Enclosure 2

B-a.

Number	Symptom/Title	Revision No./Date
E-3	STEAM GENERATOR TUBE RUPTURE	Basic 1 Sept. 1981

STEP	ACTION/EXPECTED RESPONSE	RESPONSE NOT OBTAINED

Caution *If all steam generators are ruptured, the steam generator with the lowest level should be used for subsequent RCS cooldown. DO NOT isolate this steam generator. Consider it non-ruptured.*

NOTE
- *Foldout page should be open.*
- *Personnel should be available for sampling during this procedure.*

1	Identify Ruptured Steam Generator(s): • Unexpected rise in any steam generator narrow range level • High radiation from any steam generator blowdown line 1) [Enter plant specific steps for opening blowdown lines sequentially to check radiation] • High radiation from any steam generator sample • High radiation from any steam generator steamline	IF NOT immediately identified, THEN continue with steps 3 through 9. WHEN ruptured steam generator(s) identified, THEN do step 2.
2	Isolate Ruptured Steam Generator(s): a. WHEN in narrow range, THEN stop all AFW flow to ruptured steam generator(s) 1) [Enter plant specific steps]	
	b. Close ruptured steam generator(s) main steamline isolation valve and bypass valve	b. Close non-ruptured steam generator main steamline isolation valves and bypass valves. Use non-ruptured steam generator PORVs for steam dump.
	c. Verify ruptured steam generator(s) PORVs closed	c. Manually close ruptured steam generator(s) PORV.
	d. Close ruptured steam generator(s) steam supply valve to turbine-driven AFW pump	

STEP	ACTION/EXPECTED RESPONSE	RESPONSE NOT OBTAINED
3	Check Pressurizer PORV Block Valves: a. Power available to block valves b. Block valves - OPEN	a. Restore power to block valves. b. Open block valve unless it was closed to isolate a faulty PORV.
4	Check Pressurizer PORVs: a. PORVs - CLOSED	a. Manually close PORVs. IF any valve cannot be closed, THEN manually close its block valve.

Caution
- *IF any pressurizer PORV opens because of high RCS pressure, repeat step 4 after pressure drops below PORV setpoint.*
- *Seal injection flow should be maintained to all RCPs.*

5	Check If RCPs Should Be Stopped: a. SI running - CHECK FOR FLOW OR PUMP BREAKER INDICATOR LIGHTS LIT • Charging/SI -OR- • High-head SI b. RCS pressure - EQUAL TO OR LESS THAN (1) PSIG c. Stop all RCPs	a. DO NOT STOP RCPs. Go to step 6. b. DO NOT STOP RCPs. Go to step 6

(1) Enter plant specific value derived from background document to E-0.

STEP	ACTION/EXPECTED RESPONSE	RESPONSE NOT OBTAINED
6	Check If Low-head SI Pumps Should Be Stopped: a. Check RCS pressure: 1) Pressure - GREATER THAN (1) PSIG 2) Pressure - STABLE OR INCREASING b. Reset SI c. Stop low-head SI pumps and place in standby	1) IF less than (1) psig, THEN go to E-1, LOSS OF REACTOR COOLANT, STEP 13. 2) IF decreasing, THEN go to step 7.

Caution *IF RCS pressure drops below (1) psig, the low-head SI pumps must be manually restarted to supply water to the RCS.*

7	Check Electrical Power And Air Supply Available To Essential Equipment: a. [Enter plant specific list]	Establish power supplies, as necessary.

(1) Enter plant specific shutoff head of low-head SI pumps.

STEP	ACTION/EXPECTED RESPONSE	RESPONSE NOT OBTAINED
8	Check Secondary System Integrity:	
	a. RCS hot leg temperature - GREATER THAN (1) °F	a. IF any hot leg temperature less than (1) °F and decreasing, THEN close all main steamline isolation valves and bypass valves. IF any steam generator pressure continues to decrease, THEN go to ES-3.3, SGTR WITH SECONDARY DEPRESSURIZATION.
	b. ALL steam generator pressures - GREATER THAN (2) PSIG	b. IF any steam generator pressure less than (2) psig, THEN close all main steamline isolation valves and bypass valves. IF any steam generator pressure continues to decrease, THEN GO TO ES-3.3, SGTR WITH SECONDARY DEPRESSURIZATION.
	Caution *Alternate water sources for AFW pumps will be necessary if CST level is low.*	
9	Check Steam Generator Levels:	
	a. Narrow range level - GREATER THAN (3) %	a. IF less than (3) %, THEN maintain full AFW flow until narrow range level is greater than (3) %
	b. Throttle AFW flow to maintain narrow range level at (4) %	
	Caution *DO NOT PROCEED to step 10 until faulted steam generator has been identified and isolated.*	

(1) Enter plant specific temperature corresponding to lowest expected hot leg temperature following a normal reactor trip.

(2) Enter plant specific value corresponding to 50 psi above maximum Tech Spec accumulator nitrogen pressure.

(3) Enter plant specific value showing level just in the narrow range including allowances for normal channel accuracy, post-accident transmitter errors and reference leg process errors.

(4) Enter plant specific value corresponding to no-load steam generator level including allowances for post-accident transmitter errors and reference leg process errors.

STEP	ACTION/EXPECTED RESPONSE	RESPONSE NOT OBTAINED
10	Cooldown Non-ruptured Steam Generators 50°. Below Ruptured Steam Generator:	
	a. Determine required non-ruptured steam generator pressure in table below:	

Ruptured Steam Generator Pressure (PSIG)	Required Non-ruptured Steam Generator Pressure (PSIG)	
	Any RCP Running	All RCPs Stopped
1200	780	610
1100	710	550
1000	640	490
900	570	430
800	500	370
700	430	320
600	350	260
500	310	210
400	230	160

	b. Rapidly dump steam to condenser from non-ruptured steam generators:	b. Rapidly dump steam with non-ruptured steam generator PORVs.
	1) [Enter plant specific steps]	
	c. Check ruptured steam generator(s) pressure – STABLE OR INCREASING	c. IF decreasing, THEN go to ES-3.3, SGTR WITH SECONDARY DEPRESSURIZATION, STEP 8.

Caution
- *If containment conditions are abnormal, go to E-1, LOSS OF REACTOR COOLANT, STEP 9.*
- *Disregard RCP trip criteria for all subsequent steps in this guideline.*

STEP	ACTION/EXPECTED RESPONSE	RESPONSE NOT OBTAINED
11	Check RCS Pressure:	
	a. RCS pressure - AT LEAST 200 PSI GREATER THAN RUPTURED STEAM GENERATOR PRESSURE	a. IF NOT, THEN go to ECA-3, SGTR CONTINGENCIES.
12	Depressurize RCS Using Normal Spray:	
	a. Verify normal spray - AVAILABLE	a. Go to step 14.
	b. Open normal spray valves	b. Go to step 14.
	c. Verify RCS pressure - DECREASING	c. Close spray valves and go to step 14.
13	Check If RCS Depressurization Should Be Stopped:	
	a. RCS pressure - LESS THAN OR EQUAL TO RUPTURED STEAM GENERATOR PRESSURE -OR- Pressurizer level - GREATER THAN (1) %	a. Continue depressurization until either condition met.
	b. Stop RCS depressurization by closing spray valves	
	c. Check pressurizer level - GREATER THAN (2) %	c. IF level less than (2) %, THEN go to ECA-3, SGTR CONTINGENCIES.
	d. Verify RCS pressure - INCREASING	d. IF RCS pressure decreasing or stable, THEN stop RCPs in loops with spray line connections.
	e. Go to step 16	
14	Depressurize RCS Using One Pressurizer PORV:	
	a. Open one pressurizer PORV	a. IF RCS cannot be depressurized using any PORV, THEN use auxiliary spray.

(1) Enter plant specific value corresponding to high pressurizer level reactor trip setpoint.
(2) Enter plant specific value showing level just in span including allowances for normal channel accuracy.

STEP	ACTION/EXPECTED RESPONSE	RESPONSE NOT OBTAINED
15	Check If RCS Depressurization Should Be Stopped:	
	a. RCS pressure - LESS THAN OR EQUAL TO RUPTURED STEAM GENERATOR PRESSURE -OR- Pressurizer level - GREATER THAN (1) %	a. Continue depressurization until either condition met.
	b. Stop RCS depressurization:	
	1) Close PORV	1) Close PORV block valve.
	2) Close auxiliary spray valve	2) Isolate auxiliary spray line.
	c. Check pressurizer level - GREATER THAN (2) %	c. IF level less than (2) %, THEN go to ECA-3, SGTR CONTINGENCIES.
	d. Verify RCS pressure - INCREASING	d. IF RCS pressure NOT increasing, THEN check PRT conditions. IF PRT conditions indicate RCS leak, THEN go to E-1, LOSS OF REACTOR COOLANT.
	Caution *If PRT integrity is lost, abnormal containment conditions may not be reliable indications of a loss of reactor coolant.*	
16	Check If SI Can Be Terminated:	
	a. RCS pressure - INCREASES BY 200 PSI	a. DO NOT TERMINATE SI. IF pressure has NOT increased by 200 psi AND pressurizer level is stable or decreasing, THEN go to ECA-3, SGTR CONTINGENCIES.
	b. Pressurizer level - GREATER THAN (2) %	b. DO NOT TERMINATE SI. Go to ECA-3, SGTR CONTINGENCIES.
	c. RCS subcooling - GREATER THAN - (3) °F	c. DO NOT TERMINATE SI.
	Caution *Do not proceed to step 17 until all conditions in step 16 are met.*	

(1) Enter plant specific value corresponding to high pressurizer level reactor trip setpoint.
(2) Enter plant specific value showing level just on span including allowances for normal channel accuracy.
(3) Enter sum of temperature and pressure measurement system errors translated into temperature using saturation tables.

STEP	ACTION/EXPECTED RESPONSE	RESPONSE NOT OBTAINED
17	Terminate SI: a. Go to ES-3.1, SI TERMINATION FOLLOWING STEAM GENERATOR TUBE RUPTURE	
18	Check If Condenser Can Be Used: a. Condenser - AVAILABLE	a. IF condenser not available, THEN attempt to restore condenser. IF condenser can NOT be restored, THEN evaluate if releases from ruptured faulted steam generator will exceed 10 CFR 20 limits. IF 10 CFR 20 limits will be exceeded, and any RCP running THEN cooldown per ES-3.2, SGTR ALTERNATE COOLDOWN.
19	Verify Adequate Shutdown Margin	Borate, as necessary.

Caution *Steps 20 through 23 must be performed simultaneously to avoid loss of pressurizer level control.*

20	Initiate RCS Cooldown To 350°F: a. Maintain cooldown rate - LESS THAN 50°F/HR b. Dump steam from non-ruptured steam generators to condenser 1) [Enter plant specific steps]	b. Dump steam with non-ruptured steam generator PORVs.

Caution *Charging and letdown flows should be compared to determine if leakage between the RCS and ruptured steam generator is stopped.*

21	Maintain Pressurizer Level In Normal Operating Range: a. Operate charging and letdown, as necessary	

STEP	ACTION/EXPECTED RESPONSE	RESPONSE NOT OBTAINED
22	Depressurize Ruptured Steam Generator(s):	
	a. Slowly release steam to condenser from ruptured steam generator	a. Slowly release steam to atmosphere with ruptured steam generator(s) PORV.
	1) [Enter plant specific steps]	
	Caution • *Maintain RCS pressure and temperature within normal cooldown limits.* • *IF RCS pressure or pressurizer level drop in an uncontrolled manner, THEN reinitiate SI and return to step 10.*	
23	Depressurize RCS:	
	a. Reduce RCS pressure to maintain RCS/ruptured steam generator pressures equal	
	1) Use normal pressurizer spray	1) IF letdown is in service, THEN use auxiliary spray. IF NOT in service, THEN use one pressurizer PORV.
24	Determine If SI Accumulators Should Be Isolated:	
	a. RCS pressure - LESS THAN OR EQUAL TO (1) PSIG	a. IF RCS pressure greater than (1) psig, THEN return to step 20.
	b. Close all SI accumulator isolation valves	b. Vent any unisolated accumulator.
25	Check If RHR System Can Be Placed In Service:	
	a. RCS hot leg temperatures - LESS THAN 350°F IN NON-RUPTURED LOOPS	a. IF greater than 350°F, THEN return to step 20.
	b. RCS pressure - APPROXIMATELY 400 PSIG	b. IF greater than 400 psig, THEN return to step 21.
	Caution *Do not collapse the pressurizer bubble.*	

(1) Enter plant specific value slightly above normal accumulator pressure.

STEP	ACTION/EXPECTED RESPONSE	RESPONSE NOT OBTAINED
26	Place RHR System In Service Per [Plant Specific Procedure]	
27	Continue Cooldown To Cold Shutdown: a. Cooldown using RHR b. At least one RCP - RUNNING	b. IF all RCPs stopped, THEN continue dumping steam from non-ruptured steam generators until they have stopped steaming.
28	Check RCS Temperature: a. Temperature - LESS THAN (1) °F b. Stop all RCPs c. Cooldown pressurizer 1) Spray pressurizer with auxiliary Spray	a. IF greater than (1) °F, THEN return to step 27.
29	Maintain Cold Shutdown Conditions.	

— END —

(1) Enter plant specific number for stopping RCPs during normal cooldown.

FOLDOUT FOR E-3 AND ES-3 GUIDELINES

1. RCP TRIP CRITERIA

- Trip any RCP if component cooling water to that pump is lost.
- If a controlled cooldown is not in progress, then trip all RCPs when BOTH conditions listed below are met:

  a. SI is ON

  b. RCS pressure - EQUAL TO OR LESS THAN (1) PSIG

2. SI REINITIATION CRITERIA FOLLOWING STEAM GENERATOR TUBE RUPTURE

Reinitiate SI if ANY ONE of the parameters listed below occurs:

(1) RCS subcooling - LESS THAN (2) PSIG

(2) Pressurizer level - LESS THAN 20%

3. SYMPTOMS OF LOSS OF REACTOR COOLANT DURING STEAM GENERATOR TUBE RUPTURE

Go to E-1, LOSS OF REACTOR COOLANT, if abnormal containment conditions persist AND are not due only to failure of PRT rupture disc.

4. SYMPTOMS OF PRIMARY TO SECONDARY LEAKAGE DURING RECOVERY ACTIONS

Charging and letdown flows should be compared to determine if leakage between the RCS and the ruptured steam generator exists.

5. SYMPTOMS FOR FR-C.1, RESPONSE TO INADEQUATE CORE COOLING

Go to FR-C.1, RESPONSE TO INADEQUATE CORE COOLING, when ALL symptoms in ANY ONE of the following symptom sets occurs:

PARAMETER	SYMPTOM SET		
	I	II	III
1. TCs	> 1200 °F	–	> 700 °F
2. Containment Condition	–	ABNORMAL	ABNORMAL
3. RCP Status	–	ANY ON	ALL OFF
4. RVLIS	–	< 100% NR	< (3) %WR NR

6. SYMPTOMS FOR FR-H.1, RESPONSE TO LOSS OF SECONDARY HEAT SINK

Go to FR-H.1, RESPONSE TO LOSS OF SECONDARY HEAT SINK, if AFW NOT AVAILABLE.

(1) Enter plant specific value derived from background document.

(2) Enter sum of temperature and pressure measurement system errors translated into temperature using saturation table.

(3) Enter plant specific value which is 3½ feet above bottom of active fuel in core with zero void fraction, plus uncertainties.

STEP	ACTION/EXPECTED RESPONSE	RESPONSE NOT OBTAINED
	Caution • *Following SI reset, automatic reinitiation of SI will not occur until reactor trip breakers are reset.* • *If loss of offsite power occurs after SI reset, manual action may be required to restart safeguards equipment.*	
	NOTE *Foldout page should be open.*	
1	Reset SI.	
2	Stop SI Pumps And Place In Standby: a. Low-head SI pumps b. High-head SI pumps c. All but one charging/SI pump	
3	Establish Charging/SI Pump Miniflow: a. Verify CCW flow to seal water heat exchanger b. Open miniflow isolation valves	
4	Isolate BIT: a. Close inlet isolation valves b. Close outlet isolation valves	
5	Verify SI Reinitiation NOT Required: a. RCS subcooling - GREATER THAN (1) °F	a. Manually operate SI pumps, as required. IF subcooling can NOT be maintained, THEN manually reinitiate SI and return to E-3, STEAM GENERATOR TUBE RUPTURE, STEP 10.
	b. Pressurizer level - GREATER THAN 20%	b. Manually operate SI pumps, as required. IF pressurizer level can NOT be maintained, THEN manually reinitiate SI and return to E-3, STEAM GENERATOR TUBE RUPTURE, STEP 10.

(1) Enter sum of temperature and pressure measurement system errors translated into temperature using saturation tables.

STEP	ACTION/EXPECTED RESPONSE	RESPONSE NOT OBTAINED
6	Verify Offsite Power Available: • [Enter plant specific list]	Try to restore offsite power: • [Enter plant specific list] IF offsite power cannot be restored, THEN manually load following equipment on the diesel generators: • [Enter plant specific list]
7	Reset Containment Isolation Phase A.	
8	Establish Charging: a. Close charging flow control valve b. Open charging line isolation valves c. Open charging flow control valve to establish desired flow	
9	Establish Letdown: a. Open letdown line containment isolation valves b. Open letdown line isolation valves c. Open letdown orifice isolation valves, as appropriate	
10	Align Charging/SI Pump Suction to VCT: a. Open VCT outlet isolation valves b. Close RWST outlet isolation valves	
11	Check VCT Makeup Control System: a. Makeup set for automatic control b. Makeup set for GREATER THAN RCS boron concentration	a. Adjust controls, as appropriate. b. Adjust controls, as appropriate.

STEP	ACTION/EXPECTED RESPONSE	RESPONSE NOT OBTAINED
12	Check RCP Cooling:	
	a. RCP seal injection flow - NORMAL	a. Adjust charging line hand control valve, as necessary.
	b. RCP CCW system flow - NORMAL	b. Establish CCW flow to RCPs: 1) Reset containment isolation Phase B, if necessary 2) Open appropriate CCW system isolation valves.
13	Check Non-Faulted Steam Generator Levels:	
	a. Narrow range level - GREATER THAN (1) %	a. IF less than (1) %, THEN maintain full AFW flow until narrow range level is greater than (1) %.
	b. Throttle AFW flow to maintain narrow range level at (2) %	
14	Check CST Level:	
	a. CST level - GREATER THAN (3) %	a. IF CST level low, THEN switch to alternate AFW water supply.
15	Establish Pressurizer Pressure Control:	
	a. Energize pressurizer heaters, as necessary to maintain pressure	

(1) Enter plant specific value showing level just in the narrow range including allowances for normal channel accuracy, post-accident transmitter errors, and reference leg process errors.

(2) Enter plant specific value corresponding to no-load steam generator level including allowances for post-accident transmitter errors and reference leg errors.

(3) Enter plant specific low level setpoint.

STEP	ACTION/EXPECTED RESPONSE	RESPONSE NOT OBTAINED
	NOTE *RCPs should be run in order of priority to provide pressurizer spray.*	
16	Check RCP Status:	
	a. At least one RCP - RUNNING	a. IF no RCP running, THEN attempt to start one RCP: 1) Establish conditions for running an RCP: [Enter plant specific list] 2) Start one RCP. IF an RCP cannot be started, THEN monitor natural circulation from trended values: (a) RCS subcooling - GREATER THAN (1) °F. (b) Steam pressure - STABLE. (c) RCS hot leg temperature - STABLE OR SLOWLY DECREASING. (d) Core exit TCs - STABLE OR SLOWLY DECREASING. (e) RCS cold leg temperature - NEAR SATURATION TEMPERATURE FOR STEAM PRESSURE.
		IF natural circulation NOT verified, THEN increase dumping steam.
17	Check Intermediate Range Flux:	
	a. Flux - BELOW (1)	a. Continue with step 18. WHEN below (1), THEN do steps 17 b and c.
	b. Verify source range detectors re-energized	b. Manually re-energize source range detectors.
	c. Transfer nuclear recorders to source range scale	

(1) Enter sum of temperature and pressure measurement + system errors translated into temperature using saturation tables.

(1) Enter plant specific upper range of source range detectors.

STEP	ACTION/EXPECTED RESPONSE	RESPONSE NOT OBTAINED
18	Shutdown Unnecessary Plant Equipment: a. [Enter plant specific list]	
19	Verify SI Reinitiation NOT Required:	
	a. RCS subcooling - GREATER THAN (1) °F	a. Manually operate SI pumps, as required. IF subcooling can NOT be maintained, THEN manually reinitiate SI and return to E-3, STEAM GENERATOR TUBE RUPTURE, STEP 10.
	b. Pressurizer level - GREATER THAN 20%	b. Manually operate SI pumps, as required, IF pressurizer level can NOT be maintained, THEN manually initiate SI and return to E-3, STEAM GENERATOR TUBE RUPTURE, STEP 10.
20	Continue With Procedure In Effect.	

— END —

(1) Enter sum of temperature and pressure measurement system errors translated into temperature using saturation tables.

FOLDOUT FOR E-3 AND ES-3 GUIDELINES

1. RCP TRIP CRITERIA

- Trip any RCP if component cooling water to that pump is lost.
- If a controlled cooldown is not in progress, then trip all RCPs when BOTH conditions listed below are met:

  a. SI is ON

  b. RCS pressure - EQUAL TO OR LESS THAN (1) PSIG

2. SI REINITIATION CRITERIA FOLLOWING STEAM GENERATOR TUBE RUPTURE

Reinitiate SI if ANY ONE of the parameters listed below occurs:

(1) RCS subcooling - LESS THAN (2) PSIG

(2) Pressurizer level - LESS THAN 20%

3. SYMPTOMS OF LOSS OF REACTOR COOLANT DURING STEAM GENERATOR TUBE RUPTURE

Go to E-1, LOSS OF REACTOR COOLANT, if abnormal containment conditions persist AND are not due only to failure of PRT rupture disc.

4. SYMPTOMS OF PRIMARY TO SECONDARY LEAKAGE DURING RECOVERY ACTIONS

Charging and letdown flows should be compared to determine if leakage between the RCS and the ruptured steam generator exists.

5. SYMPTOMS FOR FR-C.1, RESPONSE TO INADEQUATE CORE COOLING

Go to FR-C.1, RESPONSE TO INADEQUATE CORE COOLING, when ALL symptoms in ANY ONE of the following symptom sets occurs:

PARAMETER	SYMPTOM SET		
	I	II	III
1. TCs	> 1200 °F	–	> 700 °F
2. Containment Condition	–	ABNORMAL	ABNORMAL
3. RCP Status	–	ANY ON	ALL OFF
4. RVLIS	–	< 100% NR	< (3) %NR OR

6. SYMPTOMS FOR FR-H.1, RESPONSE TO LOSS OF SECONDARY HEAT SINK

Go to FR-H.1, RESPONSE TO LOSS OF SECONDARY HEAT SINK, if AFW NOT AVAILABLE.

(1) Enter plant specific value derived from background document.
(2) Enter sum of temperature and pressure measurement system errors translated into temperature using saturation tables.
(3) Enter plant specific value which is 3 to 5 ft above bottom of active fuel in core with zero void fraction, plus uncertainties.

Enclosure 1

C-e

TASK I.C OPERATING PROCEDURES

A. OBJECTIVE: Improve the quality of procedures to provide greater assurance that operator and staff actions are technically correct, explicit and easily understood for normal, transient, and accident conditions. The overall content, wording, and format of procedures that affect plant operation, administration, maintenance, testing, and surveillance will be included. A principal part of this work is to improve procedures for dealing with abnormal conditions and emergencies by improving the delineation of symptoms, events, and plant conditions that identify emergency or off-normal situations that confront the operators and, once identified, to assure consistency with operator training.

B. NRC ACTIONS: NRC has taken action, and will take further action, to assure immediate improvement of selected emergency operating and some other operating procedures for operating reactors and near-term operating license applicants. Specific actions are being taken for near-term operations, and actions that will lead to new and better procedures will then be considered for the longer term. In the long term, symptoms-oriented approaches to abnormal and emergency procedures will be evaluated. This effort will be coordinated with control room, simulator, and training improvements. These actions will be integrated with new operating instruments for diagnostic purposes based on the assumption that adequately trained personnel can perform the specified actions. The need for coordination and training of plant personnel is recognized.

1. Short-term accident analysis and procedures revision.

a. Description: There is an ongoing three-phase program for improving the analysis of design basis and off-normal transients and accidents and the procedures for handling such transients and accidents (see NUREG-0578, Sec. 2.1.9).

(1) Small-break loss-of-coolant accidents (LOCAs). NRR sent letters on September 13 and 27, October 10 and 30, and November 9, 1979 referencing Section 2.1.9 of NUREG-0578 to licensees of operating plants, pending operating

license applicants, licensees of plants under construction, and applicants for construction permits. The staff required that analyses be performed and guidelines prepared to develop emergency operating instructions for handling small-break loss-of-coolant accidents. Appropriate retraining of operators was also required (see also Item I.A.2.1). Guidelines were prepared for each class of operating plants and were reviewed and approved by the NRR staff.

Detailed emergency operating procedures have been or are being prepared for each operating and near-term operating plant to implement the approved guidelines for handling small-break LOCAs. An NRC audit team (with NRR leading and IE participating) performed reviews of procedures for lead plants designed by each reactor manufacturer. Procedures for the remaining operating plants will be reviewed by IE. For each plant that is being reviewed for an operating license, NRR and IE will review the small-break LOCA emergency operating instructions.

(2) Inadequate core cooling. In letters of September 13 and 27, October 10 and 30, and November 9, 1979, NRR required operating licensees, pending operating license applicants, licensees of plants under construction, and applicants for construction permits to perform analyses, including preparation of emergency procedure guidelines, and to develop procedures and conduct training to assist the plant operating staff to (a) recognize and prevent impending core uncovering and (b) recover from a condition in which the core has experienced inadequate core cooling (see also Item I.A.2.1). An NRR team, with IE members, will review these procedures on an audit basis for lead operating plants. IE will review the procedures for the remaining operating plants.

(3) Transients and accidents. In letters of September 13 and 27, October 10 and 30, and November 9, 1979, NRR required licensees of operating plants, operating license applicants, licensees of plants under construction, and pending construction permit applicants 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 three 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 B&W designed plants, the staff will followup on the Bulletin and Orders matters relating to analysis methods and results, as listed in 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, 42, 47, 55, 57.

(4) Confirmatory analyses of selected transients. In addition to the analyses performed by the reactor vendors, analyses of selected transients will be performed by NRR, using the best available computer codes, to provide the basis for comparisons with the analytical methods being used by the reactor vendors. These comparisons, together with comparisons to other data, will constitute the short-term verification effort to assure the adequacy of the analytical methods being used to generate emergency procedures. (See also Item II.E.2.2.) These analyses in the case of the B&W design will also be used to establish whether core-barrel check valves have been adequately modeled in the analysis by the vendor since the Three Mile Island accident.

b. Schedule.

(1) Guidelines for handling small-break LOCAs at operating reactors were reviewed and approved by NRR Bulletins and Orders Task Force in late 1979. Reviews of lead operating plants were performed as indicated in NUREG-0645. IE will conduct reviews of remaining operating plants by June 1, 1980. Reviews of operating license applicants will be consistent with operating license review schedules.

(2) Audits of lead operating plants will be completed in FY80. Reviews for the remaining operating plants will be conducted by IE by April 1, 1981. Reviews of operating licensee applicants will be consistent with operating license review schedules.

(3) NRR will clarify the scope of the task and issue a revised schedule for task completion by July 1980. It is expected that this requirement will be coupled with Task I.C.9.

(4) Confirmatory analyses of selected transients are to be complete by June 1980.

c. Resources: NRR FY80 - 3.5 my and $50,000, FY81 - 6.0 my; IE FY80 - 5.0 my, FY81 - 4.0 my; ADM FY80 - 0.1 my and $17,000, FY81 - 0.1 my and $17,000.