TENNESSEE VALLEY AUTHORITY
CHATTANOOGA, TENNESSEE 37401
400 Chestnut Street Tower II

June 25, 1985

Director of Nuclear Reactor Regulation
Attention: Ms. E. Adensam, Chief
Licensing Branch No. 4
Division of Licensing
U.S. Nuclear Regulatory Commission
Washington, D.C. 20555

Dear Ms. Adensam:

In the Matter of the Application of	)	Docket Nos. 50-390
Tennessee Valley Authority	)	50-391

Please refer to your letter to H. G. Parris dated September 14, 1984 which requested that additional information be provided on the Watts Bar Safety Parameter Display System (SPDS) in the areas of instrumentation and control systems and human factors engineering. TVA's letter of March 27, 1985 provided the requested information pertaining to instrumentation and control systems. Enclosed is the requested information regarding human factors engineering.

If you have any questions concerning this matter, please get in touch with D. B. Ellis of my staff at FTS 858-2682 in Chattanooga.

Very truly yours,

TENNESSEE VALLEY AUTHORITY

J. A. Domer

J. A. Domer, Chief
Nuclear Licensing Branch

Sworn to and subscribed before me
this 25th day of June 1985

Paulette H. White
Notary Public
My Commission Expires 8-24-88

Enclosure
cc: U.S. Nuclear Regulatory Commission (Enclosure)
Region II
Attn: Dr. J. Nelson Grace, Regional Administrator
101 Marietta Street, NW, Suite 2900
Atlanta, Georgia 30323

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ENCLOSURE

WATTS BAR NUCLEAR PLANT
HUMAN FACTORS ENGINEERING INFORMATION

Question 620.01 Human Factors Program

Provide a description of the display system, its human factored design, and the methods used and results from a human factors program to ensure that the displayed information can be readily perceived and comprehended so as not to mislead the operator.

TVA Response

The Safety Parameter Display System (SPDS) consists of the block type critical safety function status trees from the upgraded Westinghouse Owners Group (WOG) Emergency Response Guidelines (ERGs). Documentation for these status trees "Emergency Response Guidelines Revision 1" were transmitted to Hugh L. Thompson, Jr., Director, Division of Human Factors Safety, by the Westinghouse Owners Group on May 4, 1984.

Each tree uses several blocks containing questions with a yes or no output which leads to a status. When a status tree branch is not satisfied, it directs the operator to an appropriate function restoration guideline.

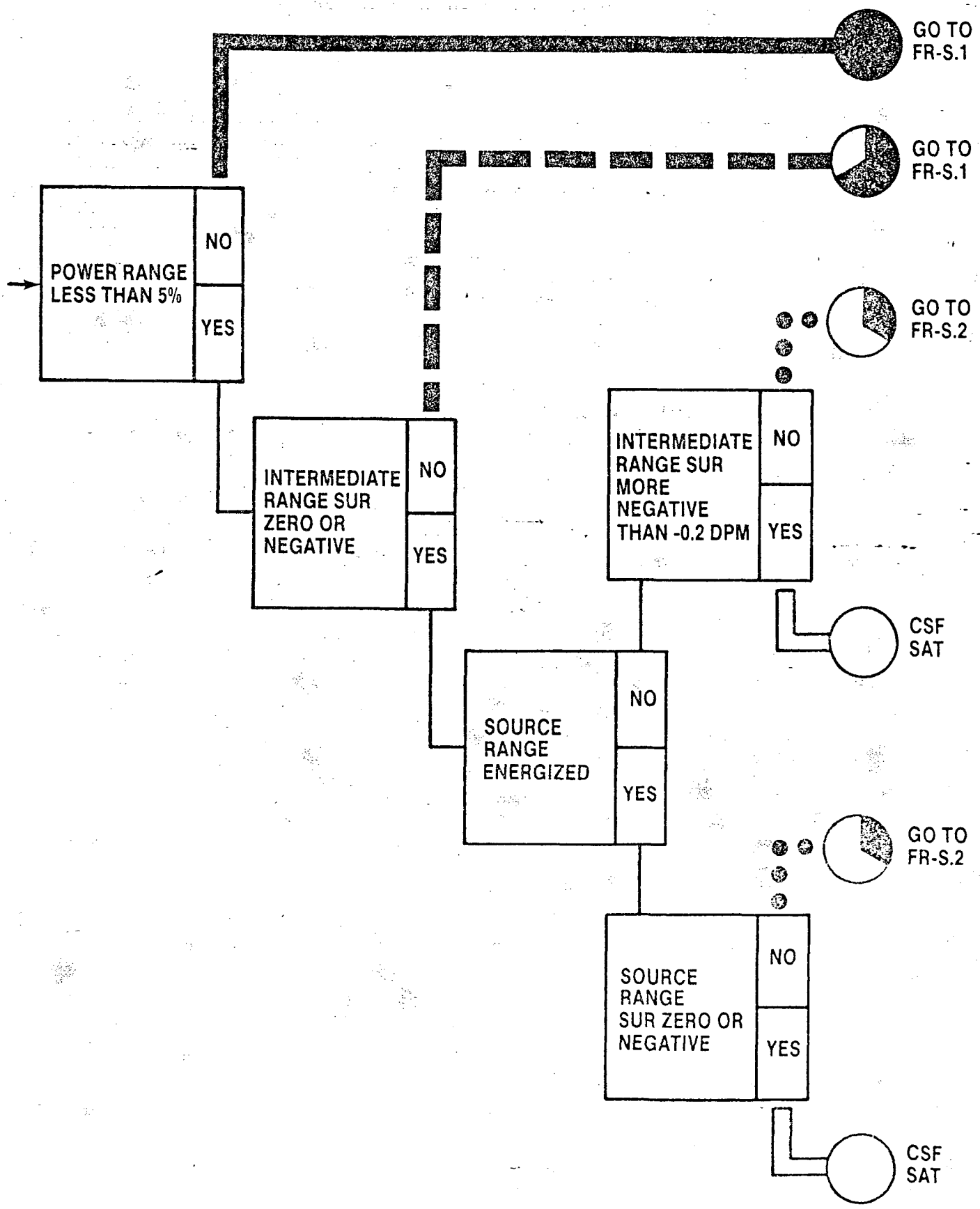
Six generic status trees from the WOG ERGs are attached. These trees will be converted to plant-specific trees for Watts Bar. The different branches are color coded to show the operator how serious any challenge is to a critical safety function. The ordering of the trees also defines priorities. The colors in order of priority are: red (solid line), magenta (dashed line), yellow (short dashed line), and green (double line).

When any status tree is displayed, colors are shown in a designated area, giving the status of the other five trees.

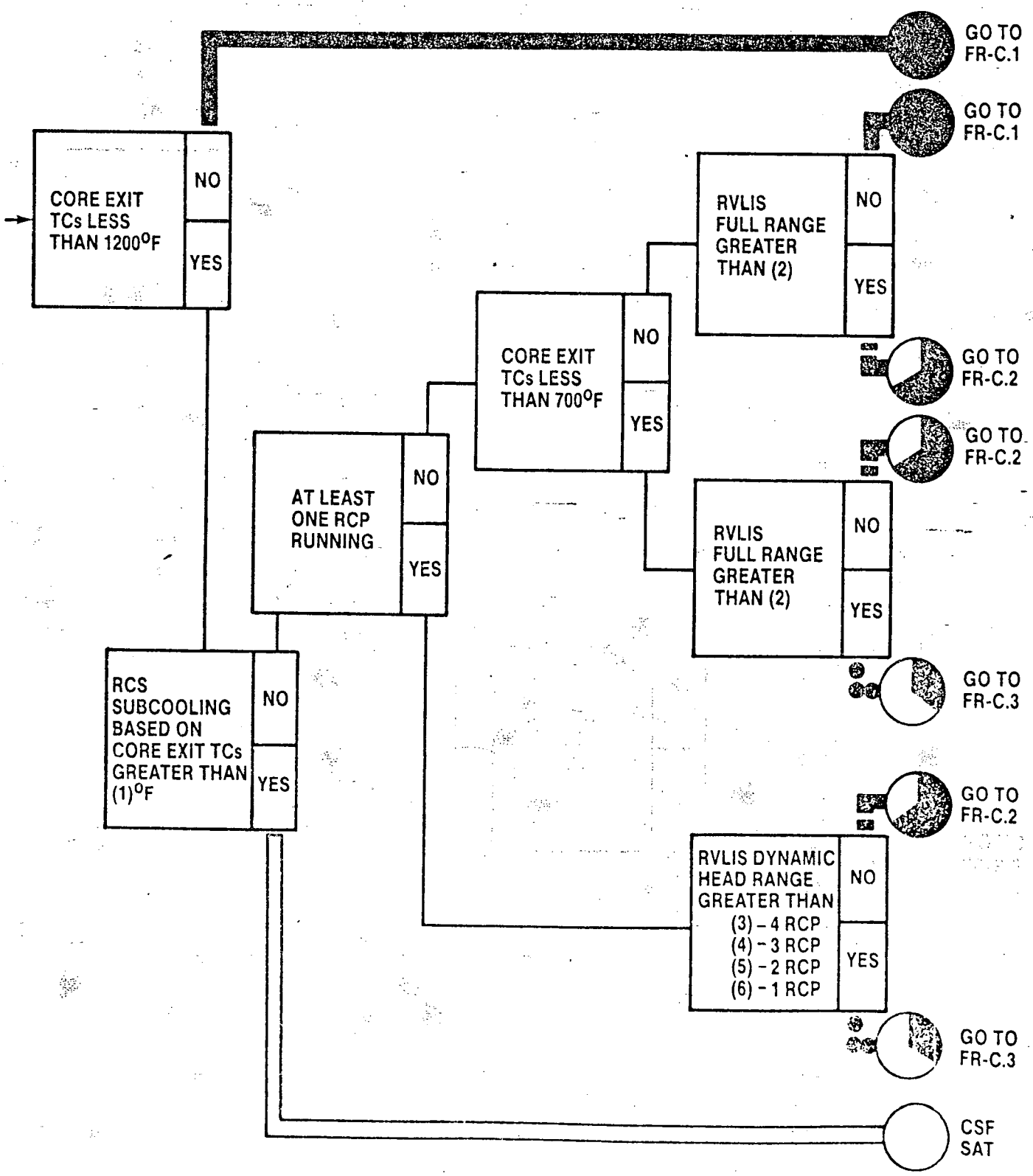
The critical safety function status trees have been developed using human factors principles. When the SPDS system is operational, the control room design review team will make a human factors review on the status tree displays.

In addition to the critical safety function status trees, a radiation monitoring display will be included. This display provides readings for important radiation monitor points (including shield building, auxiliary building, steam generator blowdown, and condenser vacuum exhaust) to supplement the containment critical safety function status trees. The critical safety function status trees along with this additional radiation monitoring display fulfill the five SPDS functions (reactivity control, reactor core cooling and heat removal from primary system, reactor coolant system integrity, radioactivity control, and containment) as identified in Supplement 1 to NUREG-0737.

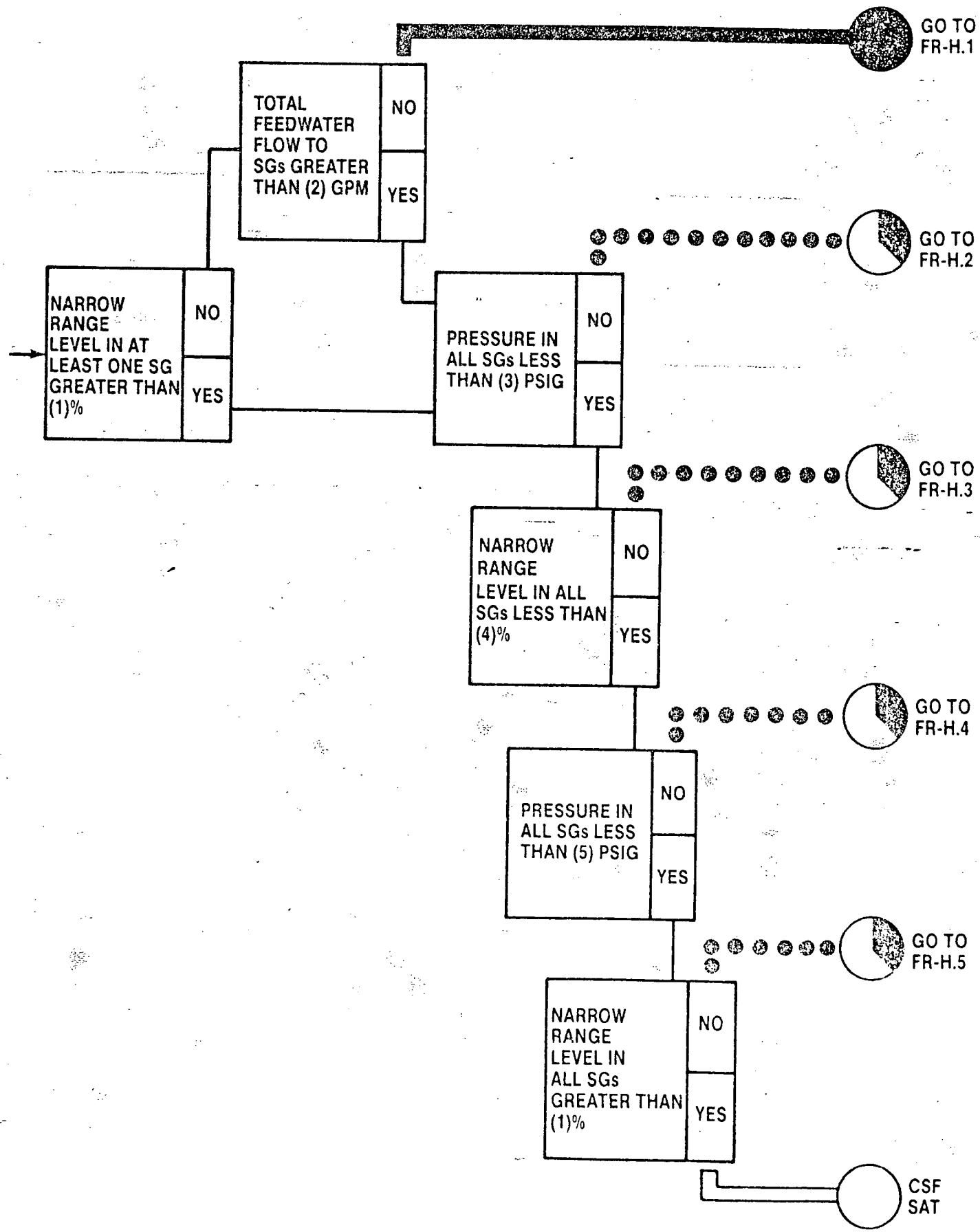
Number:	Title:	Rev. Issue/Date:
F-0.1	SUBCRITICALITY	HP/LP, REV. 1 1 Sept., 1983



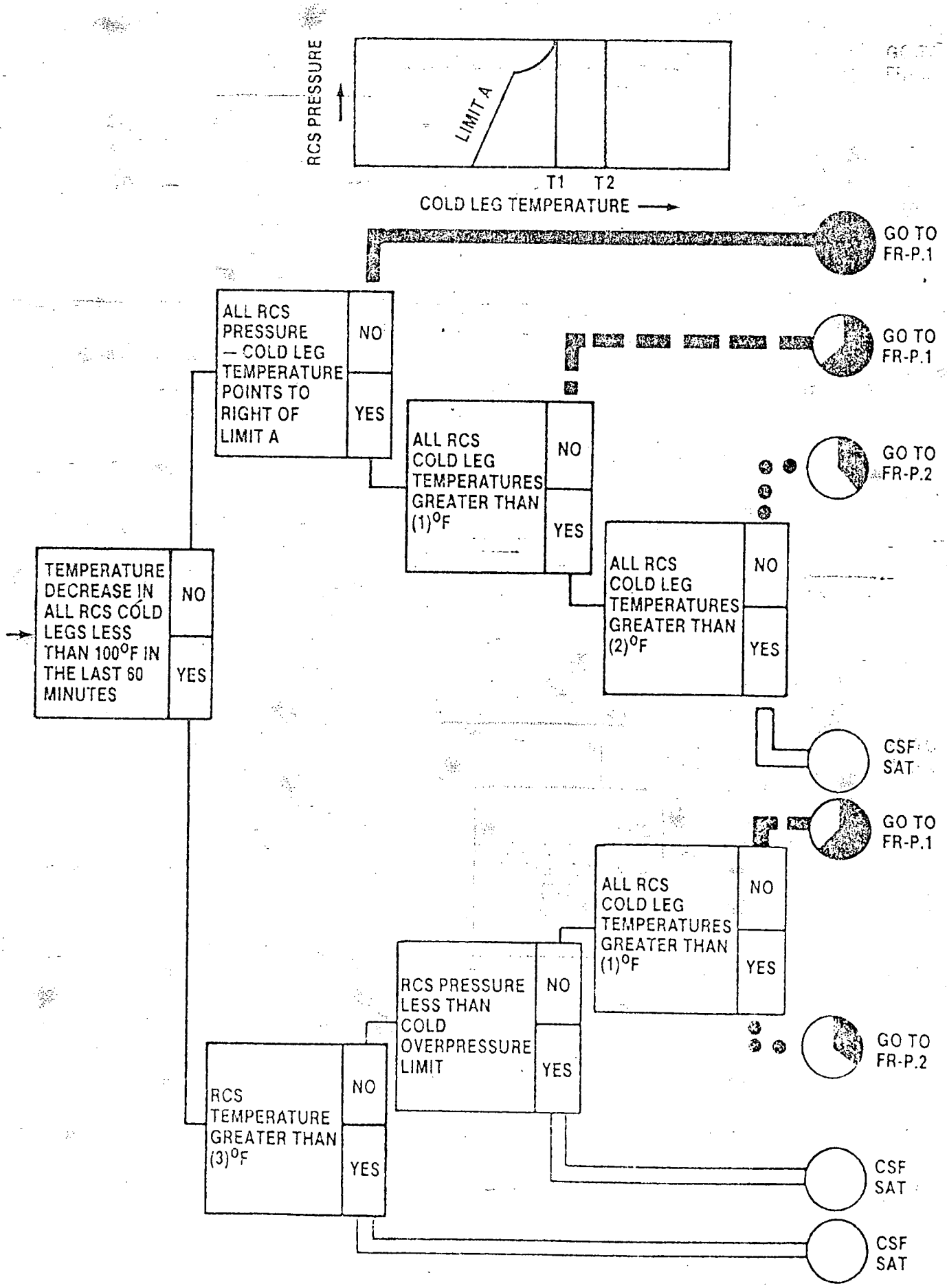
Number:	Title:	Rev. Issue/Date:
F-0.2	CORE COOLING	HP/LP, REV. 1 1 Sept., 1983



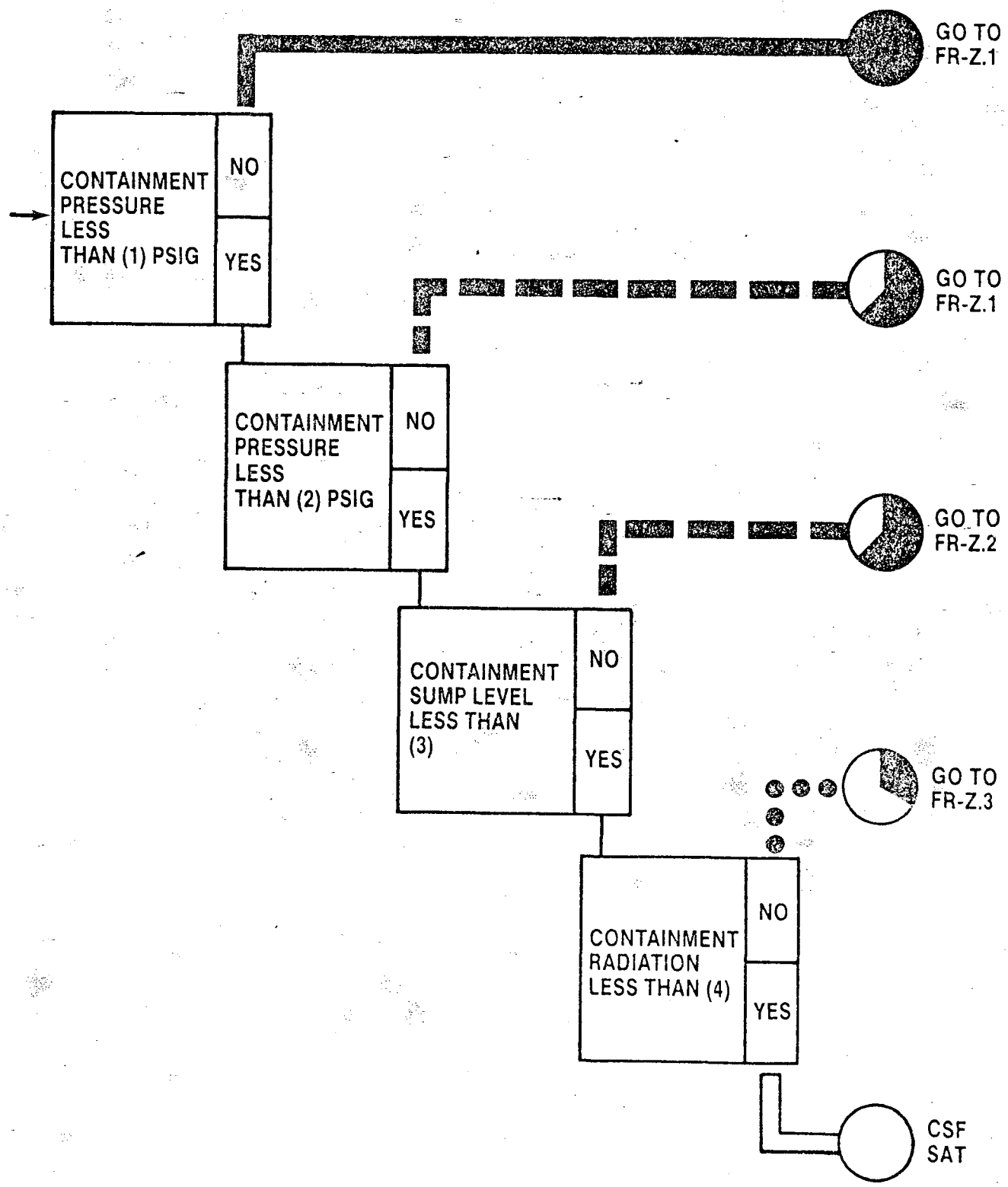
Number:	Title:	Rev. Issue/Date:
F-0.3	HEAT SINK	HP/LP, REV. 1 1 Sept., 1983



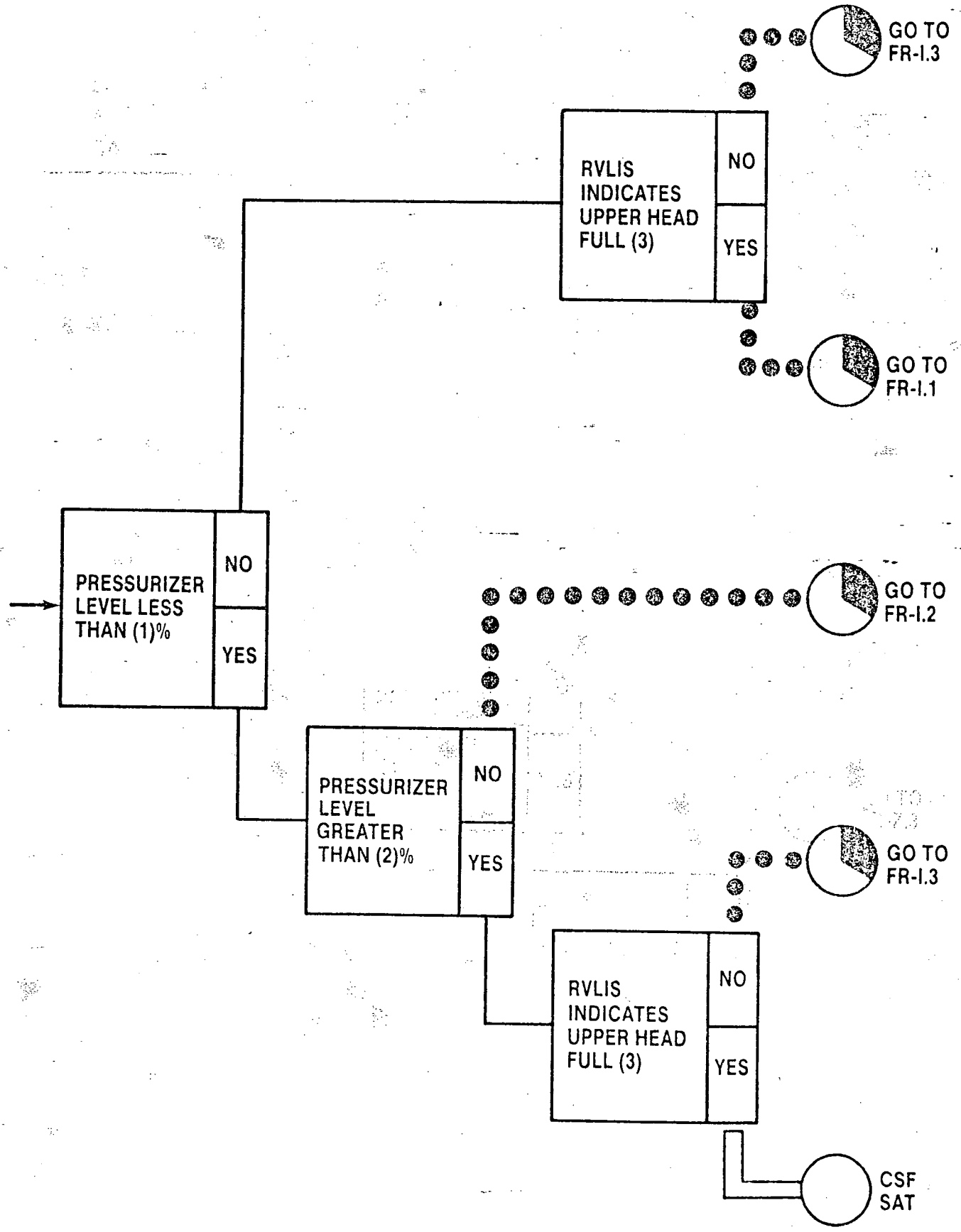
Number:	Title:	Rev. Issue/Date:
F-0.4	INTEGRITY	HP/LP, REV. 1 1 Sept., 1983



Number:	Title:	Rev. Issue/Date:
F-0.5	CONTAINMENT	HP/LP, REV. 1 1 Sept., 1983



Number:	Title:	Rev. Issue/Date:
F-0.6	INVENTORY	HP/LP, REV. 1 1 Sept., 1983



FOOTNOTES

F-0.2 CORE COOLING

(1) Enter sum of temperature and pressure measurement system errors, including allowances for normal channel accuracies and post accident transmitter errors, translated into temperature using saturation tables.

(2) Enter plant specific value which is 3-1/2 feet above the bottom of active fuel in core with zero void fraction, plus uncertainties.

(3) Enter plant specific value corresponding to an average system void fraction of 50 percent with 4 RCPs running, plus uncertainties.

(4) Enter plant specific value corresponding to an average system void fraction of 50 percent with 3 RCPs running, plus uncertainties.

(5) Enter plant specific value corresponding to an average system void fraction of 50 percent with 2 RCPs running, plus uncertainties.

(6) Enter plant specific value corresponding to an average system void fraction of 50 percent with 1 RCP running, plus uncertainties.

F-0.3 HEAT SINK

(1) Enter plant specific value showing SG level just in the narrow range, including allowances for normal channel accuracy, post-accident transmitter errors, and reference leg process errors, not to exceed 50%.

(2) Enter the minimum safeguards AFW flow requirement for heat removal, plus allowances for normal channel accuracy (typically one MD AFW pump capacity at SG design pressure).

(3) Enter plant specific pressure for highest steamline safety valve setpoint.

(4) Enter plant specific value for SG high-high level feedwater isolation setpoint.

(5) Enter plant specific pressure for lowest steamline safety valve setpoint.

FOOTNOTES (Continued)

F-0.4 INTEGRITY

(1) Enter plant specific value corresponding to temperature T_1. Refer to background document for status tree F-0.4.

(2) Enter plant specific value corresponding to temperature T_2. Refer to background document for status tree F-0.4.

(3) Enter plant specific temperature setpoint below which cold overpressure protection system is in service.

F-0.5 CONTAINMENT

(1) Enter plant specific containment design pressure.

(2) Enter plant specific containment high-2 pressure setpoint.

(3) Enter plant specific containment water level just below design flood level minus allowances for normal channel accuracy.

(4) Enter plant specific value corresponding to radiation level alarm setpoint for post accident containment radiation monitor.

F-0.6 INVENTORY

(1) Enter plant specific pressurizer high level reactor trip setpoint.

(2) Enter plant specific pressurizer low level letdown isolation setpoint.

(3) Enter plant specific instrument channel and setpoint which indicates upper head is full.

WATTS BAR NUCLEAR PLANT
HUMAN FACTORS ENGINEERING INFORMATION

Question 620.02 Data Validation

Describe the method used to validate data displayed in the SPDS. Also describe how invalid data is defined to the operator.

TVA Response

With the block-type status tree displays, computer points are displayed below a block, where applicable. The point and the yes/no outputs will be shown as bad or suspect when internal software checks show the data to be questionable. There are four quality classifications:

a. Good data.

b. Sensor data inconsistent with the majority of redundant sensor values.

c. Data evaluated as bad because it is outside the process sensor or data acquisition system span, or because hardware checks indicate a malfunctioning input device.

d. Data which is operator entered.

Further validation of data is accomplished by field verification tests which are performed after system installation. This verifies that the system will properly display the input signals and that the inputs are connected correctly.

WATTS BAR NUCLEAR PLANT
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Question 620.03 Verification and Validation Program

Define and discuss the Verification and Validation Program Plan which was 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.

TVA Response

A. General

This document outlines the plan by which Watts Bar's SPDS will be verified and validated and is based upon NSAC 39, "Verification and Validation for Safety Parameter Display Systems." Any revisions to the verification and validation (V&V) plan will be submitted to NRC.

The objectives, methods of verification and validation (V&V), personnel, and documentation to support the program will be discussed. It is intended that this will be an ongoing program; therefore, if significant modifications are made to SPDS, a similar V&V will be conducted. In addition, changes may be made to this V&V program as dictated from experience.

It should be noted that Watts Bar will utilize the Westinghouse supplied technical support complex computer system to support the SPDS critical safety function status trees (see NRC SER on W Generic Technical Support Complex transmitted via letter from D. Crutchfield to E. P. Rahe on February 2, 1984). Based on this, we will be referencing the V&V activities accomplished by Westinghouse to support our V&V of the SPDS.

B. Objectives

The verification/validation process will include the following:

1. System requirements review
2. Hardware/software verification review
3. Validation tests
4. Field verification tests, and
5. Final report

C. Responsibilities

A reviewer or review team will be responsible for verifying that the criteria of each objective are met and that discrepancies are documented.

D. Method of Verification/Validation

To ensure each objective is met, the V&V will be performed as follows:

1. System Requirements Review

This objective will be met by performing a tabletop review of the SPDS to ensure the system will satisfy the functional requirements. The reviewer(s) shall be familiar with plant equipment, operations, technical requirements, operator knowledge level, emergency operating procedures, and human factors. Additionally, the reviewer(s) should not include implementation personnel.

2. Hardware/Software Verification Review

This objective will be met by performing a tabletop review of the SPDS hardware and software to ensure the correct implementation of the system requirements. The reviewer(s) should be familiar with the computer system hardware and software.

3. Validation Tests to Conform that the System Satisfies the Functional Requirements

This objective will be met by performing status tests of the system performance. The reviewer(s) should be familiar with the computer system and the functional requirements. This testing will demonstrate that the hardware and software function acceptably. This testing will be performed using status simulated data to ensure that the SPDS performs as intended.

4. Field Verification Tests

This objective will be met by performing testing after system installation to ensure that the system was installed properly. The reviewer(s) shall be familiar with the computer system and the functional requirements. Field verification will consist of ensuring that each input signal is properly connected and that the signal range is consistent with the design.

5. Final Report

A final report documenting the SPDS V&V requirements and how they were met will be prepared.

WATTS BAR NUCLEAR PLANT
HUMAN FACTORS ENGINEERING INFORMATION
(Continued)

E. Discrepancy Detection

The purpose of the V&V program is to ensure the SPDS aids the control room personnel during abnormal and emergency conditions in determining the safety status of the plant and in assessing whether corrective actions by operators to avoid a degraded core are required. A reviewer or review team will be assigned to address each objective listed above. It will be the responsibility of the reviewer or review team to ensure that the criteria of the objectives are met and discrepancies are identified and corrected when appropriate.

F. Discrepancy Resolution

When a discrepancy is identified, a resolution will be developed. A solution will be written on the appropriate disposition.

G. Documentation

The discrepancies, including their disposition, will be maintained for the life cycle of the system. Existing division procedures will be utilized for system configuration management.

WATTS BAR NUCLEAR PLANT
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Question 620.04 Unreviewed Safety Questions

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

TVA Response

A preliminary 10 CFR 50.59 evaluation has been performed, and TVA does not consider the SPDS an unreviewed safety question.

On Technical Specification Improvement, NUREG 1024, NRC referenced statements by the Atomic Safety and Licensing Appeals Board (ALAB-531 in the matter of Portland General Electric, ET AL Trojan Nuclear Plant). In part, the Appeal Board stated:

> Technical Specifications are to be reserved for those matters as to which the imposition of rigid conditions or limitations upon reactor operation is deemed necessary to obviate the possibility of an event giving rise to an immediate threat to the public health and safety.

Inoperability of the SPDS would not pose an immediate threat to the health and safety of the public. TVA does not plan to submit technical specifications for the SPDS. This decision will enhance regulatory performance in regard to compliance with existing technical specifications.

WATTS BAR NUCLEAR PLANT
HUMAN FACTORS ENGINEERING INFORMATION

Question 620.05 Implementation Plan

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

TVA Response

As stated in the Watts Bar SER and required by the unit 1 draft license (dated May 20, 1985), we will have the unit 1 SPDS fully operational and operators trained in its usage prior to startup following the first refueling outage.