Westinghouse Owners Group

EMERGENCY RESPONSE GUIDELINES VALIDATION PROGRAM

Westinghouse Owners Group

SEABROOK STATION

WESTINGHOUSE

Nuclear Technology Division

PROGRAM PLAN

WESTINGHOUSE OWNERS GROUP

EMERGENCY RESPONSE GUIDELINES REVISION 1

VALIDATION PROGRAM PLAN

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VALIDATION PROGRAM PLAN

TABLE OF CONTENTS

Topic	Page
1.0 Introduction	5
2.0 Objectives	8
3.0 Validation Process	10
3.1 Preparation Phase	10
3.1.1 Scope of Validation	10
3.1.2 Validation Method	12
3.1.3 Validation Criteria	12
3.1.3.1 EOPs - Operator	14
3.1.3.2 EOPs - Control Room	14
3.1.3.3 EOPs - Training	16
3.1.3.4 Training - Operators	16
3.1.3.5 Training - Control Room	16
3.1.3.6 Operator - Control Room	20
3.1.3.7 Procedure - specific criteria	20
3.1.4 Procedure Writing	21
3.1.4.1 Validation Test EOPs	21
3.1.4.2 EOP Verification	24
3.1.5 Test Scenarios	24
3.1.6 Simulator Capability	27
3.1.7 Data Collection	32
3.1.7.1 Computer data	32
3.1.7.2 Videotape recordings	32
3.1.7.3 Transition flow charts	37
3.1.7.4 Debriefing sessions	37
3.1.8 Observation Teams	42
3.1.9 Test Crews	43
3.1.10 Training	43

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TABLE OF CONTENTS (Continued)

Topic		Page
3.2 Assessment Phase		47
3.3 Resolution Phase		50
3.4 Documentation Phase		50
4.0 Application to Plant-Specific Validation		50
5.0 References		51
Appendix A	Definitions	A-1
Appendix B	NUREG 0737, Item IC.1	B-1
Appendix C	Acceptance Criteria For Individual Guidelines	C-1
Appendix D	Example Run Sheets	D-1
Appendix E	References ERGs and Seabrook Test EOPs	E-1
Appendix F	Seabrook Plant Description	F-1
Appendix G	Example Lesson Plan	G-1

List of Tables

No	Title	Page
1	Validation Program Schedule	11
2	Evaluation Criteria for EOP - Operator Validation	15
3	Evaluation Criteria for EOP - Control Room Validation	17
4	Evaluation Criteria for EOP - Training Validation	18
5	Evaluation Criteria for for Training - Operator Validation	19
6	Seabrook Validation Test EOP List	22
7	Major Seabrook/ERG Reference Plant Differences	25
8	Validation Test Scenarios	28
9	ERG Rev. 1 Validation Test Parameters	33
10	Validation Training Schedule	45
11	Behavior Symptoms to Aid an Observer	48

List of Figures

No	Title	Page
1	Emergency Response Capability SYSTEM Interfaces	13
2	Expected EOP Usage In Test Scenarios	30
3	Magnetic Tape Data Block	36
4	Control Room Video Coverage	38
5	Composite Video Image	39
6	Transition Flow Chart For E-Series EOPs	40
7	Transition Flow Chart For F-Series EOPs	41

1.0 Introduction

On March 28, 1979, a commercial nuclear powered electrical generating plant located at Three Mile Island, near Harrisburg, Pennsylvania, experienced the most severe accident in the history of commercial nuclear power. Although physical equipment damage in the plant was limited almost exclusively to the reactor core, and radiation release to the environs was well within current licencing requirements, the event was viewed as indicative of a serious inability of in management of serious plant accidents.

A series of regulatory guidance and requirements followed, ranging in scope from improved instrumentation and diagnostic tools, to control room staffing and offsite emergency support facilities. Utilities operating nuclear power plants joined together to form "owner's groups," each group having in common the generally uniform system layout and design philosophy of a common NSSS supplier. This similarity in plant design allowed groups of utilities to respond to the Three Mile Island regulatory requirements in a generic manner, resulting in reduced cost to member utilities, and simplifying the regulatory/review process by providing generic response.

One of the more significant items of regulation was NUREG-0737, "Classification of the TMI Action Plan Requirements (Nov 1980). Within that document, item I.C.1, calls for improved emergency operating procedures, and lists certain specific requirements of those procedures. (Appendix B) Generic letter 82-33 (supplement 1 to NUREG-0737) subsequently reaffirmed schedular requirements and encouraged prompt utility implementation.

The Westinghouse Owners Group (WOG) responded to this particular requirement by generating a comprehensive package of generic operating instructions for coping with plant emergencies. The package, referred to as the Emergency Response Guidelines (ERGs), provides operating instructions for two distinct types of situations: For those events which can be diagnosed by an unambiguous set of symptoms, specific guidelines have been developed to allow optimal plant recovery (Optimal Recovery Guideline - ORGs). For other events

and/or malfunctions which are not amenable to diagnosis, a separate set of guidelines is provided whose function is to maintain satisfied a limited set of Critical Safety Functions (Function Restoration Guidelines - FRGS). The set of Critical Safety Functions was defined to be those necessary to protect the three primary boundaries preventing fission product release: The fuel matrix/cladding, primary system boundary, and containment building. The ORGs inherently contain instruction steps to maintain all of the Critical Safety Functions satisfied. However the FRGs only contain actions to restore and maintain the Critical Safety Functions satisfied, and contain limited guidance, if any, on recovering the plant.

In response to any protection system or safeguards system actuation, the Optimal Recovery Guidelines are entered first to obtain a diagnosis and perform the subsequent optimal recovery. Whether the event is diagnosed or not, the status of the Critical Safety Function (CSF) set is monitored using a corresponding set of Status Trees. Each tree consists of a series of binary decision points (branches) leading to a unique status condition for the CSF based on existing plant symptoms. Each unique status condition is color-coded to define the required action level, and all conditions other than "satisfied" provide a transition to the appropriate FRG. Details on status tree usage, action level priorities, and FRG implementation are contained in a separate users guide.

The ERGs are solidly based on detailed systems response and plant transient analyses, coupled with sound engineering judgement. They were subjected to detailed review by a WOG subcommittee consisting of operationally experienced members from several different utilities prior to issuance. However the initial set of ERGs, called "BASIC," was developed over an extended period of time (approximately 2.5 years). Over that period, additional analysis, experimental and operational data, and personnel changes, necessarily resulted in some differences between the later guidelines and earlier ones.

In response to available guidance at the time, the BASIC set of ERGs was subjected to a thorough validation test program in June of 1982. Details of that test program and the major conclusions are summarized in Reference 1. As a consequence of that initial validation test program, and also to incorporate accumulated utility/vendor/industry input, a revision to the ERGs was authorized by the WOG. Revision 1 was from the outset extremely well documented and laboriously reviewed to assure that every potential change for BASIC was duly addressed. As a result, the Revision 1 set is considered to be a much improved and better structured product than BASIC set.

As a final demonstration of the effectiveness of Revision 1 to the ERGs, the guidelines will be subjected to a comprehensive validation test program as described in the remainder of this document. Whenever possible, the program will follow appropriate guidelines available to the nuclear industry, and use previously published terminology and definitions. (Appendix A)

2.0 Objectives

The originally stated program objectives were very brief and are paraphrased here:

1. Validate Revision 1 of the ERGs
2. Compare Revision 1 to BASIC in light of the validation results
3. Document the validation program in such a manner that it might be referenced by any plant writing EOPs based on the ERGs

These objectives have been expanded to better utilize all of the information being generated by the program, and also to allow correlation with the separate elements of System Validation as described in Reference 3.

It must be realized that the ERGs are generic in nature and contain no plant specific data other than general system characteristics of a reference plant. In order to be tested, they must first be converted into Emergency Operating Procedures (EOPs). Since this conversion can be made with minimal perturbation to the structure, wording, logic, and usage of the ERGs, the validation results obtained can be claimed to be just as applicable to the ERGs as to the EOP set actually tested. Once written, the EOP set will need to be verified. Verification is the process which assures that the EOPs are correctly written (as required by the EOP Writers Guide) and contain the correct plant-specific technical data (as obtained from plant design, license, or test data). Validation is the process which demonstrates that actions specified in the EOPs can be followed by trained operators to manage emergency conditions in the plant.

This last definition mentions several additional aspects which are integral with the concept of EOP usage. First, there must be operators to perform the required actions. Then there must be training, so that the actions are performed as intended. And finally, there is the plant which is being controlled. For the operators, the plant exists (primarily) as it is monitored and operated from the control room.

The inseparability of these items during actual EOP usage in response to a plant emergency has resulted in the definition of SYSTEM validation: the overall evaluation of the man/machine system to determine that it works together to accomplish the desired results. The validation test program described below will exercise the operator/training/EOP/control room SYSTEM for a broad range of emergency situations.

From the test data, conclusions about the validation of each SYSTEM element will be reached. For the purposes of this program, emphasis will be placed on the EOP (ERG) and training aspects of the SYSTEM.

3.0 Validation Process

This section will describe the various aspects of the actual validation test as it relates to the operator/training/EOP/control room SYSTEM. The format is derived from on the guidance presented in Reference 3. Certain modifications have been made, however, to maintain the emphasis on ERGs as generic guidance, and to reduce the plant specific aspects. Table 1 presents the overall program schedule.

3.1 Preparation Phase

This phase identifies the resources required for the program, selects the method to be used, develops test scenarios, and determines the extent and application of validation criteria.

3.1.1 Scope of Validation

This test program will evaluate the effectiveness of the operator/training/EOP/control room SYSTEM in responding to a selected set of major plant transients. The EOPs to be used will be based on, and closely resemble, the reference ERG Revision 1 set, and the training involved will be specially developed for the same Revision 1.

Table 1

Validation Program Schedule

WOG Authorization	5-25-83
Preparation Phase	5-26-83 to 10-23-83
Operator Training	10-24-83 to 10-28-83
Validation Phase	10-31-83 to 11-4-83
Resolution Phase	11-5-83 to 12-31-83
Documentation Phase	12-15-83 to 1-15-84

3.1.2 Validation Method

The actual validations will be done on a plant-specific, full-scale control room simulator. A normal operating crew complement will use simulator-specific EOPs to guide their actions in response to control room (plant/simulator) indications during major plant casualties. Detailed observations will be made of operator performance, procedure usage, and plant response, by a specially trained observation team.

3.1.3 Validation Criteria

Since the stated objective of this SYSTEM validation is to test how well the separate elements work together to mitigate major plant transients, it follows that criteria are required for each element's interaction with each other. This implies the following six areas of interaction:

1. EOPs - Operator
2. EOPs - Control room
3. EOPs - Training
4. Training - Operator
5. Training - Control room
6. Operator - Control room

These interfaces are well-shown on the pyramidal structure shown in Figure 1 (taken from Reference 3). Furthermore, the effectiveness of the integral SYSTEM, with perhaps compounding, perhaps offsetting deficiencies, must have some criteria for acceptability. This final, global, objective is the most straightforward: The plant must be placed in a "safe," "stable" condition, regardless of imposed structural and equipment failures. "Safe" in this context means the reactor is adequately shut down (subcritical) and cooled. "Stable" means either steady-state, or changing in response to operator control (i.e., controlled cooldown). Satisfaction of this objective can be determined completely by monitoring process parameter trends (and other internal simulator computer variables).

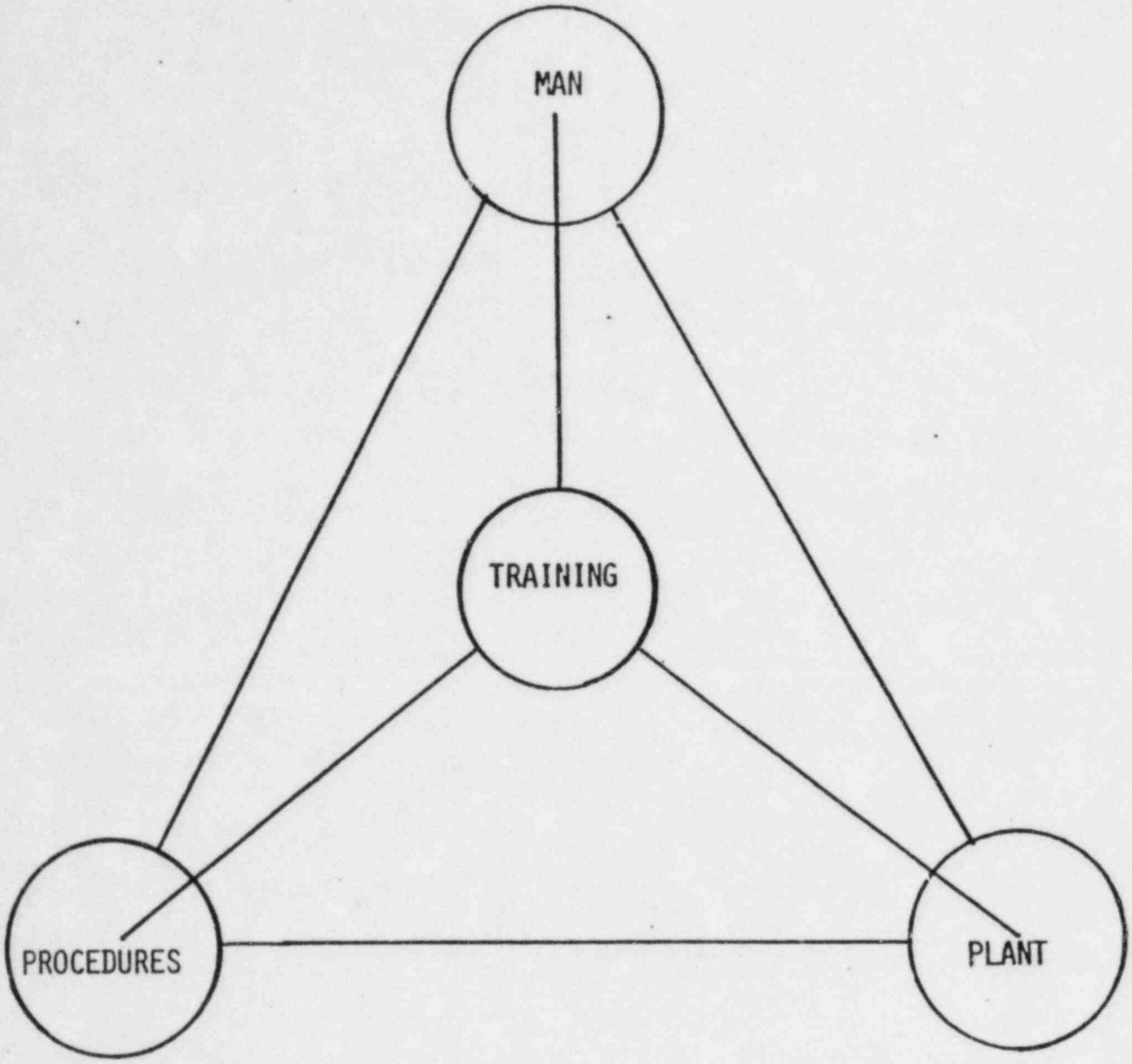


Figure 1 Emergency Response Capability SYSTEM Interfaces

Of the six areas of interaction within the SYSTEM, it should be noted that five concern EOPs and training, which are of primary concern in this Program. the sixth involves interaction between the operator and the control room, which is of only secondary concern in this Program.

3.1.3.1 EOPs-Operator

The first interaction, EOPs - operator, is concerned with the ability of the operator to use the EOPs. This implies that the EOPs properly convey what is intended at each step in a manner understood by the operator. This aspect of the EOPs was referred to as "human factors" in the BASIC ERG validation program, and validity was indicated if the operator was able to "accomplish the required tasks in an orderly and efficient manner." Detailed criteria to identify any deficiencies in the EOPs were presented in Reference 2, under the headings "level of detail" and "understandibility" and Reference 3. A composite list specifically for ERG - based EOPs is presented in Table 2.

3.1.3.2 EOPs - Control Room

This interaction is concerned with the correct incorporation of the plant, as seen and controlled by the operators in the control room, into the EOPs. In the BASIC validation, this aspect was part of "human factors" and required that "instrumentation and control devices which are referrenced actually exist in the control room and are correctly indentified." Specific criteria to identify deficiencies were presented in Reference 2 under the headings "plant compatibility" and "operator compatibility", and Reference 3. Although this Program is trying to de-emphasize the "plant" aspect, it is appropriate to demonstrate that plant-specific modifications can be properly made in ERG-based EOPs without interfering with the basic structure and usage. A composite list of acceptance criteria for the ERG-based EOPs is presented in Table 3.

Table 2

Evaluation Criteria For EOP-Operator Validation

- o Did each step contain sufficient information?
- o Were alternative actions explicit (use of "OR")?
- o Were contingency actions sufficient (RNO)?
- o Were procedures easily identified?
- o Could procedure transitions be made correctly
  - out of a procedure?
  - into another procedure at the correct step?
  - within a procedure?
- o Was the organization of the EOP set understood?
- o Was the organization within a procedure understood?
- o Were CAUTIONs and NOTEs recognized and understood?
- o Were internal procedure loops performed correctly?
- o Was the typeface easy to read?
- o Was the two-column format easy to use?
- o Were the LOGICAL statements understood (RNO)?
- o Could the foldout page be accessed and used properly?
- o Could Figures and Tables be read accurately?
- o Were CSF Status Trees properly monitored and used to control procedure (FRG) implementation?
- o Did CRT implementation of Status Trees conform to expected rules of usage?

3.1.3.3 EOPs - Training

This interaction is concerned with the presentation and emphasis placed on the EOPs in the operator training program. As part of this Validation Program, a special ERG training package was developed to respond to the criteria listed in Table 4. Details of the training program are described in a later section.

3.1.3.4 Training - Operator

This interaction is concerned with the actual training given to the operators. It assumes that the training program materials already contain complete and accurate information on both the EOPs and control room operation. The criteria for validating this interface are listed in Table 5, and were used to develop the initial ERG/operator training program lesson plans. For the purposes of this Program, it was asumed that the operators were already familiar with normal plant operations in the control room. Consideration of administrative, maintenance, and testing procedures is specifically excluded for the purposes of this Program.

3.1.3.5 Training - Control Room

This interaction involves the complete coverage of plant operations from the control room in the operator training program. Since this aspect of the operator training was assumed to be already complete, no special attempt to validate it will be made as part of this Program. However, deficiencies may be noted in SYSTEM performance due to some weakness in this area, and will be so noted in the Program findings.

Table 3

Evaluation Criteria For EOP-Control Room Validation

- o Were plant conditions (symptoms) used in the EOPs readily available to the operators?
- o Could the CSF Status Trees be properly monitored?
- o Were plant conditions consistent with EOP assumptions?
- o Were instruments and controls referenced by the EOPs available in the control room?
- o Were instruments able to be read to the accuracy required in the EOPs?
- o Could the required actions be performed by the control room crew (staffing)?
- o Could proper step sequence be maintained (control room layout)?
- o Did individual operators understand areas of cognizance in performing actions?
- o Was communication to local operators available as required?
- o Were the EOPs physically usable in the Control Room?
- o Could the EOPs be readily distinguished from other plant procedures?

Table 4

Evaluation Criteria for EOP-Training Validation

- o Are the following topics included in the Training program:
  - EOP structure?
  - EOP basis?
  - EOP usage as a set?
  - Individual procedure usage?
  - Use of NOTES and CAUTIONS?
  - Entry conditions?
  - Transitions?
  - Barrier Concept?
  - Critical Safety Functions?
  - Status Trees?
  - Status Tree usage?
  - Priorities of colors?
- o Are individual guidelines discussed in detail?
- o Is sufficient text material available for the operators to research any specific step?
- o Is a control room simulator available to exercise the EOPs?

Table 5

Evaluation Criteria For Training-Operator Validation

o Were lesson plans available for all topics listed under EOP-Training and Training-Control Room Validation criteria?

o Were knowledgable instructors available for all topics?

o Were all lessons presented in a timely fashion?

o Was a feedback mechanism (exam) employed to check operator understanding?

o Was a control room simulator employed to give real-time experience in EOP usage?

o Were all EOPs presented to the operators in some form to assure familiarity?

3.1.3.6 Operator - Control Room

This interaction is commonly referred to as the man-machine interface. Again, this Program does not intend to make Human Engineering judgements about the adequacy of the control room used for the test. Seabrook Station has performed a control room design review. The results of this review have been recorded and submitted to the NRC. All of the identified deficiencies have been evaluated for corrective changes. however, these changes have not yet been implemented on the main control board. If any deficiency in system performance is attributed to control room design, it will be so noted in the Program findings. These findings will first be screened with the present HEDs, and if not already identified, they will be added to the existing list and evalauated by the utility.

3.1.3.7
Procedure-Specific-Criteria

Because the primary objective of this Program is validation of the ERGs, special criteria have been developed for each individual procedure expected to be exercised on the simulator. These criteria encompass the two areas "EOP-Operator," and "EOP-Control Room," and can be evaluated for an individual guideline by using recorded observations and process parameter data. The entire set of procedure-specific acceptance criteria is presented in Appendix C.

3.1.4 Procedure Writing

The effort to convert the ERG Revision 1 set into plant-specific EOPs was performed by the Operations staff at Seabrook Station. Evaluation of plant-specific instrument settings and uncertainties were generated by the plant and utility engineering staff, while interpretation of required plant-specific additions were made in conjunction with the training staff.

3.1.4.1 Validation Test EOPs

A listing of the complete set of EOP titles is presented in Table 7. Every effort was made to keep the final EOPs as unchanged from the reference ERGs as possible to facilitate transfer of validation results. Examples of two reference ERGs and the Seabrook Test EOPs are presented in Appendix E. The procedure writing effort was facilitated by having a solid understanding of the BASIC version of the ERGs. Still, it is estimated that 2 man years of effort went into the present set of EOPs.

Table 6

SEABROOK VALIDATION TEST EOP LIST

E-0	Reactor Trip or Safety Injection	
	ES-0.0	Rediagnosis
	ES-0.1	Reactor Trip Response
	ES-0.2	Natural Circulation Cooldown
	ES-0.3	Natural Circulation Cooldown With Steam Void in Vessel (with RVLIS)
E-1	Loss of Reactor or Secondary Coolant	
	ES-1.1	SI Termination
	ES-1.2	Post-LOCA Cooldown and Depressurization
	ES-1.3	Transfer to Cold Leg Recirculation
	ES-1.4	Transfer to Hot Leg Recirculation
E-2	Faulted Steam Generator Isolation	
E-3	Steam Generator Tube Rupture	
	ES-3.1	Post-SGTR Cooldown Using Backfill
	ES-3.2	Post-SGTR Cooldown Using Blowdown
	ES-3.3	Post-SGTR Cooldown Using Steam Dump
ECA-0.0		Loss of All AC Power
ECA-0.1		Loss of All AC Power Recovery Without S.I. Required
ECA-0.2		Loss of All AC Power Recovery With S.I. Required
ECA-1.1		Loss of Emergency Coolant Recirculation
ECA-1.2		LOCA Outside Containment
ECA-2.1		Uncontrolled Depressurization of All Steam Generators
ECA-3.1		SGTR With Loss Of Reactor Coolant-Subcooled Recovery Desired
ECA-3.2		SGTR With Loss Of Reactor Coolant-Saturated Recovery Desired
ECA-3.3		SGTR Without Pressurizer Pressure Control

Table 6 (Cont)

SEABROOK VALIDATION TEST EOP LIST

F-0	The Critical Safety Function Status Trees
FR-S.1	Response to Nuclear Power Generation/ATWS
FR-S.2	Response to Loss of Core Shutdown
FR-C.1	Response to Inadequate Core Cooling
FR-C.2	Response to Degraded Core Cooling
FR-C.3	Response to Saturated Core Cooling Condition
FR-H.1	Response to Loss of Secondary Heat Sink
FR-H.2	Response to Steam Generator Overpressure
FR-H.3	Response to Steam Generator High Level
FR-H.4	Response to Loss of Normal Steam Release Capabilities
FR-H.5	Response to Steam Generator Low Level
FR-P.1	Response to Imminent Pressurized Thermal Shock Conditions
FR-P.2	Response to Anticipated Pressurized Thermal Shock Conditions
FR-Z.1	Response to High Containment Pressure
FR-Z.2	Response to Containment Flooding
FR-Z.3	Response to High Containment Radiation Level
FR-I.1	Response to High Pressurizer Level
FR-I.2	Response to Low Pressurizer Level
FR-I.3	Response to Voids in Reactor Vessel

3.1.4.2 EOP Verification

EOP Verification is defined as the comparative evaluation between the final EOPs and their source documents. Detailed guidance for performing this evaluation is presented in References 3 and 4.

The EOP Source documents are as follows:

- o Seabrook Station Writers Guide for Emergency Operating Procedures
- o Emergency Response Guideline Writers Guide, Revision 1
- o FSAR, Unit 1, Seabrook Station
- o Westinghouse Owners Group Emergency Response Guidelines - Revision 1
- o Seabrook Plant Description

Since the EOPs were written to minimize differences from the ERGs, and the ERGs were extensively reviewed for consistency with the ERG Writers Guide, the format comparisons were limited to random checks.

Major emphasis during the verification process was on the addition of plant specific information, and changes required because of differences in plant systems from the "reference" plant. A summary of the major systems for Seabrook was compiled into a "Seabrook Plant Description" which is included as Appendix F for reference. The detailed check against the ERGs was performed jointly by Seabrook and Westinghouse personnel, and any deviations were immediately noted and corrected. A brief summary of major differences between the Seabrook plant and the "Reference" plant assumed in the ERGs is presented in Table 7.

3.1.5 Test Scenarios

The general guidance provided in selecting event scenarios in Reference 3 was applied for this Program. The main objective of scenario selection was to exercise a major fraction of the EOP set. To accommodate and exercise the many branching paths within the set, variations of individual events were included. Events range from the most simple reactor trip, to multiple pressure boundary failures with coincident multiple equipment/control failures. In all cases, a sequence of initiating events was constructed to arrive at the final plant configuration.

Table 7

MAJOR SEABROOK/ERG REFERENCE PLANT DIFFERENCES

Reference HP Plant	Seabrook
1. AFW system has two motor-driven pumps and one turbine driven pump. Each motor-driven pump feeds 2 SGs while the turbine driven pump feeds all 4 SGs	1. EFW system has one motor-driven pump and one turbine driven Each pump is full capacity and feeds all 4 SGs. Startup feed pump can feed through EFW lines.
2. PRZR PORVs are air-operated	2. PRZR PORVs are electrically operated
3. SG PORVs are air-operated	3. SG PORVs are electrically operated
4. Air-operated valves inside containment require air supply from outside containment	4. Air-operated valves inside containment are powered by containment air compressor
5. RCP thermal barriers cooled by CCW	5. Separate thermal barrier cooling system has heat exchangers cooled by PCCW
6. Single ultimate heat sink	6. Ultimate heat sink is either Atlantic Ocean or mechanical draft cooling towers
7 Spray additive tank feeds spray pump discharge	7. Spray additive tank gravity feeds to RWST
8. Safety grade containment fan coolers	8. Containment fan coolers NOT safety equipment

Table 7 (Cont)

MAJOR SEABROOK/ERG REFERENCE PLANT DIFFERENCES

Reference HP Plant	Seabrook
9. Containment recirculation sumps are maintained empty.	9. Recirculaton sumps are maintained full. Containment (water) level instrument is qualified
10. SG narrow range level instruments are qualified (Wide range is not)	10. SG narrow range <u>and</u> wide range level instruments are qualified
11. RCS wide range pressure transmitters are located inside containment	11. Wide range RCS pressure transmitters are located outside containment
12. RVLIS has 3 ranges Full range, Dynamic head range, and Upper range.	12. RVLIS has 2 ranges. (does not have upper range)
13. Automatic switchover of AFW pump suction or low CST level.	13. Manual makeup to CST from Demineralized Water Storage Tank.

Details of the initial conditions, plant equipment status, failures to be initiated, and equipment to be made "inoperable" were summarized on individual "Test Scenario Run Sheets" for each individual test scenario. Additional information about timing of sequential failures and incorporation of unrelated "distractions" were also included on the Run Sheets.

The initial list of test scenarios was developed without consideration of actual simulator capabilities. Later versions were changed to accommodate simulator upgrades. The final list of Test Scenarios is presented as Table 8. Example Run Sheets are presented in Appendix D.

Since a stated objective of scenario selection was to exercise as many of the EOPs as possible, a matrix showing expected precedure usage is presented as Figure 2. The FRGs designated as "Y" in the figure summary indicate a YELLOW action priority, which means that action is at the operators discretion. Although the symptoms corresponding to entry conditions are expected to be exhibited frequently during the test scenarios, it cannot be stated, as in the case of the other EOPs, that the operator is expected to use them.

3.1.6 Simulator Capability

The Seabrook Control Room Simulator is an exact duplicate of the Seabrook Station Unit 1 Control Room. The entire simulator was supplied by the Link-Simulator division of Singer Corporation, and is located in the Training Center at the site in Seabrook, New Hampshire. The system configuration includes 2 GOULD/SEL model 32/55 computers and a 300 Megabyte disc. Also coupled to the plant simulation is a MODCOMP CLASSIC computer with 2,300 Megabyte discs, which duplicates the functions of the plant process computer.

Table 8

VALIDATION TEST SCENARIOS

1 Loss of offsite power - Reactor trip
1A Natural Circulation cooldown
2 Spurious SI

3 Small LOCA (1000 gpm)
3A Post-LOCA cooldown
4 Intermediate - size LOCA (5000 gpm)
4A Post-LOCA cooldown
5 DBA LOCA - No RHR pumps
6 ICC
7 Small LOCA plus subsequent SGTR
8 Small LOCA - No EFW
9 Small LOCA - No HHSI - Return to critical

10 Secondary break outside containment
11 Secondary break - All MSIVs fail to close
12 Secondary break plus subsequent secondary break
13 Secondary break - MSIV failure (all) - plus LOCA
14 Secondary break - MSIV failure (all) - plus SGTR
15 Secondary break inside containment - plus LOCA
16 Secondary break plus SGTR in faulted SG
17 Secondary break in 3 SGs plus SGTR
18 Secondary break in 2 SGs plus SGTR (1 intact)

19 SGTR
19A Post SGTR cooldown using backfill
20 SGTRs in different SGs (subsequent)

Table 8 (Cont)

VALIDATION TEST SCENARIOS

21	SGTR plus secondary break in non-ruptured SG
22	SGTR, loss of HHSI - Return to Critical
23	SG tube leak plus spurious SI
24	SGTR plus loss of EFW
25	SGTR plus secondary overpressure, all safety valves fail to close
26	Loss of all ac power
26A	Loss of all ac power recovery plus SGTR
27	ATWS from full power
28	Loss of all feedwater - power available
29	Loss of all feedwater - offsite power lost

TEST SCENARIO NUMBER (TABLE 8)

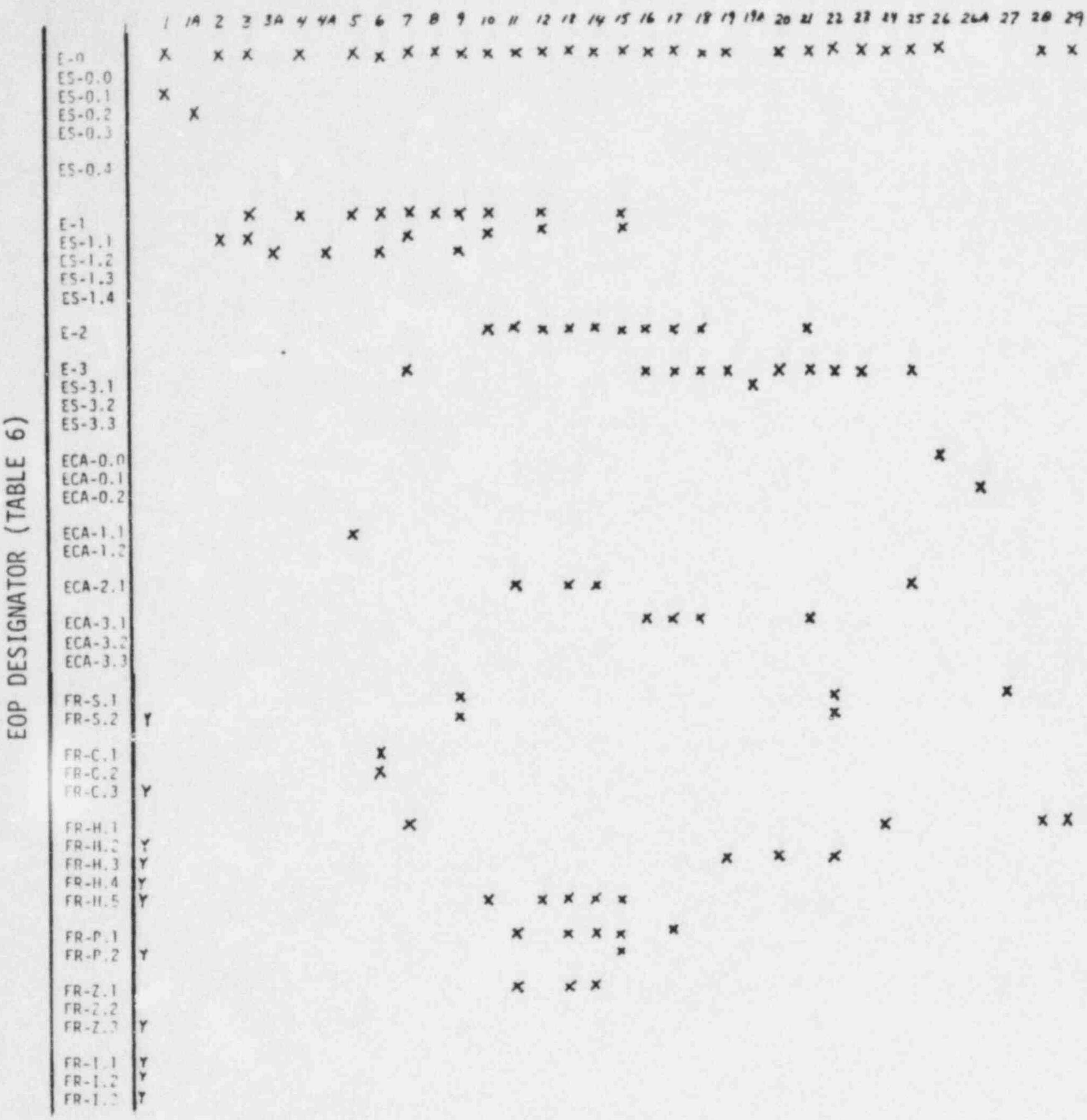


Figure 2 Expected EOP Usage in Test Scenarios

For training purposes, a broad range of malfunctions is available for use by the instructors. For the major event initiators, for example, the instructor has the capability to input:

- o a single LOCA, variable to some maximum size, or 2 other smaller, fixed - location LOCAs
- o a single steamline break inside containment
- o a single steamline break outside containment (upstream of the MSIV), and
- o a single tube rupture

These failures are consistent with the licensing basis "single-failure events" historically required for training prior to the TMI event. However, to exercise the new EOPs, and create the conditions required by the test scenarios, several changes were made in the software by Seabrook simulator support personnel.

- o The maximum size of the variable-size LOCA was increased.
- o Multiple steam generator failures (breaks inside and outside containment) can be input
- o MSIVs on any or all steamlines can fail to close.
- o SG tube ruptures can be input in multiple generators, and the maximum leak size has been increased.

The simulation software includes calculations of an "RCS void-fraction," effectively a system inventory parameter, which is displayed as equivalent reactor vessel level on a CRT display.

Another computer routine handles evaluation of the Critical Safety Function Status Trees, using the real-time plant process parameters. A separate display routine allows monitoring of the Status Trees directly on one of the main control board CRTs.

The MODCOMP (plant process) computer serves both information base and alarm handling functions. The information section of the computer has a graphics capability that allows dynamic representation of all the major systems of the plant as well as trending of individual points or groups of points on the

control room CRTs. Both types of graphics are controlled by the operators via keyboard commands. The alarm function presents the operator with prioritized CRT displays of the off-normal events to facilitate interpretation of severity.

3.1.7 Data Collection

Several different types of data will be collected during the validation test to support the multiple conclusions which will result.

3.1.7.1 Computer data

To preserve a record of plant behavior during each test scenario, a special software routine was written by Seabrook personnel to save a comprehensive list of process parameters and equipment status at 5 second intervals during each test run. The parameters are obtained from the simulator data "pool" and copied to available storage space on a 300 Megabyte disc. Following each test scenario, the same data is copied (off-line) to a magnetic tape. Data for each transient scenario will be kept on a separate tape. A listing of the parameters being recorded is presented in Table 9. An example of the tape format is shown in Figure 3.

3.1.7.2 Videotape recordings

Two permanently installed video cameras are already available in the Seabrook control room, and will be used during the test. Both camera images are recorded simultaneously (split screen) on tape to effectively cover most of the control board at all times. A separate digital time is superimposed on the frame to provide the time reference. The simulator instructor controls the video recorder, digital time clock, and computer data recording. In this way, the computer clocks and video tape clock can be synchronized for each run. Figure 4 shows the approximate control room coverage by the two cameras. Figure 5 is a sketch of the video image composite.

Table 9

ERG-REVISION 1 VALIDATION TEST PARAMETERS

System	Parameter Description	Number of Values	Type*
NIS	S.R. Count Rate	1	A
	S.R. Startup Rate	1	A
	I.R. Amps	1	A
	I.R. Startup Rate	1	A
	Nuclear Power	4	A
Pressurizer	N.R. Pressure	1	A
	Level	1	A
	Relief Line Temperature	4	A
	Relief Line Flow	1	A
	Liquid Temperature	1	A
	Steam Temperature	1	A
	Boron Concentration	1	A
	Spray Flow	1	A
	Auxiliary Spray Flow	1	A
	PORV Position	2	D
	PORV Block Valve Position	2	D
	Heater Status	4	D
Pressurizer Relief Tank	Pressure	1	A
	Level	1	A
	Temperature	1	A
RCS	W.R. Pressure	1	A
	Auctioneered Tavg	1	A
	Hot Leg Temperature	4	A
	Cold Leg Temperature	4	A
	Loop Flow	4	A
	Core Aug. Water Density	1	A
	Boron Concentration	1	A
	Subcooling (from TCs)	1	A
	Core Exit TCs	20	A
	Upper Range Vessel Level	1	A
	"WR" Vessel Level	1	A
	Core Decay Heat	1	A
CVCS	Makeup Flow to VCT	1	A
	Charging Flow	1	A
	Letdown Flow	1	A
	Seal Injection Flow	4	A
	Seal Leakoff Flow	4	A
	VCT Level	1	A
	Emergency Borate Flow	1	A

* A = analog
D = digital

Table 9 (Cont)

ERG-REVISION 1 VALIDATION TEST PARAMETERS

System	Parameter Description	Number of Values	Type*
Secondary	SG Pressure	4	A
	SG Steam Flow	4	A
	SG Main Feedwater Flow	4	A
	SG Emergency Feedwater Flow	4	A
	SG N.R. Level	4	A
	SG W.R. Level	4	A
	SG PORV Position	4	D
	Main Feedwater Isolation Valve Position	4	D
	MSIV Position	4	D
	MSIV Bypass Valve Position	4	D
	Steam Dump to Condenser-Demand	1	A
	Steam Header Pressure	1	A
	Steam Header Flow	1	A
	Feed Header Pressure	1	A
	Feed Header Flow	1	A
	Main Generator, MWe	1	A
	Condenser "A" Vacuum	1	A
	CST Level	1	A
	Main Feed Pump Status (Startup)	2 (1)	D
	Emergency Feed Pump Status	2	D
	Condensate Pump Status	3	D
Radiation Monitoring	RCS Activity	1	A
	SG Activity-Sample	4	A
	Containment Activity	1	A
	SG Blowdown Activity	1	A
	Condenser Air Discharge	1	A
	Auxiliary Building Radiation	1	A
	Stack	1	A
ECCS	RWST Level	1	A
	BIT Flow to RCS	1	A
	SI Pump Flow	2	A
	RHR Pump Flow	2	A
	Containment Spray Pump Flow	2	A
	Accumulator Pressure	4	A
	Charging Pump Status	2	D
	SI Pump Status	2	D
	RHR Pump Status	2	D
	PD Pump Status	1	D
	Containment Recirc. Sump Isolation Valve Status	2	D

Table 9 (Cont)

ERG-REVISION 1 VALIDATION TEST PARAMETERS

System	Parameter Description	Number of Values	Type*
PCCW	RCP Thermal Barrier Flow	4	A
	RCP Flow	4	A
	RHR Heat Exchanger Flow	2	A
	Flow to SI Pumps	2	A
	Flow to CCP	2	A
	Flow to RHR Pumps	2	A
Containment	Pressure	1	A
	Temperature	1	A
	Humidity	1	A
	Recirculation Sump Levels	2	A
	Normal Sump Levels	2	A
	Hydrogen	1	A
	Fan Cooler Status	6	D
	Control Air Isolation Valve Status	1	D
	Control Air Pressure Inside Cont.	1	A
Service Water	Fan Cooler Water Flow	4	A
	FC Inlet Temperature	1	A
	FC Outlet Temperature	1	A
	CCW HX Flow	2	A
	CCW HX Temperatire In	1	A
	CCW HX Temperature Out	1	A
Electrical	D/G Status	2	D
	Safeguards Bus Volts	2	A
	Safeguards Bus Amps	2	A
	Service Bus Volts	4	A
Break	RCS Leak Rate	4	A
	SGTR Leak Rate	4	A
	Steamline Break Flow	4	A
	Feedline Break Flow	4	A

HOUR

MINUTES

SECONDS

DATE

ANALOG DATA POINTS (206)

0 0 0 09/19/83

.491E+02	.491E+02	.489E+02	.491E+02	.489E+02	.489E+02	.592E+01
.595E+01	.595E+01	.592E+01	.833E+01	.777E+02	.781E+02	.781E+02
.777E+02	.829E+01	.740E+00	.744E+00	.740E+03	.744E+03	.592E+01
.595E+01	.000E+00	.100E+03	.131E+05	.200E+02	.151E+03	.305E+02
.443E+00	.443E+00	.443E+00	.269E-02	.513E+03	.512E+03	.513E+03
.513E+03	.513E+03	.513E+03	.513E+03	.512E+03	.513E+03	.513E+03
.513E+03	.513E+03	.513E+03	.513E+03	.513E+03	.513E+03	.513E+03
.513E+03	.513E+03	.513E+03	.513E+03	.513E+03	.513E+03	.513E+03
.512E+03	.513E+03	.513E+03	.512E+03	.513E+03	.513E+03	.512E+03
.513E+03	.513E+03	.513E+03	.512E+03	.513E+03	.512E+03	.512E+03
.512E+03	.512E+03	.512E+03	.512E+03	.512E+03	.512E+03	.512E+03
.512E+03	.512E+03	.512E+03	.512E+03	.512E+03	.512E+03	.512E+03
.512E+03	.512E+03	.512E+03	.512E+03	.512E+03	.512E+03	.000E+00
.000E+00	.493E+02	.335E+00	.343E+00	.000E+00	.000E+00	.000E+00
.278E+00	.278E+00	.278E+00	.278E+00	.000E+00	.000E+00	.156E+04
.000E+00	.000E+00	.000E+00	.000E+00	.116E+03	.124E+03	.204E+01
.000E+00	.000E+00	.000E+00	.000E+00	.000E+00	.000E+00	.000E+00
.000E+00	.152E+00	.160E+00	.163E+00	.157E+00	.204E+01	-.546E-10
.316E-07	.311E-07	.311E-07	.308E-07	.130E+04	.131E+04	.487E+02
.487E+02	.134E+00	.000E+00	.567E+03	.567E+03	.567E+03	.567E+03
.544E+02	-.135E+03	.103E+03	.103E+03	.103E+03	.103E+03	.170E+04
.513E+03	.513E+03	.513E+03	.513E+03	.513E+03	.513E+03	.513E+03
.513E+03	.141E+04	.567E+03	.567E+03	.530E+03	.100E+03	.000E+00
.595E+02	.000E+00	.000E+00	.000E+00	.000E+00	.481E+02	.505E+02
.516E+02	.496E+02	.000E+00	.000E+00	.000E+00	.000E+00	.481E+02
.505E+02	.516E+02	.496E+02	.745E+03	.507E+02	.860E+02	.745E+03
.498E+02	.858E+02	.745E+03	.500E+02	.859E+02	.745E+03	.506E+02
.860E+02	.000E+00	.000E+00	.000E+00	.000E+00	.000E+00	.000E+00
.154E+01	-.110E+02	.727E+03	.000E+00	.000E+00	.000E+00	.636E+03
.636E+03	.636E+03	.636E+03				

0 0 0 0 0 0 0 0 0 0 0 0 0 0 0 0 0 0

DIGITAL DATA POINTS (18)

Figure 3 Magnetic Tape Data Block

3.1.7.3 Transition flow charts

In order to track the operators' progress through the EOP network, transition flow charts will be used. These charts are like large road maps, showing all planned transitions in the entire network. For each test scenario, flowcharts will be marked with the "expected" path through the EOPs. These charts will serve as the reference for the observation team when noting deviations from expected procedure usage. The charts will provide space for describing deviations, and also for recording elapsed time in the scenario. Figures 6 and 7 are reduced versions of the flow charts, showing the general EOP presentation. Figure 6 shows the ORG or E-series EOPs, while Figure 7 shows the status trees and F-series EOPs. These flowcharts will be generated specifically for the test EOPs, and contain the wording for <u>each</u> high level operator action step within each procedure. Actual size of each flowchart is approximately 2 feet by 3 feet.

3.1.7.4 Debriefing sessions

Immediately following each test scenario, the operating crew and observation team will meet in a classroom to discuss any noted deviations and EOP usage in general. The observation crew will use a special debriefing questionnaire as well as their annotated transition flow charts to evaluate deviations and document operator comments on EOP usage. Data from these sessions will be written on the special debriefing questionnaires, and all discussions will be recorded on tape. Standard size audio cassettes will be used for this purpose; again, a separate tape will be used for each separate test scenario debriefing session.

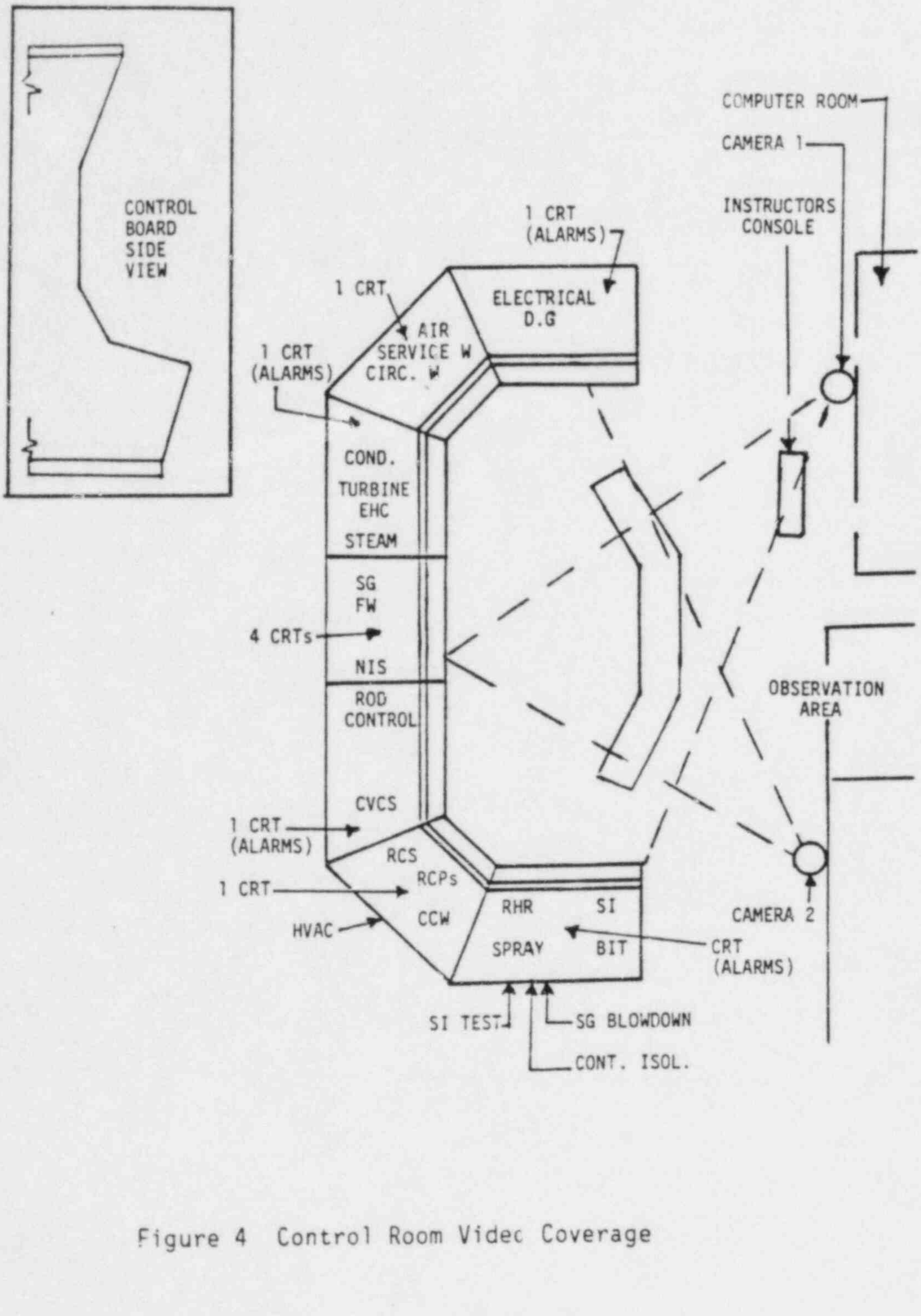
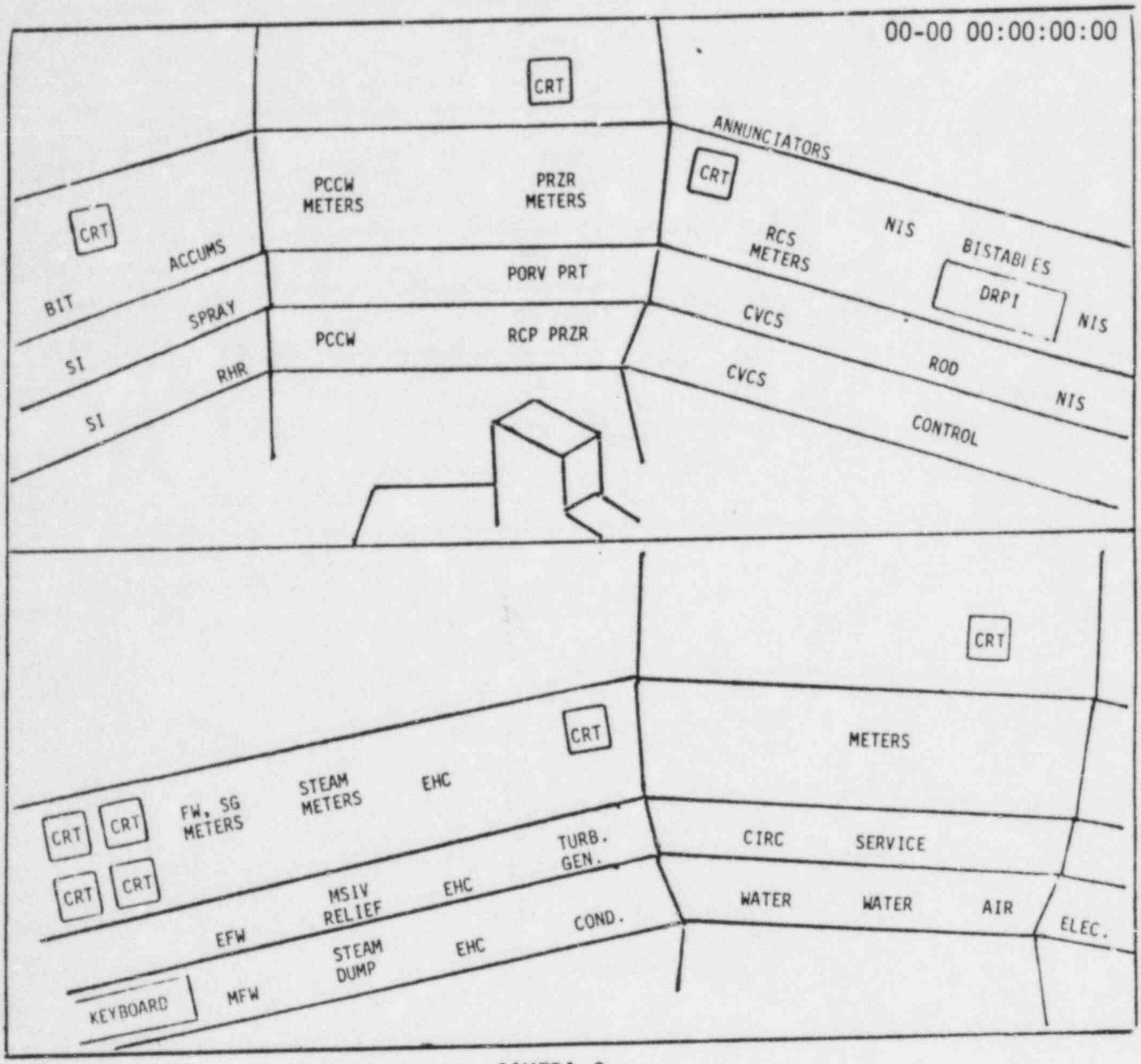


Figure 4 Control Room Video Coverage

CAMERA 1



CAMERA 2

Figure 5 Composite Video Image

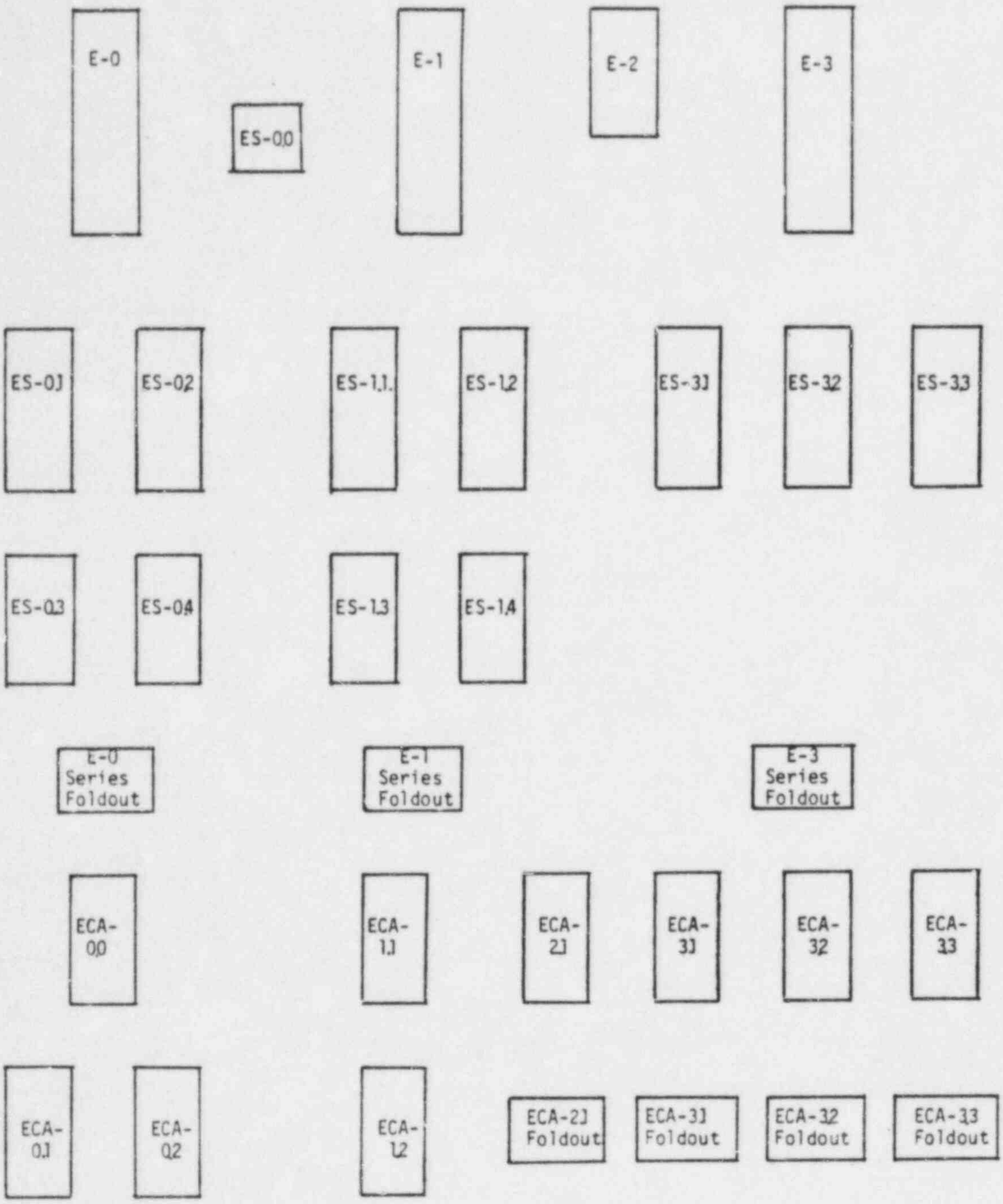


Figure 6 Transition Flow Chart For E-Series EOPs

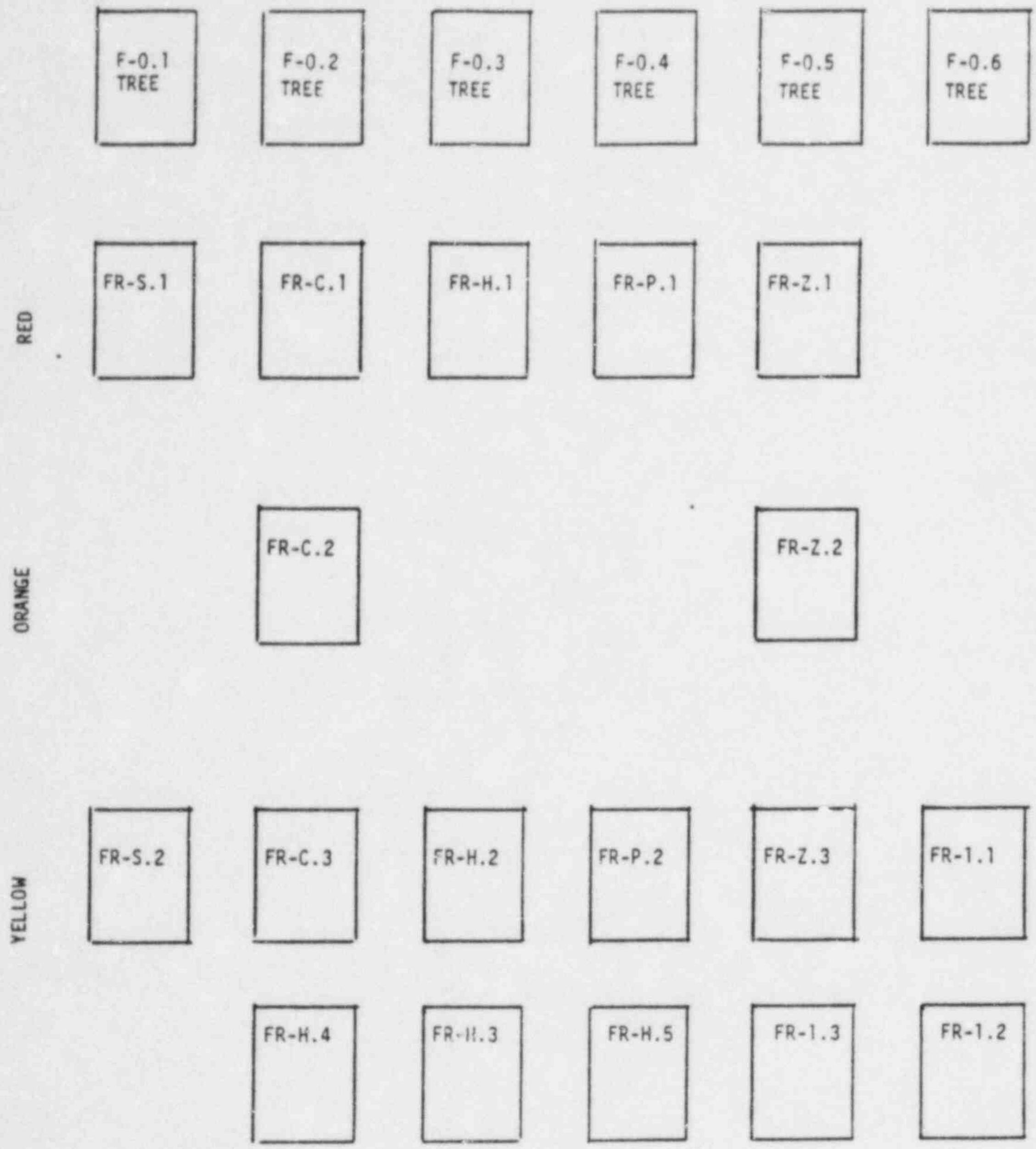


Figure 7 Transition Flow Chart For F-Series EOPs

3.1.8 Observation Teams

Each test scenario being performed in the control room will be observed by a specially selected and trained observation team. This team will be responsible for documenting all deviations from nominal (expected) performance observed during a scenario. A team will be made up of, at least, the following personnel.

- o One member familiar with ERG development, power plant operations, and the training program developed for this test.
- o One member familiar with ERG development and its analytical basis.
- o One member familiar with ERG development and skilled in human factors evaluations.
- o One member from the Seabrook Operations staff, familiar with the control room, the operators, and the EOP development.
- o One member from the Seabrook training staff, familiar with simulator operations, the simulator control room, and the Seabrook operator training program.

Each individual on an observation team will be carefuly instructed in his duties as an observer. Special checklists stating the validation criteria will be provided as constant reminders of the expected nominal performance. Team members will be briefed in the usage of the transition flow charts which will be available for each test scenario, and of the special debriefing questionnaire.

It will be the observation team's responsibility to record all observed deviations, real or suspected, discuss them subsequently with operators during the debriefing session, make an initial evaluation of the cause, and with the operators input, suggest possible resolutions.

3.1.9 Test Crews

Two full crews of operators will be provided by Seabrook - Operations. Each crew will consist of a unit shift supervisor, senior control room operator, and control room operator, all normally in the control room, and a shift superintendent, not normally in the control room, but on shift with the operating crew. In the event of an emergency, the shift superintendent acts as the Shift Technical Advisor for the operating crew, and fulfills the assigned duties of that position. For the purposes of this test, the STA has the sole function of monitoring the CSF Status Trees.

All the operators on one crew have previous plant operating experience (although not at Seabrook). Both crews are familiar with the Seabrook control room and normal plant operations.

3.1.10 Training

A special training program for the operating crews has been developed. This program uses the established validation criteria to assure that the proper interfaces with the EOPs, control room, and operators are considered. Much of the program structure closely follows the corresponding training program used in the BASIC Validation (Reference 1). Again, however, the training is compressed into a single week because of personnel and equipment constraints.

The training program covers basically two areas. Obviously, the new set of EOPs are covered as thoroughly as time permits, both in the classroom and on the simulator. Special emphasis is placed on procedure usage, since strict adherence to proper usage will be requested. However, equally important to the success of the Program, is to make the operators fully aware of the overall validation being performed. For this reason, the training program covers such things as the observation team's responsibilities, transition flow chart usage, video tape recordings, and especially the debriefing sessions. The operators will see the full list of test scenarios, and will experience many similar transients on the simulator during the training week.

The operators will also be familiar with the full list of validation criteria being used by the observers/evaluators during and after the actual test. This should encourage them to point out any difficulties which they experience in actually using the EOPs.

Most importantly, the operators will be constantly reminded that their personal abilities are not being judged, but that they are performing a valuable service in the overall Program implementation.

Table 10 shows the daily schedule, both classroom and simulator, for the training program. Not obvious from the table is the fact that the simulator exercises will be gradually increased in scope to function exactly as intended during the test. By the last training day, the simulator exercises will be identical to the conduct of the actual test scenarios, complete with data taking, observation teams, and debriefing sessions.

Appendix G presents a sample Lesson Plan for EOP (ERG) Rules of Usage. During the training week, each operator will have his own complete copy of the EOPs to aid familiarization. These EOPs will be identical to the ones used during the test, so that continuity is maintained.

Table 10

Validation Training Schedule

	Classroom	Simulator
Day 1	Validation Test Program Structure of ERGs Transition Flow Chart Rules of Usage - FORMAT Critical Safety Functions Status Trees Non-Accident EOPs o E-0 series o ECA-0 series	o Reactor Trip o Spurious SI o Loss of All AC Power
Day 2	Loss of Coolant Accident EOPs o E-1 series o ECA-1 series Inadequate Core Cooling EOPs o FR-C.1 o FR-H.1	o DBA LOCA (with switchover) o Small LOCA with cooldown o ICC (small LOCA without ECCS)
Day 3	Secondary Break EOPs o E-2 o ECA-2.1 Function Restoration EOPs o S-series o P-series o Z-series o I-series	o Secondary break outside containment o Secondary break inside containment (MSIVS fail to close) o ATWS

Table 10 (Cont)

Validation Training Schedule

	Classroom	Simulator
Day 4	SG Tube Rupture EOPs o E-3 series o ECA-3 series	o Small SGTR with steam release cooldown o Large SGTR with backfill cooldown o SGTR with secondary break in ruptured SG.
Day 5	Function Restoration EOPs o H-series o C-series	(Full test process duplication) o Loss of all FW with loss of offsite power
	Quiz on EOP usage Final Procedure Check	o Surprise transients

3.2 Assessment Phase

This phase comprises the actual simulator exercises of the test scenarios including EOP usage by the operators and will total approximately 40 hours of operation. Observation teams will be present and all validation data wil be recorded.

All personnel orientation for test participants will be complete prior to this phase, and all necessary materials will be on hand. Selection of the test scenarios to be run each day will be made randomly. Similarly, personnel in each observation team will be rotated daily.

For each test scenario, the following sequence will be observed:

1. The simulator instructor will use the supplied Test Scenario Run Sheet to initialize the simulation and properly align the control board.

2. One full operating crew will enter the control room and be briefed on current plant operating and equipment status, and be given operating instructions (i.e., raise or lower power, synchronize generator, start up the reactor, etc.).

3. The observation crew will take up positions in the simulator room observation booth, which provides them an unobstructed view of the entire control room. They will have Transition Flow Charts, appropriately pre-marked for the scenario, on which to record their observations and clock times. Also available in the observation booth will be several sets of EOPs, lists of the validation criteria, and a list of "behavioral symptoms" to watch for (from Reference 3), shown here as Table 11. The booth will be equipped with a TV monitor showing the images being recorded by the two cameras, and the digital time; the monitor will also supply the audio track being recorded, so the observers can hear what is being said by the operators.

TABLE 11
BEHAVIORAL SYMPTOMS TO AID AN OBSERVER

COMMISSION TYPE ERROR INDICATORS

- o does not walk to correct area of control room on first try
- o does not look at correct display or does not look in correct direction on first try
- o does not touch correct control on first try
- o does not set control to correct value on first try
- o performs an action not in the procedure
- o selects wrong procedures
- o selects too many procedures

SEQUENCE TYPE ERROR INDICATOR

- o performs action out of sequence

OMISSION TYPE ERROR INDICATORS

- o does not perform an action or step
- o allows a limit to be exceeded
- o fails to detect key signal
- o fails to perform task within allotted time

UNCERTAINTY INDICATORS

- o has to interpolate from charts, graphs, etc.
- o has to re-read procedures
- o takes excessive time to read procedures
- o takes excessive time to complete action
- o cannot remember what to do once procedures have been read
- o does not use procedures (when procedures are available is tentative, confused)
- o cannot find key information in procedures

4. The simulator instructor will synchronize the digital clock on the video screen with the computer clock for this scenario (zero both), and then activate the simulation and the computer data recording program.

5. After a minute or two of "steady-state" operation, the malfunctions specified on the Run Sheet for this test scenario will be input at the instructor's console.

6. The operators will respond to indicated plant conditions using the EOPs. Observations and plant data are recorded.

7. At some appropriate time, determined by the observation team leader, the test scenario is terminated.

8. Operators and observation team proceed to debriefing room, taking along the video tape and all observation notes. In the computer room, the stored plant data is copied to magnetic tape.

9. In the debriefing session, the tape cassette recorder is turned on, and discussion of the scenario follows. A standard questionnaire is used, plus all noted deviations are discussed. Comments are recorded on forms provided for this purpose. Copies of the EOPs are available in the debriefing room, plus a video player, if the tape needs to be replayed for reference. At the same time, a second operating crew and observation team are beginning the next test scenario in the control room.

10. All data for this scenario is cataloged by assigned number for future reference and analysis:

    - Magnetic tape
    - Video tape
    - Debriefing cassette
    - Run sheet
    - Transition flow charts
    - Comment sheets

11. At this point, the operating crew and observation team are free until called for their next test scenario.

A separate classroom in the training building will be equipped with a video monitor(s) showing the control room activity, for any non-participating observers. A copy of the EOPs, and sample transition flow charts and questionnaires will be provided for reference.

3.3 Resolution Phase

This phase will involve review and resolution of all noted deviations. It will include analysis of plant transient data/video tapes/debriefing recordings as necessary to clarify the context of a deviation, and possibly to suggest or modify a resolution. Out of this phase will come statements of SYSTEM validity, and also quantitative validation results for each element in the SYSTEM.

3.4 Documentation Phase

Documentation for the Validation Program will include:

- o Discussion of all test scenarios as run on the simulator
- o Listing of all deviations and resolutions
- o Recommendations for improvements in training
- o Recommendation for improvements in Validation (simulator) testing
- o Summary of the program

4.0 Application to Plant-Specific Validation

This section will address application of the "reference" method of validation to be used by any utility generating EOPs based on the ERGs.

5.0 References

1. Summary Report - Emergency Response Guidelines Validation Program, WCAP-10204, September, 1982.

2. Emergency Operating Procedures Validation Guidelines, INPO-83-006, July, 1983.

3. DRAFT - Component Verification And System Validation Guideline, May 9, 1983 (NUTAC).

4. Emergency Operating Procedures Verification Guideline, INPO-83-004, March, 1983.

5. Westinghouse Owners Group Emergency Response Guidelines - Revision 1, XXX, 1983.

APPENDIX A

DEFINITIONS

APPENDIX A

DEFINITIONS

Control Room Simulator - Dynamic device which imitates functions of control room hardware in real, fast, or slowed time.

Emergency Operating Procedures (EOPs) - Plant procedured directing operator actions necessary to mitigate consequences of transients and accidents that cause plant parameters to exceed reactor protection setpoints, engineered safety feature setpoints, or other appropriate technical limits.

Emergency Operating Procedure Guidelines (EPGs) - Guidelines that provide technical bases for the development of EOPs.

Emergency Response Guidelines (ERG) - A complex and detailed network of generic emergency guidance for W plants.

Function Restoration Guideline (FRG) - Those sets of operator action steps which are specifically intended to respond to a Critical Safety Function challenge as determined by plant symptoms.

Operator-Plant-Procedure-Training System (System) - To address Emergency Response Capabilities (ERC) the system elements used to mitigate the consequences of an emergency condition are as follows:

- o "operator" consists of the control room operating crew.
- o "plant" consists of the plant as seen from its control room with its instruments and controls. It may either include or not include a Safety Parameter Display System (SPDS).
- o "procedure" consists of the EOP set and supporting system operating procedures (EOP Network).
- o "training" consists of the EOP training program

Mock-Up - Static device (e.g., 3-D photos, 2-D photos, drawings) which portrays control room hardware and configuration.

Optimal Recovery Guideline (ORG) - Those sets of operator action steps in the ERG network which respond to a specific, diagnosed event. Guidance is provided to recover the plant from the event in the most efficient manner.

Paced Simulator Performance - Method of validation whereby actions are carried out by control room operating personnel in response to cues from simulated equipment in real, fast, or slowed time.

Plant Functions - Performance requirements and objectives of the plant design, such as core cooling, reactivity control, inventory control and electricity generation.

Real Equipment - On-line, functional hardware contained in a nuclear power plant control room.

Real Performance - Method of validation whereby actions are carried out by control room operating personnel in response to cues from functional on-line equipment in real time.

Reference Validation - Method of validation whereby data developed in a common EOP validation program is referenced by similar plants.

(Critical) Safety Functions - A limited set of plant functions which, if maintained, will prevent core damage and/or radioactivity release to the environment. An activity which assures the integrity of the physical barriers against radiation release.

Source Documents - Documents or records upon which the System components are based.

Status Tree - Graphical device to quickly evaluate the condition of a Critical Safety Function. Identifies off-normal conditions and the appropriate FRG for restoration of the function.

Symptoms - Displayed plant characteristics which directly or indirectly indicate plant status.

System Operational Correctness - A characteristic of the System which indicates the degree to which the components are compatible.

System Validation - The overall System (operator, control room, EOPs and training) evaluation performed to determine that the system components work together to accomplish the desired results.

Table-Top - Method of validation whereby an operating crew explains their step-by-step actions during a proposed event scenario to an observer/review team.

Verification - The evaluation performed to ensure consistency between any System element and its appropriate source documents.

Walk-Through - Method of validation whereby an operating crew conducts a step-by-step enactment of their actions during a proposed event scenario without carrying out the actual control functions.

Westinghouse Owners Group (WOG) - Organization of utilities which own nuclear power plants with Westinghouse-supplied Nuclear Steam Supply Systems. Activities involve generic engineering, licensing, and operational issues relating to Westinghouse-designed nuclear units.

Writers Guide for EOPs - A plant document that provides instructions for writing EOPs, emphasizing the incorporation of good writing principles.

Appendix B

Item I.C.1 from

NUREG 0737

(Clarification of TMI Action Plant Requirement, November, 1980)

I.C.1 GUIDANCE FOR THE EVALUATION AND DEVELOPMENT OF PROCEDURES FOR TRANSIENTS AND ACCIDENTS

Position

In letters of September 13 and 27, October 10 and 30, and November 9, 1979, the Office of Nu-lear Reactor Regulation required licensees of operating plants, applicants for operating licenses and licensees of plants under construction to perform analyses of transients and accidents, prepare emergency procedure guidelines, upgrade emergency procedures, including procedures for operating with natural circulation conditions, and to conduct operator retraining (see also item I A.2 1). Emergency procedures are required to be consistent with the actions necessary to cope with the transients and accidents analyzed. Analyses of transients and accidents were to be completed in early 1980 and implementation of procedures and retraining were to be completed 3 months after emergency procedure guidelines were established however, some difficulty in completing these requirements has been experienced. Clarification of the scope of the task and appropriate schedule revisions are being developed. In the course of review of these matters on Babcock and Wilcox (B&W)-designed plants, the staff will follow up on the bulletin and orders matters relating to analysis methods and results, as listed in NUREG-0660, Appendix C (see Table C.1, items 3, 4, 16, 18, 24, 25, 26, 27; Table C.2, items 4, 12, 17, 18, 19, 20; and Table C.3, items 6, 35, 37, 38, 39, 41, 47, 55, 57).

Changes to Previous Requirements and Guidance

A. Modification to Clarification

(1) Addresses owners' group and vendor submittals.

(2) References to task action plan items I.C.8 and I.C.9.

(3) Scope of procedures review is explained.

(4) Establishes configuration control of guidelines for emergency procedures.

B. Modification to Implementation

(1) Deleted reference to NUREG-0578. Recommendation 2.1.9 for item I.C.I(a)2, inadequate core cooling.

Clarification

The letters of September 13 and 27, October 10 and 30, and November 9, 1979. required that procedures and operator training be developed for transients and accidents. The initiating events to be considered should include the events presented in the final safety analysis report (FSAR) loss of instrumentation buses, and natural phenomena such as earthquakes, floods, and tornadoes. The purpose of this paper is to clarify the requirements and add additional requirements for the reanalysis of transients and accidents and inadequate core cooling.

Based on staff reviews to date, there appear to be some recurring deficiencies in the guidelines being developed Specifically, the staff has found a lack of justification for the approach used (i.e , symptom-, event-, or function-oriented) in developing diagnostic guidance for the operator and in procedural development. It has also been found that although the guidelines take implicit credit for operation of many systems or components, they do not address the availability of these systems under expected plant conditions nor do they address corrective or alternative actions that should be performed to mitigate the event should these systems or components fail.

The analyses conducted to date for guideline and procedure development contain insufficient information to assess the extent to which multiple failures are considered. NUREG-0578 concluded that the single-failure criterion was not considered appropriate for guideline development and called for the consideration of multiple failures and operator errors. Therefore, the analyses that support guideline and procedure development should consider the occurrences of multiple and consequential failures. In general, the sequence of events for the transients and accidents and inadequate core cooling analyzed should postulate multiple failures such that, if the failures were unmitigated, conditions of inadequate core cooling would result.

Examples of multiple failure even s include:

(1) Multiple tube ruptures in a single steam generator and tube rupture in more than one steam generator:
(2) Failure of main and auxiliary feedwater;
(3) Failure of high-pressure reactor coolant makeup system:
(4) An anticipated transient without scram (ATWS) event following a loss of offsite power, stuck-open relief valve or safety/relief valve, or loss of main feedwater; and
(5) Operator errors of omission or commission

The analyses should be carried out far enough into the event to assure that all relevant thermal/hydraulic/neutronic phenomena are identified (e.g., upper head voiding due to rapid cooldown, steam generator stratification). Failures and operator errors during the long-term cooldown period should also be addressed.

The analyses should support development of guidelines that define a logical transition from the emergency procedures into the inadequate core cooling procedure including the use of instrumentation to identify inadequate core cooling conditions. Rationale for this transition should be discussed. Additional information that should be submitted includes:

(1) A detailed description of the methodology used to develop the guidelines

(2) Associated control function diagrams, sequence-of-event diagrams, or others. if used;
(3) The bases for multiple and consequential failure considerations;
(4) Supporting analysis, including a description of any computer codes used; and
(5) A description of the applicability of any generic results to plant-specific applications.

Owners' group or vendor submittals may be referenced as appropriate to support this reanalysis. If owners' group or vendor submittals have already been forwarded to the staff for review, a brief description of the submittals and justification of their adequacy to support guideline development is all that is required.

Pending staff approval of the revised analysis and guidelines, the staff will continue the pilot monitoring of emergency procedures described in Task Action Plan item I.C.8 (NUREG-0660). For PWRs, this will involve review of the loss of coolant, steam-generator-tube rupture, loss of main feedwater, and inadequate core cooling procedures. The adequacy of each PWR vendor's guidelines will be identified to each NTOL during the emergency-procedure review. Since the analysis and guidelines submitted by the General Electric Company (GE) owners' group that comply with the requirements stated above have been reviewed and approved for trial implementation on six plants with applications for operating licenses pending, the interim program for BWRs will consist of trial implementation on these six plants.

Following approval of analysis and guidelines and the pilot monitoring of emergency procedures, the staff will advise all licensees of the adequacy of the guidelines for application to their plants. Consideration will be given to human factors engineering and system operational characteristics, such as information transfer under stress, compatibility with operator training and control room design, the time required for component and system response, clarity of procedural actions, and control-room-personnel interactions. When this determination has been made by the staff, a long-term plan for emergency procedure review, as described in task action plan item I.C.9, will be made available. At that time, the reviews currently being conducted on NTOLs under item I.C.8 will be discontinued, and the review required for applicants for operating licenses will be as described in the long-term plan. Depending on the information submitted to support development of emergency procedures for each reactor type or vendor, this transition may take place at different times. For example, if the GE guidelines are shown to be effective on the six

plants chosen for pilot monitoring, the long-term plan for BWRs may be complete in early 1981. Operating plants and applicants will then have the option of implementing the long-term plan in a manner consistent with their operating schedule: provided they meet the final date required for implementation. This may require a plant that was reviewed for an operating license under item I.C.8 to revise its emergency procedures again prior to the final implementation date for Item I.C.9. The extent to which the long-term program will include review and approval of plant-specific procedures for operating plants has not been established. Our objective, however, is to minimize the amount of plant-specific procedure review and approval required. The staff believes this objective can be acceptably accomplished by concentrating the staff review and approval on generic guidelines. A key element in meeting this objective is the use of staff-approved generic guidelines and guideline revisions by licensees to develop procedures. For this approach to be effective, it is imperative that, once the staff has issued approval of a guideline, subsequent revisions of the guideline should not be implemented.by licensees until reviewed and approved by the staff. Any changes in plant-specific procedures based on unapproved guidelines could constitute an unreviewed safety issue under 10 CFR 50.59. Deviations from this approach on a plant-specific basis would be acceptable provided the basis is submitted by the licensee for staff review and approval. In this case, deviations from generic guidelines should not be implemented until staff approval is formally received in writing. Interim implementation of analysis and procedures for small-break loss-of-coolant accident and inadequate core cooling should remain on the schedule contained in NUREG-0578, Recommendation 2.1.9.

Applicability

This requirement applies to all operating reactors and applicants for operating license.

Implementation

Reanalysis of transients and accidents and inadequate core cooling and preparation of guidelines for development of emergency procedures should be completed and submitted to the NRC for review by January 1, 1981. The NRC staff will review the analyses and guidelines and determine their acceptability by July 1, 1981, and will issue guidance to licensees on preparing emergency procedures from the guidelines. Following NRC approval of the guidelines, licensees and applicants for operating licenses issued prior to January 1, 1982, should revise and implement their emergency procedures at the first refueling outage after January 1, 1982. Applicants for operating licenses issued after January 1, 1982 should implement the procedures prior to operation. This schedule supersedes the implementation schedule included in NUREG-0578, Recommendation 2.1.9 for item I.C.1(a)3, Reanalysis of Transients and Accidents. For those licensees and/or owners groups that will have difficulty in attaining the January 1, 1981 due date for submittal of guidelines, a comprehensive program plan, proposed schedule, and a detailed justification for all delays and problems shall be submitted in lieu of the guidelines.

Type of Review

A preimplementation review of guidelines will be performed.

A preimplementation review of procedures will be performed.

Documentation Required

See above, "Implementation."

Technical Specification Changes Required

Changes to technical specifications will not be required

References

(Deleted)

Appendix C

Acceptance Criteria for Individual Guidelines

E-0: Reactor Trip or Safety Injection

Purpose:

- o To verify proper response of automatic protection systems following manual or automatic actuation of reactor trip or safety injection.
- o To identify appropriate optimal recovery guideline for subsequent recovery.

Criteria for Technical Validation:

- o Optimal plant status can be monitored and maintained:
  - CSFs not unduly challenged
  - Automatic actions and equipment verified operating as designed
  - RCPs tripped as necessary
  - Low head SI pumps and diesel generators stopped when not required.
  - SG levels maintained

- o An Appropriate Optimal Recovery Guideline Selected:

  - Reactor trip without SI (ES-0.1)
  - Loss of all AC power (ECA-0.0)
  - Secondary break (E-2)
  - SG tube rupture (E-3)
  - Loss of reactor coolant without tube rupture (E-1)
  - LOCA outside containment (ECA-1.2)

- o Appropriate Function Restoration Guideline Selected Before Monitoring of Status Trees Initiated

  - ATWS (FR-S.1)
  - Loss of Feedwater (FR-H.1)

ES-0.1 Reactor Trip Response

Purpose:

To provide necessary instructions to stabilize and control the plant following a reactor trip without a safety injection.

Criteria For Technical Validation

o Optimal Plant Status Can Be Monitored and Maintained

- Plant stabilized at no-load conditions
- Necessary plant equipment checked and restoration attempted if necessary
- RCPs restarted if possible
- CSFs not unduly challenged

o Appropriate Long-Term Action Selected

- Natural circulation cooldown (ES-0.2)
- Applicable normal plant procedure

ES-0.2 Natural Circulation Cooldown

Purpose:

To provide actions to perform a natural circulation RCS cooldown and depressurization to cold shutdown without upper-head void formation.

Criteria For Technical Validation

- o Optimal Plant Status Can Be Monitored and Maintained

  - RCP restarted if possible without loss of RCS pressure control
  - Boration to Cold Shutdown Boration completed before cooldown
  - CRDM fans started if possible
  - Cooldown under prescribed limits completed
  - Depressurization completed without void formation
  - CSFs not unduly challenged

- o Appropriate Long-Term Action Selected

  - Completion of this guideline for no void growth
  - ES-0.3 or ES-0.4 if cooldown with void is necessary

ES-0.3 Natural Circulation Cooldown With Steam Void in Vessel (With RVLIS)

Purpose:

To provide actions to continue natural circulation cooldown and depressurization to cold shutdown under conditions that allow for the potential formation of a steam void in the vessel with a vessel without a vessel level system available to monitor void growth.

Criteria for Technical Validation

o Optimal Plant Status Can Be Monitored and Maintained:

- RCP restarted if possible without loss of RCS pressure control
- Cooldown under prescribed limits completed
- Depressurization completed with void maintained within specified limits
- CSFs not unduly challenged

ES-0.4 Natural Circulation Cooldown With Steam Void in Vessel (Without RVLIS)

Purpose:

To provide actions to continue natural circulation cooldown and depressurization to cold shutdown under conditions that allow for the potential formation of a steam void in the vessel without a vessel level system available to monitor void growth.

Criteria for Technical Validation:

o Optimal Plant Status Can be Monitored and Maintained:

  - RCP restarted if possible without loss of RCS pressure control
  - Cooldown under prescribed limits completed
  - Depressurization completed with void maintained within specified limits
  - CSFs not unduly challenged

E-1: Loss of Reactor or Secondary Coolant

Purpose:

- o To maintain all critical safety functions satisfied
- o To place the plant in an optimal condition following a loss of reactor or secondary coolant

Criteria for Technical Validation:

- o Optimal Plant Status Can Be Monitored and Maintained

  - CSFs not unduly challenged
  - RCPs tripped as necessary
  - SG levels maintained
  - Low head SI and containment spray stopped when not required

- o Coincident or Subsequent Failures can be Detected and Addressed in a Reasonable Time:

  - Secondary break (E-2)
  - SG tube rupture (E-3)
  - Loss of all AC power (ECA-0.0)
  - Critical safety function challenges (appropriate FRGs)
  - Loss of ECR capability (ECA-1.1)

- o Appropriate Long Term Recovery Method can be Selected and Performed:

  - SI termination (ES-1.1)
  - Cooldown/depressurization to RHR (ES-1.2)
  - Emergency Coolant Recirculation (ES-1.3 and ES-1.4)

ES-1.1 SI Termination

Purpose:

To provide instructions to terminate safety injection and stabilize plant conditions.

Criteria for Technical Validation

o Optimal Plant Status can be Monitored and Maintained:

- CSFs not unduly challenged
- SI pumps can be stopped and plant condition stabilized in acceptable time period
- Necessary plant equipment checked and restoration attempted if necessary
- RCPs restarted if possible without loss of RCS pressure control

o Appropriate Long Term Recovery Selected

- SI required (E-1, ES-1.2)
- SI not required (appropriate plant procedure)

o Coincident or Subsequent Failures can be Detected and Addressed in a Reasonable Time

- Secondary break (E-2)
- SG tube rupture (E-3)
- Loss of AC power (ECA-0.0)
- CSF challenges (appropriate FRGs)

ES-1.2 Post LOCA Cooldown and Depressurization

Purpose:

To provide actions to cool down and depressurize the RCS to cold shutdown conditions following a loss of reactor coolant inventory.

Criteria for Technical Validation

- o Optimal Plant Status can be Monitored and Maintained:
  - CSFs not undully challenged
  - SI pumps can be stopped and plant depressurized to cold shutdown in acceptable time period
  - Necessary plant equipment checked and restoration attempted if necessary
  - SG levels maintained
  - Cooldown completed within prescribed limits
  - RCSP restarted if possible
  - SI reinitiation performed if required

- o Coincident or Subsequent Failures can be detected and addressed in a Reasonable Time
  - Secondary break (E-2)
  - SG tube rupture (E-3)
  - Loss of all AC power (ECA-0.0)
  - CSF challenges (appropriate FRG)
  - RWST inventory depletion (ES-1.3)

ES-1.3 Transfer to Cold Leg Recirculation

Purpose

To provide instructions for transferring the safety injection system and containment spray system to the recirculation mode.

Criteria for Technical Validation

o Optimal system switchover can be completed

o Loss of ECR identified and ECA-1.1 guideline implemented

E-2: Faulted Steam Generator Isolation

Purpose:

o To identify and isolate a faulted steam generator

Criteria for Technical Validation:

o Identification and Isolation of faulted SG can be performed:

o Appropriate long term recovery method can be selected and performed:
  - Loss of reactor or secondary coolant alone or in combination (E-1)
  - SG tube rupture (E-3)
  - Uncontrolled depressurization of all steam generators (ECA-2.1)

E-3: Steam Generator Tube Rupture

Purpose:

- To stop primary-to-secondary leakage for a steam generator tube rupture event and determine the proper recovery guideline

Criteria for Technical Validation:

- All steam generators with failed tubes are identified

- Steam flow from all ruptured steam generators is terminated or the operator is transitioned into ECA-3.1 to minimize primary-to-secondary leakage

- Primary-to-secondary leakage is terminated or sufficiently managed to control steam generator inventory, or the operator is transitioned to ECA-3.1 to minimize primary-to-secondary leakage

- Recovery actions directed by the E-3 guideline should not result in an orange or red path challenge to any Critical Safety Function

- RCS subcooling and pressurizer level are maintained greater than instrument uncertainties after SI flow is terminated.

ES-3.1, ES-3.2, and ES-3.1, Post-SGTR Cooldown

Purpose:

- To cooldown and depressurize the RCS to cold shutdown conditions following a steam generator tube rupture event.

- To control RCS pressure and reactor coolant makeup flow to maintain indications of adequate coolant inventory and RCS subcooling, while minimizing primary-to-secondary leakage.

Criteria for Technical Validation

- Pressure in the RCS and ruptured steam generators and reactor coolant temperature should decrease.

- Recovery actions should not cause water relief from any ruptured steam generator.

- RCS subcooling and pressurizer level are maintained greater than instrument uncertainties.

- Subsequent loss of coolant events, including steam generator tube failures, are detected and the operator is transitioned to the appropriate optimal recovery guideline.

ECA-3.1 SGTR With Loss of Reactor Coolant - Subcooled Recovery Desired

ECA-3.2 SGTR With Loss of Reactor Coolant - Saturated Recovery Desired

Purpose:

o To cool and depressurize the RCS to cold shutdown conditions with an unisolatable loss of reactor coolant when symptoms of a SGTR are present

Criteria for Technical Validation:

o Pressurizer level and RCS subcooling are maintained greater than instrument uncertainties, or core exit temperatures continue to decrease with RVLIS indicating level above the top of the core

o RCS pressure and temperature continue to decrease toward cold shutdown conditions

o Cycling of SI pumps is minimized

o RCS pressure and reactor coolant makeup flow are controlled to limit RCS subcooling

o The operator transitions to ECA-3.2 to further reduce reactor coolant leakage if makeup water supply is critically low or steam generator overfill is imminent.

FR-S Series

Purpose:

- o To respond to a challenge to the Subcriticality CSF
- o To respond to an ATWS event (FR-S.1)

Criteria for Technical Validation:

- o Function Restoration Can be Implemented
  - Reactor trip checked
  - Emergency boration established, if possible
  - Excessive RCS cooldown identified and addressed
  - Other CSFs not unduly challenged
- o Appropriate optimal recovery guideline selected for long-term recovery
  - Guideline and step in effect if CSF restored
- o Optimal plant status can be monitored and maintained for ATWS event.

FR-C Series

Purpose:

To restore core cooling after a challenge to the core cooling critical safety function

Criteria for Technical Validation

- o Function restoration can be implemented
  - SI flow established, if possible
  - Cooldown/depressurization can be completed within prescribed limits
  - RCPs restarted, if necessary
  - Other CSFs not unduly challenged
- o Appropriate optimal recovery selected for long-term recovery
  - Guideline and step in effect
  - Loss of Reactor or Secondary Coolant (E-1)

FR-H Series

Purpose:

- To respond to a challenge to the Heat Sink CSF
- To respond to a loss-of-all-feedwater event

Criteria for Technical Validation

o Function restoration can be implemented:

  - Source of secondary feed established, if possible
  - Bleed and feed heat removal mode established if necessary
  - Bleed and feed terminated properly after restoration of secondary heat sink
  - Overpressure SG condition can be addressed properly
  - High level or low level condition can be addressed properly
  - Other CSFs not unduly challenged

o Appropriate optimal recovery can be selected for long-term recovery:

  - Guideline and step in effect
  - SI Termination (ES-1.1)

o Optimal plant status can be monitored and maintained for Loss-of-All-Feedwater Event

FR-P Series

Purpose:

To respond to a challenge to the integrity CSF

Criteria for Technical Validation:

- o Function restoration can be implemented:
  - Excessive cooling can be identified and addressed properly
  - Excessive pressurization can be identified and addressed properly
  - RCS depressurization completed within prescribed limits
  - Other CSFs not unduly challenged
- o Appropriate optimal recovery can be selected for long-term recovery
  - Guideline and step in effect
  - Any subsequent cooldown performed within prescribed limits.

Appendix D

Example Test Scenario Run Sheets

ERG REV. 1 VALIDATION PROGRAM

TEST SCENARIO RUN SHEET

TITLE: Spurious SI

INITIAL
CONDITIONS: 100% Power, Equilibrium, MOL I.C. No. 15

SEQUENCE OF MALFUNCTIONS

	DESCRIPTION	No.	OPTION
1.	Inadvertent SIS	147	N/A

SPECIAL INSTRUCTIONS FOR THIS SCENARIO

1. Inform operators on shift turnover that I&C performing tests in protection racks.

TEST SCENARIO NO. EXAMPLE 2

ERG REV. 1 VALIDATION PROGRAM

TE. SCENARIO RUN SHEET

TITLE: Loss of All (High Pressure) Feedwater

INITIAL
CONDITIONS: 100% Power, Equilibrium, EOL I.C. No. 17

SEQUENCE OF MALFUNCTIONS

DESCRIPTION	No.	OPTION
1. Trip of Both MFW Pumps	014	N/A
2. Trip of S/U FWP	056	N/A
3 Loss of Emergency FW	152	N/A

SPECIAL INSTRUCTIONS FOR THIS SCENARIO

1. Inform operators at shift turnover that maintenance is being done on steam packing exhauster (HX)

2. Tag out DG-IB (lube oil level was low, estimate 2 more hours to lift tag)

TEST SCENARIO NO. EXAMPLE 3

ERG REV. 1 VALIDATION PROGRAM

TEST SCENARIO RUN SHEET

TITLE: Loss of All AC Power - Recovery with SI Required

INITIAL
CONDITIONS: 15% Power, Ready to Synchronize I.C. NO. 9

SEQUENCE OF MALFUNCTIONS

	DESCRIPTION	No.	OPTION
1.	Total loss of offsite power	114	N/A
2.	EDG 1A Fails to Auto Start	118	N/A
3.	EDG 1B Low Lube Oil Trip	119	N/A
4.	RCS Leak - SG 3 Tube Rupture	025	100% (Ramp over 30 minutes)

SPECIAL INSTRUCTIONS FOR THIS SCENARIO

1. Shift turnover instructions - Synchronize and pick up 600 MWe over 1 hour.
2. Initiate loss-of-offsite power coincident with main breaker closure. Diesels should not start
3. Start SGTR at +10 minutes. Restore one DG at +25 minutes. Restore second DG at +35 minutes.

Appendix E

Reference ERGs and Seabrook Test EOPs

SEABROOK TEST PROCEDURE E-0

A. PURPOSE

This procedure provides actions to verify proper response of the automatic protection systems following manual or automatic actuation of a reactor trip or safety injection, to assess plant conditions, and to identify the appropriate recovery procedure.

B. SYMPTOMS OR ENTRY CONDITIONS

1. Any symptom that requires a manual reactor trip listed in ATTACHMENT A, if one has not occurred.

2. The following are symptoms of a reactor trip:

   a. Any reactor trip annunciator lit.

   b. Rapid decrease in neutron level indicated by nuclear instrumentation.

   c. All shutdown and control rods are fully inserted. Rod bottom lights are lit.

3. Any symptom that requires a manual reactor trip and safety injection listed in ATTACHMENT B, if one has not occurred.

4. The following are symptoms of a reactor trip and safety injection:

   a. Any SI annunciator or status lamp lit.

   b. ECCS pumps in service.

   c. Phase A isolation.

STEP	ACTION/EXPECTED RESPONSE	RESPONSE NOT OBTAINED

NOTE
- Steps 1 through 14 are IMMEDIATE ACTION steps.
- Initiate monitoring of critical safety function status trees at Step 27 OR if exiting from this procedure.

STEP	ACTION/EXPECTED RESPONSE	RESPONSE NOT OBTAINED
1	Verify Reactor Trip: • Rod bottom lights - LIT • Reactor trip and bypass breakers - OPEN • Rod position indicators - AT ZERO ON DRPI • Neutron flux - DECREASING	Manually trip reactor. IF reactor will NOT trip, THEN go to FR-S.1, RESPONSE TO NUCLEAR POWER GENERATION/ATWS, Step 1.
2	Verify Turbine Trip:	
	a. All turbine stop valves - CLOSED	a. Manually trip turbine.
3	Verify Power To AC Emergency Busses:	
	a. AC emergency busses - AT LEAST ONE ENERGIZED • E-5 Voltmeter • E-6 Voltmeter	a. Try to restore power to at least one ac emergency bus. IF power can NOT be restored to at least one ac emergency bus, THEN go to ECA-0.0, LOSS OF ALL AC POWER, Step 1.
	b. AC emergency busses - ALL ENERGIZED	b. Try to restore power to deenergized ac emergency busses.

STEP	ACTION/EXPECTED RESPONSE	RESPONSE NOT OBTAINED
4	Check If SI Is Actuated: • SI Annunciator Lit • SI Status Monitor Light • ECCS Pumps Running • Phase A Isolation • Auto Start of EDGs	Check if SI is required. SI IS REQUIRED IF: 1) PRZR PRESSURE $\leq$ 1850 PSIG 2) CONTAINMENT PRESSURE $\geq$ 4.3 PSIG 3) STEAMLINE PRESSURE $\leq$ 585 PSIG IF SI is required, THEN manually actuate. IF SI is NOT required, THEN go to ES-0.1, REACTOR TRIP RESPONSE, Step 1.
5	Verify FW Isolation: • Flow control valves - CLOSED • Flow control bypass valves - CLOSED • FW isolation valves - CLOSED	Manually close valves as necessary.
6	Verify Containment Isolation Phase A:	
	a. Phase A - ACTUATED	a. Manually actuate Phase A.
	b. Phase A valves - CLOSED AS INDICATED BY STATUS PANEL • TRAIN A • TRAIN B	b. Manually close valves as necessary.
7	Verify EFW Pumps Running:	
	a. MD pump - RUNNING	a. Manually start pump.
	b. Turbine-driven pump - RUNNING	b. Manually open steam supply valves.

STEP	ACTION/EXPECTED RESPONSE	RESPONSE NOT OBTAINED
8	Verify ECCS Pumps Running: • CCPs - TRAIN A AND B • SI pumps - TRAIN A AND B • RHR pumps - TRAIN A AND B	Manually start pumps.
9	Verify PCCW Pumps - RUNNING: a. Loop A - ONE PUMP RUNNING b. Loop B - ONE PUMP RUNNING c. Thermal barrier cooling pumps - AT LEAST ONE PUMP RUNNING	Manually start pumps.
10	Verify Ultimate Heat Sink Operation: a. Train A - RUNNING 1) One SW pump - OR - 2) One CT pump AND CT fan in TA mode b. Train B - RUNNING 1) One SW pump - OR - 2) One CT pump AND CT fan in TA mode	Manually start pumps and align valves as necessary.

STEP	ACTION/EXPECTED RESPONSE	RESPONSE NOT OBTAINED
11	Verify SW Cooling To DG Jacket Water Coolers:	
	a. Train A cooling established	
	1) SW-V16 - OPEN	1) OPEN SW-V16.
	2) Flow indicated	2) Continue to Step 11b.
	b. Train B cooling established	
	1) SW-V18 - OPEN	1) OPEN SW-V18.
	2) Flow indicated	2) Continue to Step 12.
12	Verify Containment Ventilation Isolation:	Close valves immediately.
	a. Containment air purge valves - CLOSED INDICATED ON STATUS PANEL • CAP-V1 • CAP-V2 • CAP-V3 • CAP-V4	
	b. Containment on-line purge valves - CLOSED INDICATED ON STATUS PANEL • COP-V1 • COP-V2 • COP-V3 • COP-V4	

STEP	ACTION/EXPECTED RESPONSE	RESPONSE NOT OBTAINED
13	Check If Main Steamlines Should Be Isolated:	
	a. Steam line isolation is REQUIRED IF: • Any steamline - LESS THAN OR EQUAL TO 585 PSIG • Containment pressure is - GREATER THAN OR EQUAL TO HI-2 SETPOINT 4.3 PSIG	a. Go to Step 14.
	b. Verify MSIV AND MSIV bypass valves - CLOSED	b. Manually close valves.

STEP	ACTION/EXPECTED RESPONSE	RESPONSE NOT OBTAINED
14	Check Containment Pressure - HAS REMAINED LESS THAN HI-3 SETPOINT, 18 PSIG, BY PRESSURE RECORDING	IF pressure has gone greater than 18 PSIG, THEN:

a. Verify containment spray initiated.

   1) Train A in operation.

      - CBS Pump A - RUNNING
      - CBS-V11 - OPEN
      - CBS Pump A discharge pressure - LESS THAN 310 PSIG
      - Miniflows - CLOSED

   2) Train B in operation.

      - CBS Pump B - RUNNING
      - CBS-V17 - OPEN
      - CBS Pump B discharge pressure - LESS THAN 310 PSIG
      - Miniflows - CLOSED

   IF NOT initiated, THEN manually initiate.

   1) HOLD BOTH manual activate switches in a train AND place to ACTUATE.

b. Verify containment Phase B ('P' signal) valves actuate to proper position on status light panel. IF status light panel does NOT indicate proper position, manually position valves.

c. Stop all RCPs.

STEP	ACTION/EXPECTED RESPONSE	RESPONSE NOT OBTAINED
15	Verify ECCS Flow:	
	a. CCP flow indicators - CHECK FOR FLOW THROUGH BIT	a. Manually start pumps and align valves. REFER TO ATTACHMENT A, ECCS VALVE ALIGNMENT - CCP VIA BIT TO RCS COLD LEGS.
	b. RCS pressure - LESS THAN 1550 PSIG • RC-PI-405 • RC-PI-403	b. Go to Step 16.
	c. SI pump flow indicators - CHECK FOR FLOW • TRAIN A • TRAIN B	c. Manually start pumps and align valves. Refer to ATTACHMENT B, ECCS VALVE ALIGNMENT - SIP TO RCS COLD LEGS.
	d. RCS pressure - LESS THAN 200 PSIG	d. Go to Step 16.
	e. RHR pump flow indicators - CHECK FOR FLOW • TRAIN A • TRAIN B	e. Manually start pumps and align valves. REFER TO ATTACHMENT C, ECCS VALVE ALIGNMENT - RHR PUMP TO RCS COLD LEGS.
16	Verify EFW Flow - GREATER THAN 470 GPM TOTAL COMBINED FLOW TO AT LEAST TWO SGs	Manually start pumps. IF proper flow can NOT be established, THEN go to FR-H.1, RESPONSE TO LOSS OF SECONDARY HEAT SINK, STEP 1.

STEP	ACTION/EXPECTED RESPONSE	RESPONSE NOT OBTAINED

NOTE High EFW flow to a faulted SG will cause automatic closure of that SG's EFW flow control valves.

17	Verify EFW Valve Alignment - PROPER EMERGENCY ALIGNMENT	Manually align valves as necessary.

SG	VALVE NOMENCLATURE	REQUIRED POSITION
A	FV-4214A, FLOW CONTROL FV-4214B, FLOW CONTROL	OPEN OPEN
B	FV-4224A, FLOW CONTROL FV-4224B, FLOW CONTROL	OPEN OPEN
C	FV-4234A, FLOW CONTROL FV-4234B, FLOW CONTROL	OPEN OPEN
D	FV-4244A, FLOW CONTROL FV-4244B, FLOW CONTROL	OPEN OPEN

STEP	ACTION/EXPECTED RESPONSE	RESPONSE NOT OBTAINED
18	Verify ECCS Valve Alignment - PROPER EMERGENCY ALIGNMENT INDICATED ON STATUS PANELS • TRAIN A • TRAIN B	Manually align valves as necessary. ECCS valve alignment checklists are provided as ATTACHMENTS C, D and E.
19	Check RCS Temperature - STABLE AT OR TRENDING TO 557°F • WR LOOP TEMPERATURE RECORDERS	IF temperature less than 557°F and decreasing, THEN: a. Stop dumping steam. b. IF cooldown continues, THEN throttle total EFW flow but not less than 470 GPM - TOTAL COMBINED FLOW. Maintain WR level above top of SG U-tubes.

LEVEL ABOVE SG U-TUBES	
ADVERSE CONTM	NORMAL CONTM
NARROW RANGE LEVEL GREATER THAN 28%	WIDE RANGE LEVEL GREATER THAN 65%

c. IF cooldown continues, THEN close MSIVs AND MSIV bypass valves.

IF temperature greater than 557°F and increasing, THEN:

- Manually dump steam to condenser

- OR -

- Manually dump steam with SG ASDVs.

STEP	ACTION/EXPECTED RESPONSE	RESPONSE NOT OBTAINED
20	Check PRZR PORVs and Spray Valves:	
	a. PORVs - CLOSED	a. IF PRZR pressure less than 2385 psig, THEN manually close PORVs. IF any valve can NOT be closed, THEN manually close its block valve. IF block valve can NOT be closed, THEN go to E-1, LOSS OF REACTOR OR SECONDARY COOLANT, Step 1.
	b. Normal PRZR spray valves - CLOSED	b. IF PRZR pressure less than 2260 psig, THEN manually close valves. IF valves can NOT be closed, THEN stop RCP(s) supplying failed spray valve(s). • PCV-455A RC-P-1C • PCV-455B RC-P-1A

NOTE Seal injection flow should be maintained to all RCPs.

STEP	ACTION/EXPECTED RESPONSE	RESPONSE NOT OBTAINED
21	Check If RCPs Should Be Stopped:	
	a. High Head ECCS Pumps - AT LEAST ONE RUNNING • Centrifugal Charging Pump - OR - • SI Pump	a. Go to Step 22.
	b. RCP Trip Parameter - LESS THAN 1375 PSIG IN RCS	b. Go to Step 22.
	c. Stop all RCPs	

STEP	ACTION/EXPECTED RESPONSE	RESPONSE NOT OBTAINED
22	Check If SGs Are Not Faulted:	
	a. Check pressures in all SGs - • NO SG PRESSURE DECREASING IN AN UNCONTROLLED MANNER • NO SG COMPLETELY DEPRESSURIZED	a. Go to E-2, FAULTED STEAM GENERATOR ISOLATION, Step 1.
23	Check If SG Tubes Are Not Ruptured: • Condenser effluent radiation - NORMAL • Main steamline radiation - NORMAL	Go to E-3, STEAM GENERATOR TUBE RUPTURE, Step 1.
24	Check If RCS Is Intact: • Containment radiation - NORMAL • Containment pressure - NORMAL • Containment building level - NORMAL	Go to E-1, LOSS OF REACTOR OR SECONDARY COOLANT, Step 1.

STEP	ACTION/EXPECTED RESPONSE	RESPONSE NOT OBTAINED
25	Check If ECCS Flow Should Be Reduced:	
	a. RCS subcooling based on core exit TCs - GREATER THAN 30°F	a. DO NOT STOP ECCS PUMPS. Go to Step 27.
	b. Secondary heat sink: • Total EFW flow to intact SGs - GREATER THAN 470 GPM TOTAL COMBINED FLOW WITH INTACT SG WR LEVELS ABOVE TOP OF U-TUBES, 65% FOR NORMAL CONTAINMENT - OR - • NR level in at least one intact SG - GREATER THAN 28% FOR ADVERSE CONTAINMENT	b. IF neither condition satisfied, THEN DO NOT STOP CENTRIFUGAL CHARGING PUMPS OR SI PUMPS. Go to Step 27.
	c. RCS pressure - STABLE OR INCREASING	c. DO NOT STOP ECCS PUMPS. Go to Step 27.
	d. PRZR level - GREATER THAN 5%	d. DO NOT STOP ECCS PUMPS. Try to stabilize RCS pressure with normal spray. Return to Step 25a.
26	Go To ES-1.1, SI TERMINATION, Step 1	
27	Initiate Monitoring Of Critical Safety Function Status Trees	

CAUTION CST makeup should commence as early as possible to avoid low inventory problems.

STEP	ACTION/EXPECTED RESPONSE	RESPONSE NOT OBTAINED
28	Check SG Levels:	
	a. NR level - GREATER THAN 5%	a. Maintain total EFW flow greater than 470 GPM to intact SGs until NR level greater than 5% in at least one SG. DO NOT allow intact SG WR level to decrease below top of U-tubes.
	b. Control EFW flow to maintain NR level - BETWEEN 5% AND 50%	b. IF NR level in any SG continues to increase in an uncontrolled manner, THEN go to E-3, STEAM GENERATOR TUBE RUPTURE, Step 1.
29	Check Secondary Radiation - NORMAL USING RDMS: • Main steamlines - OR - • Condenser effluent	Go to E-3, STEAM GENERATOR TUBE RUPTURE, Step 1.

LEVEL ABOVE SG U-TUBES	
ADVERSE CONTM	NORMAL CONTM
NARROW RANGE LEVEL GREATER THAN 28%	WIDE RANGE LEVEL GREATER THAN 65%

STEP	ACTION/EXPECTED RESPONSE	RESPONSE NOT OBTAINED
30	Check Auxiliary Building Radiation - NORMAL USING RDMS	Evaluate cause of abnormal conditions. IF the cause is a loss of RCS inventory outside containment, THEN go to ECA-1.2, LOCA OUTSIDE CONTAINMENT, Step 1.
31	Check PRT Conditions - NORMAL • PORV OR SAFETY valve tailpipe temperature - LESS THAN 140°F • Pressure - BETWEEN 2 PSIG AND 4 PSIG • Level - BETWEEN 60% AND 86% • Temperature - LESS THAN 120°F	Evaluate cause of abnormal conditions. • Pressurizer PORV's • RHR relief valves • Letdown relief valve • Seal return header relief valve

CAUTION If offsite power is lost after SI reset, manual action may be required to restart safeguard equipment.

32	Reset SI	
33	Reset Containment Isolation Phase 'A' AND Phase 'B'	

STEP	ACTION/EXPECTED RESPONSE	RESPONSE NOT OBTAINED

NOTE If a LOP has occurred, reset EPS-RMO.

34 Reestablish Instrument Air Supplies While Continuing With This Guideline:

a. Restart service air compressors as follows:

1) Open SW isolation valves to SCCW

- SW-V4
- SW-V5

IF on CT, OPEN SW-V74 AND SW-V76, CLOSE SW-V75

2) Start ONE SCCW pump

3) Start at least one service air compressor

RESPONSE NOT OBTAINED: a. Open fire water cooling to service air compressors and start at least one service air compressor.

b. Restart containment air compressors as follows:

1) Open PCCW containment isolation valves

TRAIN A, LOOP A	TRAIN B, LOOP B
CC-V168	CC-V175
CC-V57	CC-V176
CC-V122	CC-V257
CC-V121	CC-V256

2) Start one containment air compressor

- IA-C-4A (Loop A cooled)
- IA-C-4B (Loop B cooled)

STEP	ACTION/EXPECTED RESPONSE	RESPONSE NOT OBTAINED
	CAUTION RCS pressure should be monitored. If RCS pressure drops below 200 PSIG, RHR pumps must be manually restarted to supply water to RCS.	
35	Check If RHR Pumps Should Be Stopped:	
	a. Check RCS pressure:	
	1) Pressure - GREATER THAN 200 PSIG	1) Go to E-1, LOSS OF REACTOR OR SECONDARY COOLANT, Step 1.
	2) Pressure - STABLE OR INCREASING	2) Go to Step 36.
	b. Stop RHR pumps and place in standby	
36	Check If Emergency Diesel Generators Should Be Stopped:	
	a. Verify AC emergency busses - ENERGIZED BY UATs OR RATs	a. Try to restore offsite power to AC emergency busses. IF offsite power can NOT be restored, THEN RESET EPS - RMO.
		1) Go to Step 37
	b. Stop emergency diesel - generators and reset for AUTO START	
	1) Stop diesel generator	
	2) Reset diesel generator	
37	Return To Step 19	

- END -

ATTACHMENT A

The following are symptoms that require a reactor trip, if one has not occurred:

FUNCTIONAL UNIT	TRIP SETPOINT
A. POWER RANGE NEUTRON FLUX:	
1) LOW SETPOINT	$\leq$ 25% OF RTP
2) HIGH SETPOINT	$\leq$ 109% OF RTP
B. POWER RANGE, NEUTRON FLUX, HIGH POSITIVE RATE	$\leq$ 5% OF RTP WITH A TIME CONSTANT $\geq$ 1 SECONDS
C. POWER RANGE, NEUTRON FLUX, HIGH NEGATIVE RATE	$\leq$ 5% OF RTP WITH A TIME CONSTANT $\geq$ 1 SECONDS
D. INTERMEDIATE RANGE, NEUTRON FLUX	$\leq$ 25% OF RTP
E. SOURCE RANGE, NEUTRON FLUX	$\leq$ 10^5 CPS
F. OVERTEMPERATURE ΔT	$\leq$ 109.95% $\pm$ PENALTIES
G. OVERPOWER ΔT	$\leq$ 109% - PENALTIES
H. PRESSURIZER PRESSURE -- LOW	$\geq$ 1945 PSIG
I. PRESSURIZER PRESSURE -- HIGH	$\leq$ 2385 PSIG
J. PRESSURIZER WATER LEVEL--HIGH	$\leq$ 92% OF INSTRUMENT SPAN
K. LOSS OF FLOW	$\geq$ 90% OF LOOP DESIGN FLOW

ATTACHMENT A
(cont.)

The following are symptoms that require a reactor trip, if one has not occurred:

FUNCTIONAL UNIT	TRIP SETPOINT
L. STEAM GENERATOR WATER LEVEL - LOW-LOW	≥ 15% OF NARROW RANGE RANGE INSTRUMENT SPAN
M. UNDERVOLTAGE - REACTOR COOLANT PUMPS	≥ 10,200 VOLTS AC
N. UNDERFREQUENCY - REACTOR COOLANT PUMPS	≥ 57.2 Hz
P. TURBINE TRIP	
1) LOW TRIP SYSTEM PRESSURE	≥ 800 PSIG
2) TURBINE STOP VALVE CLOSURE	ALL VALVES CLOSED
SAFETY INJECTION INPUT FROM ESF	NA

ATTACHMENT B

The following are symptoms that require a reactor trip and safety injection, if one has not occurred:

FUNCTIONAL UNIT	SI SETPOINT
A. PRESSURIZER PRESSURE - LOW	$\leq$ 1850 PSIG
B. CONTAINMENT PRESSURE - HIGH	$\geq$ 4.3 PSIG
C. STEAMLINE PRESSURE - LOW	$\leq$ 585 PSIG

ATTACHMENT C

ECCS VALVE ALIGNMENT - CCP
VIA BIT TO RCS COLD LEGS

VALVE	NOMENCLATURE	POSITION
CS-V142	CHARGING ISOL.	CLOSED
CS-V143	CHARGING ISOL.	CLOSED
CS-LCV-112B	CVCT OUTLET	CLOSED
CS-LCV-112C	CVCT OUTLET	CLOSED
CS-LCV-112D	RWST OUTLET	OPEN
CS-LCV-112E	RWST OUTLET	OPEN
CS-V844	BIT INLET	OPEN
CS-V65	BIT INLET	OPEN
CS-V845	BIT INLET	OPEN
CS-V66	BIT INLET	OPEN
CS-V846	BIT BYPASS	CLOSED
CS-V847	BIT BYPASS	CLOSED
CS-V165	BIT RECIRC. PUMP DISCHG.	CLOSED
CS-V173	BIT RECIRC. ISOL.	CLOSED
CS-V174	BIT RECIRC. ISOL.	CLOSED
SI-V138	BIT OUTLET TO RCS	OPEN
SI-V139	BIT OUTLET TO RCS	OPEN

ATTACHMENT D

- ECCS VALVE ALIGNMENT - SIP -
TO RCS COLD LEGS

VALVE	NOMENCLATURE	POSITION
CBS-V47	SI PUMP A SUCTION FROM RWST	OPEN
CBS-V49	SI PUMP A SUCTION FROM RWST	OPEN
SI-V90	SI PUMP A MIN FLOW TO RWST	OPEN
SI-V102	SI TO HOT LEGS	CLOSED
SI-V112	SI TO COLD LEGS	OPEN
SI-V114	SI TO COLD LEGS	OPEN
CS-V460	SI PUMP A SUCTION CROSSOVER	CLOSED
CS-V461	SI PUMP A SUCTION CROSSOVER	CLOSED
CS-V475	SI PUMP A SUCTION CROSSOVER	OPEN
CBS-V51	SI PUMP B SUCTION FROM RWST	OPEN
CBS-V53	SI PUMP B SUCTION FROM RWST	OPEN
SI-V89	SI PUMP B MIN FLOW TO RWST	OPEN
SI-V93	SI PUMP A & B MIN FLOW TO RWST	OPEN
SI-V111	SI TO COLD LEGS	OPEN
SI-V77	SI TO HOT LEGS	CLOSED

ATTACHMENT E

- ECCS VALVE ALIGNMENT - RHR -
PUMP TO RCS COLD LEGS

VALVE	NOMENCLATURE	POSITION
CBS-V8	CONTM SUMP TO RHR PUMP A & CBS PUMP A	CLOSED
CBS-V2	RWST TO CBS PUMP A & RHR PUMP A	OPEN
RC-V23	RHR SUCTION FROM RCS	CLOSED
RC-V88	RHR SUCTION FROM RCS	CLOSED
RC-V87	RHR SUCTION FROM RCS	CLOSED
RC-V22	RHR SUCTION FROM RCS	CLOSED
CBS-V5	RWST TO CBS PUMP B & RHR PUMP B	OPEN
CBS-V14	CONTM SUMP TO RHR PUMP B & CBS PUMP B	CLOSED
RH-V36	RHR TRAIN B TO SUCTION OF SI PUMP B	CLOSED
RH-V35	RHR TRAIN A TO SUCTION OF SI PUMP A	CLOSED
RH-V21	RHR SYSTEM B TO HOT LEGS	OPEN
RH-V32	RHR A/B TO HOT LEGS	CLOSED
RH-V26	RHR TRAIN B TO COLD LEGS	OPEN
RH-V22	RHR SYSTEM A TO HOT LEGS	OPEN
RH-V70	RHR A/B TO HOT LEGS	CLOSED
RH-V14	RHR TRAIN A TO COLD LEGS	OPEN

(T)

OPERATOR ACTION SUMMARY FOR E-0 SERIES PROCEDURES

1. RCP TRIP CRITERIA

Trip all RCPs if BOTH conditions listed below occur:

a. CCPs or SI pumps - AT LEAST ONE RUNNING

b. RCP Trip Parameter - LESS THAN 1375 PSIG

2. SI ACTUATION CRITERIA

Actuate SI and go to E-0, REACTOR TRIP OR SAFETY INJECTION, Step 1, if EITHER condition listed below occurs:

- RCS subcooling based on core exit TCs - LESS THAN 30°F
- Pressurizer level - CANNOT BE MAINTAINED GREATER THAN 5% [(30)% FOR ADVERSE CONTAINMENT]

3. RED PATH SUMMARY

a. SUBCRITICALITY - Nuclear power greater than 5%

b. CORE COOLING - Core exit TCs greater than 1200°F

- OR -

Core exit TCs greater than 700°F AND RVLIS full range less than 40% with no RCPs running

c. HEAT SINK - SG narrow range level in all SGs less than 28% AND total feedwater flow less than 470 gpm

d. INTEGRITY - Cold leg temperature decrease greater than 100°F in last 60 minutes AND RCS cold leg temperature less than 250°F

e. CONTAINMENT - Containment pressure greater than 52 PSIG

4. EFW SUPPLY

Commence CST makeup as soon as possible to avoid low inventory problems.

WOG EMERGENCY RESPONSE GUIDELINE E-0

A. PURPOSE

This guideline provides actions to verify proper response of the automatic protection systems following manual or automatic actuation of a reactor trip or safety injection, to assess plant conditions, and to identify the appropriate recovery guideline.

B. SYMPTOMS OR ENTRY CONDITIONS

1) The following are symptoms that require a reactor trip, if one has not occurred:
[Enter plant specific setpoints and requirements]

2) The following are symptoms of a reactor trip:

a. Any reactor trip annunciator lit.

b. Rapid decrease in neutron level indicated by nuclear instrumentation.

c. All shutdown and control rods are fully inserted. Rod bottom lights are lit.

3) The following are symptoms that require a reactor trip and safety injection, if one has not occured:
[Enter plant specific setpoints and requirements]

4) The following are symptoms of a reactor trip and safety injection:

a. Any SI annunciator lit.

b. SI pumps running:

[Enter plant specific list]

NOTE: o Steps 1 through 14 are IMMEDIATE ACTION steps.

o Foldout page should be open.

1	Verify Reactor Trip: o Rod bottom lights - LIT o Reactor trip and bypass breakers - OPEN o Rod position indicators - AT ZERO o Neutron flux - DECREASING	Manually trip reactor. IF reactor will NOT trip, THEN go to FR-S.1, RESPONSE TO NUCLEAR POWER GENERATION/ATWS, Step 1.
2	Verify Turbine Trip:	
	a. All turbine stop valves - CLOSED	a. Manually trip turbine.
3	Verify Power To AC Emergency Busses:	
	a. AC emergency busses - AT LEAST ONE ENERGIZED	a. Try to restore power to at least one ac emergency bus. IF power can NOT be restored to at least one ac emergency bus, THEN go to ECA-0.0, LOSS OF ALL AC POWER, Step 1.
	b. AC emergency busses - ALL ENERGIZED	b. Try to restore power to deenergized ac emergency busses.
4	Check If SI Is Actuated: [Enter plant specific means]	Check if SI is required. IF SI is required, THEN manually actuate. IF SI is NOT required, THEN go to ES-0.1, REACTOR TRIP RESPONSE, Step 1.

5	Verify FW Isolation: o Flow control valves - CLOSED o Flow control bypass valves - CLOSED o FW isolation valves - CLOSED o SG blowdown isolation valves - CLOSED o SG sample isolation valves - CLOSED	Manually close valves as necessary.
6	Verify Containment Isolation Phase A:	
	a. Phase A - ACTUATED	a. Manually actuate Phase A.
	b. Phase A valves - CLOSED	b. Manually close valves.
7	Verify AFW Pumps Running:	
	a. MD pumps - RUNNING	a. Manually start pumps.
	b. Turbine-driven pump - RUNNING IF NECESSARY	b. Manually open steam supply valves.
8	Verify SI Pumps Running: o Charging/SI pumps - RUNNING o High-head SI pumps - RUNNING o Low-head SI pumps - RUNNING	Manually start pumps.
9	Verify CCW Pumps - RUNNING	Manually start pumps.

10	Verify Service Water Pumps - RUNNING	Manually start pumps.
11	Verify Containment Fan Coolers - RUNNING IN EMERGENCY MODE	Manually start fan coolers in emergency mode.
12	Verify Containment Ventilation Isolation:	
	a. Dampers - CLOSED	a. Manually close dampers.

[Appropriate steps for verification of other essential equipment as required by the specific plant design should be placed after Step 12.]

13	Check If Main Steamlines Should Be Isolated:	
	a. [Enter plant specific means or setpoints]	a. Go to Step 14.
	b. Verify main steamline isolation and bypass valves - CLOSED	b. Manually close valves.
14	Verify Containment Spray Not Required:	
	a. Containment Pressure - HAS REMAINED LESS THAN (1) PSIG	a. Perform the following: 1) Verify containment spray initiated. <u>IF</u> <u>NOT</u>, <u>THEN</u> manually initiate. 2) Verify containment isolation Phase B valves closed. <u>IF</u> <u>NOT</u>, <u>THEN</u> manually close valves. 3) Stop all RCPs.

15	Verify SI Flow:	
	a. Charging/SI pump flow indicators - CHECK FOR FLOW	a. Manually start pumps and align valves.
	b. RCS pressure - LESS THAN (2) PSIG [(3) PSIG FOR ADVERSE CONTAINMENT]	b. Go to Step 16.
	c. High-head SI pump flow indicators - CHECK FOR FLOW	c. Manually start pumps and align valves.
	d. RCS pressure - LESS THAN (4) PSIG [(5) PSIG FOR ADVERSE CONTAINMENT]	d. Go to Step 16.
	e. Low-head SI pump flow indicators - CHECK FOR FLOW	e. Manually start pumps and align valves.
16	Verify AFW Flow - GREATER THAN (6) GPM	Manually start pumps and align valves as necessary. IF AFW flow greater than (6) gpm can NOT be established, THEN go to FR-H.1, RESPONSE TO LOSS OF SECONDARY HEAT SINK, Step 1.
17	Verify AFW Valve Alignment - PROPER EMERGENCY ALIGNMENT	Manually align valves as necessary.
18	Verify SI Valve Alignment - PROPER EMERGENCY ALIGNMENT	Manually align valves as necessary.

19	Check RCS Average Temperature - STABLE AT OR TRENDING TO (7)°F	IF temperature less than (7)°F and decreasing, THEN: a) Stop dumping steam. b) IF cooldown continues, THEN control total feed flow. Maintain total feed flow greater than (6) gpm until narrow range level greater than (8)% [(9)% FOR ADVERSE CONTAINMENT] in at least one SG. c) IF cooldown continues, THEN close main steamline isolation and bypass valves. IF temperature greater than (7)°F and increasing, THEN: o Dump steam to condenser. -OR- o Dump steam using SG PORVs.

20 Check PRZR PORVs And Spray Valves:

a. PORVs - CLOSED

a. IF PRZR pressure less than (10) psig, THEN manually close PORVs. IF any valve can NOT be closed, THEN manually close its block valve. IF block valve can NOT be closed, THEN go to E-1, LOSS OF REACTOR OR SECONDARY COOLANT, Step 1.

b. Normal PRZR spray valves - CLOSED

b. IF PRZR pressure less than (11) psig, THEN manually close valves. IF valves can NOT be closed, THEN stop RCP(s) supplying failed spray valve(s).

NOTE: Seal injection flow should be maintained to all RCPs.

21 Check If RCPs Should Be Stopped:

a. SI pumps - AT LEAST ONE RUNNING

a. Go to Step 22.

- o Charging/SI

-OR-

- o High-head SI

b. RCP Trip Parameter - LESS THAN (12) [(13) FOR ADVERSE CONTAINMENT]

b. Go to Step 22.

c. Stop all RCPs

22	Check If SGs Are Not Faulted:	
	a. Check pressures in all SGs - o NO SG PRESSURE DECREASING IN AN UNCONTROLLED MANNER o NO SG COMPLETELY DEPRESSURIZED	a. Go to E-2, FAULTED STEAM GENERATOR ISOLATION, Step 1.
23	Check If SG Tubes Are Not Ruptured: o Condenser air ejector radiation - NORMAL o SG blowdown radiation - NORMAL	Go to E-3, STEAM GENERATOR TUBE RUPTURE, Step 1.
24	Check If RCS Is Intact: o Containment radiation - NORMAL o Containment pressure - NORMAL o Containment recirculation sump level - NORMAL	Go to E-1, LOSS OF REACTOR OR SECONDARY COOLANT, Step 1.

25	Check If SI Flow Should Be Reduced:	
	a. RCS subcooling based on core exit TCs - GREATER THAN (14)°F	a. DO NOT STOP SI PUMPS. Go to Step 27.
	b. Secondary heat sink: o Total feed flow to SGs - GREATER THAN (6) GPM -OR- o Narrow range level in at least one SG - GREATER THAN (8)%	b. IF neither condition satisfied, THEN DO NOT STOP SI PUMPS. Go to Step 27.
	c. RCS pressure - STABLE OR INCREASING	c. DO NOT STOP SI PUMPS. Go to Step 27.
	d. PRZR level - GREATER THAN (15)%	d. DO NOT STOP SI PUMPS. Try to stabilize RCS pressure with normal PRZR spray. Return to Step 25a.
26	Go To ES-1.1, SI TERMINATION, Step 1	

27	Initiate Monitoring Of Critical Safety Function Status Trees	

CAUTION: Alternate water sources for AFW pumps will be necessary if CST level decreases to less than (16).

28	Check SG Levels:	
	a. Narrow range level - GREATER THAN (8)%	a. Maintain total feed flow greater than (6) gpm until narrow range level greater than (8)% in at least one SG.
	b. Control feed flow to maintain narrow range level between (8)% and 50%	b. IF narrow range level in any SG continues to increase in an uncontrolled manner, THEN go to E-3, STEAM GENERATOR TUBE RUPTURE, Step 1.
29	Check Secondary Radiation - NORMAL [Enter plant specific means]	Go to E-3, STEAM GENERATOR TUBE RUPTURE, Step 1.
30	Check Auxiliary Building Radiation - NORMAL	Evaluate cause of abnormal conditions. IF the cause is a loss of RCS inventory outside containment, THEN go to ECA-1.2, LOCA OUTSIDE CONTAINMENT, Step 1.
31	Check PRT Conditions - NORMAL	Evaluate cause of abnormal conditions.

CAUTION: If offsite power is lost after SI reset, manual action may be required to restart safeguards equipment.

32	Reset SI	
33	Reset Containment Isolation Phase A And Phase B	
34	Establish Instrument Air To Containment	Start one air compressor and establish instrument air to containment.

CAUTION: RCS pressure should be monitored. If RCS pressure decreases to less than (4) psig the low-head SI pumps must be manually restarted to supply water to the RCS.

35	Check If Low-Head SI Pumps Should Be Stopped:	
	a. Check RCS pressure:	
	1) Pressure - GREATER THAN (4) PSIG	1) Go to E-1, LOSS OF REACTOR OR SECONDARY COOLANT, Step 1.
	2) Pressure - STABLE OR INCREASING	2) Go to Step 36.
	b. Stop low-head SI pumps and place in standby	

36	Check If Diesel Generators Should Be Stopped:	
	a. Verify ac emergency busses - ENERGIZED BY OFFSITE POWER	a. Try to restore offsite power to ac emergency busses. IF offsite power can NOT be restored, THEN load the following equipment on ac emergency busses: [Enter plant specific list]
	b. Stop any unloaded diesel generator and place in standby	
37	Return To Step 19	

- END -

FOOTNOTES

(1) Enter plant specific containment pressure setpoint for spray actuation.

(2) Enter plant specific value for the shutoff head pressure of the high-head SI pumps, plus allowances for normal channel accuracy.

(3) Enter plant specific value for the shutoff head pressure of the high-head SI pumps, plus allowances for normal channel accuracy and post accident transmitter errors, not to exceed 2000 psig.

(4) Enter plant specific value for the shutoff head pressure of low-head SI pumps, plus allowances for normal channel accuracy.

(5) Enter plant specific value for the shutoff head pressure of the low-head SI pumps, plus allowances for normal channel accuracy and post accident transmitter errors.

(6) 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).

(7) Enter plant specific no-load temperature.

(8) Enter plant specific value showing SG level just in the narrow range, including allowances for normal channel accuracy.

(9) 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%.

(10) Enter PRZR PORV pressure setpoint.

(11) Enter PRZR Spray pressure setpoint.

(12) Enter plant specific RCP trip parameter and setpoint, including allowances for normal channel accuracy. Refer to Generic Issues section of Executive Volume.

(13) Enter plant specific RCP trip parameter and setpoint, including allownaces for normal channel accuracy and post accident transmitter errors. Refer to Generic Issues section of Executive Volume.

FOOTNOTES (Continued)

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

(15) Enter plant specific value showing PRZR level just in range, including allowances for normal channel accuracy.

(16) Enter plant specific value corresponding to CST low level switchover setpoint in plant specific units.

SEABROOK TEST PROCEDURE FR-H.1

Code:	Symptom/Title:	Procedure No./ Revision No./Date:
FR-H.1 Rev. 1-T	RESPONSE TO LOSS OF SECONDARY HEAT SINK	OS-1353.1 0-T / 10/11/83

A. PURPOSE

This procedure provides actions to respond to a loss of secondary heat sink in all steam generators.

B. SYMPTOMS OR ENTRY CONDITIONS

This procedure is entered from:

1) E-0, REACTOR TRIP OR SAFETY INJECTION, Step 16, when minimum EFW flow is not verified.

2) F-0.3, HEAT SINK Critical Safety Function Status Tree on a RED condition.

STEP	ACTION/EXPECTED RESPONSE	RESPONSE NOT OBTAINED

NOTE If ECA-2.1, UNCONTROLLED DEPRESSURIZATION OF ALL STEAM GENERATORS, is in effect and total feed flow is less than 470 gpm due to operator action, then this procedure should not be performed.

CAUTION

- If RCS temperature and pressure are increasing due to loss of secondary heat sink, RCPs should be tripped and Steps 10 through 16 should be immediately initiated for bleed and feed.
- Feed flow should not be reestablished to any faulted SG if a non-faulted SG is available.

STEP	ACTION/EXPECTED RESPONSE	RESPONSE NOT OBTAINED
1	Check If Secondary Heat Sink Is Required:	
	a. RCS pressure - GREATER THAN ANY NON-FAULTED SG PRESSURE	a. Go to E-1, LOSS OF REACTOR OR SECONDARY COOLANT, Step 1.
	b. RCS hot leg temperature - GREATER THAN 350°F [320°F FOR ADVERSE CONTAINMENT]	b. Try to place RHR System in service while continuing in this guideline. Refer to OS-1013.03 and OS-1013.04, RHR TRAIN A and RHR TRAIN B STARTUP AND OPERATION. <u>IF</u> adequate cooling with RHR System established, <u>THEN</u> return to procedure and step in effect.

Code:	Symptom/Title:	Procedure No./ Revision No./Date:
FR-H.1 Rev. 1-T	RESPONSE TO LOSS OF SECONDARY HEAT SINK	OS-1353.1 0-T / 10/11/83

STEP	ACTION/EXPECTED RESPONSE	RESPONSE NOT OBTAINED
2	Try To Establish EFW Flow To At Least One SG:	
	a. Check control room indications for cause of EFW failure: • CST level low • EFW pump power supply • EFW valve alignment • EFW pump failure	
	b. Try to restore EFW flow	
	c. Check total flow to SGs - GREATER THAN 470 GPM	c. Dispatch operator to locally restore EFW flow. Check suction and discharge valve lineup. Go to Step 3.
	d. Return to procedure and step in effect	
3	Stop All RCPs	
4	Check CCP Status - AT LEAST ONE AVAILABLE	Go to Step 10.

CAUTION If offsite power is lost after SI reset, manual action may be required to restart safeguards equipment.

STEP	ACTION/EXPECTED RESPONSE	RESPONSE NOT OBTAINED
5	Try To Establish SUFP Flow To At Least One SG:	
	a. IF offsite power is NOT available, THEN power SUFP from Bus E5	
	b. Check CST inventory adequate for SUFP operation on lower suction. IF adequate, THEN transfer to lower suction - CO-V142 OPEN	b. Break condenser vacuum and take suction from condenser hotwell - CO-V145 OPEN.
	c. Establish feed path through main feed header:	c. Establish feed path through EFW headers:
	1) Reset FW isolation	1) Open FW-V163.
	2) Place main feed regulating valve controllers in manual and close valves	2) Open FW-V156. IF no feed is obtained, THEN go to Step 9.
	3) Open FW isolation valves	
	IF no FW isolation valve can be opened, THEN go to Step 9	
	d. Start the SUFP and control feed using the FW bypass valves	d. Go to Step 7.

STEP	ACTION/EXPECTED RESPONSE	RESPONSE NOT OBTAINED
6	Check SG Levels:	
	a. WR level - ABOVE TOP OF U-TUBES IN AT LEAST ONE SG, 65% [95% FOR ADVERSE CONTAINMENT]	a. IF feed flow to at least one SG verified, THEN maintain flow to restore SG level above U-tubes.

LEVEL ABOVE SG U-TUBES	
ADVERSE CONTM	NORMAL CONTM
NARROW RANGE LEVEL GREATER THAN 28%	WIDE RANGE LEVEL GREATER THAN 65%

IF feed flow NOT verified, THEN go to Step 7.

b. Return to procedure and step in effect.

STEP	ACTION/EXPECTED RESPONSE	RESPONSE NOT OBTAINED
	CAUTION Following block of automatic SI actuation, manual SI actuation may be required if conditions degrade.	
7	Try To Establish Feed Flow From Condensate System:	
	a. Depressurize RCS to less than 530 PSIG:	
	1) Check letdown - IN SERVICE	1) Use one PRZR PORV. <u>IF NOT</u>, <u>THEN</u> use auxiliary spray. Go to Step 7b.
	2) Use auxiliary spray	2) Use one PRZR PORV.
	b. Block SI signals: • Low steamline pressure SI • Low PRZR pressure SI	
	c. Depressurize at least one SG to less than 530 PSIG:	
	1) Dump steam to condenser at maximum rate	1) Manually or locally dump steam using SG ASDVs. <u>IF</u> SG ASDVs <u>NOT</u> available, <u>THEN</u> go to Step 9.
	d. Establish condensate flow:	
	1) Locally open FW-V103, SGFP bypass valve	
	2) Close <u>both</u> SGFP discharge valves	
	3) Close <u>both</u> SGFP recirc valves	
	4) Start at least one condensate pump	4) Go to Step 9
	5) Close condensate header long term recirc valve	
	6) Reset FW isolation	6) Establish flow through EFW header: a) Open FW-V163 b) Open FW-V156
	7) Open FW isolation valves	
	8) Control flow using FW bypass valves	

STEP	ACTION/EXPECTED RESPONSE	RESPONSE NOT OBTAINED

8 Check SG Levels:

a. WR level - ABOVE TOP OF U-TUBES IN AT LEAST ONE SG, 65% [95% FOR ADVERSE CONTAINMENT]

a. IF feed flow to at least one SG verified, THEN maintain flow to restore SG level above U-tubes.

LEVEL ABOVE SG U-TUBES	
ADVERSE CONTM	NORMAL CONTM
NARROW RANGE LEVEL GREATER THAN 28%	WIDE RANGE LEVEL GREATER THAN 65%

IF feed flow NOT verified, THEN go to Step 9.

b. Return to procedure and step in effect

9 Check For Loss Of Secondary Heat Sink:

a. SG WR level - LESS THAN TOP OF U-TUBES WITH NO FEED FLOW AVAILABLE

a. Return to Step 1.

LEVEL ABOVE SG U-TUBES	
ADVERSE CONTM	NORMAL CONTM
NARROW RANGE LEVEL GREATER THAN 28%	WIDE RANGE LEVEL GREATER THAN 65%

CAUTION Steps 10 through 16 must be performed quickly in order to establish RCS heat removal by RCS bleed and feed.

10 Actuate SI

STEP	ACTION/EXPECTED RESPONSE	RESPONSE NOT OBTAINED
11	Verify RCS Feed Path: a. Check pump status: • CCPs - AT LEAST ONE RUNNING - OR - • SI pumps - AT LEAST ONE RUNNING b. Check valve alignment for operating pumps - PROPER EMERGENCY ALIGNMENT ON STATUS PANEL	Manually start pumps and align valves as necessary per Attachment A to establish feed path. IF a feed path can NOT be established, THEN continue attempts to establish feed flow. Return to Step 5.
12	Reset SI	
13	Reset Containment Isolation Phase A And Phase B	
14	Establish RCS Bleed Path: a. Verify power to PRZR PORV block valves - AVAILABLE b. Verify PRZR PORV block valves - ALL OPEN c. Open all PRZR PORVs	 a. Restore power to block valves. b. Open block valves.

STEP	ACTION/EXPECTED RESPONSE	RESPONSE NOT OBTAINED
15	Verify Adequate RCS Bleed Path:	
	a. PRZR PORVs - AT LEAST TWO OPEN	a. Perform the following: 1) Depressurize at least one intact SG to atmospheric pressure using SG ASDV. 2) Align any available low pressure water source to the depressurized SG(s).
16	Maintain RCS Heat Removal: • Maintain ECCS flow - GREATER THAN REQUIRED ON FIGURE OS-1353.1-1 • Maintain PRZR PORVs - AT LEAST TWO OPEN	

CAUTION If RWST level reaches decreases to less than 23.5%, the ECCS system should be aligned for cold leg recirculation using ES-1.3, TRANSFER TO COLD LEG RECIRCULATION.

STEP	ACTION/EXPECTED RESPONSE	RESPONSE NOT OBTAINED
17	Continue Attempts To Establish Secondary Heat Sink In At Least One SG: • EFW flow • SUFP flow • Condensate flow	
18	Check For Adequate Secondary Heat Sink: a. SG WR level - ABOVE TOP OF U-TUBES IN AT LEAST ONE SG, 65% [95% FOR ADVERSE CONTAINMENT]	a. Return to Step 17.
19	Check RCS Temperatures: • Core exit TCs - DECREASING • RCS hot leg temperatures - DECREASING	Return to Step 17.

STEP	ACTION/EXPECTED RESPONSE	RESPONSE NOT OBTAINED

NOTE
- After stopping any ECCS pump, RCS pressure should be allowed to stabilize before stopping another ECCS pump.
- The charging pumps and SI pumps should be stopped on alternate ECCS trains when possible.

20 Check If One CCP Should Be Stopped:

a. Two CCPs - RUNNING — a. Go to Step 21.

b. Determine required RCS subcooling from table:

REQUIRED SUBCOOLING (°F)

SI PUMP STATUS	W/RCPs		WO/RCPs	
	NORMAL CONTAINMENT	ADVERSE CONTAINMENT	NORMAL CONTAINMENT	ADVERSE CONTAINMENT
NONE RUNNING	121°	121°	135°	135°
ONE RUNNING	68°	68°	75°	75°
TWO RUNNING	59°	59°	65°	65°

c. RCS subcooling based on core exit TCs - GREATER THAN REQUIRED SUBCOOLING — c. DO NOT STOP CCP. Go to Step 23.

d. PRZR level - GREATER THAN 5% [30% FOR ADVERSE CONTAINMENT] — d. DO NOT STOP CCP. Go to Step 23.

e. Stop one CCP

STEP	ACTION/EXPECTED RESPONSE	RESPONSE NOT OBTAINED
21	Check If One SI Pump Should Be Stopped:	
	a. Any SI pump - RUNNING	a. Go to Step 22.
	b. Determine required RCS subcooling from table:	

SI PUMP STATUS	W/RCP				WO/RCP			
	ONE CCP		NO CCP		ONE CCP		NO CCP	
	NORMAL	ADVERSE	NORMAL	ADVERSE	NORMAL	ADVERSE	NORMAL	ADVERSE
	CONTAINMENT		CONTAINMENT		CONTAINMENT		CONTAINMENT	
ONE RUNNING	236°	236°	---	---	236°	236°	---	---
TWO RUNNING	70°	70°	104°	104°	77°	77°	117°	117°

STEP	ACTION/EXPECTED RESPONSE	RESPONSE NOT OBTAINED
	c. RCS subcooling based on core exit TCs - GREATER THAN REQUIRED SUBCOOLING	c. DO NOT STOP SI PUMP. Go to Step 23.
	d. PRZR level - GREATER THAN 5% [30% FOR ADVERSE CONTAINMENT]	d. DO NOT STOP SI PUMP. Go to Step 23.
	e. Stop one additional SI pump	
	f. Return to Step 21a	

Code:	Symptom/Title:	Procedure No./ Revision No./Date:
FR-H.1 Rev. 1-T	RESPONSE TO LOSS OF SECONDARY HEAT SINK	OS-1353.1 0-T / 10/11/83

STEP	ACTION/EXPECTED RESPONSE	RESPONSE NOT OBTAINED
22	Check If Normal Charging Should Be Established:	
	a. Check the following: • SI pumps - STOPPED • CCPs - ALL BUT ONE STOPPED	a. DO NOT ISOLATE BIT. Return to Step 20.
	b. RCS subcooling based on core exit TCs - GREATER THAN 100°F	b. DO NOT ISOLATE BIT. Go to Step 23.
	c. PRZR level - GREATER THAN 5% [30% FOR ADVERSE CONTAINMENT]	c. DO NOT ISOLATE BIT. Go to Step 23.
	d. Go to Step 24	

NOTE After closing a PRZR PORV, it may be necessary to wait for RCS pressure to increase to permit stopping SI pumps in Steps 20 and 21.

STEP	ACTION/EXPECTED RESPONSE	RESPONSE NOT OBTAINED
23	Check PRZR PORV Status:	
	a. PRZR PORVs - ANY OPEN	a. Go to E-1, LOSS OF REACTOR OR SECONDARY COOLANT, Step 1.
	b. Close one PRZR PORV	b. Close associated PORV block valve. IF block valve can NOT be closed, THEN go to E-1, LOSS OF REACTOR OR SECONDARY COOLANT, Step 1.
	c. Return to Step 20	

STEP	ACTION/EXPECTED RESPONSE	RESPONSE NOT OBTAINED
24	Establish 60 GPM Charging Flow: a. Open charging line isolation valves b. Establish 60 GPM charging flow using flow control valve c. Adjust seal injection flow as necessary using HCV-182	
25	Isolate BIT: a. Close inlet isolation valves b. Close outlet isolation valves c. Check BIT bypass valves closed	
26	Check PRZR PORVs - ALL CLOSED	Close PRZR PORVs. IF any valve can NOT be closed, THEN manually close its block valve.
27	Control Charging Flow To Maintain PRZR Level	
28	Go To ES-1.1, SI TERMINATION, Step 11	

- END -

ATTACHMENT A

ECCS INJECTION ALIGNMENT

- CCPs VIA BIT TO RCS COLD LEGS

VALVE	NAME	POSITION
CS-LCV-112D	RWST OUTLET	OPEN
CS-LCV-112E	RWST OUTLET	OPEN
CS-V142	CHARGING ISOL.	CLOSED
CS-V143	CHARGING ISOL.	CLOSED
CS-LCV-112B	CVCT OUTLET	CLOSED
CS-LCV-112C	CVCT OUTLET	CLOSED
CS-V844	BIT INLET	OPEN
CS-V65	BIT INLET	OPEN
CS-V845	BIT INLET	OPEN
CS-V66	BIT INLET	OPEN
CS-V846	BIT BYPASS	CLOSED
CS-V847	BIT BYPASS	CLOSED
CS-V165	BIT RECIRC. PUMP DISCH.	CLOSED
CS-V173	BIT RECIRC. ISOL.	CLOSED
CS-V174	BIT RECORC. ISOL.	CLOSED
SI-V138	BIT OUTLET TO RCS	OPEN
SI-V139	VIT OUTLET TO RCS	OPEN

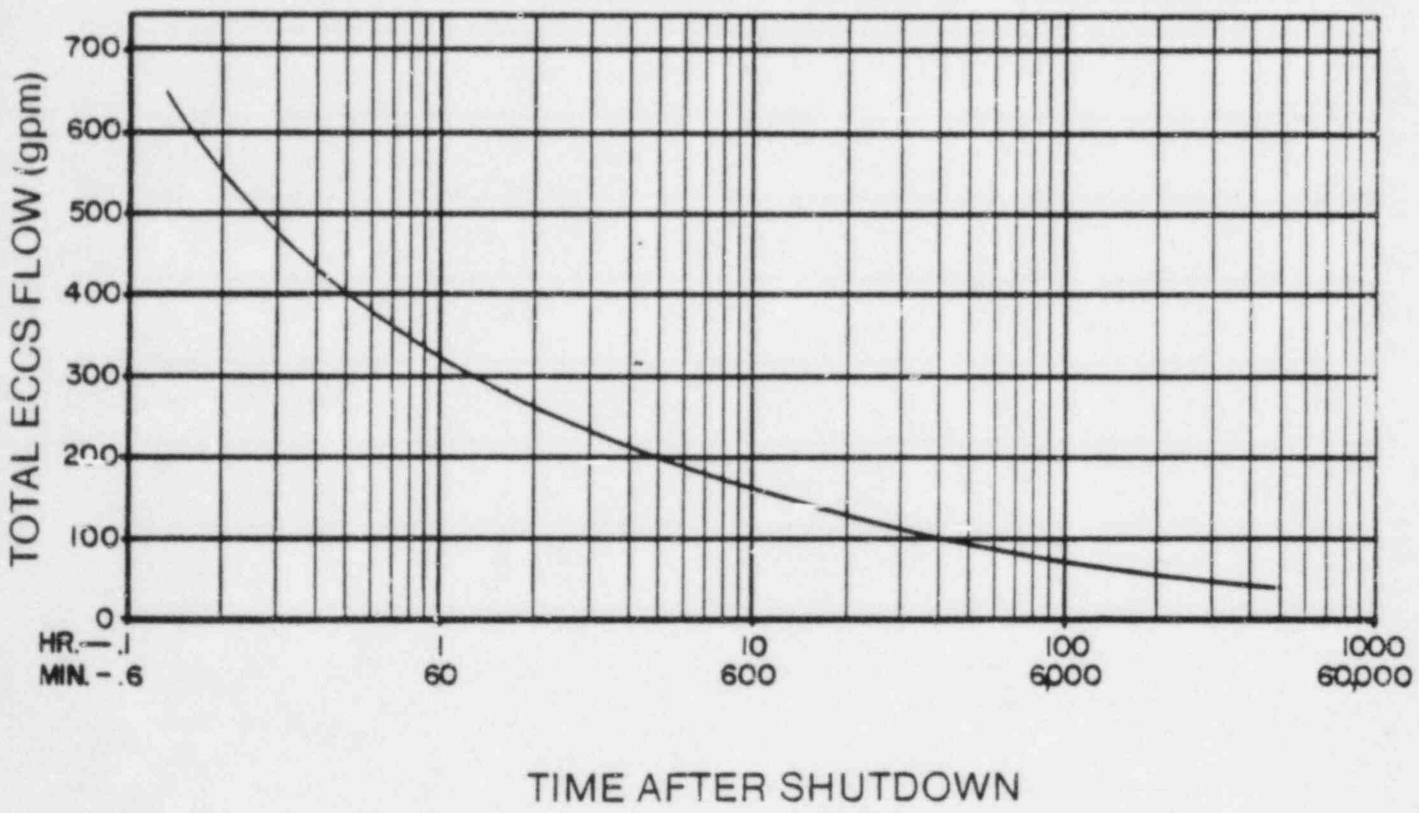
ATTACHMENT A
(CONTINUED)

- SI PUMPS TO RCS COLD LEGS

VALVE	NAME	POSITION
CBS-V47	SI-P-6A SUCTION FROM RWST	OPEN
CBS-V49	SI-P-6A SUCTION FROM RWST	OPEN
SI-V90	SI-P-6A MIN FLOW TO RWST	OPEN
SI-V102	SI TO HOT LEGS	CLOSED
SI-V112	SI TO COLD LEGS	OPEN
SI-V114	SI TO COLD LEGS	OPEN
CS-V460	SI-P-6A SUCTION CROSSOVER	CLOSED
CS-V461	SI-P-6A SUCTION CROSSOVER	CLOSED
CS-V475	SI-P-6A SUCTION CROSSOVER	OPEN
CBS-V51	SI-P-6B SUCTION FROM RWST	OPEN
CBS-V53	SI-P-6B SUCTION FROM RWST	OPEN
SI-V89	SI-P-6B MIN FLOW TO RWST	OPEN
SI-V93	SI-P-6A & B MIN FLOW TO RWST	OPEN
SI-V111	SI TO COLD LEGS	OPEN
SI-V77	SI TO HOT LEGS	CLOSED

CODE:	SYMPTOM / TITLE:	PROCEDURE NO. REVISION NO. / DATE:
FR-H.1 REV.1-7	**RESPONSE TO LOSS OF SECONDARY HEAT SINK**	OS-1353.1 00T/ 09/22/83

FIGURE OS-1353.1-1
Required ECCS Flow
for
Core Bleed and Feed Cooling to Remove Decay Heat



NOTES: RCPs NOT RUNING ASSUMED
88°F ECCS EOL CONDITIONS

WOG EMERGENCY RESPONSE GUIDELINE FR-H.1

A. PURPOSE

This guideline provides actions to respond to a loss of secondary heat sink in all steam generators.

B. SYMPTOMS OR ENTRY CONDITIONS

This guideline is entered from:

1) E-0, REACTOR TRIP OR SAFETY INJECTION, Step 16, when minimum AFW flow is not verified.

2) F-0.3, HEAT SINK Critical Safety Function Status Tree on a RED condition.

NOTE If ECA-2.1, UNCONTROLLED DEPRESSURIZATION OF ALL STEAM GENERATORS, is in effect and total feed flow is less than (1) gpm due to operator action, this guideline should not be performed.

CAUTION
- o If parameter (2) [(3) FOR ADVERSE CONTAINMENT] is exceeded due to loss of secondary heat sink, RCPs should be tripped and Steps 10 through 16 should be immediately initiated for bleed and feed.
- o Feed flow should not be reestablished to any faulted SG if a non-faulted SG is available.

1	Check If Secondary Heat Sink Is Required:	
	a. RCS pressure - GREATER THAN ANY NON FAULTED SG PRESSURE	a. Go To E-1, LOSS OF REACTOR OR SECONDARY COOLANT, Step 1.
	b. RCS temperature - GREATER THAN (4)°F [(5)°F FOR ADVERSE CONTAINMENT]	b. Try to place RHR System in service while continuing in this guideline. Refer to [Enter plant specific procedure number and title]. <u>IF</u> adequate cooling with RHR System established, <u>THEN</u> return to guideline and step in effect.

2	Try To Establish AFW Flow To At Least One SG:	
	a. Check control room indications for cause of AFW failure: o CST level o AFW pump power supply o AFW valve alignment	
	b. Try to restore AFW flow	
	c. Check total flow to SGs - GREATER THAN (1) GPM	c. Dispatch operator to locally restore AFW flow. Go to Step 3.
	d. Return to guideline and step in effect	
3	Stop All RCPs	
4	Check Charging/SI Pump Status - AT LEAST ONE AVAILABLE	Go to Step 10.

CAUTION If offsite power is lost after SI reset, manual action may be required to restart safeguards equipment.

5	Try To Establish Main FW Flow To At Least One SG:	
	a. Check Condensate System - IN SERVICE	a. Try to place Condensate System in service. <u>IF</u> <u>NOT</u>, <u>THEN</u> go to Step 9.
	b. Check FW isolation valves - OPEN	b. Perform the following: 1) Reset SI if necessary. 2) Reset FW isolation. 3) Open FW isolation valves. <u>IF</u> no FW isolation valve can be opened, <u>THEN</u> go to Step 9.
	c. Establish Main FW flow: [Enter plant specific means]	c. Go to Step 7.

6 Check SG Levels:

a. Narrow range level in at least one SG - GREATER THAN (6)% [(7)% FOR ADVERSE CONTAINMENT]

a. IF feed flow to at least one SG verified, THEN maintain flow to restore narrow range level to greater than (6)% [(7)% FOR ADVERSE CONTAINMENT]. IF NOT verified, THEN go to Step 7.

b. Return to guideline and step in effect

CAUTION Following block of automatic SI actuation, manual SI actuation may be required if conditions degrade.

7 Try To Establish Feed Flow From Condensate System:

a. Depressurize RCS to less than (8) PSIG:

1) Check letdown - IN SERVICE

1) Use one PRZR PORV. IF NOT, THEN use auxiliary spray. Go to Step 7b.

2) Use auxiliary spray

2) Use one PRZR PORV.

b. Block SI signals:

- o Low Steamline Pressure SI
- o Low PRZR Pressure SI

c. Depressurize at least one SG to less than (9) PSIG:

1) Dump steam to condenser at maximum rate

1) Manually or locally dump steam from SGs:

- o Use PORV.

-OR-

- o [Enter plant specific means].

IF NOT, THEN go to Step 9.

d. Establish condensate flow: [Enter plant specific means]

d. Go to Step 9.

Step	Action/Expected Response	Response Not Obtained
8	Check SG Levels:	
	a. Narrow range level in at least one SG - GREATER THAN (6)% [(7)% FOR ADVERSE CONTAINMENT]	a. IF feed flow to at least one SG verified, THEN maintain flow to restore narrow range level to greater than (6)% [(7)% FOR ADVERSE CONTAINMENT]. IF NOT verified, THEN go to Step 9.
	b. Return to guideline and step in effect	
9	Check For Loss Of Secondary Heat Sink:	
	a. Parameter (2) [(3) FOR ADVERSE CONTAINMENT] - EXCEEDED	a. Return to Step 1.

CAUTION Steps 10 through 16 must be performed quickly in order to establish RCS heat removal by RCS bleed and feed.

Step	Action/Expected Response	Response Not Obtained
10	Actuate SI	
11	Verify RCS Feed Path:	Manually start pumps and align valves as necessary to establish feed path. IF a feed path can NOT be established, THEN continue attempts to establish AFW flow. Return to Step 5.
	a. Check pump status: o Charging/SI pumps - AT LEAST RUNNING -OR- o High-head SI pumps - AT LEAST ONE RUNNING	
	b. Check valve alignment for operating pumps - PROPER EMERGENCY ALIGNMENT	
12	Reset SI	
13	Reset Containment Isolation Phase A And Phase B	

14	Establish Instrument Air To Containment	Start one air compressor and establish instrument air to containment.
15	Establish RCS Bleed Path:	
	a. Verify power to PRZR PORV block valves - AVAILABLE	a. Restore power to block valves.
	b. Verify PRZR PORV block valves - ALL OPEN	b. Open block valves.
	c. Open all PRZR PORVs	
16	Verify Adequate RCS Bleed Path:	
	a. PRZR PORVs - AT LEAST TWO OPEN	a. Perform the following: 1) Open all RCS high point vents: [Enter plant specific list] 2) Depressurize at least one intact SG to atmospheric pressure using SG PORV. 3) Align any available low pressure water source to the depressurized SG(s).
17	Maintain RCS Heat Removal: o Maintain SI flow o Maintain PRZR PORVs - AT LEAST TWO OPEN	

CAUTION If RWST level decreases to less than (10), the SI system should be aligned for cold leg recirculation using ES-1.3, TRANSFER TO COLD LEG RECIRCULATION.

18 Continue Attempts To Establish Secondary Heat Sink In At Least One SG:

- o AFW flow
- o Main FW flow
- o Condensate flow
- o Other low pressure flow

19 Check For Adequate Secondary Heat Sink:

a. Narrow range level in at least one SG - GREATER THAN (6)% [(7)% FOR ADVERSE CONTAINMENT] — a. Return to Step 18.

20 Check RCS Temperatures: — Return to Step 18.

- o Core exit TCs - DECREASING
- o RCS hot leg temperatures - DECREASING

NOTE

- o After stopping any SI pump, RCS pressure should be allowed to stabilize before stopping another SI pump.
- o The charging/SI pumps and high-head SI pumps should be stopped on alternate ECCS trains when possible.

21 Check If One Charging/SI Pump Should Be Stopped:

a. Two charging/SI pumps - RUNNING

a. Go to Step 22.

b. Determine required RCS subcooling from table:

HIGH-HEAD SI PUMP STATUS	RCS SUBCOOLING (°F)	
	ONE CHARGING/SI PUMP RUNNING	NO CHARGING/SI PUMP RUNNING
NONE RUNNING	(11)°F	(11)°F
ONE RUNNING	(11)°F	(11)°F
TWO RUNNING	(11)°F	(11)°F

c. RCS subcooling based on core exit TCs - GREATER THAN REQUIRED SUBCOOLING

c. DO NOT STOP CHARGING/SI PUMP. Go to Step 24.

d. PRZR level - GREATER THAN (12)% [(13)% FOR ADVERSE CONTAINMENT]

d. DO NOT STOP CHARGING/SI PUMP. Go to Step 24.

e. Stop one charging/SI pump

22 Check If One High-Head SI Pump Should Be Stopped:

a. Any high-head SI pump - RUNNING

a. Go to Step 23.

b. Determine required RCS subcooling from table:

HIGH-HEAD PUMP STATUS	RCS SUBCOOLING (°F)			
	ONE CHARGING/SI PUMP RUNNING		NO CHARGING/SI PUMP RUNNING	
	ANY RCP RUNNING	NO RCP RUNNING	ANY RCP RUNNING	NO RCP RUNNING
ONE RUNNING	(11)°F	(11)°F	(11)°F	(11)°F
TWO RUNNING	(11)°F	(11)°F	(11)°F	(11)°F

c. RCS subcooling based on core exit TCs - GREATER THAN REQUIRED SUBCOOLING

c. DO NOT STOP SI PUMP. Go to Step 24.

d. PRZR level - GREATER THAN (12)% [(13)% FOR ADVERSE CONTAINMENT]

d. DO NOT STOP SI PUMP. Go to Step 24.

e. Stop one additional high-head SI pump

f. Return to Step 22a

23	Check If Normal Charging Should Be Established:	
	a. Check the following: o High-head SI pumps - STOPPED o Charging/SI pumps - ALL BUT ONE STOPPED	a. DO NOT ISOLATE BIT. Return to Step 21.
	b. RCS subcooling based on core exit TCs - GREATER THAN (11)°F	b. DO NOT ISOLATE BIT. Go to Step 24.
	c. PRZR level - GREATER THAN (12)% [(13)% FOR ADVERSE CONTAINMENT]	c. DO NOT ISOLATE BIT. Go to Step 24.

NOTE After closing a PRZR PORV, it may be necessary to wait for RCS pressure to increase to permit stopping SI pumps in Steps 21 and 22.

24	Check PRZR PORV Status:	
	a. PRZR PORVs - ANY OPEN	a. Go to E-1, LOSS OF REACTOR OR SECONDARY COOLANT, Step 1.
	b. Close one PRZR PORV	b. Close associated PORV block valve. IF block valve can NOT be closed, THEN go to E-1, LOSS OF REACTOR OR SECONDARY COOLANT, Step 1.
	c. Return to Step 21	

25	Establish (14) GPM Charging Flow: a. Close charging line flow control valve b. Open charging line hand control valve c. Open charging line isolation valves d. Establish (14) GPM charging flow using flow control valve	
26	Isolate BIT: a. Close inlet isolation valves b. Close outlet isolation valves	
27	Check PRZR PORVs - ALL CLOSED	Close PRZR PORVs. IF any valve can NOT be closed, THEN manually close its block valve.
28	Control Charging Flow To Maintain PRZR Level	
29	Go To ES-1.1, SI TERMINATION, Step 11	

- END -

FOOTNOTES

(1) 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.)

(2) Enter plant specific paramenter and setpoint for diagnosing loss of secondary heat sink, including allowances for normal channel accuracy. Refer to background document.

(3) Enter plant specific parameter and setpoint for diagnosing loss of secondary heat sink, including allowances for normal channel accuracy and post accident transmitter errors.

(4) Enter plant specific temperature requirement, including allowances for normal channel accuracy, for placing RHR in service.

(5) Enter plant specific temperature requirement, including allowances for normal channel accuracy and post accident transmitter errors, for placing RHR in service.

(6) Enter plant specific value showing SG level just in the narrow range, including allowances for normal channel accuracy.

(7) 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%.

(8) Enter plant specific RCS pressure 50 psi below permissive to block SI.

(9) Enter plant specific shutoff head pressure of condensate pumps.

(10) Enter plant specific value corresponding to RWST switchover alarm in plant specific units.

FOOTNOTES (Continued)

(11) Enter plant specific subcooling criteria. Refer to "SI Reduction Sequence Evaluation" section of Executive Volume.

(12) Enter plant specific value showing PRZR level just in range, including allowances for normal channel accuracy.

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

(14) Enter a charging flow rate comparable to normal charging/SI pump miniflow, i.e., 60 gpm.

Appendix F

Seabrook Plant Description

EMERGENCY RESPONSE GUIDELINE
SEABROOK PLANT DESCRIPTION

TABLE OF CONTENTS

SECTION	PAGE
1. INTRODUCTION	1
2. PLANT SYSTEMS	2-17
3. ATTACHMENT A - CRITICAL SAFETY FUNCTION INSTRUMENTATION REQUIREMENTS	
4. APPENDIX B - INSTRUMENTATION AND CONTROL REQUIREMENTS	

Control and Protection Actuation Systems

- 2.1 Reactor Trip Actuation System
- 2.2 ESF Actuation System

Instrumentation Systems

- 2.3 Nuclear Instrumentation System
- 2.4 Control Rod Instrumentation System
- 2.5 Radiation Monitoring System
- 2.6 Containment Instrumentation System
- 2.7 Critical Safety Function Monitoring System

Process Control Systems

- 2.8 Reactor Coolant System
- 2.9 Emergency Core Cooling System
- 2.10 Residual Heat Removal System
- 2.11 Chemical and Volume Control System
- 2.12 Component Cooling Water System (Primary)
- 2.13 Service Water System
- 2.14 Containment Spray System
- 2.15 Containment Atmosphere Control System
- 2.16 Main Steam System
- 2.17 Main Feedwater and Condensate System
- 2.18 Emergency Feedwater System
- 2.19 Steam Generator Blowdown System
- 2.20 Primary Sampling System and Post Accident Sampling
- 2.21 Spent Fuel Storage and Cooling System
- 2.22 Control Rod Drive Mechanism Cooling System
- 2.23 Control Rod Control System
- 2.24 Turbine Control System

Support Systems

- 2.25 Electrical Power System
- 2.26 Pneumatic Power System

2.1 Reactor Trip Actuation System

The reactor trip actuation system monitors specified process parameters and equipment status and actuates reactor trip if conditions exceed specified limits. The reactor trip actuation system includes automatic actuations that occur concurrent with actuation of reactor trip.

The process parameters and equipment status monitored by the reactor trip actuation system are plant-specific. A reactor trip is automatically actuated if any condition exceeds its specified limit. Concurrent with actuation of reactor trip, a P-4 signal is generated and provides the following:

- o Turbine trip
- o Input signal for feedwater isolation
- o Input signal for SI block logic (concurrent with manual SI reset signal)
- o Steam dump control logic

2.2 Engineered Safeguards Features Actuation System

The engineered safety features (ESF) actuation system monitors specified process parameters and actuates engineered safety features operation if conditions exceed specified limits. The ESF Actuation System consists of the following automatic actuation signals:

Safety Injection Signal

The safety injection (SI) signal is the primary ESF actuation signal. It is automatically actuated on any of the following:

- o Low pressurizer pressure
- o Low steamline pressure
- o High-1 containment pressure
- o Manual operator actuation

The following is automatically actuated by an SI signal:

- o Reactor Trip
- o Feedwater isolation (Closure of feedwater isolation, flow control and bypass valves.)
- o Emergency feedwater pump start (Start motor - driven and turbine-driven EFW Pumps)
- o Diesel - generator start
- o Safety injection system start
- o Containment isolation Phase A ("T" signal)
- o Containment ventilation isolation

Containment Spray Signal

The containment spray signal automatically actuates on any of the following:

- o High - 3 containment pressure
- o Manual operator actuation

The following plant equipment automatically actuates on a containment spray signal.

- o Containment spray pumps start
- o Containment isolation phase B valves close
- o Containment spray system valves reposition for spray and chemical addition

The containment spray signal actuation logic includes the following reset capability.

- o Manual reset for containment spray system actuation signal
- o Manual reset capability for containment isolation phase B valve closure signal

Emergency Feedwater System

The Emergency Feedwater System (EFW) consists of two pumps, one motor-driven and one turbine-driven. Either pump is full capacity and feeds all four steam generators. The system design provides for automatic isolation of a single line asociated with a faulted steam generator upon sensing high flow.

Both EFW pumps automatically start on any of the following conditions:

- o SI signal
- o Loss of power signal
- o Low-Low level in any steam generator
- o Manual operator actuation

Any EFW start signal also actuates isolation of steam generator blowdown.

The EFW pumps are not used for plant startup and normal plant shutdowns. To accomplish this task, a single Startup Feedwater Pump is provided. This pump receives an automatic start signal an loss of both main feedwater pumps only. The Startup Feedwater Pump is also used as a backup to the EFW system. It can provide feedwater to all steam generators through the main feedwater lines or through the EFW lines.

Containment Isolation Phase A Signal ("T" Signal)

The containment isolation Phase A signal automatically isolates non-essential containment penetrations to prevent or minimize the release of the radioactive material outside containment This signal automatically actuates on any of the following:

- o SI signal
- o Manual operator actuation of Phase A signal

The containment isolation Phase A signal closes most valves in plant systems that penetrate containment. The actuation logic includes separate reset capability for each actuation signal.

Containment Isolation Phase B Signal ("P" Signal)

The containment isolation phase B signal automatically isolates remaining containment penetrations to prevent the release of radioactive material outside containment. The signal automatically actuates on any of the following:

- o High - 3 containment pressure
- o Manual operator actuation of containment spray signal

The containment isolation Phase B signal automatically closes the containment isolation valves in the component cooling water lines to the reactor coolant pumps. The actuation logic includes separate reset capability for each actuation signal.

Main Steamline Isolation

Main steamline isolation is actuated on any of the following:

- o HI - 2 containment pressure (same setpoint as HI-1)
- o Low steamline pressure in any steamline when pressurizer pressure above P-11
- o High steam pressure rate in any main steamline when pressurizer pressure below P-11
- o Manual operator actuation

The main steamline isolation logic includes the following reset capability:

- o Manual reset/block for low steamline pressure actuation signal (concurrent with a P-11 low pressurizer pressure permissive)
- o Automatic block for high steam pressure rate actuation signal when pressurizer pressure above P-11 setpoint
- o Manual reset for main steamline isolation signal

The following plant equipment automatically positions due to a main steamline isolation signal:

- o Main steamline isolation valves close
- o Main steamline isolation bypass valves close

Containment Ventilation Isolation

The containment ventilation isolation signal automatically isolates containment ventilation penetrations to prevent the release of radioactive material outside contaiment. This signal automatically actuates on any of the following:

o SI signal

o High containment radiation

The containment ventilation isolation signal closes valves in the ventilation system. The actuation logic includes separate reset capability for each actuation signal.

Main Feedwater Isolation Signal

The main feedwater isolation signal automatically isolates the main feedwater lines to prevent excessive cooling of the RCS and filling of the steam generators. The signal automatically actuates on any of the following input signals.

o SI signal

o High - High level (P-14 signal) in any steam generator

o Reactor trip signal (P-4 coincidnet wil a low reactor coolant system T avg signal

The feedwater isolation signal closes the following valves:

o Main feedwater isolation valves

o Main feedwater flow control valves

o Main feedwater bypass valves

The main Feedwater isolation signal includes the following reset capability.

o Separate reset capability for the reactor trip signal coincident with RCS low T avg signal

o Shared reset capability for the SI actuation signal. This capability requires reset of reactor trip signal (P-4) and reset of SI signal. Also SG high-high level condition cannot exist.

2.3 Nuclear Instrumentation System

The nuclear instrumentation system (NIS) monitors the reactivity state of the reactor core. It consists of instrumentation that monitors neutron flux and startup rate. Neutron flux is monitored over the source, intermediate and power range. Startup rate is monitored over the source and intermediate range. The NIS includes a neutron flux recorder that can be switched to record different ranges. The source range neutron flux detectors automatically re-energize when flux decreases below the source range high flux trip (P-6) setpoint following a reactor trip, permitting the neutron flux recorder to be manually transferred to the source range scale. For post-accident harsh environment conditions, a qualified NIS consisting of wide-range neutron flux, flux rate and shutdown margin monitor is used.

2.4 Control Rod Instrumentation System

The control rod instrumentation system monitors the position of the reactor core control rods. It consists of control rod position and bottom light instrumentation.

2.5 Radiation Monitoring System

The radiation instrumentation system monitors the radiation levels in specified process systems and specified areas internal and external to the containment.

2.6 Containment Instrumentation System

The containment instrumentation system monitors the environmental condition of the contaiment. It consists of containment pressure and temperature instrumentation, containment building level instrumentation and position indication for containment ventilation valves.

2.7 Critical Safety Function Monitoring System

The critical safety function monitoring system consists of instrumentation used to monitor plant variables associated with the following critical safety function:

- o Subcriticality
- o Core cooling
- o Heat sink
- o Reactor vessel integrity
- o Contaiment
- o RCS inventory

This instrumentation system consists of post accident qualified instruments which display those variables necessary to monitor the critical safety functions. All variables are displayed on hard-wired indicators and/or recorders in the main control room. These same variables also input to the plant computer system which provides automatic monitoring of the status of each critical safety function.

2.8 Reactor Coolant System

The reactor system (RCS) transfers heat from the reactor core to the main steam system or residual heat removal system and provides a barrier against the release of reactor coolant or radioactive material to the containment enviroment.

The RCS consists of four identical heat transfer loops (connected in parallel to the reactor vessel) a pressurizer and a pressurizer relief tank. Flow from the RCS hot leg and cold leg enter the bypass loop and returns via a common bypass header to the RCP suction. Each bypass loop contains a hot leg and cold leg manifold that includes RTD temperature sensors used in plant control and protection. RTDs are also installed in wells in each loop's hot and cold legs to provide accurate temperature measurement and indication while in natural circulation or RHR modes.

The pressurizer is connected to the hot leg of one loop via the pressurizer surge line and the cold legs of two loops via the pressurizer spray lines. The pressurizer has two power operated relief valves (with associated block valves), three code safety valves and heaters. RCS pressure is controlled by use of the pressurizer where water and steam are maintained in equilibrium through use of the heaters, water spray and steam release.

The pressurizer PORVs and safety valves discharge to the pressurizer relief and (PRT) where steam discharge is condensed and cooled by mixing with water. The PRT system has an external circulating pump and heat exchanger to assist in post-discharge cooldown.

The pressurizer PORVs are electrically operated and do not require pneumatic power sources. Each PORV is powered from independent, train associated power supplies.

A low temperature-overpressurization protection (LTOP) system is provided that operates the PORV's. Below an RCS temperature of 305°F, the PORV pressure setpoint is automatically decreased as a function of temperature.

Reactor coolant pumps (RCPs) are powered from 13.8 KV buses with two RCPs on each bus. An RCP that drives pressurizer spray is assigned to each bus.

Non-condensible gas is vented from the RCS via a single train vessel head vent or the train associated pressurizer PORVs.

NOTES: Pressurizer PORVs and SG PORVs do not use air for operation.

Letdown and charging line valves are pneumatic valves but fail in safest position, i.e.;

- o Letdown line isolation valves fail closed
- o Charging line isolation valves fail open
- o Excess letdown isolation valves fail closed
- o Auxiliary pressurizer spray valve fails closed
- o RCP #1 seal leakoff isolation valves fail open

Pneumatic power would be required to re-establish operability.

Air operated valves (AOVs) inside containment are powered pneumatically from the containment air compressors. AOVs outside containment are powered from the plant air compressors.

2.9 Emergency Core Cooling System (ECCS)

The ECCS system provides coolant to the reactor coolant system and introduces negative reactivity for events that require engineered safety features operation.

The ECCS is designed to operate in three modes depending on plant transient:

- Cold leg injection mode (short term core cooling mode
  The Cold leg injection mode is defined as that period during which borated water is delivered from the refueling water storage tank (RWST) and accumulators to the RCS cold legs.

- Hot leg recirculation mode is defined as that period during which borated water is recirculated from the containment sump to the RCS cold legs.
  The Hot leg recirculation mode is that period during which borated water is recirculated from the containment sump to both the RCS hot legs and RCS cold legs.

The ECCS consists of four major subsystems:

- Centrifugal Charging Pump Subsystem
  The charging subsystem consists of two centrifugal charging pumps and a boron injection tank. These pumps are part of the chemical and volume control system and provide charging and RCP seal injection flow during normal operation. Upon receipt of an SI signal, these pumps are automatically isolated from the normal charging function and aligned in the cold leg injection mode. In this mode the charging pumps take suction from the RWST and discharge through the BIT and RCP seals to all RCS cold legs. During recirculation modes, the charging pump's discharge remains through the BIT path to the RCS cold legs.

The discharge shutoff pressure of the charging pumps is greater than RCS normal operating pressure.

The BIT contains 12 weight percent boric acid solution. During normal operation the BIT contents are isolated by parallel inlet and outlets sets of motor operated valves. The BIT is bypassed in the recirculaton mode to reduce radiation levels in the PAB.

- SI Subsystem

  The SI subsystem consists of two centrifugal SI pumps. These pumps are part of the SI system and are aligned in the SI cold leg injection mode alignment. Upon receipt of an SI signal, the SI pumps automatically start in the SI cold leg injection mode. In this mode the SI pumps take suction from the RWST and discharge to all RCS cold legs (through the accumulator discharge lines). During recirculation modes the SI pumps are aligned to take suction from the RHR pump discharge and discharge to all RCS cold legs or hot legs depending on recirculation mode.

  The discharge shutoff pressure of the SI pump is approximately 1600 PSIG.

- RHR Subsystem

  The RHR subsystem consists of two centrifugal pumps and two heat exchangers. These pumps and heat exchangers are part of the RHR system and provide normal plant shutdown heat removal. During plant operation, these pumps are aligned in the cold leg injection mode alignment. Upon receipt of an SI signal, the RHR pumps automatically start in the SI cold leg injection mode alignment. In this mode the RHR pumps take suction from RWST and discharge to all RCS cold legs (through the accumulator discharge lines). During recirculation modes, the RHR pumps are aligned to take suction from the containment recirculation sump and to discharge to the suction of the centrifugal charging pumps and SI pumps as well as to all RCS cold legs or two RCS hot legs depending on recirculation mode.

  The discharge shutoff pressure of the RHR pumps is approximately 200 PSIG.

The RHR heat exchangers are supplied with primary component cooling water (PCCW) during recirculation modes.

- Accumulator Subsystem

  The accumulator subsystem consists of four accumulator tanks, each connecting to one RCS cold leg via an accumulator injection line. Each tank contains borated water and is pressurized to a nominal 650 PSIG with a nitrogen cover gas. A single isolation valve is provided in each accumulator injection line and series vent valves are provided to vent the accumulators if necessary. During plant operation the injection isolation valves are open with power removed from the valve operators. The accumulators are available to deliver their contents to the RCS cold legs during the injection mode of any emergency transient that is accompanied by RCS depressurization below the accumulator pressure.

2.10 Residual Heat Removal System

The residual heat removal (RHR) system removes residual heat from the reactor coolant system during plant shutdown operations at low reactor coolant system pressures.

The RHR system consists of two RHR pumps and RHR heat exchangers. The RHR system provides normal shutdown heat removal when RCS pressure and temperature are reduced to approximately 400 psig and 350°F. During normal shutdown heat removal operations, the RHR pump suction is aligned to the RCS hot legs and the RHR pump discharge is aligned to the RCS cold legs.

Portions of the RHR system also function as part of the ECCS. This shared function is described in Section 2.8.

2.11 Chemical and Volume Control System

The chemical and volume control system (CVCS) provides coolant to the reactor coolant system and provides core reactivity control for normal operation and any event that does not require engineered safety features operation.

The CVCS consists of charging and letdown capability for control of RCS inventory. Letdown capability is provided by two letdown paths (letdown line and excess letdown line). Charging capability is provided by three charging pumps (two centrifugal pumps that also function as ECCS pumps charging line and RCP seal injection lines. The RCP seal injection lines supply each RCP and provide RCP seal cooling.

Suction flow to the charging pumps is provided by the chemical volume control tank (CVCT) or by the refueling water storage tank (RWST). The charging pumps suction is normally aligned to the CVCT, but is automatically transferred to the RWST on the following:

- o SI signal
- o CVCT low-low level signal

The CVCS includes two redundant boric acid tanks and associated boric acid pumps. Each boric acid tank contains a four weight percent boric acid solution. This solution is fed to the suction of the charging pumps in metered quantities for core reactivity control.

The CVCS also includes a boron thermal regeneration system (BTRS) which is used to control core reactivity.

Portions of the CVCS also function as part of the SI system. This shared function is described in Sectin 2.8.

2.12 Primary Component Cooling Water System

The primary component cooling water (PCCW) system provides heat removal from primary system processes and equipment.

The system consists of two train associated loops, each of which serves one train of ECCS and containment spray related components and other plant related system processes and equipment used during normal operation. Each PCCW loop cools two reactor coolant pump motors.

In addition to the two train associated loops, the system includes a dedicated train associated thermal barrier cooling system that cools all four reactor coolant pump thermal barriers. This substem consists of two heat exchangers. Each heat exchanger is cooled by one PCCW loop. Either thermal barrier cooling pump and either heat exchanger provides cooling for all four reactor coolant pump thermal barriers.

2.13 Service Water System

The service water (SW) system provides heat removal from the PCCW system and the emergency diesel-generator jacket cooling water system. Like the PCCW system, the SW system consists of two train associated loops, each serving its respective train associated heat loads. Either or both SW loops can cool secondary plant heat loads but are isolated on SI actuation.

The ultimate heat sink can be either the Atlantic Ocean or the atmosphere. When the Atlantic Ocean is the ultimate heat sink, each SW loop consists of two redundant SW pumps per train, piping loops, heat exchangers and intake and discharge points for seawater. When the atmosphere is the ultimate heat sink, each SW loop consists of a mechanical draft cooling tower, cooling tower pump, heat exchangers and cooling tower fans.

The normal heat sink is the Atlantic Ocean. Should the flow of seawater become restricted, the cooling tower is automatically placed in service.

2.14 Containment Spray System

The containment spray system provides containment pressure suppression and airborne fission product removal for events that require engineered safety features actuation. Each train associated containment spray system consists of a containment spray pump, heat exchanger, spray header and a shared spray additive tank. The spray additive tank contains a twenty percent sodium hydorxide solution. This tank gravity feeds the RWST upon receipt of a Hi-3 ESF actuation signal. The RWST provides suction to all ECCS and containment spray pumps.

2.15 Containment Atmosphere Control System

The containment atmosphere control system provides containment heat removal and combustible gas mixture control. It consists of the containment fan coolers, containment electric hydrogen recombiners and containment ventilation equipment that provide for mixing of the containment atmosphere. During normal oerating conditions five of the six containment fan coolers operate to keep internal containment temperature within environmental limits.

In accident conditions requiring containment spray, the function of heat removal is accomplished by spray system operation. Combustible gas is mixed by two train associated air mixing fans. Recombination of H_2 AND O_2 is accomplished by two train associated electric recombiners inside the containment. Containment fan coolers are not used for any accident mitigation function.

2.16 Main Steam System

The main steam system provides controlled heat removal from the reactor coolant system via the steam genrators. It consists of separate main steamlines from each steam generator that join to a common steam header to the turbine-generator/condenser. The steam generators can be isolated from the main steam header by main steamline isolation and bypass valves located in the individual main steamlines.

Main steam release capabiltiy is provided via the condenser steam dump system and the atmospheric steam release system. The condenser steam dump system uses the main steam header and steam dump valves to the condenser. The atmospheric steam dump system uses power operated relief valves upstream of the main steam isolation and bypass valves to release steam to atmosphere. The atmospheric steam dump valves are hydraulically operated and do not require plant air for operation.

Each main steam line contains ASME code safety valves for over-pressure protection.

Steam supply lines from the main steam lines to the turbine-driven EFW pump are provided from two steam generators. The steam supply lines include isolation valves for initiation and isolation of steam supply to the turbine-driven EFW pump.

2.17 Main Feedwater and Condensate System

The main feedwater and condensate system provides coolant to the secondary side of the steam generators during plant power operation. It consists of separate main feedwater lines to each steam generator that originate from a commmon main feedwater header. The steam generators can be isolated from the main feedwater header by feedwater flow control valves, bypass valves and isolation valves located in the individual main feedwater lines.

The condensate portion of the system consists of three fifty percent condensate pumps and two fifty percent heater drain pumps.

2.18 Emergency Feedwater System

The emergency feedwater (EFW) system provides secondary coolant to the steam genrators for events that require engineered safety features operation or when the startup feedwater pump is inoperable.

The EFW system consists of two centrifugal pumps, one trubine-driven and one motor driven. Either or both pumps supply secondary makeup to all four steam genrators. Flow control valves allow the operator to throttle faulted steam generator, will terminate flow to meet the generic flow requirements.

Connections are made into the EFW discharge headers to supply secondary coolant makeup from the startup feed pump.

The EFW pumps are always aligned to the condensate storage tank (CST). Makeup to the CST is from the demineralized water storage tank (DWST) or the water treatment facility.

The CST is a 400,000 gallon tank with 200,000 gallons dedicated for EFW. The DWST is sized at 200,000 gallons.

2.19 Steam Generator Blowdown System

The steam generator blowdown system provides letdown from the secondary side of the steam generators. It consists of separate blowdown lines from each steam generator that terminate in the steam generator blowdown tank. The blowdown water is either cleaned up and recycled, discharged or processed in the blowdown evaporators should a steam generator tube leak develope.

Steam generator blowdown is automatically terminated when the EFW pumps are called upon to automatically start.

2.20 Primary Sampling System

The sampling system provides means for sampling process systems. It consists of the sampling system equipment that can be used to sample the RCS and the containment recirculation sump.

2.21 Spent Fuel Storage and Cooling System

The spent fuel storage and pool cooling system controls fuel storage positions to ensure a subcritical geometric configuration and provides heat removal to maintain stored fuel within specified temperature limits. It includes the level instrumentation for the spent fuel pool.

2.22 Control Rod Drive Mechanism Cooling System

The control rod drive mechanism (CRDM) cooling system provides heat removal from the control rod drive mechanisms. It consists of the ventilation fans used to circulate air around the control rod drive mechanisms.

2.23 Control Rod Control System

The control rod control system controls the positions of the control rods in the reactor core. It includes those controls used to manually insert control rods.

2.24 Turbine Control System

The turbine control system controls the turbine generator. It includes those controls used to manually trip the turbine generator supplied by General Electric.

2.25 Electrical Power System

The electrical power system provides AC and DC electircal power to equipment that requires electrical power to accomplish their functins. It consists of two independent off-site circuits, either of which can supply all on-site power requirements. The on-site emergency AC power supply is a two train system powered by separate diesel generators. The DC power system is a two train system. Each vital DC train consists of redundant batteries, battery charges and DC buses. Vital AC instrument power can be supplied by either AC from two sources or from the DC power supply inverters.

The emergency diesel generators automatically start on the following:

o SI signal (runs but does not connect to the bus unless loss of power occurs.)
o Loss of Power (as sensed by undervoltage relays.)

The diesel generators automatically energize their respective ac emergency buses if both off-site power circuits are unavailable. The following major loads are sequenced on the associated buses in accordance with plant conditions as follows:

a. Loss of Power (LOP) without SI

- o Charging pumps
- o PCCW pumps
- o Service water pumps or cooling tower pumps/fans
- o RHR pumps if running prior to LOP
- o Inverters and battery chargers
- o Emergency lighting
- o HVAC systems
- o Air compressors (plant and containment)
- o Boric acid transfer pumps
- o BIT recirculation pumps and heaters
- o MOVs
- o Emergency feedwater pump (motor driven)

b. Loss of Power (LOP) with S

- o Charging pumps
- o SI pumps
- o RHR pumps
- o Containment spray pumps
- o PCCW pumps
- o Service water pumps or cooling tower pumps/fans
- o Inverters and battery chargers
- o Emergency lighting
- o HVAC systems
- o Hydrogen mixing fans
- o Power receptacle for H_2 recombiners
- o Pressurizer heaters
- o Emergency feedwater pump (motor driven)
- o MOVs

2.26 Pneumatic Power System

The pneumatic power system supplies pneumatic power to equipment that requires pneumatic power to accomplish its functions. Equipment in this category includes:

- o Condenser steam dump valves
- o Repostioning certain valves from their safety related position to their normal non-safety related position

ATTACHMENT A

PROCEDURAL INSTRUMENTATION REQUIREMENTS
FOR
SEABROOK CRITICAL SAFETY FUNCTION MONITORING
AND STATUS TREE LOGIC INPUTS

BASED IN PART ON THE
WESTINGHOUSE OWNER'S GROUP
GENERIC EMERGENCY RESPONSE GUIDELINES

Rev. 00
June 1983

THIS INFORMATION MEETS THE FOLLOWING CRITERIA FOR SEABROOK CRITICAL SAFETY FUNCTION MONITORING AND STATUS TREE LOGIC INPUTS:

- <u>All</u> variables are hard wired to main control room indicators, recorders or displays for manual monitoring of critical safety functins. This provides on-the-spot monitoring ability should the Main Plant Computer System or CRTs be off line.

- <u>All</u> variables are hard wired to main control room indicators, recorders or displays to direct operator actions.

- <u>All</u> variables provide input directly to the Critical Safety Function Status Tree Logics which direct Emergency Action Levels.

- These variables are classified as Type A, Category 1 by Regulatory Guide 1.97.

VARIABLE	UNITS	DISPLAY
SUBCRITICALITY:		
Intermediate Range Flux Level	%	Indicator
Intermediate Range Flux Rate	DPM	Indicator
Shutdown Margin Monitor	CPS	Indicator - of source range counts with shutdown margin alarm
CORE COOLING:		
Core Exit Temperature	°F	Indicator - averaged T/C value of center and hot channel in each quadrant
RCS Subcooling	°F	Indicator - computed from RCS wide pressure and core exit temperature above
Reactor Vessel Level	%	Indicator
Reactor Vessel W Level	%	Indicator
HEAT SINK:		
Steam Generator Level - NR	%	Indicator
Steam Generator Level - WR	%	Indicator and recorder
Steam Generator	PSIG	Indicator and recorder
Emergency Feedwater Flow	GPM	Indicator
INTEGRITY:		
RCS Cold Leg Temperature - WR	°F	Indicator and recorder
RCS Pressure	PSIG	Indicators - 0-3000 PSIG 0-700 PSIG
CONTAINMENT:		
Containment Pressure	PSIG	Indicator and recorder
Containment Building Level	GALLONS	Indicator and recorder
Containment Post Accident Radiation	R/HR	Indicator
INVENTORY:		
Pressurizer Level	%	Indicator

ATTACHMENT B

PROCEDURAL INSTRUMENTATION
AND
CONTROL REQUIREMENTS
FOR
SEABROOK EMERGENCY RECOVERY PROCEDURES

BASED IN PART ON THE
WESTINGHOUSE OWNER'S GROUP
GENERIC EMERGENCY RESPONSE GUIDELINES

Rev. 00
June 1983

PROCEDURAL INSTRUMENTATION
AND
CONTROL REQUIREMENTS
FOR THE
SEABROOK EMERGENCY RECOVERY PROCEDURES

I - Indication
C - Control

ITEM	I	C
Reactor Trip Actuation System		
Reactor Trip and Bypass Breakers	X	X
Reactor Trip Signal	X	X
Turbine Trip Signal	X	X
ESF Actuation System		
SI Signal	X	X
SI Signal Block	X	X
Si Signal Reset	X	X
Containment Isolation Phase A Signal Reset	X	X
Feedwater Isolation Signal Reset	X	X
Containment Spray Signal	X	X
Containment Spray Signal Reset	X	X
Main Steamline Isolation Signal	X	X
Tower Actuation Signal	X	X
Nuclear Instrumentation System		
Intermediate Range Neutron Flux	X	-
Intermediate Range Flux Rate	X	-
Source Range Counts	X	-
Source Range Shutdown Margin Alarm	X	-
Containment Instrumentation System		
Containment Pressure	X	-
Containment Temperature (averaged value)	X	-
Containment Building Level	X	-
Phase A Containment Isolation Valves	X	X
Phase B Containment Isolation Valves	X	X
Containment Ventilation Isolation Valves (CAP, COP)	X	X
Containment Hydrogen Concentration	X	X
Containment Building Level Recorder	X	-

ITEM	I	C
Reactor Coolant System		
RCS Wide Range Pressure	X	-
Pressurizer Pressure	X	-
RCS Hot Leg Wide Range Temperature	X	-
RCS Cold Leg Wide Range Temperature	X	-
RCS Average Temperature	X	-
Core Exit Temperature (averaged value)	X	-
Pressurizer Water Temperature	X	-
Pressurizer Level	X	-
RCS Subcooling (^{o}F)	X	-
Reactor Vessel Level Instrumentation System (RVLIS) o Vessel Level o Vessel W Pressure	X	-
Reactor Coolant Pumps	X	X
Pressurizer PORVs	X	X
Pressurizer PORV Block Valves	X	X
Pressurizer spray Valves	X	X
Reactor Vessel Vent Valves	X	X
Pressurizer Heaters	X	X
Pressurizer Normal Spray	X	X
Pressurizer Pressure Control	X	X
PRT Pressure	X	-
PRT Level	X	-
PRT Temperature	X	-
Relief Valve Tail Pipe Temperatures o POressurizer PORVs o Letdown Line	X	-
Pressurizer Safety Valve Tail Pipe Temperature	X	-
RCS Vent Paths o RV Head Vent Valves o Pressurizer PORVs	X	X
RCS Dilution Paths o CVCS Makeup System o BTRS	X	X
Emergency Core Cooling System		
Accumulator Pressure	X	-
SI Pump Discharge Pressure	X	-
Boron Injection Tank (BIT) Temperature	X	-
Refueling Water Storage TanK (RWST) Level	X	-
RWST Level Recording	X	-
CCP Flow to BIT	X	-
SI Flow	X	-
Throttle SI Flow	X	X

ITEM	I	C
Emergency Core Cooling System (Cont)		
Throttle CCP Flow via BIT	X	X
CCPs	X	X
RHR Pumps	X	X
SI Pumps	X	X
All ECCs Related Remotely Operated Valves (SI, RHR, CS)	X	X
Residual Heat Removal System		
RHR Flow	X	-
RHR Temperature Control	X	X
RHR Suction Valves from rCS	X	X
Chemical and Volume Control System		
VCT Level	X	-
Boric Acid Tank Temperature	X	-
Charging Flow	X	-
RCP Seal Injection Flow	X	-
RCP Seal Leakoff Flow	X	-
CVCT Outlet Isolation Valves	X	-
Charging Line Isolation Valves	X	X
Charging Line Flow Control Valve (FCV-121)	X	X
Charging Line Hand Control Valve (HCV-182)	X	X
Pressurizer Auxiliary Spray Valve	X	X
CCP Miniflow Valves	X	X
CCP Suction Valves from RWST	X	X
Letdown Pressure Control Valve (PCV-131)	X	X
RCP #1 Seal Leakoff Valvs	X	X
Letdown Flow	X	-
Letdown Isolation Valves	X	X
Letdown Flow Control Valves (NCV-189, HCV-190)	X	X
Excess Letdown Isolation Valves	X	X
Excess Letdown Flow Control Valve (HCV-123)	X	X
RCP Seal Injection Outside Containment Isolation Valves	X	X
Emergency Borate Valve (CS-V426)	X	-
Boric Acid Pumps	X	X
CVCT Makeup Control System	X	X
o RMW Pumps		
o Makeup Flow Control Valves and Controllers FCV-110A, 110B, 111A and 111B		
BTRS Master Control Switch		

ITEM	I	C
Component Cooling Water System		
PCCW Pumps	X	X
PCCW Temperature Control Valves	X	X
Thermal Barrier Cooling Pumps	X	X
RCP Thermal Barrier Outlet Valves	X	X
PCCW Valves to Components and Systems	X	X
o Letdown Heat Exchanger		
o RHR Heat Exchangers		
o Containment Spray Heat Exchangers		
o Excess Letdown Heat Exchangers		
o Spent Fuel Pool Heat Exchangers		
o PCCW Radiation Monitors		
PCCW Head Tank Level	X	X
PCCW Flow to RCP Motors	X	-
Ultimate Heat Sink		
Service Water Pumps	X	X
Cooling Tower Pumps	X	X
All Service Water Remote Operated Valves	X	X
Cooling Tower Fans	X	X
Cooling Tower Level	X	-
Service Water Pump Bay Level	X	-
Flow Through Diesel Generator Jacket Water Heat Exchanger	X	-
Containment building Spray System		
Containment Spray Pumps	X	X
Containment Spray Valves to Spray Rings	X	X
Containment Spray Pump Discharge Pressure	X	-
Combustible Gas Control System		
Containment Ventilation Isolation Valves (COP, CAP)	X	X
Hydrogen Monitoring Systems	X	X
Hydrogen Concentration	X	-
Hydrogen Recombiner Systems	X	X
Hydrogen Mixing Fans	X	X
Containment Enclosure System		
Containment Enclosure Emergency Exhaust Filter Systems	X	X
Containment Enclosure Pressure	X	-

ITEM	I	C
Main Steam System		
Steam Generator Pressure	X	-
Steam Generator Narrow Range Level	X	-
Steam Generator Wide Range Level	X	-
Steam Generator ASDVs	X	X
Condenser Steam Dump Valves	X	X
Main Steamline Isolation Valves	X	X
Main Steamline Isolation Bypass Valves	X	X
Steam Supply Valves to Turbine Driven EFW Pump	X	X
Turbine Stop Valves	X	-
Condenser Steam Dump Controls (T_{avg} & Pressure Modes)	X	X
Main Steamline Radiation	X	-
Main Feedwater and Condensate System		
Feedwater Flow Control Valves	X	X
Feedwater Flow Control Bypass Valves	X	X
Feedwater Isolation Valves	X	X
Main Feedwater Flow	X	X
Condensate Pump Discharge Header Pressure	X	-
Startup Feed pump	X	X
Startup Feed Pump Discharge Valves (FW-V156 & V163)	X	X
Emergency Feedwater System		
EFW Flow to Each SG	X	-
Condensate Storage Tank Level	X	-
Motor Driven EFW Pump	X	X
Steam Driven EFW Pump	X	X
EFW Flow Control Valves	X	X
Diesel Driven Fire Pumps (Emergency Makeup to CST)	X	X
Steam Generator Blowdown System		
Steam Generator Blowdown Isolation Valves	X	X
Electric Power System		
Emergency Bus Voltage (E-5 & E-6)	X	-
UAT Breakers	X	X
RAT Breakers	X	X
Diesel Generators	X	X
DC Bus Voltage	X	-
DC Bus Current	X	-
Grid Voltagr	X	-

ITEM	I	C
Air Systems		
Instrument Air Header Pressure	X	-
Containment Air Header Pressure	X	-
Service Air Compressors	X	X
Containment Air Compressors	X	X
Control Rod Instrumentation System		
Control Rod Position	X	-
Control Rod Bottom Lights	X	-
Radiation instrumentation System		
Process Monitors		
o SG Blowdown	X	-
o Steamlines (upstream or ASDVs)		
o Condenser Effluent		
o PCCW		
o Plant Vent Radiogas (low and high range)		
o Plant Vent Air Particulate (low and high range)		
o PAB Air Particulate		
o PAB Radiogas		
o Containment Enclosure Air Exhaust		
o Containment Enclosure Emergency Exhaust		
o PAB Misc. Ventilation		
o Air Intakes		
Area Monitors o RHR Vault 1	X	-
o RHR Vault 2		
o Charging Pump CS-P-2A		
o Charging pump CS-P-2B		
o Charging Pump CS-P-128		
o CVCT Area		
o Sampling Room		
o PAB Lower level		
Turbine Control System		
Turbine Runback	X	-

ITEM	I	C
Control Rod Drive Mechanism Cooling System		
Control Rod Drive Mechanism Fans	X	X
Sampling System		
RCS Sampling	X	X
Containment Recirculation Sump Sampling	-	X
Spent Fuel Storage and Cooling System		
Spent Fuel Pool Level	X	-

Appendix G
Example Lesson Plan

CLASSROOM DAY 1

LESSON OUTLINE	NOTES AND REFERENCES
4. Rules of Use for ERGs	
4.1 ERG consists of o ORGs o CSF Status Trees o FRGs	Generic Users Guide Format
4.2 ORGs and FRGs o identical formats o same rules of usage apply to both	1 TP 4.2
4.3 CSF Status Trees o branching pattern o series of questions o each question depends on answer to previous question	1 TP 4.3
4.4 How to use ORGs/FRGs	
4.4.1 Cover sheet (Entry point of guideline) o purpose of guideline o symptoms or entry conditions	1 TP 4.4.1
4.4.2 Two-column Format o left hand column for operator action o right hand column for contingency instructions	1 TP 4.4.2
4.4.3 Left Hand Column o written for operator to proceed down left column unless response not obtained (RNO) o contains expected conditions, actions, and checks o accomplishes guideline purpose o move down column with satisfactory results	

LESSON OUTLINE		NOTES AND REFERENCES
4.4.4	Right Hand Column (Response Not Obtained)	
	o contains contingency instructions	
	o high level contingency good for any series of left hand column subtasks	1 TP 4.4.4a
	o first contingency does not have high-lighted logic terms (if-then construction of two column format)	1 TP 4.4.4b
	o subsequent or indirect contingencies always expressed in logical terms	
	o after contingency action completed return to next left hand column step or substep	
	o if contingency action cannot be successfully done and further instruction not provided-return to next left hand column step or substep	
4.4.5	Highlighted High Level Step	1 TP 4.4.5a
	o describes task to be performed	
	o expected result in capital letters and separated by dash	
	o no expected result for simple control manipulations or actions	1 TP 4.4.5b
4.4.6	Subtasks	
	o provides multiple actions under high level step	
	o expected response in CAPITAL LETTERS	
	o letters or numbers designate sequence of performance when important	1 TP 4.4.6a
	o bullets (o) designate subtasks when sequence not important	1 TP 4.4.6b
4.4.7	Purpose of Notes	
	o administrative/advisory information to support operator action	

4.4.8 Purpose of Cautions

- o inform of potential hazards to personnel or equipment
- o advise on actions/transitions necessary due to plant condition changes

4.4.9 How to Use Notes/Cautions

- o introduced by bold lettered NOTE or CAUTION
- o text extends across page
- o multiple items identified by bullet (o)
- o precede applicable step
- o applicable to entire guideline if precedes first operator step of guideline
- o NOTES/CAUTIONS preceding transition steps to other guidelines are applicable

4.4.10 Immediate Actions

- o can be performed without using written guideline
- o note advises of the immediate steps

4.4.11 Task Completion Requirements

- o need not be completed before proceeding unless specified
- o sufficient to begin task and know progression is satisfactory
- o allows efficient procedure use with lengthy steps

4.4.12 Transitions

- o different steps in the same guideline or
- o other guidelines
- o NOTES or CAUTIONS preceding step transitioned to are applicable

LESSON OUTLINE		NOTES AND REFERENCES
4.4.13	Guideline Ends	
	o transition to another guideline or	
	o transition to a normal plant procedure	
4.4.14	Foldout Page	
	o actions/transitions applicable to all steps in guideline series	
	o note in each guideline reminds operator to open foldout page	
4.4.15	Example of Guideline Usage (overall review)	1 TP 4.4.15 a-c
4.5	CSF Status Trees	
4.5.1	Trees	
	o six differant trees	1 TP 4.5.1 a
	o tree branches color coded or line patterned	1 TP 4.5 1 b&c
	o mechanism for safety status evaluation of plant	
	o each tree evaluates a separate safety aspect	1 TP 4.5.1 d
	o at any given time, plant status represented by single path through each tree	
	o each path unique and uniquely labeled at end point	
	o if safety status is normal then no transition specified	
	o plant is safe when safety functions are shown to be satisfied	
4.5.2	Use of CSF Status Trees	1 TP 4.5.2
	o enter at left-hand side arrow	
	o tree complete when coded terminus reached	

- o user should log color and instruction and continue to next tree in sequence
- o RED or ORANGE FRG entered, perform to completion unless pre-empted by a higher priority condition

4.5.3 Red terminus encountered

- o immediately stop ORG in progress
- o perform required FRG
- o if higher priority Red arises, suspend lower Red and address higher Red

4.5.4 Orange terminus encountered

- o monitor all remaining trees
- o if no Red encountered, suspend ORG and perform required FRG
- o if Red condition or higher priority Orange condition arises, suspend Orange FRG and address Red or Orange condition

4.5.5 Yellow terminus encountered

- o immediate operator action not required
- o indicative of off-normal and/or temporary condition (may be restored to normal by actions in progress)
- o Yellow conditions FRGs are implemented based on operator evaluation

4.5.6 Green terminus encountered

- o indicative of normal plant conditions (CSF of that tree not challenged)
- o no operator actions required

4.5.7 Tree Monitoring

4.5.7.1 continuous if any status coded Red or Orange exist

LESSON OUTLINE		NOTES AND REFERENCES
4.5.7.2	no condition worse than yellow	
	o reduce monitoring frequency to 10-20 minutes intervals	
	o dependent upon no significant change in plant status	
4.5.7.3	Terminate tree monitoring	
	o Rx protection system restored (trip breakers closed)	
	o Engineering Safeguards System restored (SI reset)	
4.5.8	Example of CSF status tree usage (subcriticality)	1 TP 4.5.8
4.5.8.1	RED terminus encountered	
	o immediate response required (no other higher priority than subcriticality (RED)	
	o go to FRG FR-S.1	
	o suspend whichever ORG in progress	
	o after FRG completed return to guideline and step in effect	
4.5.8.2	ORANGE terminus encountered	
	o prompt response required	
	o go to FRG FR-S.1	
	o continue monitoring status trees	
	o any RED terminus takes priority over the subcriticality ORANGE	
4.5.8.3	YELLOW terminus encountered	
	o action not required immediately	
	o continue monitoring trees	
4.6	Use of ERG network in control room	
4.6.1	Direct entry	Use large transition charts for this presentation
	o Rx trip or SI (E-0)	
	o Loss of all ac (ECA-0.0)	

LESSON OUTLINE | NOTES AND REFERENCES

4.6.2 Entry into E-0
- o remains in E-0 and directed by action step to monitor trees or
- o transition to other guideline, at which point trees are to be monitored

4.6.3 Tree monitoring and actions

4.6.3.1 Use standard rules of usage

4.6.3.2 May be done by
- o control room operators
- o other member of shift assigned to control room
- o dedicated computer routine (backup paper required if computer goes down)

4.6.3.3 Operator in charge of recovery
- o be immediately informed of RED or ORANGE priority status condition
- o be regularly advised of YELLOW and GREEN conditions

4.6.3.4 ORG actions
- o suspended if either RED or ORANGE condition detected
- o NO ORG actions are to be performed while CSF (RED or ORANGE condition) is being restored (unless specificied by FRG)

4.6.3.5 CSF restored
- o normally return to guideline and step in effect before FRG implemented
- o at times FRG require transitions to ORG other than one in effect (due to conditions created by FRG)

LESSON OUTLINE	NOTES AND REFERENCES
4.6.3.6 FRG to ORG o operator judgement required to prevent inadvertent reinstatement of RED or ORANGE condition o ORGs optimal assuming safety equipment available o RED or ORANGE condition implies some safety equipment or function not available o some ORG adjustments may be required	Example: FR-H.1 establishes alternate feed path to SGs. ES-0.1 requires main feedline be isolated. Main feed or condensate feed may be alternate path in FR-H.1. The operator would not want to isolate the main feed line.
4.6.3.7 Certain ECAs take precendence over FRGs based on specific initiating events (identified by NOTE at beginning of guidelines)	Example: ECA-0.0 deals with loss of electrical power to ac emergency busses. No safeguards equipment used to restore CSFs is operable. Trees should be monitored for information only. ECA-0.0 provides best strategy for maintaining CSFs.
4.6.4 ERG use ends o transition to normal plant operating procedure o transition to "appropriate" procedure while on RHR at cold shutdown conditions o cold leg recirculation or hot hot leg recirculation with longer term recovery actions being analyzed	

0040V/1:101183

LESSON OUTLINE		NOTES AND REFERENCES
4.7	ERGs Modes of Applicability	
	o originally written for transients a "hot" or "at power" conditions	1 TP 4.7 a-c
	o transients would result in a protective function	
	o guidance based upon availability of safety equipment defined in Tech Specs for Mode 1 and 2 operation	
4.7.1	Other modes of operation	
	o safety equipment defined for modes 1 and 2 may not be available	
	o some detailed instructions in ERGs not applicable	
4.7.2	CSF trees	
	o assumes mode 1 or 2 initial conditions	
	o tree use may be expanded to other modes if intent of each tree is understood	