manner, would add very little to the usefulness of the SPDS. It would not provide any more information to the operators by its inclusion in SPDS than by its current availability through the plant process computer displays.

As is the case with any variable, hydrogen concentration may be included in the critical safety function status determination scheme by revision of the status trees by the Westinghouse Owners' Group. If this occurs, we will review the changes to determine if the revision should apply to the Millstone Unit No. 3 Emergency Operating Procedures. If a revision is appropriate, the SPDS design will also be revised. However, such a revision will not occur prior to fuel load.

4. Hot Leg Temperature

Hot leg temperature has not been included as an SPDS variable for critical safety function monitoring. However, hot leg temperature is available as an input to the plant process computer and can be displayed, as required, from any SPDS console. Availability of hot leg temperature from any SPDS console is consistent with the fourth option identified by the NRC Staff.

While hot leg temperature is an important parameter for monitoring natural circulation, it is a secondary parameter for monitoring core cooling. Natural circulation is one of several ways of removing decay heat. By monitoring reactor coolant system subcooling, core exit temperature and reactor vessel level, all modes of core cooling are addressed. The set of core cooling variables monitored by the SPDS covers failures, including loss of natural circulation. In keeping with the philosophy of a minimum, yet complete, set of variables for SPDS, hot leg temperature need not be included.

The highest hot leg temperature has been included in the Pressure - Temperature plot for the ICC displays on the SPDS. However, these displays and the associated variables in these displays are not necessarily

related to critical safety function monitoring. As stated in the Safety Analysis Report, these displays were included to resolve a potential Human Engineering Discrepancy associated with the Class 1E ICC displays. The inclusion of hot leg temperature in the ICC display does not affect the core cooling critical safety function status determination.

5. Cold Leg Temperature

Cold leg temperature is a variable monitored by SPDS for the RCS Integrity Critical Safety Function. It is monitored directly for cold over pressurization concerns and indirectly through cooldown rate. It is included in the supplementary SPDS displays that show the sensors used to determine critical safety function status.

6. Intermediate Range Neutron Flux

Intermediate range neutron flux is indirectly used in the critical safety function monitoring of subcriticality as an input to startup rate. As such, it is included in the supplementary SPDS displays of sensor input to the critical safety function status determination.

7. Source Range Neutron Flux

Source range neutron flux is indirectly used in the critical safety function monitoring of subcriticality as an input to startup rate. As such, it is included in the supplementary SPDS display of sensor input to the critical safety function status determination.

In addition, the NRC Staff indicated in Reference (6) that we had not demonstrated how radiation in the secondary system (steam generators and steamlines) is monitored by the SPDS when the steam generators and/or their steamlines are isolated. The main steamline radiation monitors listed in Appendix B to the SAR are located upstream of the main steamline isolation valves and, as such, provide adequate indication of radiation levels in the secondary system even if the isolation valves are closed.

Docket No. 50-423

Attachment No. 4

Millstone Nuclear Power Station, Unit No. 3

Safety Analysis Report

Revision 1

May, 1985

Table of Contents

Table of Contents (Cont.)

1.0 INTRODUCTION

1.1 Summary of the Safety Analysis

This report provides a written safety analysis for the Millstone Unit No. 3 Safety Parameter Display System (SPDS). Information is provided to show that the SPDS is being designed to meet the provisions of Supplement 1 to NUREG-0737.

The SPDS is part of an Emergency Response Information System (ERIS) that combines all plant process computer functions for emergency response tasks. The critical safety functions were selected to be consistent with the Westinghouse Owners' Group Emergency Response Guidelines from which Millstone Unit No. 3 Emergency Operating Procedures (EOPs) are being developed. The SPDS displays are being developed with the consideration of human factors principles. Signals input to SPDS shall be evaluated for quality and validation. A verification and validation program will be conducted, including an independent review of the SPDS.

In this manner, a SPDS design is being developed that will provide an effective aid to the operators in determining the safety status of the plant during abnormal and emergency conditions.

1.2 Discussion

The SPDS is one part of an integrated emergency response capability. It will be consistent with the Emergency Operating Procedures (EOPs) and the Operators' Training Program. For Millstone Unit No. 3, the EOPs will be based upon the Westinghouse Owners' Group Emergency Response Guidelines.

The Emergency Response Guidelines (ERGs) are composed of:

- o Optimal Recovery Guidelines and Emergency Contingencies
- o Critical Safety Function Status Trees and Restoration Guidelines

The Optimal Recovery Guidelines provide guidance for the operator to recover the plant from nominal design basis faulted and upset conditions. The Function Restoration Guidelines, when used with the Critical Safety Function Status Trees, provide a systematic means for addressing any challenge to plant critical safety functions, which is entirely independent of initiating event or plant state.

The structure of the Critical Safety Function Status Trees has been carefully chosen to be compatible with the existing basis for operator training, since the status trees provide an explicit tool to re-emphasize the necessity for the operator to be always aware of the state of his plant safety functions. An additional advantage derived from the introduction of the status tree concept directly into the procedures structure is that the operator is provided with a performance aid, to reinforce his training

and assist his memory, particularly during high-stress situations typical of transient or emergency conditions.

From this discussion of the Critical Safety Function Status Trees and the SPDS, it is clear that they perform the same functions and must be compatible. Thus, the Critical Safety Functions and Variables selection for SPDS has been based upon the Critical Safety Function Status Trees of the Emergency Response Guidelines.

1.3 NRC Criteria

1.3.1 Supplement 1 of NUREG-0737

Regarding the SPDS, Section 4.1 of Supplement 1 to NUREG-0737 identifies the following NRC criteria:

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

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

c. The SPDS shall be suitably isolated from electrical or electronic interference with equipment and sensors that are in use for safety systems. Procedures which describe the timely and correct safety status assessment when the SPDS is and is not available, will be developed by the licensee in parallel with the SPDS. Furthermore, operators should be trained to respond to accident conditions both with and without the SPDS available.

d. The selection of specific information that should be provided for a particular plant shall be based on engineering judgment of individual plant licensees, taking into account the importance of prompt implementation.

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

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

(i) Reactivity control

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

(iii) Reactor coolant system integrity

(iv) Radioactivity control

(v) Containment conditions

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

The remainder of this report defines the extent of compliance of the Millstone Unit No. 3 SPDS with the above NRC criteria.

1.3.2 Regulatory Guide 1.97

The variables needed to determine the status of the Critical Safety Functions (CSFs) and Radioactivity Releases are identical to the majority of the Types A, B, C and E variables of Regulatory Guide 1.97, and therefore these variables will be a major part of the SPDS data base. Type D variables, those needed to assess the performance and availability of safety systems, are not part of the SPDS data base. Although not part of the SPDS, most Type D variables will be part of the plant process computer data base.

The design criteria stated in Regulatory Guide 1.97 for Category 1 sensors infers that a third channel may be required if a failure of one channel results in information ambiguity. The SPDS will have the capability, if necessary, to use techniques such as analytic redundancy, to determine the valid reading and avoid the need to install a third channel.

1.3.3 Generic Letter 82-28 (NUREG-0737 Item II.F.2)

Another designated function of the SPDS is to monitor the overall status of core cooling adequacy. The Class 1E display for Inadequate Core Cooling (ICC) is presently provided in the instrument rack room. To resolve this potential Human Engineering Discrepancy (HED), the primary ICC display will be provided via the SPDS. In the event that the SPDS is not available during accident conditions, the ICC information will still be available on Class 1E qualified devices (ICC panels). As a minimum, the SPDS will include the capability to display the following ICC information.

a. Core map of all core exit thermocouples (CETs).

b. Pressure/Temperature Plots with the saturation curve, subcooling to 300°F, superheat to 45°F.

c. Time history plots of all ICC related variables including reactor vessel level and selected temperature inputs.

d. Water level in the reactor vessel head and upper plenum.

2.0 SPDS DESIGN DESCRIPTION

2.1 Overview

One function of the Millstone Unit No. 3 plant process computer system is the supplying of information required for responses to an emergency condition. Because of the need to develop integrated emergency response facilities and data systems to aid in accident management, an Emergency Response Information System (ERIS) is being developed for Millstone Unit No. 3. This system will display information in the Technical Support Center and Emergency Operations Facility. This report covers only those functions of ERIS related to SPDS.

2.2 SPDS Definition

SPDS aids the control room operating crew in monitoring the status of the CSFs that constitute the basis of the plant-specific, symptom-oriented EOPs. Its principal purpose is to aid the control room personnel during abnormal and emergency conditions in determining the safety status of the plant and in assessing whether abnormal conditions warrant corrective action by operators to avoid a degraded core.

2.3 SPDS Availability

Although the SPDS need not be a safety-grade system, implementation of a highly reliable, state-of-the-art SPDS is an important design objective.

As a design objective, the availability of the SPDS will be greater than 99 percent during normal plant operation. In this context, design availability is understood to encompass the following minimal functional capabilities:

1) The ability to monitor and display the status of all critical safety functions.

2) The ability to determine the value of all variables which are used in the CSF status determination.

2.4 SPDS Use and Location

SPDS displays of CSF status and supporting displays of CSF-related parameters will be accessible to operators in the vicinity of the main control board. SPDS displays that include EOP logic, prompts and algorithm information will be available at the control room location where CSF monitoring will occur.

2.5 Modes of Operation

The CSFs defined for Millstone Unit No. 3 are not appropriate for all modes of operation. Specifically, it is assumed that a status tree is entered from either a Start-up or Power Operation mode and not from a Refueling or Cold Shutdown mode.

The design of the SPDS for Millstone Unit No. 3 therefore only requires the availability of the SPDS in modes 1, 2, 3 and 4 (power operation, startup, hot standby, and hot shutdown).

2.6 Display Flexibility

The SPDS hardware and software will have the capability to display plant information in the following types of common formats, both singly and mixed formats:

Alphanumeric prompts, messages and labels

EOP status trees

Horizontal or vertical bar graphs, if necessary

Mimic/P&ID displays

Multivariable plots vs. time

Variable vs. variable plots

2.7 Data Storage

Capability will be provided to store up to 375 SPDS variables for the interval from two hour pre-event to twelve hours post-event.

2.8 Signal Validation

The SPDS will have the capability of validating individual signals used in SPDS displays and algorithms by use of simple analysis, checking and comparative methods to be specified for each SPDS variable.

2.9 Electric Power Sources

The SPDS, as part of the plant process computer system, will be powered from an emergency power supply in the event of loss of offsite power.

2.10 Electrical Separation

The SPDS, as part of the plant process computer system, will receive signals from both Class 1E and non-1E sources. Adequate electrical separation in accordance with the guidance of Regulatory Guide 1.75 will be provided for all signals, power sources and output devices.

3.0 SPDS CRITICAL SAFETY FUNCTION AND VARIABLE SELECTION

3.1 Selection Process

The SPDS is being designed to complement the EOPs, that is, to aid the operator in implementing the EOPs. It is not intended to require the operator to use the SPDS displays in the transient identifications. The major user of the SPDS during a transient would be the senior reactor operator to "see" the overall plant condition and how actions taken by the operator under his direction affect the maintenance of the six critical safety functions.

The plan for operator response to an Engineered Safeguards System actuation is shown in Figure 1. If the specific event can be diagnosed, the operator is directed to use a defined set of procedural steps to effect plant recovery. If no diagnosis is possible, the operator is trained to monitor certain critical safety functions which indicate overall plant safety status. If any safety function is challenged, the operator is directed to a contingency action through an evaluation and identification scheme of the critical safety functions. To complement this plan, the SPDS can be most effectively used to continuously monitor the critical safety functions and assist the operator in the evaluation scheme to determine the appropriate contingency action. In this manner, the SPDS will be consistent with the W Emergency Response Guidelines.

The W Emergency Response Guidelines have identified the critical safety functions (CSFs) and have developed critical safety function status trees for critical safety function evaluation.

The Critical Safety Functions were selected to monitor three barriers to the release of radioactivity. The Critical Safety Functions are associated with the barriers in the following manner:

Barrier	Critical Safety Function
Fuel Matrix and Fuel Clad	Maintenance of SUBCRITICALITY (minimize energy production in the fuel) Maintenance of CORE COOLING (provide adequate reactor coolant for heat removal from the fuel) Maintenance of a HEAT SINK (provide adequate secondary coolant for heat removal from the fuel) Control of Reactor Coolant INVENTORY (maintain enough reactor coolant for effective heat removal and pressure control)

Reactor Coolant System Pressure Boundary	Maintenance of a HEAT SINK (provide adequate heat removal from the RCS) Maintenance of Reactor Coolant System INTEGRITY (prevent failure of RCS) Control of Reactor Coolant INVENTORY (prevent flooding and loss of pressure control)
Containment Vessel	Maintenance of CONTAINMENT Integrity (prevent failure of containment vessel)

Situations can arise in which the integrity of a barrier is lost and cannot be restored even though all Critical Safety Functions are satisfied. The classic double-ended guillotine break of reactor coolant system piping constitutes an irrevocable failure of the reactor coolant system pressure boundary barrier. In this situation the reactor coolant system pressure boundary barrier is recognized to be failed, and all available resources are directed toward minimizing further degradation of the failed barrier and keeping the fuel matrix/cladding barrier and the containment barrier intact.

The SPDS will be used to assist in the CSF evaluation by monitoring the CSFs, using the same logic as the CSF status trees. This is necessary to facilitate operator use of the SPDS in support of the Millstone Unit No. 3 Emergency Operating Procedures. These status trees are shown in Figures 2a-2f.

The SPDS will also display information for Radioactivity Release. A display summarizing Radioactivity Release has been identified to aid the shift supervisor in performing his emergency response function prior to the staffing of the Emergency Response Facilities. Radioactivity Release is not a critical safety function, however, since radioactivity assessment has already been factored into the containment CSF.

3.2 Critical Safety Functions

The critical safety functions are shown in Table 1 in order of priority. The status of the critical safety function will be indicated by four states:

- o Green - critical safety function is satisfied
- o Yellow - critical safety function is not fully satisfied
- o Orange - critical safety function is under severe challenge
- o Red - critical safety function is in jeopardy

The state of the critical safety functions will be determined using the status tree logic given in Figures 2a-2f.

3.3 Critical Safety Function Variables

The variables for determining critical safety function status will be the decision points in the critical safety function status trees. These variables are listed in Table 2, grouped by safety function.

3.4 Radioactivity Release Function

The status of the radioactivity release function will also be indicated by four states:

- o Green - no abnormal releases
- o Yellow - releases exceed unusual event (Delta-2) criteria
- o Orange - releases exceed alert (Charlie-1) criteria
- o Red - releases exceed site area emergency (Charlie-2) criteria

These states were selected to correspond to the Emergency Action Levels identified in the Millstone Nuclear Power Station Emergency Plan.

3.5 Radioactivity Release Variables

The variables for determining the radioactivity release status were selected by identifying all potential release paths for radioactivity. These variables are listed in Table 3.

3.6 Instrumentation

The instruments used in measuring the critical safety and radioactivity function variables are given in the Appendices A and B.

3.7 Analytical Basis for Critical Safety Function and Variable Selection

The SPDS critical safety functions and variables have been chosen to be identical to the critical safety functions developed for the Emergency Response Guidelines. Thus, the analytical basis for the SPDS selection is the same as the basis for the ERGs. These ERG critical safety function status trees were reviewed and approved for implementation by the NRC in its Safety Evaluation of "Emergency Response Guidelines" (Generic Letter 83-22).

3.8 Emergency Response With and Without SPDS

The Emergency Response Guidelines contain CSF evaluations that are simple enough to allow manual evaluations. This manual evaluation will

be performed if the SPDS is not available. Since the SPDS is entirely compatible with the ERGs, only one set of procedures (EOPs) are required.

4.0 SPDS DISPLAYS

4.1 Display Philosophy

Each display location provides independent access to SPDS displays. Displays selected at one CRT can be different from those displays selected elsewhere. During an emergency, for example, this would allow operators to select SPDS displays that aid process control actions and permit supervisory personnel to simultaneously view SPDS displays oriented toward overview and safety assessment.

In order to maintain CSF status indication at all times, one SPDS display will include indication of the status of each CSF in a format that is common to all SPDS displays. CSF status will be supplemented on each display with a unique set of information and plant data developed to aid one or more of the following:

a. Assessment/Control of CSF plant variables.

b. EOP Entry Condition Indication.

c. CSF Status Tree Assessment.

The set of SPDS displays and access controls will be implemented with a hierarchy or structure that facilitates and systematizes passage between displays.

4.2 Primary Displays

At least one (1) control room CRT will continuously monitor the status of all CSFs during Modes 1, 2, 3 & 4. Other information may be displayed simultaneously as long as the status of the CSFs are still able to be determined. CSF monitoring will include indication of the need to enter a specific Function Recovery Procedure as defined in the ERGs and EOPs.

Each SPDS display will show a common set of indications of the status of the six CSFs and of Radioactivity Release. Status indication colors will correspond to the status colors in the ERGs and EOPs. When any Function Recovery entry condition is met, this will be indicated by the CSF to which it applies. The format for presenting this information will be common to all SPDS displays.

4.3 Secondary Displays

During normal, transient and accident conditions access will be provided to a certain number of predefined displays. These secondary displays will

support the CSF status indicators and enable the operating crew to determine/evaluate the reasons for changes in CSF status and the potential need to enter a Function Recovery Procedure.

The set of secondary displays will consist of at least one display oriented to each of the following functions.

a. Subcriticality CSF Variables and Status Tree.

b. Core Cooling CSF Variables and Status Tree.

c. Heat Sink CSF Variables and Status Tree.

d. Integrity CSF Variables and Status Tree.

e. Containment CSF Variables and Status Tree.

f. Coolant Inventory CSF Variables and Status Tree.

4.4 Other Displays

A set of supporting displays will be generated for displaying other important information such as:

a) Plant variable information to aid CSF assessment and EOP execution, if necessary.

b) Inadequate core cooling variables not included in the primary display of core cooling CSF variables.

4.5 Display Change

Each secondary display will be accessible through a menu.

Once a secondary display is presented on the CRT, other supporting displays can be accessed in a timely manner.

All display page changes will be operator initiated and not computer initiated.

4.6 Variable Status Indication

All SPDS variables will be displayed with a visual indication of the associated quality level as determined by SPDS data processing and validation, e.g., invalid or unvalidated variables could be tagged. Appropriate visual indication will also be available on displays of SPDS variables when out-of-scan, substituted or dummy signals are involved.

5.0 SIGNAL VALIDATION

5.1 Introduction

The use of misleading data by the SPDS should be avoided since it can adversely affect the quality of many variables. Sources of misleading data include sensors that fail, peg, or are removed from scan and instrumentation that drifts. Signal validation techniques will be incorporated into the software processing to reduce the chance of using inappropriate data.

5.2 The Validation Process

Sensor signals used by the SPDS will undergo pass/fail processing, range limit checking and signal validation, as appropriate, before being used in the algorithms which determine the status of the safety functions. The quality of a plant parameter is indicated by its quality tag. The validation process is as described below:

a. Pass/fail processing determines whether or not a sensor signal is in scan, the multiplexor communication interface is operating within design limits, and the analog/digital converter drift is within design limits. A sensor signal failing pass/fail processing is assigned an invalid quality tag.

b. Range limit checking assures that a sensor signal is above the lower five percent (typical value) and below the upper five percent (typical value) of its instrument range. A sensor signal not within the range limit is assigned an unvalidated quality tag.

c. Signal validation determines whether or not a sensor signal is consistent with other redundant signals within a specified error band. A sensor signal failing signal validation is assigned an unvalidated quality tag and one passing is assigned a validated quality tag.

Validated parameters will be used by the SPDS to evaluate the status of the safety functions. Presentation of information for the SPDS will be associated with quality tags which will indicate the quality of the processed sensor signal and the quality of calculated variables. Four distinct quality levels will be used:

Validated - Applies when redundant sensor signals or analytically derived variables are compared within a specified error band, pass limit checking, and pass Pass/Fail.

Unvalidated - Applies when a sensor signal is correctly processed through Pass/Fail but is not validated by comparison with another sensor(s) or analytically derived variables, or fails limit checking.

Invalid - Applies when a sensor signal fails Pass/Fail.

Substituted - Applies when a substituted value is used instead of the actual sensor signal. Substituted values are treated as Invalid by the SPDS algorithms.

The approach to signal validation implemented on the Millstone Unit No. 3 SPDS is based on the parity space concept for fault detection and isolation developed at C.S. Draper Laboratory for nuclear plant applications. The PARITY software module is adapted for use on the Millstone Unit No. 3 plant process computer. The standard use of PARITY is to evaluate each plant parameter based on three to five redundant sensor signals, and to provide a composite best estimate of the parameter along with an indication of the quality of the estimate. Additional software was developed to make non-standard decisions, to revise the quality tag for each inconsistent sensor signal and to estimate parameters having only two redundant sensor signals.

It is believed that the described use of signal validation will provide input to the SPDS that:

a. is purged of inconsistent signals when remaining signals are consistent,

b. is chosen using pre-established decisions if sufficient consistency is lacking, and

c. is tagged to inform the operator of its quality status.

Thus, the process is designed to provide extra reliability and to reduce decision-making-overhead in emergency situations.

6.0 VERIFICATION AND VALIDATION (V&V)

6.1 Verification and Validation Overview

This section provides an overview of the system verification and validation program. The objective of the Verification and Validation (V&V) program is to provide a quality SPDS through independent technical review and evaluation conducted in parallel with SPDS development.

When V&V is integrated with the SPDS development process it provides a means for:

a. independent technical evaluation of the system

b. assuring formally documented implementation

c. improved integration of system hardware and software

d. regulatory review and approval

A team approach will be used for accomplishing V&V. The team composition will be multi-discipline, will include both user, systems and design functions, and will be independent from system development activities. The V&V team will develop and document a V&V plan as one of its initial activities.

6.2 SPDS Verification and Validation

Key overall elements of SPDS V&V will be to assure:

a. Comprehensive technical review of system functional requirements to determine that the SPDS will perform appropriate functions.

b. Comprehensive technical evaluation of the implementation process to establish that tasks are a consistent, complete and correct translation of previous tasks.

c. Adequate documentation of the system, as well as for system implementation.

d. Adequate configuration management to document and control system and implementation changes.

6.2.1 SPDS Design Verification

The objective of SPDS design verification is to review the system functional and design requirements to determine that they are adequate and technically correct, and then to review the following design activities to verify that the translation of requirements is adequate and technically correct throughout the ensuing design steps.

System functional requirements are the foundation on which the SPDS will be designed, built, installed and accepted. The system design will also be validated against the functional requirements. SPDS functional requirements will be verified against the criteria of Supplement 1 to NUREG-0737 and any other criteria that are identified to serve as the basis for SPDS functional definition.

After verification of the functional and design requirements, other design documentation will be verified for accurate and complete translation o. the requirements from various tasks in the design process to the subsequent ones. Verification will include a correlation between the design features and the requirements.

6.2.2 SPDS Validation

SPDS validation will be conducted using a combination of the three levels listed below and will assure that the system meets functional requirements and will aid control room use of EOPs.

a. Factory Testing

SPDS software and hardware may be integrated for functional testing prior to site installation. Testing will be conducted for appropriate hardware, software and system functions in accordance with a systematic test plan.

b. Installation and Acceptance Testing

After SPDS installation in the plant has been completed, functional testing will be performed to demonstrate correct operation of the installed SPDS hardware and software. End-to-end checkouts of all SPDS inputs and outputs will be performed. These checkouts will cover from sensor signal input to SPDS variable display.

c. Man-in-the-Loop Evaluation

Operations personnel, trained in EOPs, will review CSF displays and interface provisions. The objective of this evaluation (not necessarily performed in the control room) will be to review the SPDS design as a potential aid to emergency response by operations personnel.

6.3 Verification and Validation of the Emergency Operating Procedures

Because the SPDS philosophy is to complement the Emergency Operating Procedures (EOPs), verification and validation of the EOPs is an important part of the V&V of the SPDS. Verification and validation has been performed for the Westinghouse ERGs and will be performed for the Millstone Unit No. 3 EOPs that are being developed from these guidelines.

6.3.1 V&V of the Westinghouse Emergency Response Guidelines

The Westinghouse Emergency Response Guidelines were submitted to the NRC Staff in 1982. The NRC reviewed these guidelines and issued a Safety Evaluation Report in June, 1983. The NRC Staff concluded that the guidelines were acceptable. In particular, they concluded that the six Critical Safety Functions were sufficient to protect the three physical barriers. While some areas of additional work on the ERGs were identified, the NRC Staff recommended implementation of the ERGs.

In addition, the Westinghouse Owners' Group developed a Validation Program for the guidelines which included a simulator test program at the Seabrook facility in October, 1983. The simulator test demonstrated that a computer-based status tree evaluation was a highly effective method of critical safety function monitoring.

Because of the extensive verification and validation performed on the ERGs, they represent a sound basis for the development of the Millstone Unit No. 3 SPDS and Emergency Operating Procedures.

6.3.2 Verification of the Millstone Unit No. 3 Emergency Operating Procedures

The Millstone Unit No. 3 Emergency Operating Procedures are being developed based upon the guidance of the Westinghouse Owners' Group Emergency Response Guidelines. To verify consistency between the ERGs and EOPs, a step by step comparison will be jointly made by the Millstone Unit No. 3 operators and the Control Room Design Review (CRDR) team. Justification for any differences will be provided and documented.

6.3.3 Validation of the Millstone Unit No. 3 Emergency Operating Procedures

When the Millstone Unit No. 3 EOPs are developed, a task analysis of the procedures will be performed by the Millstone Unit No. 3 operators and the CRDR team. Actions required to perform the steps in the EOPs will be defined and assessed. The control room operators' ability to efficiently and correctly perform the stated actions considering controls, instrumentation and physical layouts will also be assessed. This task analysis will therefore be used for verification of the selection of the instrumentation to be used for the critical safety function variables.

The Millstone Unit No. 3 Emergency Operating Procedures will be validated upon completion. A description of the Emergency Operating Procedure Validation Program will be submitted as part of the Emergency Operating Procedures Generation Package.

7.0 HUMAN FACTORS ENGINEERING

7.1 Human Factors Engineering

The fundamental SPDS design objective is to serve as an operator aid to monitor the overall safety status of the plant. Human factors considerations must be an integral part of a program to successfully develop such a system.

This section describes the role of the primary SPDS user, the context of use, and the human factors principles that will be incorporated into the SPDS design.

7.1.1 SPDS Use

The Millstone Unit No. 3 control room staff includes four licensed operators (i.e., two Senior Reactor Operators (SROs) and two Reactor Operators (ROs)). One of the SROs will be the Shift Supervisor (SS). The SS/SRO will be the primary SPDS user. The SPDS is intended to help the SS/SRO in managing the plant during unusual situations where problem detection and problem solving on a plant wide scale are involved. The major role of the SPDS is to help the operating crew maintain the plant in a safe condition or to show how to return the plant to a safe condition if it has departed from normality.

The present control room and the resources available to the SS/SRO are sufficient to carry out these tasks. The SPDS is intended as an aid to the SS/SRO, not as a replacement for necessary safety equipment. The SPDS serves as a concentrated data source and thus permits the SS/SRO to obtain desired information without walking the boards to check readings.

The role of the SS/SRO is as a decision maker and manager of the plant. The role of ROs and the other SRO is to assist the SS/SRO by carrying out the tasks deemed necessary by the SS/SRO. Although ROs are carrying out specific tasks such as maintaining levels, starting pumps, or checking instrument readings, they need to be cognizant of the impact their operations have on overall plant condition. SPDS displays will be accessible to RO personnel to help maintain the needed understanding of the overall picture and to foster a team approach to plant emergency response.

7.1.2 Control Room Design

The arrangement and number of SPDS display stations in the control room will provide separate SPDS stations for the SS/SRO (away from the boards) and for operators (visible from operating stations at the boards). This arrangement will provide the SS/SRO with a good view of the SPDS from his work station (he can see both the SPDS and the boards at the same time) and by the operators from their stations at the boards. Thus the arrangement will permit a flexible use pattern which is weighted towards the needs of the SS/SRO while still permitting RO use.

7.2 Human Factors Design Guidelines

The following is a discussion of the human factors activities to be accomplished during the development of the SPDS computer generated displays.

7.2.1 Task Definition

This activity is designed to acquaint the designer with the reasoning behind the display requirements and to give him a feel for how and when the displays will be used. The designer determines how each task is presently performed, the information needed to accomplish it, and how the display can assist in plant performance.

7.2.2 Determine Equipment Considerations

The purpose of this activity is to assure that any limitations which may be imposed by the equipment are known to the display designer. For example, the designer needs to determine the amount of information that will fit on one CRT screen, colors available, controls, brightness, etc.

7.2.3 Determine Viewing Environment

The purpose of this activity is to become familiar with the location and environment in which the equipment is to be used. It is also necessary to determine the positions (e.g., standing, sitting, viewing distances) from which the user will want to read the information on the displays.

7.2.4 Determination of Human Factors Criteria

This activity is to obtain a definition of existing human factors criteria that apply to the specific environmental conditions or display features. Most of the criteria utilized for CRT displays can be found in Section 6.7.2 of NUREG-0700 (Cathode Ray Tube Displays).

7.2.5 Develop Display Concept

The display concept will be developed to give the display designer an overall idea of how he is going to accomplish the total task, how many displays will be used and how each one fits into the total picture. It will enable the design to be in accordance with user capabilities so that the resulting displays mesh with user needs. In general, the designer will develop the following information:

a. Identify user needs
b. How many displays are needed
c. Define the task to be accomplished with each display
d. How they should be set up (hierarchy)
e. How the displays are to be accessed
f. How any required data is to be entered
g. How the user can recover from any errors

h. Define user capabilities (e.g., a newly licensed operator)
i. Develop a prompt philosophy based on operator capabilities

7.2.6 Design Review

The purpose of this activity is to insure that the overall plan for display design is satisfactory. This is also another control point in the design process. It permits the designer to be sure that his product is going to meet all requirements when it is completed.

7.2.7 Develop Displays

This is the actual design of the displays. All of the activities above are designed to get the designer to this point with enough knowledge of user needs, equipment capabilities, and the environmental constraints so that the resulting product is compatible with all requirements. In general, the following activities are performed as part of this process:

a. Determine how the needed information is to be shown.
b. Determine the appearance of each display element.
c. Determine the colors to be used.
d. Determine the dynamics of each variable element.
e. Determine access to each display.
f. Determine how the user can recover from errors.
g. Determine what prompts are to be used and where.

7.2.8 Display Review

The purpose of this step is to insure that the detailed design meets all the original requirements. An important step in this process is a review of the displays by typical users (i.e., plant operators).

7.2.9 Issue System Specification

This is the final control point for the display design before its release for implementation. It also provides clear guidance to programming personnel regarding the final product.

8.0 CONCLUSION

The SPDS for Millstone Unit No. 3 is being designed to adequately address the provisions of Supplement 1 to NUREG-0737. Specifically:

a) The SPDS will provide a concise display of critical plant variables to aid the control room operators in determining the safety status of the plant that is consistent with the Westinghouse Emergency Response Guidelines and the Millstone Unit No. 3 Emergency Operating Procedures.

b) The SPDS will display CSF information on colorgraphic terminals located in the control room. The SPDS will display the status of the CSFs continuously. The SPDS will be part of the plant process computer system and is being designed to meet availability considerations consistent with the SPDS function.

c) Since the SPDS will be completely consistent with the Westinghouse Emergency Response Guidelines, only one set of procedures is required for emergency response with and without the SPDS. Adequate electrical separation will be provided in accordance with the guidelines of Regulatory Guide 1.75.

d) The critical safety functions and variables have been selected to be consistent with the analytical basis of the Emergency Response Guidelines. In general, the Regulatory Guide 1.97 instruments will be the source of the variables.

e) The SPDS displays are being designed to meet human factors principles.

f) The SPDS provides information about:

    (1) reactivity control

    (2) core cooling and heat removal

    (3) RCS integrity

    (4) radioactivity control

    (5) containment conditions

This safety analysis shows that the SPDS will be consistent with Emergency Response Guidelines and the Millstone Unit No. 3 Emergency Operating Procedures, and provides an integrated approach to abnormal and emergency conditions. Human factors principles are being considered in the design to assure that the operators can use the SPDS effectively. A Verification and Validation Program will assure that independent reviews are conducted to assure proper implementation of the SPDS design.

The development of the SPDS will be an effective aid for the control room operators to determine the safety status of the plant during abnormal and emergency conditions.

TABLES

TABLE 1: Critical Safety Functions

I.	Subcriticality	Highest Priority ↑
II.	Core Cooling	
III.	Heat Sink	
IV.	Integrity	
V.	Containment	
VI.	Inventory	↓ Lowest Priority

TABLE 2: Critical Safety Function Variables

SAFETY FUNCTION		VARIABLE	
I.	Subcriticality	1.	Reactor trip signal
		2.	Power level
		3.	Startup rate
		4.	Source range energized
II.	Core Cooling	1.	Core exit temperature
		2.	RCS subcooling
		3.	RV level
III.	Heat Sink	1.	S/G level
		2.	Total FW flow rate
		3.	S/G pressure
IV.	Integrity	1.	Cooldown Rate
		2.	RCS temperature
		3.	RCS pressure
V.	Containment	1.	Containment pressure
		2.	Containment level
		3.	Containment radiation
VI.	RCS Inventory	1.	Pressurizer level
		2.	Reactor vessel level

TABLE 3: Variables for Radioactivity Release

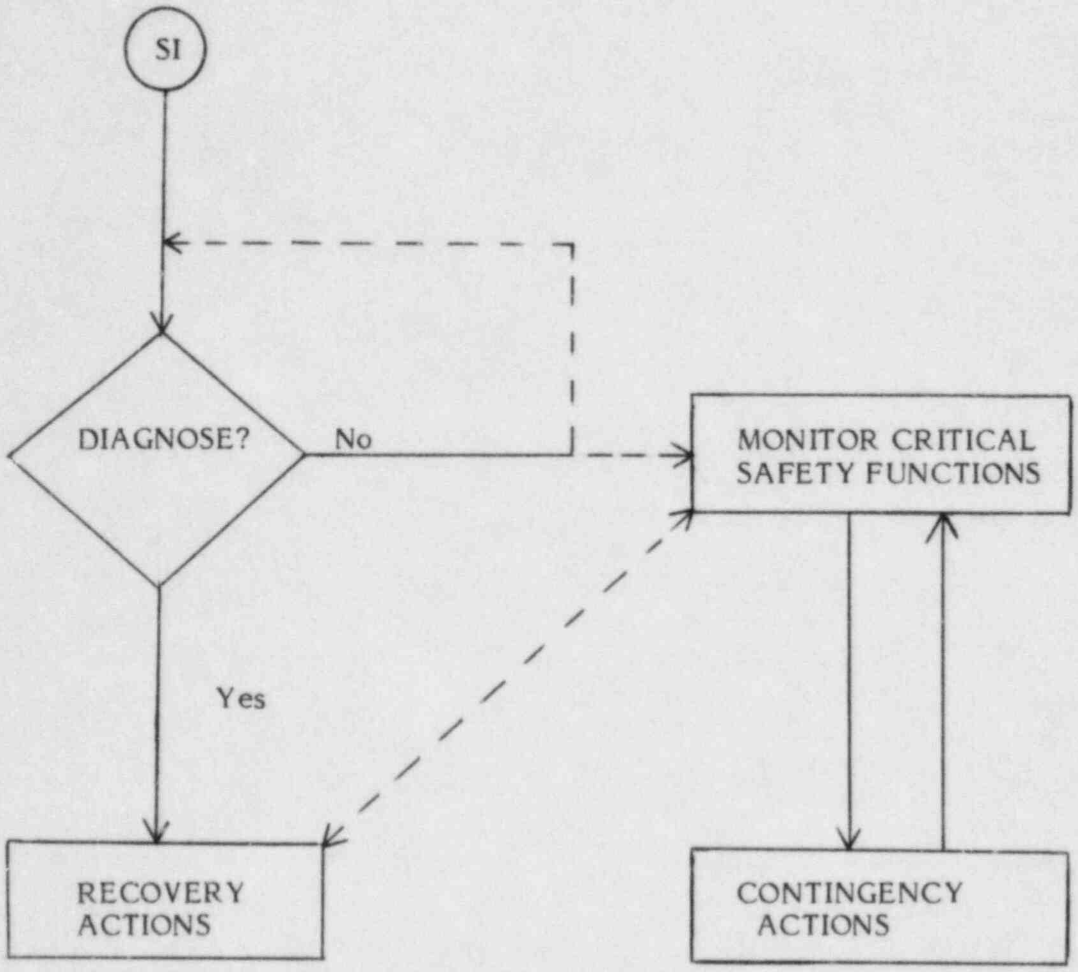
1. Main Steam Line Radiation

   a) main steam line radiation monitor
   b) steam generator safety valve status
   c) atmospheric dump valve status
   d) auxiliary feedwater pump radiation monitor

2. Effluent Radiation

   a) ventilation vent gas monitor
   b) SLCRS radiation monitor

FIGURES



OPERATOR RESPONSE LOGIC FOLLOWING

ACTUATION OF ENGINEERED SAFEGUARDS SYSTEM

FIGURE 1

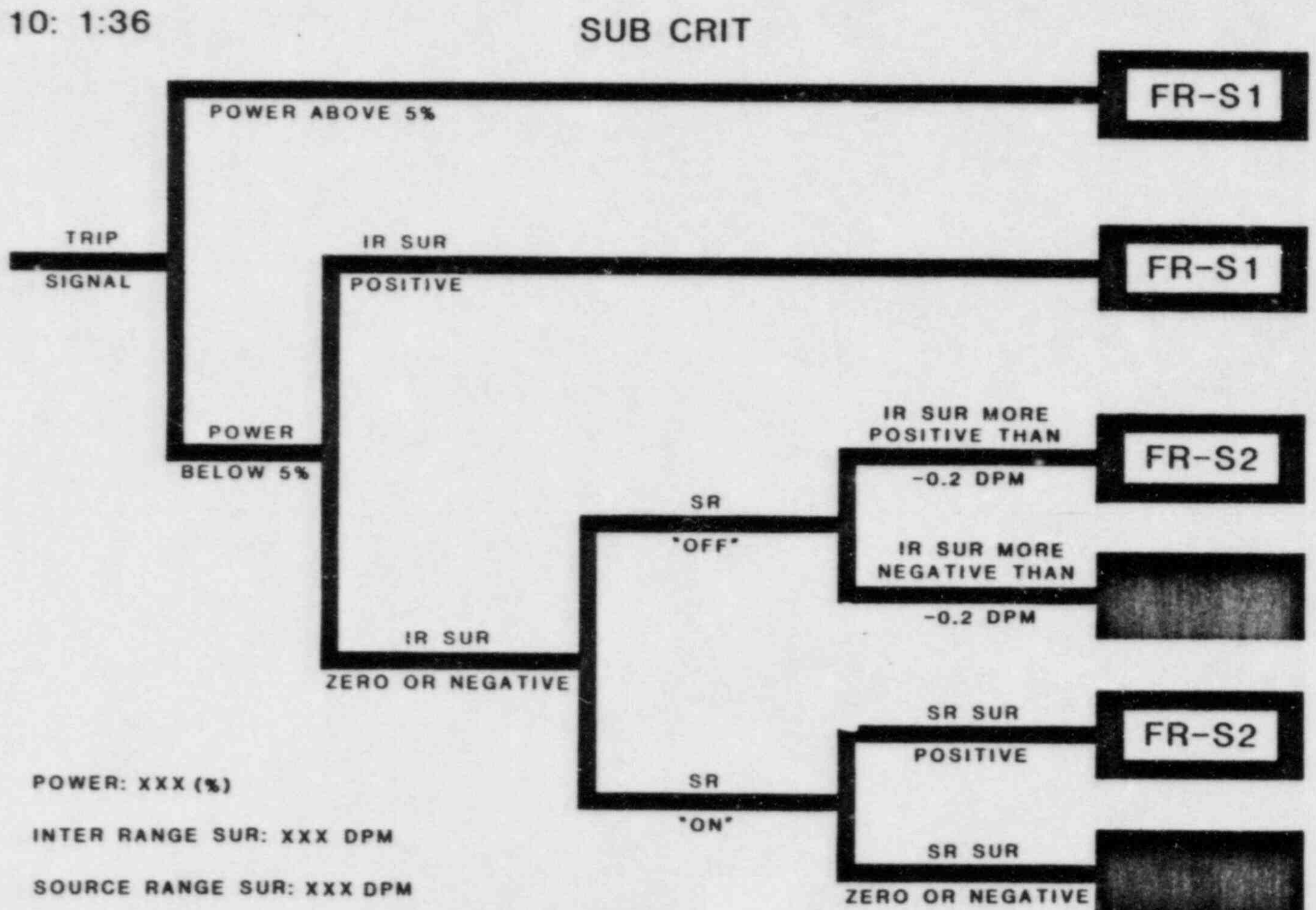

10: 1:36
SUB CRIT
TRIP
SIGNAL
POWER ABOVE 5%
FR-S1
POWER
BELOW 5%
IR SUR
POSITIVE
FR-S1
IR SUR
ZERO OR NEGATIVE
SR
"OFF"
IR SUR MORE
POSITIVE THAN
-0.2 DPM
FR-S2
IR SUR MORE
NEGATIVE THAN
-0.2 DPM
SR
"ON"
SR SUR
POSITIVE
FR-S2
SR SUR
ZERO OR NEGATIVE
POWER: XXX (%)
INTER RANGE SUR: XXX DPM
SOURCE RANGE SUR: XXX DPM

FIGURE 2a

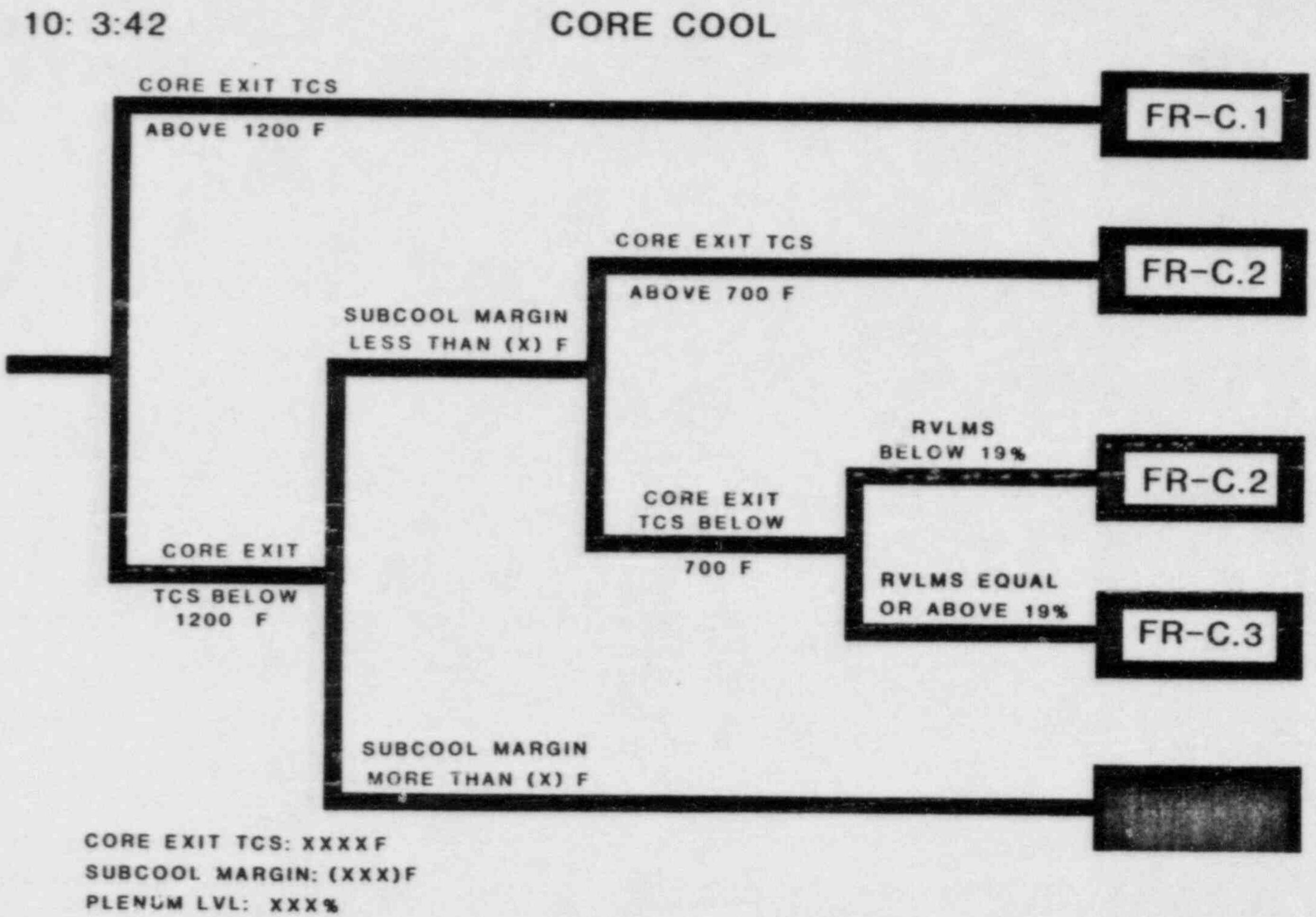

10: 3:42
CORE COOL
CORE EXIT TCS
ABOVE 1200 F
FR-C.1
CORE EXIT TCS
ABOVE 700 F
FR-C.2
SUBCOOL MARGIN
LESS THAN (X) F
RVLMS
BELOW 19%
FR-C.2
CORE EXIT
TCS BELOW
700 F
CORE EXIT
TCS BELOW
1200 F
RVLMS EQUAL
OR ABOVE 19%
FR-C.3
SUBCOOL MARGIN
MORE THAN (X) F
CORE EXIT TCS: XXXXF
SUBCOOL MARGIN: (XXX)F
PLENUM LVL: XXX%

FIGURE 2b

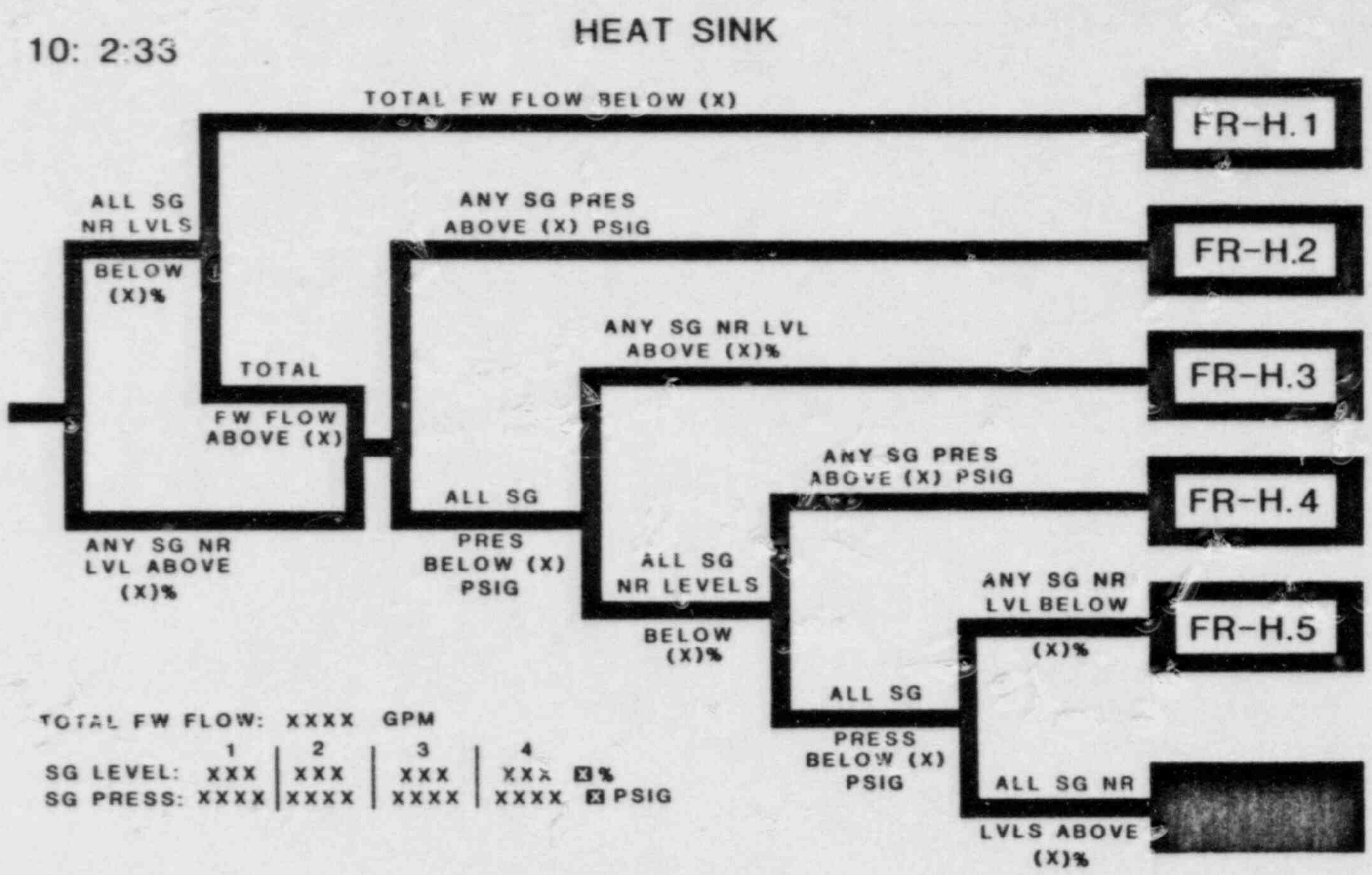

10: 2:33
HEAT SINK
TOTAL FW FLOW BELOW (X)
FR-H.1
ALL SG NR LVLS BELOW (X)%
ANY SG PRES ABOVE (X) PSIG
FR-H.2
ANY SG NR LVL ABOVE (X)%
FR-H.3
TOTAL FW FLOW ABOVE (X)
ANY SG PRES ABOVE (X) PSIG
FR-H.4
ALL SG PRES BELOW (X) PSIG
ANY SG NR LVL ABOVE (X)%
ALL SG NR LEVELS BELOW (X)%
ANY SG NR LVL BELOW (X)%
FR-H.5
ALL SG PRESS BELOW (X) PSIG
ALL SG NR LVLS ABOVE (X)%
TOTAL FW FLOW: XXXX GPM
1 2 3 4
SG LEVEL: XXX XXX XXX XXX %
SG PRESS: XXXX XXXX XXXX XXXX PSIG

FIGURE 2c

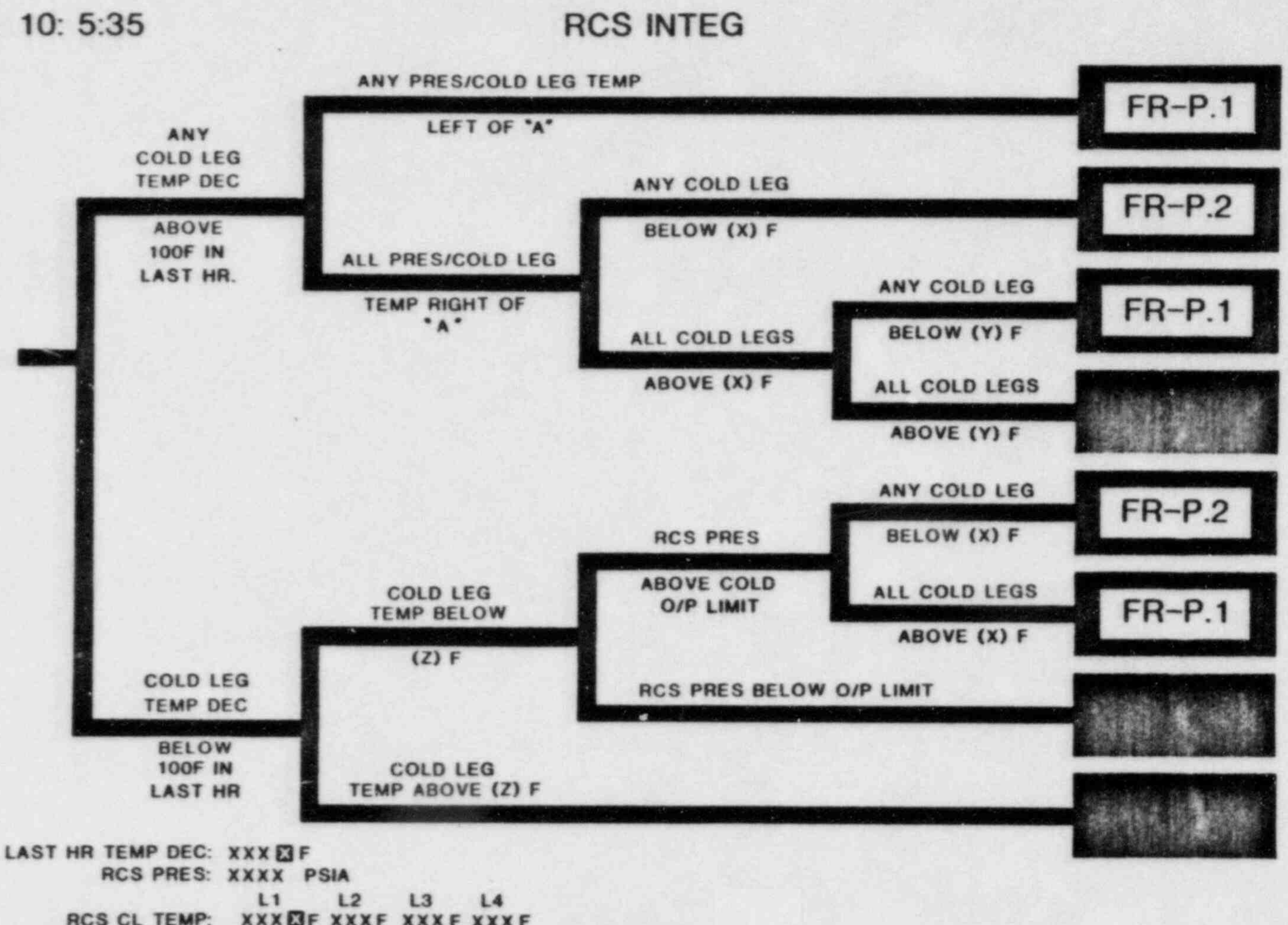

10: 5:35
RCS INTEG
ANY PRES/COLD LEG TEMP
LEFT OF "A"
FR–P.1
ANY COLD LEG TEMP DEC
ABOVE 100F IN LAST HR.
ANY COLD LEG
BELOW (X) F
FR–P.2
ALL PRES/COLD LEG
TEMP RIGHT OF "A"
ANY COLD LEG
BELOW (Y) F
FR–P.1
ALL COLD LEGS
ABOVE (X) F
ALL COLD LEGS
ABOVE (Y) F
ANY COLD LEG
BELOW (X) F
FR–P.2
RCS PRES
ABOVE COLD O/P LIMIT
ALL COLD LEGS
ABOVE (X) F
FR–P.1
COLD LEG TEMP BELOW
(Z) F
RCS PRES BELOW O/P LIMIT
COLD LEG TEMP DEC
BELOW 100F IN LAST HR
COLD LEG TEMP ABOVE (Z) F
LAST HR TEMP DEC: XXX F
RCS PRES: XXXX PSIA
L1 L2 L3 L4
RCS CL TEMP: XXX F XXX F XXX F XXX F

FIGURE 2d

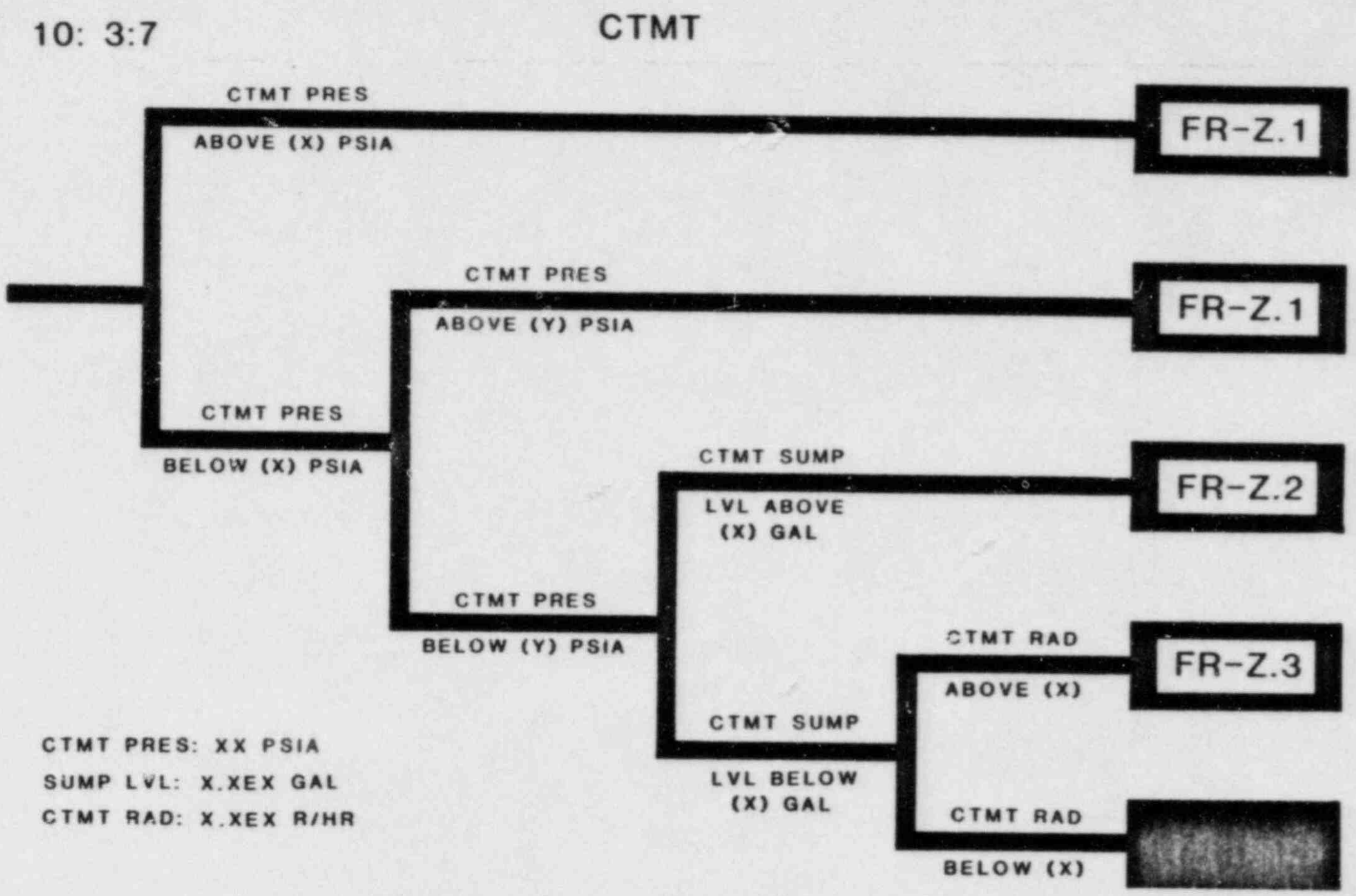

10: 3:7
CTMT
CTMT PRES
ABOVE (X) PSIA
FR-Z.1
CTMT PRES
ABOVE (Y) PSIA
FR-Z.1
CTMT PRES
BELOW (X) PSIA
CTMT SUMP
LVL ABOVE
(X) GAL
FR-Z.2
CTMT PRES
BELOW (Y) PSIA
CTMT RAD
ABOVE (X)
FR-Z.3
CTMT SUMP
LVL BELOW
(X) GAL
CTMT RAD
BELOW (X)
CTMT PRES: XX PSIA
SUMP LVL: X.XEX GAL
CTMT RAD: X.XEX R/HR

FIGURE 2e

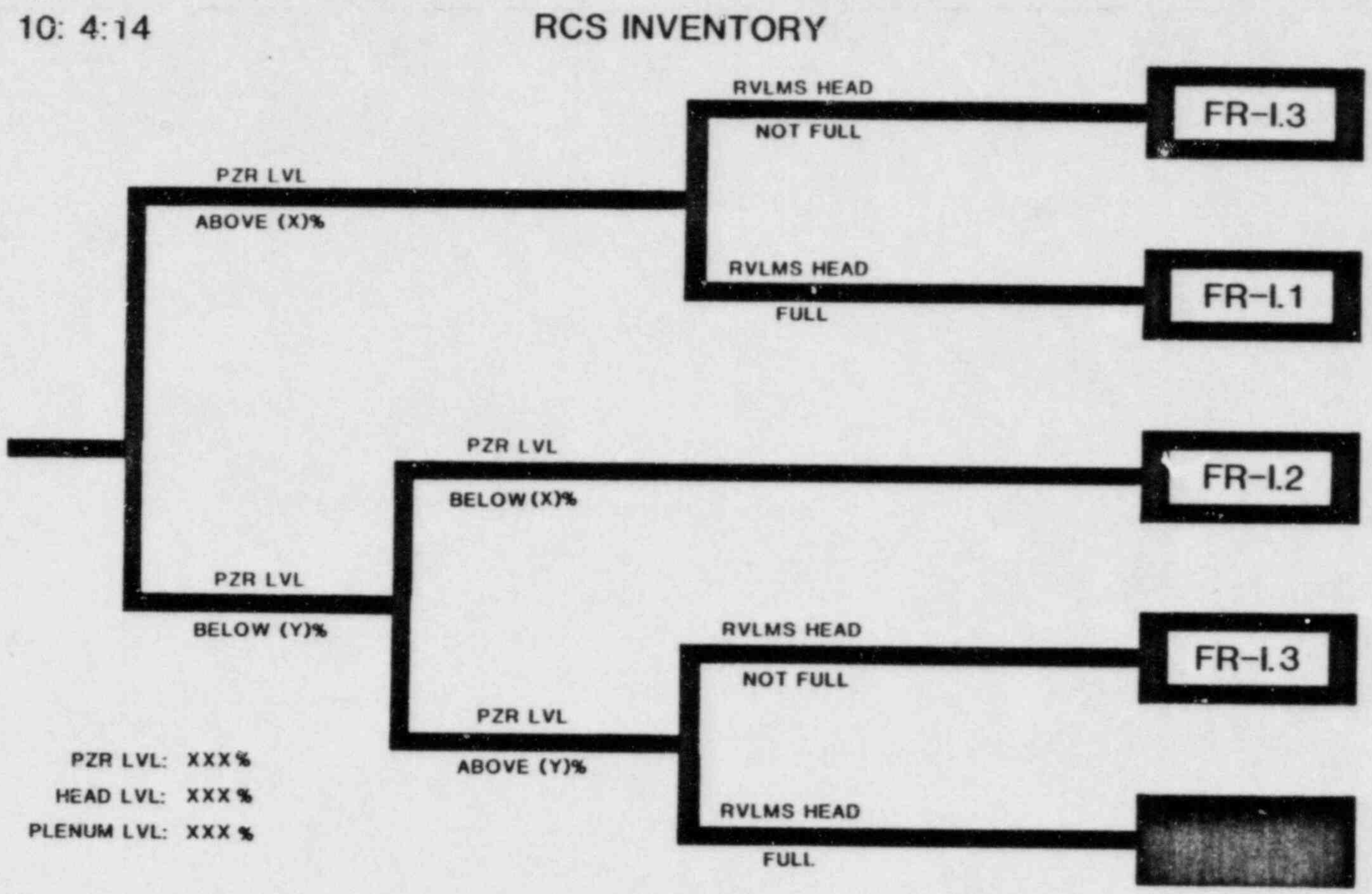

10: 4:14
RCS INVENTORY
RVLMS HEAD
NOT FULL
FR-I.3
PZR LVL
ABOVE (X)%
RVLMS HEAD
FULL
FR-I.1
PZR LVL
BELOW (X)%
FR-I.2
PZR LVL
BELOW (Y)%
RVLMS HEAD
NOT FULL
FR-I.3
PZR LVL
ABOVE (Y)%
RVLMS HEAD
FULL
PZR LVL: XXX%
HEAD LVL: XXX %
PLENUM LVL: XXX %

FIGURE 2f

APPENDIX A

INSTRUMENTATION FOR CRITICAL SAFETY FUNCTION MONITORING

CRITICAL SAFETY FUNCTION: Subcriticality

VARIABLE: Reactor Trip

Description	Instrument No.
Reactor Trip Signal	TMB-RX
Trip Annunciators	MB4E-1-1 MB4E-1-2 MB4E-1-3 MB4E-1-4 MB4E-1-6
	MB4E-2-3 MB4E-2-4 MB4E-2-6 MB4E-2-7
	MB4E-3-3 MB4E-3-4 MB4E-3-5 MB4E-3-6
	MB4E-4-1 MB4E-4-2 MB4E-4-5 MB4E-4-6
	MB4E-5-1 MB4E-5-2 MB4E-5-3 MB4E-5-5 MB4E-5-7
	MB4E-6-3 MB4E-6-5 MB4E-6-7
Rod Bottom	3RD1-RB 3RD1-RB2

CRITICAL SAFETY FUNCTION: Subcriticality

VARIABLE: Power

Description	Instrument No.
Power Range Monitors	3NMP-NM41F 3NMP-NM42F 3NMP-NM43F 3NMP-NM44F
Wide Range Fission Channels A and B	3NME*DET1WR 3NME*DET2WR

CRITICAL SAFETY FUNCTION: Subcriticality

VARIABLE: Startup Rate

Description	Instrument No.
Intermediate Range Monitor	3NMI-NM35B 3NMI-NM36B
Source Range Monitor	3NMS-NM31F 3NMS-NM32F
Wide Range Fission Channels A and B	3NME*DET1SR 3NME*DET2SR

CRITICAL SAFETY FUNCTION: Subcriticality

VARIABLE: Source Range Energized

Description	Instrument No.
Source Range	3NMS-NC31H
Loss of Voltage	3NMS-NC32H

CRITICAL SAFETY FUNCTION: Core Cooling

VARIABLE: Core Exit Temperature

Description	Instrument No.
Core Exit Thermocouples	3CTS*TE1 through 3CTS*TE50

CRITICAL SAFETY FUNCTION: Core Cooling

VARIABLE: RCS Subcooling

Description	Instrument No.
Core Exit Thermocouples	3CTS*TE1 through 3CTS*TE50
Pressurizer Pressure	3RCS*PT455A 3RCS*PT456 3RCS*PT457 3RCS*PT458
RCS Pressure	3RCS*PT403 3RCS*PT405

CRITICAL SAFETY FUNCTION: Core Cooling

VARIABLE: RC Pump Status

DELETED

CRITICAL SAFETY FUNCTION: Core Cooling

VARIABLE: RV level

Description	Instrument No.
Head Level	Train A Train B
Plenum Level	Train A Train B

CRITICAL SAFETY FUNCTION: Heat Sink

VARIABLE: S/G level

Description	Instrument No.
Narrow Range S/G Level	
#1	3FWS*LT517/LT518/LT519
#2	3FWS*LT527/LT528/LT529
#3	3FWS*LT537/LT538/LT539
#4	3FWS*LT547/LT548/LT549

CRITICAL SAFETY FUNCTION: Heat Sink

VARIABLE: Total FW Flow

Description		Instrument No.
MFW	1	3FWS-FT510/511
	2	3FWS-FT520/521
	3	3FWS-FT530/531
	4	3FWS-FT540/541
AFW	1	3FWA*FT51A
	2	3FWA*FT33B
	3	3FWA*FT33C
	4	3FWA*FT51D

CRITICAL SAFETY FUNCTION: Heat Sink

VARIABLE: S/G pressure

Description	Instrument No.
S/G Outlet Pressure	
1	3MSS*PT514/PT515/PT516
2	3MSS*PT524/PT525/PT526
3	3MSS*PT534/PT535/PT536
4	3MSS*PT544/PT545/PT546

CRITICAL SAFETY FUNCTION: Integrity

VARIABLE: Cooldown Rate

Description	Instrument No.
Cold Leg RTD Loop #1	3RCS*TE413B
2	3RCS*TE423B
3	3RCS*TE433B
4	3RCS*TE443B

CRITICAL SAFETY FUNCTION: Integrity

VARIABLE: RCS Temperature

Description	Instrument No.
Cold Leg RTD Loop #1	3RCS*TE413B
2	3RCS*TE423B
3	3RCS*TE433B
4	3RCS*TE443B

CRITICAL SAFETY FUNCTION: Integrity

VARIABLE: RCS pressure

Description	Instrument No.
Pressurizer Pressure	3RCS*PT455A 3RCS*PT456 3RCS*PT457 3RCS*PT458
Wide Range RCS Pressure	3RCS*PT403 3RCS*PT405

CRITICAL SAFETY FUNCTION: Containment

VARIABLE: Containment Pressure

Description	Instrument No.
Wide Range Pressure	3LMS*PT24A 3LMS*PT24B
Narrow Range Pressure	3LMS*PT934 3LMS*PT935 3LMS*PT936 3LMS*PT937

CRITICAL SAFETY FUNCTION: Containment

VARIABLE: Water Level

Description	Instrument No.
Wide Range Sump Level	3RSS*LT22A 3RSS*LT22B

CRITICAL SAFETY FUNCTION: Containment

VARIABLE: Containment Area Radiation

Description	Instrument No.
Wide Range Monitors	3RMS*RE04 3RMS*RE05

CRITICAL SAFETY FUNCTION: Inventory

VARIABLE: Pressurizer Level

Description	Instrument No.
Pressurizer Level	3RCS*LT459 3RCS*LT460 3RCS*LT461

CRITICAL SAFETY FUNCTION: Inventory

VARIABLE: RV level

Description	Instrument No.
Head Level	Train A Train B
Plenum Level	Train A Train B

APPENDIX B

INSTRUMENTATION FOR RADIOACTIVITY RELEASE DISPLAY

RADIOACTIVITY RELEASE DISPLAY

VARIABLE: Main Steam Line Radiation

Description	Instrument No.
Main Steam Line Radiation Monitors	3MSS*RE75 3MSS*RE76 3MSS*RE77 3MSS*RE78
Safety Valve Flow Switches	3SVV*FE28A/FE28B/FE28C/FE28D 3SVV*FE29A/FE29B/FE29C/FE29D 3SVV*FE30A/FE30B/FE30C/FE30D 3SVV*FE31A/FE31B/FE31C/FE31D 3SVV*FE32A/FE32B/FE32C/FE32D
Atmospheric Dump Valve Position Switches	3MSS*Z20A# 3MSS*Z20B# 3MSS*Z20C# 3MSS*Z20D#
Auxiliary Feedpump Radiation Monitor	3MSS*RE79

RADIOACTIVITY RELEASE DISPLAY

VARIABLE: Effluent Radiation

Description	Instrument No.
Ventilation Vent Monitor	3HVR*RE10
SLCRS Monitor	3HVR*RE19