Filed: July 15, 1983

APPLICANTS' DIRECT TESTIMONY No. 1

<u>Members of the Panel</u>:

David N. Merrill
George S. Thomas
John DeVincentis
David A. Maidrand
Peter L. Anderson
James A. MacDonald
Robert J. Merlino

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Introduction: Members of the Panel

David N. Merrill is the Executive Vice President of Public Service Company of New Hampshire. He has been employed by PSNH since 1949; he was elected a vice president in 1965 and was elected Executive Vice President in 1973. Mr. Merrill has responsibility for Engineering, Production, Fuel Procurement and Supply, Energy Management and Research, and all aspects of Seabrook contruction and licensing. Mr. Merrill's qualifications appear in Appendix 13A of the FSAR.

Mr. George S. Thomas is Vice President - Nuclear Production of Public Service Company of New Hampshire. Mr. Thomas began employment with Yankee Atomic Electric Company in 1969, and with PSNH in 1980. Mr. Thomas has been delegated the responsibilities of the Executive Vice President insofar as they include the operation of Seabrook Station, and Mr. Thomas therefore has overall responsiblity for the operation and associated support of Seabrook Station from the PSNH corporate office. Mr. Thomas's qualifications appear in Appendix 13A of the FSAR.

Mr. John DeVincentis is the Seabrook Station Project Manager for Yankee Atomic Electric Company. Mr. DeVincentis joined Yankee in 1963 and has carried several responsibilities for several nuclear projects continuously since that time except for the four years from 1975 to 1979, during which time he was assigned to New England Power Company in connection with its proposed units in Charlestown, Rhode Island. Mr. DeVincentis' qualifications are set forth in Appendix C to the FSAR.

Mr. David Maidrand is a Senior Project Engineer in the Seabrook Project Department of Yankee Atomic Electric Company. Mr. Maidrand commenced his employment with Yankee Atomic in 1974 following 9 years employment with the New England Electric System. Mr. Maidrand's qualifications are set forth In Exhibit 3 hereto.

Mr. Peter L. Anderson is the Lead Seabrook Systems Engineer. Mr. Anderson became employed by Yankee Atomic Electric Company in 1981, following eleven years experience with Maine Yankee Atomic Power Company. Mr.

Anderson is a senior systems engineer for Seabrook and his qualifications appear in Appendix 13C of the FSAR.

Mr. James A. MacDonald is the Manager of the Yankee Radiation Protection Group. Mr. MacDonald began employment with Yankee in 1970 and has been Manager of the Radiation Protection Group since 1973. Mr. MacDonald's qualifications appear in Appendix 13C of the FSAR.

Robert J. Merlino is the president and co-founder of HMM Associates of Waltham, Massachusetts. HMM Associates has been actively involved in radiological emergency planning since the firm was founded, approximately five years ago. To date, HMM Associates has compiled evacuation time estimate studies for fourteen nuclear power plant sites in various parts of the country. Mr. Merlino has directed all emergency planning related work conducted by HMM Associates for PSNH, including the evacuation time estimate study. A copy of Mr. Merlino's resume follows on the next three pages:

ROBERT J. MERLINO

Education

B.S.	Civil Engineering, Tufts University, 1963
	Certificate in Reactor Engineering, Bettis Reactor Engineering School, 1965
M.B.A.	Business Administration, Babson College, 1970

Summary of Experience

Mr. Merlino has broad experience in emergency planning, nuclear licensing and project management. A nationally known authority on emergency planning for nuclear facilities, he has been involved in emergency planning projects for over ten years, and has been active in AIF and EEI emergency planning activities. He has appeared as an expert witness on emergency planning, before NRC Atomic Safety and Licensing Boards. He has served as project manager for licensing activities for a number of facilities.

Experience

1978 - Present	HMM Associates, principal and project manager. He has led emergency planning activities on behalf of several nuclear utilities. This has included evacuation studies, plan and procedures writing for stations, state and local plans, and reviews and audits of various types. He provides frequent consultation to nuclear utility executives on regulatory and management matters. Most recently has provided technical support to Arizona Public Service Co. and Florida Power and Light in meeting emergency preparedness requirements. His involvement has included assistance with plant emergency plans, corporate plans, procedures, and state and local plans. He has provided a computer model for accident dose calculation, conducted training sessions in use of the model, and assisted in scenario development and exercise evaluation.
1977-1978 and 1973-1976	Environmental Research & Technology, Inc. (ERT). Mr. Merlino held a number of positions at ERT as Project Manager, Manager of Air Quality Programs and Manager of Nuclear Services Division. He directed projects involving site selection, air quality and meteorological monitoring, radiological impact assessments and environmental baseline investigations. He represented clients at meetings with regulatory agencies and at public hearings.

Among the projects he directed were:

- o Performance of environmental studies, preparation of environmental impact report and permitting for a pulp and paper mill expansion.
- o Evacuation analyses for two proposed nuclear power stations.
- o A probabilistic analysis of loss of coolant accident doses.
- o Development of computer models for calculating atmospheric dispersion.
- o Installation and operation of meteorological and air quality monitoring networks and data acquisition systems around several fossil-fired and nuclear power stations.

1976-1977 Tera Corporation, Senior Project Manager. Performed and managed engineering and environmental studies. This included the development of a sea breeze fumigation model for calculating ground level concentrations from stack releases.

1968-1973 Yankee Atomic Electric Co., Manager of Safety Analysis. Responsibilities included nuclear power plant site evaluations and participation in and direction of preparation of site-related portions of safety analysis and environmental reports for four nuclear power stations. Topics included land use, meteorology, population distribution, evaluation of potential hazards from nearby industrial and military facilities and radiological safety. Prepared testimony and participated extensively in public hearings as an expert witness before state and federal regulatory bodies.

1967-1968 Pioneer Service and Engineering Co., Nuclear Engineer. Performed safety and analyses for nuclear power stations and developed design requirements for engineered safety systems.

1963-1967 U.S. Navy, Division of Naval Reactors, Staff Engineer. While on active duty, directed government contractors in areas of nuclear propulsion plant mechanical systems design and testing.

ROBERT J. MERLINO

Professional Affiliations/Registrations

Registered Professional Engineer (Nuclear), State of California
American Nuclear Society

Environmental Qualification: Time Duration

(NECNP Contention I.B.2)

As admitted by the Board, this contention reads as follows:

> "The Applicant has not satisfied the requirements of GDC 4 that all equipment important to safety be environmentally qualified because it has not specified the time duration over which the equipment is qualified."

This contention can be confusing, because it does not distinguish between two discrete concepts: (i) how long the equipment can be run, under normal circumstances, without losing its ability to withstand the harsh accident environment (should an accident occur), and (ii) how long the equipment can withstand the harsh environment after the accident has occurred. For ease of reference, we shall refer to these concepts (which together constitute the "qualified life" of an item of equipment, as defined in IEEE 323-1974) as the "pre-accident qualification duration" and the "post-accident qualification duration." From NECNP's interrogatories, it appears that the focus of this contention is the "post-accident qualification

duration." See interrogatories 14 and 29 of "NECNP's First Set of Interrogatories and Request for Documents to Applicants on Contentions I.A.2, I.B.1, I.B.2 and I.C" (filed 10/13/82).

As a matter of fact, the environmental qualification time duration standard for Seabrook Station electrical equipment is as follows: as to pre-accident qualification duration, the equipment in question is qualified either to the life of the plant or some shorter period, and if a shorter period is specified, then the equipment must be replaced or requalified before the period elapses. As to post-accident qualification duration, all equipment is qualified to withstand accident environmental conditions for one year (the conditions being those set forth in "Service Environment Chart", Figure 3.11(B)-1, at FSAR § 3.11), and any equipment that cannot be qualified for one year is then reviewed on a case-by-case basis to determine whether, for the particular duration that equipment is required to remain operational in the case of an accident in order to perform its safety function, a shorter period is

sufficient. This standard fully meets (and in substantial part it exceeds) the requirement as to equipment environmental qualification time duration of General Design Criterion 4.

One other point of potential confusion should be cleared up at the outset. Our response to Interrogatory No. 1 of NECNP's second set of interrogatories explained that our definition of the terms "important to safety" and "safety related" were the same and are used interchangeably to identify structures, systems, or components that perform a safety function. We further stated that there is no equipment designated "important to safety" but not safety-related. The following discussion, therefore, refers to all equipment that must be environmentally qualified to assure it will perform its safety function. We use the term "safety related" to refer to all such equipment.

Seabrook was in the preliminary design stage when Reg. Guide 1.89 was issued. Reg. Guide 1.89, which endorsed IEEE 323-1974, provided guidance for the first time on the requirement that electrical equipment be

qualified to withstand an accident environment after having been exposed to pre-accident conditions for a qualified duration. It was decided that all safety-related electrical equipment not supplied by Westinghouse under the Nuclear Steam Supply System ("NSSS") contract would be qualified to perform its safety function in the harsh environment. Rather than determine specific accident scenarios for each application, all equipment was specified for a 40-year normal life followed by one year post-accident conditions. This common post-accident duration permits generic qualification of identical equipment irrespective of the requirements of the actual application. There were several reasons we chose this method. First and most importantly, we were not "backfitting" an existing design. By that, we mean that we were not trying to show that an existing design was acceptable. We were designing and purchasing systems to meet the requirements of IEEE-323-1974 and Reg. Guide 1.89. We felt that we could eliminate any potential for error that might exist if one tried to identify specific operating durations and accident

environments for each piece of equipment. By this approach, we achieved the flexibility of locating equipment and interchanging like equipment among systems without jeopardizing equipment qualification. We also knew that in most cases, vendors either qualified equipment for in-containment accident conditions or for mild environments and there was little financial benefit to attempting to identify conditions which fell between these extremes.

The result is that much of our safety-related equipment will be qualified to operate longer and in environments that are more severe than is required by GDC 4.

As we proceed with our detailed review of the equipment qualification data packages, we may find equipment that cannot be qualified to operate for one year in the extreme environment. Should this situation arise, we will identify the time duration over which the equipment is required to perform its safety function, with a margin. That particular equipment will then be qualified to that specific duration. This

will be done on a case-by-case basis and will be so identified in the equipment qualification data file.

To date, we have not identified any equipment that cannot be qualified for one year of post-accident environmental conditions.

The NSSS safety-related electrical equipment is qualified using specific qualification times based on the accident scenarios for each specific equipment application. These are set forth in Table 3.11(N)-3 in FSAR § 3.11. Applicability to the Seabrook conditions will be verified by comparing the qualification profile to ensure that the test profiles envelope the Seabrook profile. This qualification meets or exceeds GDC 4, Reg. Guide 1.89, and IEEE 323-1974.

For these reasons, the Seabrook environmental qualification program meets or exceeds the requirements of GDC 4.

Classification Scheme

(Contention NECNP III.1 & NH-20)

Introduction

The Seabrook Station Radiological Emergency P. (SS-REP) includes a system for emergency recognition and classification as the basis for the activation of the Applicant's emergency response organization and the notification to and activiation of the emergency response organizations of the federal, state, and local authorities. Emergency conditions are categorized by the SS-REP into one of the following four emergency classes:

1. Unusual Event
2. Alert
3. Site Area
4. General

These four emergency classes are the same as those incorporated into federal, state, and local radiological emergency plans and which govern their response to an emergency notification by Seabrook Station personnel. This uniformity in the classification system used by the Applicants with that

used by the federal, state, and local authorities is in accordance with 10 CFR § 50.47(b)(4).

Classification System Descripton

The four emergency classes cover a graded scale of severity from a potential degradation of plant safety margins at the Unusual Event level to substantial core degradation or melting with potential for loss of containment integrity at the General Emergency level. An explanation of all four emergency classes to show this graded scale of emergency severity is given in Section 5.0 of the SS-REP.

That section, as well as its companion Appendix A of the SS-REP, was transmitted to the parties by PSNH letter to USNRC, SBN-525 entitled Emergency Classification System, dated June 27, 1983, and a copy is annexed to this testimony as Exhibit 1.[1] The

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[1] It has also been included in Amendment 49 of the FSAR, as indicated in that letter. It should be noted that Table A.5 in Exhibit 1 contains certain typgraphical corrections from the document transmitted on June 27, 1983.

revised Section 5.0 describes the manner by which emergency conditions are categorized into the four emergency classes (Subsections 5.3.1 through 5.3.4). It also describes the use of the symptomatic approach being incorporated into the Emergency Operating Procedures (EOPs) by the Applicants as an aid to emergency recognition and classification by the operator. This is described in the revised Subsection 5.2 and is set forth in detail in the information specified in the revised Appendix A.

This symptomatic approach being utilized in the development of the Seabrook Station EOPs is a result of the more than three year effort of the Westinghouse Owners Group (which includes Seabrook Station representation) to improve the methods by which EOPs were based and written and operators oriented and trained to respond to emergency conditions. Improvement was jointly recognized by the industry and the NRC in the analysis of the TMI accident response. The Critical Safety Functions (CSFs) (those functions that must be successfully maintained to insure adequate safety margins), and the symptoms that represent

various degrees of challenge to them, were identified to concentrate the operators' attention on that information that is most important in an emergency situation.

As the materials in SS-REP Section 5.0 and Appendix A demonstrate, the Applicant's emergency classification system fully utilizes this symptomatic approach to CSF challenges. Subsection 5.2 describes the concept of color-coded status trees associated with the symptomatic EOPs and Appendix A shows in detail the relationship between status trees associated with the five CSFs and the categorization of the condition into one of the four emergency classes. These CSF status trees are used to 1) monitor station safety status, 2) alert operators to potential emergency conditions, and 3) direct operators to appropriate CSF restoration procedures. These CSF status trees are available on the Safety Parameter Display System of the plant computer and are also avialable as hard copy for backup.

Additionally, Appendix A also shows that events that are not represented, at least initially, as a CSF

threat are categorized into the emergency classification system. These types of events are occurrences that are mainly external to the systems and equipment associated with the maintenance of CSFs. In this way, the various challenges to CSFs and all the other appropriate actual or potential emergency conditions are incorporated into the Applicant's emergency classification system.

Conclusion

The information provided to the Shift Superintendent to recognize emergency conditions and categorize them in accordance with the emergency classification system as described above and presented in detail in the referenced submittal is a means to smoothly transfer between emergency condition recognition and the categorization and classification step. The incorporation of the symptom-based CSF status tree approach used for the EOPs into this classification system aids in this process.

Evacuation Times

(NECNP Contentions III.12 and .13)

This Board's Order of June 30, 1983, disposed of the entirety of this contention except for two respects. The Board has reframed the remaining aspects of the contention thus:

> "NECNP III.12/III.13 Evacuation Time Estimates
>
> "The evacuation time estimates provided by Applicants in Appendix C of the Radiological Emergency Plan are deficient in failing to include an estimate of:
>
> "1. the times for evacuation during adverse weather conditions deveoping on a busy summer weekend; and
>
> "2. the times for simultaneous evacuation of beach areas lying NE to SSE of the Seabrook site."

This testimony is limited to those two issues.

Attached hereto as Exhibit 2 is a study entitled "Evacuation Clear Time Estimates for Areas Near Seabrook Station" (updated 1981, revised July 1983). This is in essentially the same format as, and uses the same data and computer program as, Appendix C to the SS-REP contained in the FSAR, and it is, in fact, the study of which Appendix C is a condensed version. The differences between Exhibit 2 and Appendix C are

threefold: one, Exhibit 2 contains a somewhat fuller description of the evacuation time estimate methodology; two, Exhibit 2 contains an evacuation time estimate for simultaneous evacuation of the entire EPZ (under the same summer scenarios initially studied); and three, Exhibit 2 contains an evacuation time estimate for simultaneous evacuation of the entire EPZ under the peak weekend population-adverse weather scenario.[2]

The "entire EPZ" scenarios for the original summer cases produce evacuation clear times of 6 hours, 5 minutes for the summer weekend-fair weather scenario and 4 hours, 10 minutes for the summer weekday-fair weather scenario.

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[2]This last item is an estimate compiled in response to this Board's Order of June 30, 1983. The original "entire EPZ" case was run at the time the study was first performed, but it was not reported in SS-REP Appendix C because, at the time Appendix C was published, NUREG-0654/FEMA-REP-1, Rev. 0, did not require such a case and the effort was to portray that called for by FEMA-REP-1.

Exhibit 2 also contains a summer weekday-fair weather scenario for each sector. These, too, were not included in Appendix C because they were not called for by FEMA-REP-1.

The "entire EPZ" case for the summer weekend-adverse weather scenario yields an evacuation clear time of 9 hours, 15 minutes.

All of the "entire-EPZ" cases account for an evacuation of the beach area from NE to SSE, plus all other areas of the EPZ, at the same time.

While we have included this last estimate (i.e., summer weekend-adverse weather), we wish to point out that, in our judgment, it overstates the time that a real life evacuation would take under peak population, adverse weather conditions. The reason for this is that, in order to model this scenario, it was assumed that none of the people at the beach areas began to depart -- notwithstanding the degrading weather -- until the signal to evacuate was given. In real life, one of two things would happen: either (1) the weather would begin to degrade before the notification was given, in which case some people would begin to leave before the signal was given, or (2) the weather would not begin to degrade until after the evacuation notification was given, in which case the effects of adverse weather upon evacuation would not appear until

the evacuation was underway and at least partially completed. Either of these real life situations would produce lower evacuation times than the case actually modelled.

In all other respects, these estimates are based upon the same methodology, the same assumptions, and the same data as did those the results of which are described in SS-REP Appendix C.

PSNH

Public Service of New Hampshire

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bcc: J.P. Cady, A.C. Cerne, S.D. Floyd, J.H. Herrin, W.P. Johnson, G.F. McDonald, D.N. Merrill, D.E. Moody, NRC Chrono, H.T. Tracy(2), Projects-All, Projects-SLA, Ropes&Gray(Dignan/Ritsher/Ga, A.M. Shepard, J.W. Singleton, T.F. B7.1.2, G.S. Thomas, J.E. Tribble, UE&C(SB-15990), ASLB, R.J. Harrison, J.A. MacDonald, T. Fuller, P.L. Anderson, W.N. Fadden, A.E. Ladieu, G. Tsouderos, D.A. Maidrand

June 27, 1983

SBN- 525
T.F. B7.1.2

United States Nuclear Regulatory Commission
Washington, D. C. 20555

Attention: Mr. George W. Knighton, Chief
Licensing Branch No. 3
Division of Licensing

References:
(a) Construction Permits CPPR-135 and CPPR-136, Docket Nos. 50-443 and 50-444
(b) USNRC Letter, dated May 12, 1983, "Issuance of Supplement No. 1 to the Safety Evaluation Report (Seabrook Station, Units 1 and 2)," G. W. Knighton to R. J. Harrison
(c) PSNH Letter, dated April 14, 1983, "Response to Generic Letter 82-33, Supplement No. 1 to NUREG-0737," J. DeVincentis to D. G. Eisenhut

Subject: Emergency Classification System

Dear Sir:

In response to the open item delineated in Supplement No. 1 to the Safety Evaluation Report (Reference (b)), we have enclosed a new Section 5.0 of the Seabrook Station Radiological Emergency Plan which provides a conceptual description of the Emergency Classification System.

Please note that the Emergency Action Level setpoints (Tables A.1-A.5) and Emergency Status Indicators (Tables A.1-A.5) color combinations are tentative (some also indicate that setpoints and color schemes will be provided later). Section 5.0 should be reviewed in light of its conceptual nature. The Westinghouse Owners Group, as of this writing, is continuing to revise its Emergency Response Guidelines (ERGs) in response to the NRC review. These ongoing changes to the ERGs are expected to effect the Emergency Action Level setpoints and Emergency Status Indicators color combinations.

Setpoints and color combinations will be provided subsequent to completion of the ERGs and Seabrook Station Emergency Operating Procedures (Reference (c) commits to December 1983 for completion of Emergency Operating Procedures).

The enclosed Section 5.0 will be incorporated in OL Application Amendment 49.

Very truly yours,

YANKEE ATOMIC ELECTRIC COMPANY

John DeVincentis

John DeVincentis
Project Manager

ALL/pf

Enclosure

cc: Atomic Safety and Licensing Board Service List

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5.0 EMERGENCY CLASSIFICATION SYSTEM

5.1 Summary

An Emergency Classification System has been defined which categorizes a wide spectrum of component or system failures and other occurrences that could potentially reduce station safety margins. The incidents are categorized according to severity in the following four classes: Unusual Event, Alert, Site Area Emergency, and General Emergency.

These predetermined emergency classes are declared by Seabrook Station personnel. They assist emergency response organizations in determining the assessment, corrective and protective actions to be taken onsite and offsite.

Emergency classifications are based upon events identified by certain measurable and observable indications of station conditions. These indications of degrading station status are called Emergency Action Levels (EALs) and are listed in detail with their associated station conditions in Appendix A. These EALs aid the operator in emergency recognition and assure the first step is completed in emergency response. It must be recognized that if conditions warrant such action, the classification of the event may change as the incident increases or reduces in severity.

5.2 Symptomatic Approach to Classification

A symptomatic approach has been developed to assist the operator in emergency recognition and classification. In order to concentrate the amount of plant process data provided to the operator to that which is necessary for event classification, use is made of the color coded status trees that are associated with the sympton based Emergency Operating Procedures. These symptomatic status tree analyses allow the operator to recognize accident severity and concentrate on the appropriate corrective actions. Symptomatic status trees which relate to Seabrook Station EALs are provided along with a description of the approach in Appendix A.

5.3 Emergency Classes

5.3.1 Unusual Event

AN UNUSUAL EVENT INDICATES A POTENTIAL DEGRADATION OF STATION SAFETY MARGINS WHICH IS NOT LIKELY TO AFFECT PERSONNEL ON-SITE OR THE PUBLIC OFF-SITE OR RESULT IN RADIOACTIVE RELEASES REQUIRING OFF-SITE MONITORING.

Unusual Events are conditions which do not cause serious damage to the station and may not require a change in operational status. For a complete list of the Unusual Event conditions, refer to Appendix A, Table A.1.

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5.3.2 Alert

AN ALERT INDICATES A SUBSTANTIAL DEGRADATION OF STATION SAFETY MARGINS WHICH COULD AFFECT ON-SITE PERSONNEL SAFETY, COULD REQUIRE OFF-SITE IMPACT ASSESSMENT, BUT IS NOT LIKELY TO REQUIRE OFF-SITE PUBLIC PROTECTIVE ACTION.

Station response and off-site notification associated with this event classification assure that sufficient emergency response personnel, both on and off site, are mobilized and respond to event conditions. Actual releases of radioactivity which substantially exceed Technical Specification limits may be involved, and thus radiation monitoring and dose projection may be an integral portion of the emergency response required. For a complete list of Alert conditions refer to Appendix A, Table A.2.

5.3.3 Site Area Emergency

A SITE AREA EMERGENCY INDICATES AN EVENT WHICH INVOLVES LIKELY OR ACTUAL MAJOR FAILURES OF STATION FUNCTIONS NEEDED FOR THE PROTECTION OF THE PUBLIC.

The events included in this Site Area Emergency category represent a potential for off-site releases which could impact the public to the extent that protective actions may be necessary. For a complete list of Site Area Emergency conditions, refer to Appendix A, Table A.3.

5.3.4 General Emergency

A GENERAL EMERGENCY INVOLVES SUBSTANTIAL CORE DEGRADATION OR MELTING WITH POTENTIAL FOR LOSS OF CONTAINMENT INTEGRITY.

For a complete list of General Emergency conditions, refer to Appendix A, Table A.4.

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APPENDIX A

EMERGENCY CLASSIFICATION SYSTEM AND EMERGENCY ACTION LEVELS

Tables A.1 through A.4 represent the station conditions and associated Emergency Action Levels (EALs) that are categorized in accordance with the four emergency classes. The emergency conditions include a wide spectrum of events that represent varying degrees of threat to station personnel onsite or the public offsite. As the tables show, full use is made of the various degrees of challenge to the five Critical Safety Functions (CSFs);

1) Subcriticality,
2) Core Cooling,
3) Heat Sink,
4) RCS Integrity, and
5) Containment Integrity.

EALs relate levels of challenge to the CSFs and other numerous process parameter indicators of emergency conditions such as system pressures, liquid levels, radiation intensity, and temperatures, to appropriate emergency classifications.*

Symptom based status trees simplify the initial emergency classification process by relating the most critical safety parameter indicators directly to the EALs. The individual status trees for each of the five emergency conditions used in conjunction with Table A.5 assist the operator in emergency classification and also directs them to the appropriate Emergency Operating Procedures for mitigation of the incident. Symptomatic status trees indicating emergency conditions are available to the operator on the plant process computer and are displayed on SPDS. Hard copies of these status trees are also available.

The CSF status trees, Figures A.1 through A.5, are based on plant events which pose a threat to the safety status of the plant. Color coding is used to identify event priorities for the individual branches of the status trees as follows:

○ GREEN - The Critical Safety Function is satisfied - no operator action is called for.

◔ YELLOW - The Critical Safety Function is not fully satisfied - operator action may eventually be needed.

◕ ORANGE - The Critical Safety Function is under severe challenge - <u>prompt</u> operator action is necessary.

● RED - The Critical Safety Function is in jeopardy - <u>immediate</u> operator action is required.

(4

Table A.5 provides a cross reference which correlates the CSFs by color priorities and other emergency conditions by category with their appropriate emergency classes. For example, if parameters of critical safety function Number 2., core cooling, complete an orange branch of the appropriate fault tree then according to Table A.5, a Site Area Emergency classification is reached. However, if an orange branch of the Core Cooling status tree is completed in conjunction with a 3 (Heat Sink) red or orange or a 5 (Containment Integrity) red, then a General Emergency is reached as shown in Column 3 CSF/Combinations of Table A.5. Combinations of CSFs and other emergency conditions are correlated with emergency classes in column 4. Column 5 lists emergency conditions which constitute emergency classifications independent of Critical Safety Functions.

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* Numbers which have been provided are tentative and will be operationally verified. Numbers which have not been provided are to be calculated and verified, and will be provided at a later time.

(9)

TABLE

UNUSUAL EVENT CLASSIFICATION

Critical Safety Function/ Emergency Condition	Emergency Action Level	Emergency Status Indicators*
Core Cooling	o Core exit TCs less than 1200°F, RCS subcooling less then (later)°F, at least one RCP is running and RVLIS wide range is greater than; (later) % 4 RCP (later) % 2 RCP (later) % 3 RCP (later) % 1 RCP	2 yellow
	o No RCPs running, RCS subcooling less than (later)°F, core exit TCs is less than 700° and RVLIS narrow range greater than (later)%;	2 yellow
RCS Integrity	o RCS Cold Leg temperature greater than (later)°F and less than (later)°F and RCS coolant system has exceeded 100°F per hour cooldown rate;	4 yellow
	o RCS pressure greater than cold overpressure limit, RCS temperature less than 305°F, and RCS temperature decrease less than 100°F per hour;	4 yellow
Heat Sink	o Pressure less than 1255 psig in all SGs but greater than 1185 psig.	3 yellow
	o Pressure less than 1255 psig in all SGs with narrow range level not less than 84.5% in all SGs.	3 yellow
	o Pressure less than 1185 psig in all SGs with narrow range level less than 20% in all SGs.	3 yellow
Containment	o Containment radiation levels greater than (later);	5 yellow
Loss of Plant Process Computer	o As indicated or observed;	13.
Loss of Offsite AC Power	o As indicated or observed;	9.
Radiological Releases	o Releases exceeding Technical Specifications	6a.
Fire	o Fire within the station protected area which requires outside fire-fighting assistance;	11a.
Control Room Evacuation	o With control remaining at remote safe shutdown panel;	12a.
Airplane crash, Train Derailment or Explosion Onsite	o By observation;	14.
Offsite Medical Assistance Required for Contaminated and Injured Worker at Local Support Hospital	o Emergency transport of the worker to local support hospital;	15.
Loss of Onsite AC Power Capability	o As indicated or observed;	16.
Severe or Natural Phenomenon	o Response spectrum seismic unit triggered;	17a.
	o 50-year flood or low water level by observation or receipt of warning from offsite authorities;	17d.
	o Tornado observed onsite;	17e.
	o Hurricane observed onsite (sustained winds of (later) for (later) period of time;	17f.
	o Security compromises;	17h.
	o Technical Specifications surpassed causing shutdown.	18.

* To be used with Table A.5 and Figures A.1-A.4.

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TABLE A.2

ALERT CLASSIFICATION

Critical Safety Function/ Emergency Condition	Emergency Action Level	Emergency Status Indicators*
Subcriticality	o Intermediate range SUR is zero or positive in power range less than 5% when the reactor should be subcritical;	1 orange
RCS Integrity	o RCS cold leg temperature is less than (later)°F, RCS pressure-temperature point to right of limit A (see Figure A.4) and RCS temperature decrease greater than 100°F per hour;	4 orange
Core Cooling and Containment Integrity	o When containment radiation is greater than (later) and, either:	5 yellow
	No RCPs are running, core exit TCS are less than 700° and RVLIS narrow is greater than (later)%; or	2 yellow
	Core exit TCs less than 1200°F, RCS subcooling less than (later)°F at least one RCP is running, and RVLIS wide range is greater than: (later)% 4 RCP (later)% 2 RCP (later)% 3 RCP (later)% 1 RCP	2 yellow
Containment Integrity and Failure to Isolate Containment	o Failure to isolate containment as indicated by Phase A and Phase B isolation indication panels when containment radiation is greater than (later);	7 and 5 yellow
Radiological Releases	o 10 time Technical Specifications;	6b.
Fire	o Controlled fire which effects only one train of safety-related equipment with the potential for affecting the other train;	11a.
Severe or Natural Phenomenon	o Earthquake greater than containment foundation DBE alarm initiating a shutdown;	17b.
	o Tornado observed onsite which has degraded safety components.	17g.

* To be used with Table A.5 and Figures A.1-A.4.

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TABLE A.3

SITE AREA EMERGENCY CLASSIFICATION

Critical Safety Function/ Emergency Condition	Emergency Action Level	Emergency Status Indicators*
Subcriticality	o Power range is greater than 5% power when reactor should be subcritical;	1 red
Core Cooling	o Core exit TCs less than 1200°F but greater than 700°F, RCS subcooling less than (later)°F, and no RCPs are running, and RVLIS narrow range is greater than (later)%;	2 orange
	o RCS subcooling less than (later)°F, no RCPs running, core exit TCs less than 700°, and RVLIS narrow range less than (later)%;	2 orange
	o Core exit TCs less than 1200°F, and RCS subcooling less than (later)°F, at least one RCP running, and RVLIS wide range less than; (later)% 4 RCP (later)% 2 RCP (later)% 3 RCP (later)% 1 RCP	2 orange
Heat Sink	o Wide range level less than top of U-tubes in all SGs and total feedwater flow to SGs less than 470 gpm;	3 red
	o Wide range level less than top of U-tubes in all SGs and total feedwater flow to SGs greater than 470 gpm and pressure greater than 1255 psig in all SGs;	3 orange
	o Wide range level greater than top of U-tubes in at least one SG and pressure greater than 1255 psig in all SGs;	3 orange
RCS Integrity	o RCS temperature decrease greater than 100°F in last 60 minutes <u>and</u> RCS Pressure-Temperature ratio exceeds Limit A (see Figure A.4)	4 red
Containment Integrity	o Containment pressure greater than 52 psig;	5 red
	o Containment pressure less than 52 psig, but greater than 5 psig;	5 orange
	o Containment pressure less than 52 psig and containment sump not less then (later);	5 orange
Radiological Releases	o Radiological releases exceed EPA PAGs at Site Boundary;	6c.
Loss of all AC Power	o As indicated or observed;	8.
Loss of all DC Power	o As indicated or observed;	10.
Fire	o Uncontrolled fire which affects safety-related equipment;	11b.
Control Room Evacuation	o Evacuation of control room without control at remote shutdown panel;	12b.
Severe or natural Phenomenon	o Earthquake with potential impact on SSE.	17c.

* To be used with Table A.5 and Figures A.1-A.4.

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TABLE A.4

GENERAL EMERGENCY CLASSIFICATION

Critical Safety Function/ Emergency Condition	Emergency Action Level	Emergency Status Indicators*
Core Cooling	o Core exit TCs higher than 1200°F;	2 red
	o Core exit TCs less than 1200°F but greater than 700°F, RCS subcooling less than (later)°F, no RCPs running and RVLIS narrow range less than (later)%;	2 red
Core Cooling and Heat Sink	o Core exit TCs less than 1200°F, RCS subcooling less than (later)°F, no RCPs running, core exit TCs greater than 700°F and RVLIS narrow range greater than (later)% combined with heat sink wide range level less than top of U-tubes in all SGs and total feedwater flow to SGs less than 470 gpm; or	2 orange 3 red
	Combined with wide range level greater than top of U-tubes in at least one SG and pressure not less than 1255 psig in all SGs; or	3 orange
	Wide range level not greater than top of U-tubes in any SGs and total feedwater flow to SGs greater than 470 gpm, and pressure not less than 1255 psig in all SGs;	3 orange
	o Core exit TCs less than 1200°F, and RCS subcooling less than (later)°F, no RCPs running, core exit TCs less than 700°F, RVLIS narrow range less than (later)% combined with heat sink wide range level less than top of U-tubes in all SGs and total feedwater flow to SGs less than 470 gpm; or	2 orange 3 red
	Combined with wide range level greater than top of U-tubes in at least one SG and pressure not less than 1255 psig in all SGs; or	3 orange
	Wide range level not greater than top of U-tubes in any SGs and total feedwater flow to SGs greater than 470 gpm, and pressure no less than 1255 psig in all SGs;	3 orange
	o Core exit TCs less than 1200°F, and RCS subcooling less than (later)°F, at least one RCP running, RVLIS wide range less than: (later)% 4 RCP (later)% 2 RCP (later)% 3 RCP (later)% 1 RCP	2 orange
	Combined with heat sink wide range level less than top of U-tubes in all SGs and total feedwater flow to SGs less than 470 gpm; or	3 red
	Combined with wide range level greater than top of U-tubes in at least one SG and pressure not less than 1255 psig in all SGs; or	3 orange
	Wide range level not greater than top of U-tubes in any SGs and total feedwater flow to SGs greater than 470 gpm, and pressure not less than 1255 psig in all SGs;	3 orange
Core Cooling and Containment Integrity	o Containment pressure greater than 52 psig combined with core exit TCs less than 1200°F, RCS subcooling less than (later)°F, no RCPs running, core exit TCs greater than 700°F and RVLIS narrow range greater than (later)%; or	5 red and 2 orange
	o With core exit TCs less than 1200°F, RCS subcooling less than (later)°F, no RCPs running, core exit TCs less than 700°F, RVLIS narrow range less than (later)%; or	2 orange
	o With core exit TCs less than 1200°F, RCS subcooling less than (later)°F, at least one RC running and RVLIS wide range less than: (later)% 4 RCP (later)% 2 RCP (later)% 2 RCP (later)% 1 RCP	2 orange
Containment Integrity and Loss of all AC Power	o Containment pressure greater than 52 psig combined with loss of all ac power as indicated or observed;	5 red and 8.
	o Containment pressure less than 52 psig but greater than 5 psig combined with loss of all ac power as indicated or observed;	5 orange and 8.
Containment Integrity and Failure to Isolate Containment	o Containment pressure greater than 52 psig combined with loss of all ac power as indicated or observed;	5 red and 7.
Loss of All AC and DC Power	o As indicated or observed.	8 and 10.

* To be used with Table A.5 and Figures A.1-A.4.

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TABLE A.5

EMERGENCY CLASS CROSS REFERENCE WITH CRITICAL SAFETY FUNCTIONS (CSFs)/EMERGENCY CONDITIONS

Emergency Class	CSF Singles	CSF/Combinations	Complications	Miscellaneous
General	2 red	2 orange/3 red 2 orange/3 orange 2 orange/5 red	5 red/8 5 orange/8 3 red/8 5 red/7	10+8
Site Area	1 red 2 orange 3 red 3 orange 4 red 5 red 5 orange			6c, 8, 10, 11b, 12b, 17c
Alert	1 orange, 4 orange	2 yellow/5 Yellow	5 yellow/7	6b, 11a, 17b, 17g
Unusual Event	2 yellow 3 yellow 4 yellow 5 yellow			6a, 9, 11c, 12a, 13, 14, 15, 16, 17a, 17d, 17e, 17f, 17h, 18

Revised from SB?-575

Critial Safety Functions

1. Subcriticality
2. Core Cooling
3. Heat Sink
4. RCS Integrity
5. Containment Integrity

Miscellaneous Emergency Conditions

6. Hi Releases
   a) Technical Specification
   b) 10 x Technical Specification
   c) Indications EPA PAGs will be exceeded at site boundary
7. Failure to isolate containment
8. Loss of all ac power
9. Loss of offsite ac power
10. Loss of all dc power
11. Fire:
   a) Controlled-affects only one train of safety-related equipment with the potential for affecting the other train
   b) Uncontrolled-affects safety-related equipment
   c) Within plant protected area which requires outside fire-fighting assistance
12. Control room evacuation
   a) With control at remote shutdown panel
   b) Without control at remote shutdown panel
13. Loss of plant process computer
14. Observation of aircraft crash, train derailment or explosion onsite
15. Emergency transport of contaminated and injured worker to local support hospital
16. Loss of onsite ac power capability
17. Severe or natural phenomenon
   a) Response spectrum seismic unit triggered
   b) Earthquake greater than containment foundation DBE alarm levels
   c) Earthquake with potential impact on SSE
   d) 50-year flood or low water level by observation or receipt of warning from offsite authorities
   e) Observation of tornado onsite
   f) Observation of hurricane onsite (sustained winds of (later) for (later) period of time)
   g) Tornado observed onsite which has degraded safety components
   h) Security compromises
18. Shutdown-Technical Specifications surpassed

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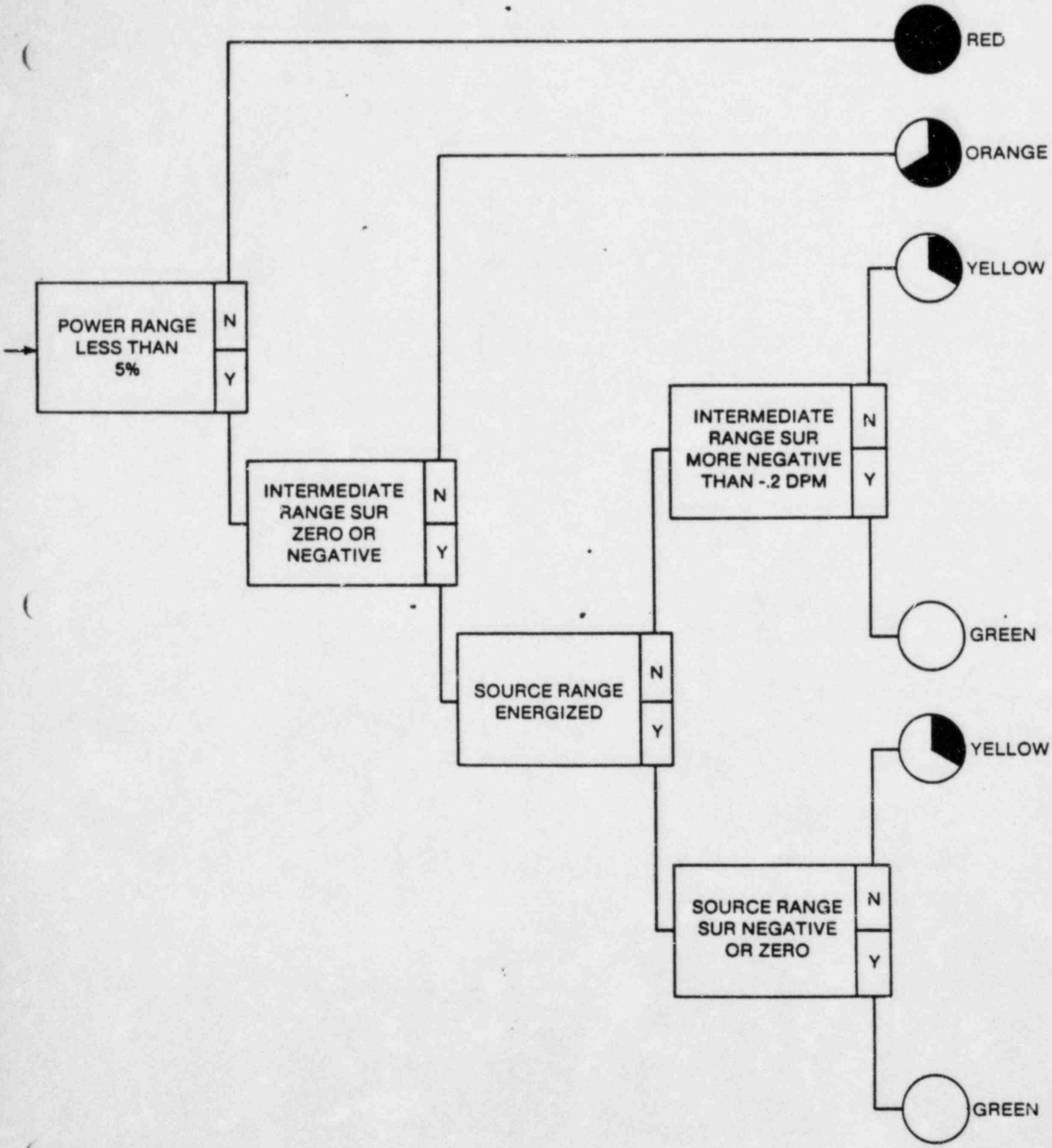
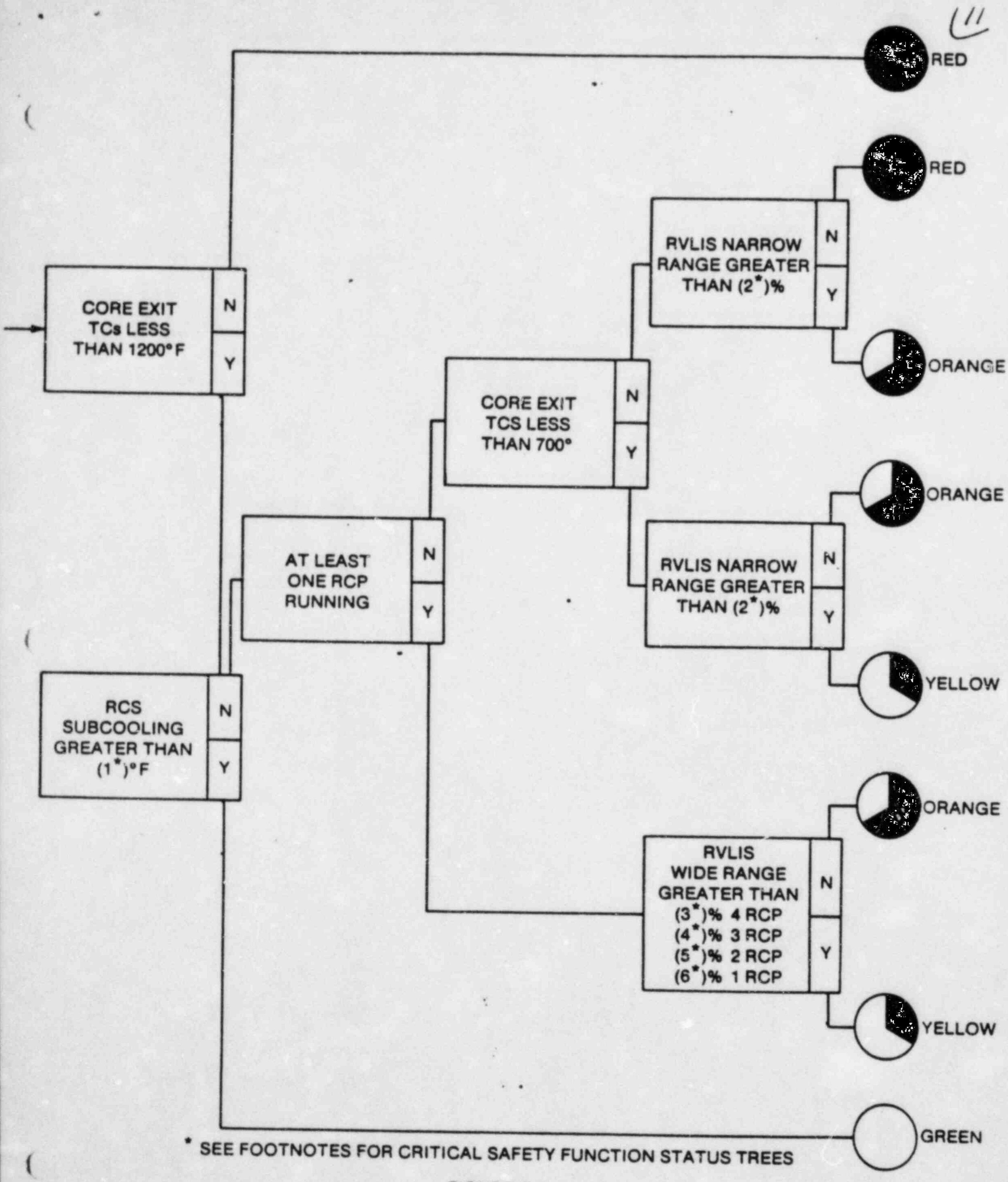


FIGURE A.1

STATUS TREE FOR CRITICAL SAFETY FUNCTION
NUMBER 1 - SUBCRITICALITY



* SEE FOOTNOTES FOR CRITICAL SAFETY FUNCTION STATUS TREES

FIGURE A.2

STATUS TREE FOR CRITICAL SAFETY FUNCTION
NUMBER 2 - CORE COOLING

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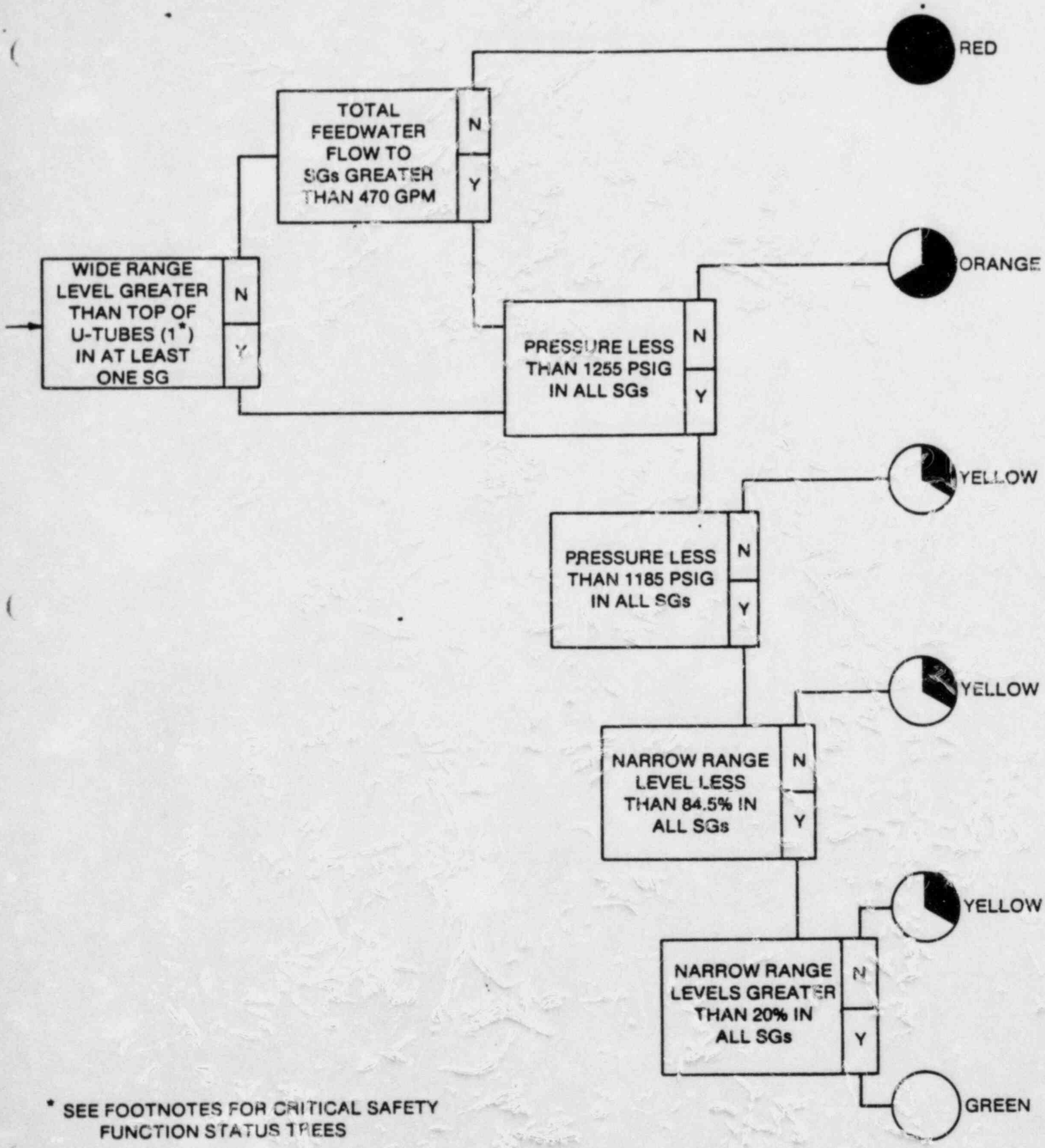
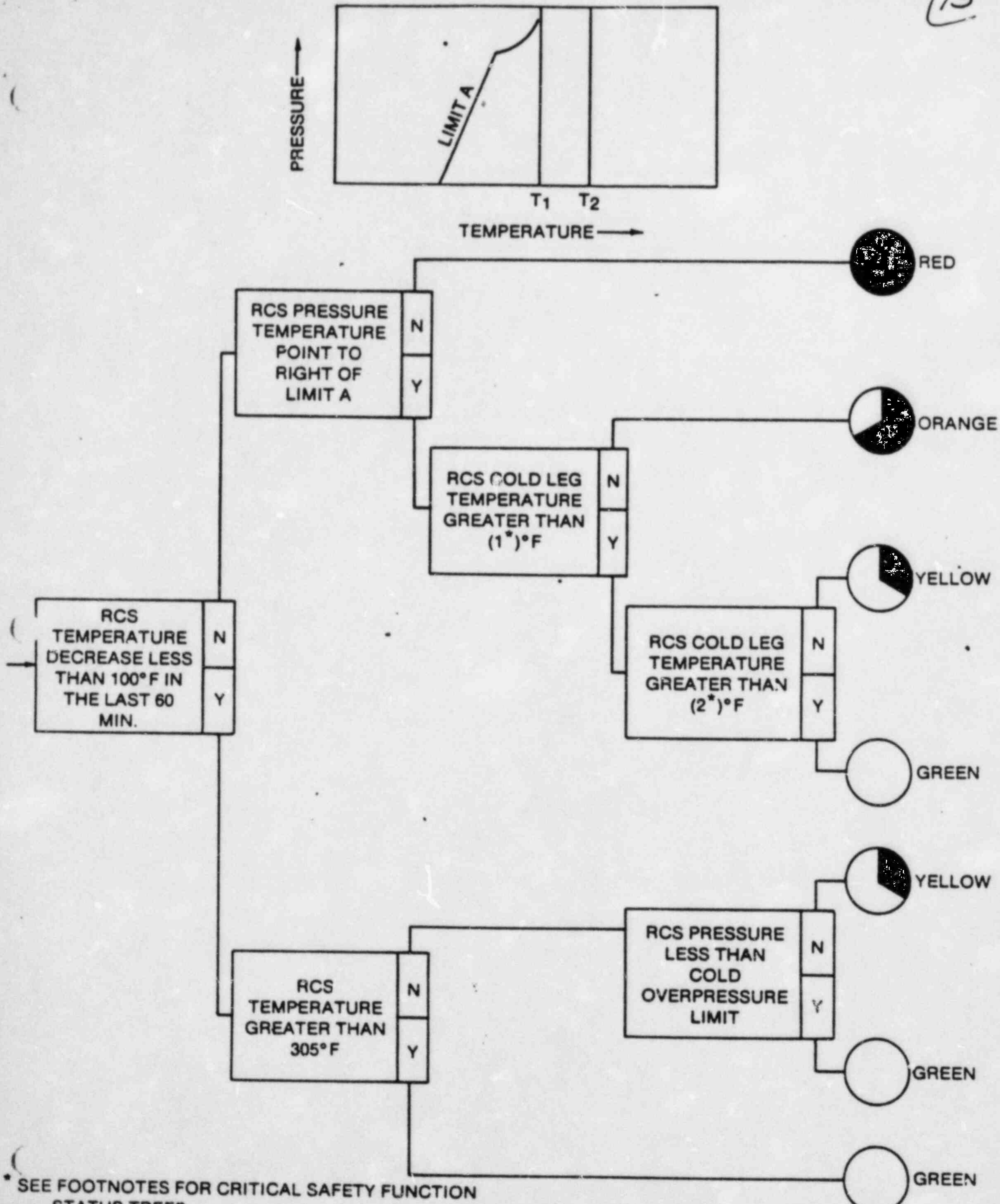


* SEE FOOTNOTES FOR CRITICAL SAFETY FUNCTION STATUS TREES

FIGURE A 3

STATUS TREE FOR CRITICAL SAFETY FUNCTION
NUMBER 3 - HEAT SINK



* SEE FOOTNOTES FOR CRITICAL SAFETY FUNCTION STATUS TREES

FIGURE A.4

STATUS TREE FOR CRITICAL SAFETY FUNCTION NUMBER 4 - RCS INTEGRITY

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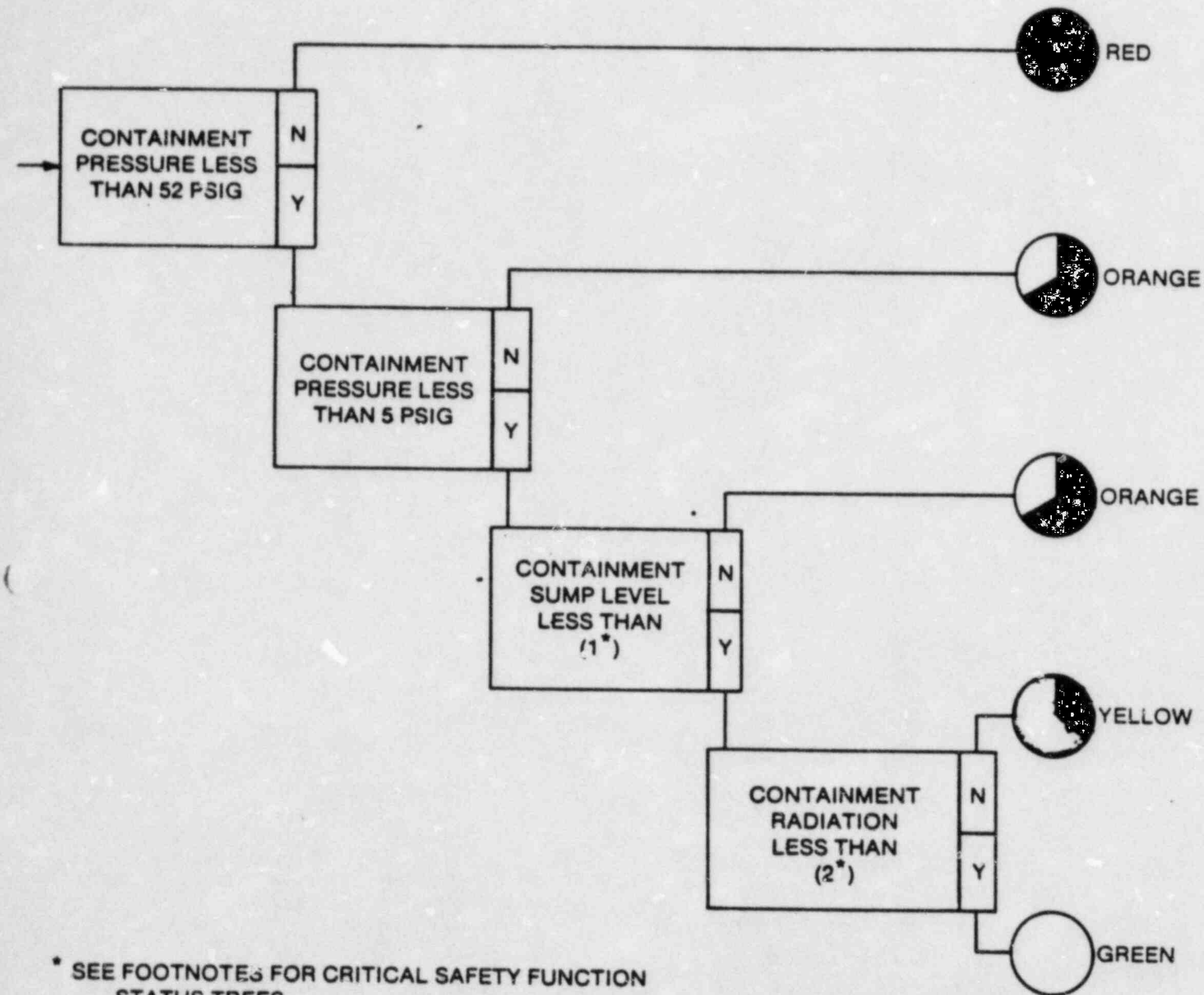

* SEE FOOTNOTES FOR CRITICAL SAFETY FUNCTION STATUS TREES

FIGURE A.5

STATUS TREE FOR CRITICAL SAFETY FUNCTION NUMBER 5 - CONTAINMENT

FOOTNOTES FOR CRITICAL SAFETY FUNCTION STATUS TREES

Figure A.2 Core Cooling

(1) Enter sum of temperature and pressure measurement system 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.

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

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

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

Figure A.3 Heat Sink

(1) Actual indicated level corresponding to top of U-tubes is dependent on calibration of wide range channel performed prior to startup.

Figure A.4 RCS Integrity

(1) Enter plant specific temperature corresponding to temperature T_1 (refer to FR-P.1 background document).

(2) Enter plant specific temperature corresponding to temperature T_2 (refer to FR-P.1 background document).

Figure A.5 Containment

(1) Enter plant specific level corresponding to the combined volumes of: RWST + Accumulators + RCS + 1/2 CST (to be calculated at a later date).

(2) Enter plant specific value corresponding to radiation level alarm setpoint for post accident containment radiation monitor (to be set prior to initial plant operation).

CERTIFICATE OF SERVICE

I, R. K. Gad III, one of the attorneys for the Applicants herein, hereby certify that on July 15, 1983, I made service of the within "Applicants' Direct Testimony No. 1" by mailing copies thereof, postage prepaid, to:

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