TVA

Tennessee Valley Authority, 1101 Market Street, Chattanooga, Tennessee 37402-2801

July 24, 2003

10 CFR 50,
Appendix E
Section V

U.S. Nuclear Regulatory Commission
ATTN: Document Control Desk
Washington, D.C. 20555-0001

Gentlemen:

In the Matter of	)	Docket Nos.	50-259	50-390
Tennessee Valley Authority	)		50-260	50-391
			50-296	50-327
				50-328

TVA CENTRAL EMERGENCY CONTROL CENTER (CECC) - EMERGENCY PLAN IMPLEMENTING PROCEDURE (EPIP) REVISIONS

In accordance with the requirements of 10 CFR Part 50, Appendix E, Section V, enclosed are copies of the Effective Page Listing and revisions to CECC EPIPs.

PROCEDURE		EFFECTIVE DATE
EPIP	EPL	7/1/03
EPIP-8	Rev. 26	7/1/03
EPIP-13	Rev. 10	7/1/03

If you have any questions, please contact Terry Knuettel at (423) 751-6673.

Sincerely,

for Mark J. Burzynski
Manager
Nuclear Licensing

Enclosures
cc: See page 2

A045

cc (Enclosures):

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NRC Senior Resident Inspector [No enclosures, by request of site resident]
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DOCUMENT RELEASE AND FILING INSTRUCTIONS

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Prepared By: Gail White
Extension: 751-2108
Organization: AS&P
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______ *Date*

Contact: ______ *Ext.* ______

Acceptance:
Janice E. Pogue 7/2/03
Signature *Date*

TENNESSEE VALLEY AUTHORITY
CENTRAL EMERGENCY CONTROL CENTER EMERGENCY PLAN
IMPLEMENTING PROCEDURES
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CECC EPIP Coversheet

Tennessee Valley Authority CENTRAL EMERGENCY CONTROL CENTER EMERGENCY PLAN IMPLEMENTING PROCEDURES	Title DOSE ASSESSMENT STAFF ACTIVITIES DURING NUCLEAR PLANT RADIOLOGICAL EMERGENCIES	CECC EPIP-8 REV. 26 Effective Date: 7/01/03

WRITTEN BY: Thomas E. Atkins
Signature

REVIEWED BY: [signature] 6/26/03
Signature Date

PLAN EFFECTIVENESS DETERMINATION: Thomas E. Atkins 6/26/03
Signature Date

CONCURRENCES

Concurrence Signature	Date
☒ Manager, EP Program Planning and Implementation David Pond	6/30/03
☒ Manager, Emergency Preparedness B.K. Marks	6/30/03
☒ Manager, Radiological and Chemistry Services [signature]	07/01/03
☐	

APPROVAL

APPROVED BY: [signature] VP Eng & Tech Svcs 7/1/03
Signature Title Organization Date

CECC-EPIP-8
DOSE ASSESSMENT STAFF ACTIVITIES
DURING NUCLEAR PLANT RADIOLOGICAL EMERGENCIES
REVISION LOG

Rev. No.	Date	Revised Pages
0	3/22/88	All (Changed from IPD to EPIP)
1	11/18/88	2-7, Apps. A, B, C, & D
2	12/12/88	Appendix A
3	4/26/89	All
4	9/19/89	App. C
5	10/26/89	All
6	5/21/91	App. B, pgs. 1-4; Appendix C, pgs. 1-2; App. D, pg. 1
7	10/17/91	App. B, pgs. 2-4; App. C, pg. 1.
8	05/13/93	1-4; App. A; App. B, pg. 1, 3, & 4; and App. G; App. C deleted. All pages issued.
9	11/22/93	Pg. 4; App. B, pgs. 1&4; App. D changed to App. C; App. E changed to App. D; App. F changed to App. E; and App. G changed to App. F.
10	11/30/93	1, 3, 4; App. A, pg. 1; App. B, pgs. 1-2; App. C, pg. 1-5; App. D, pg. 1; App. E, pg. 1; App. F, pg. 1; App. G, pgs. 1-6.
11	06/24/94	App. B, pg. 1; App. D, pgs. 2-5; App. F; App. J added. All pages issued.
12	6/27/95	Pg. 1; App. A; App. B, p.3; App. C, p. 5; App. D, p. 2; App. G, pgs. 4 and 6
13	1/17/96	App. B, pg. 2, editorial changes, add table for BFN stack release; App. C, pgs. 1 & 3, Add new criteria for Type I and Type II releases; App. D, pgs. 2-5, add nonogram alignment checks
14	5/30/96	Pg. 3, App. A, App. B, App. C, App. D, App. F, App. G; annual review; ground level release tables and nomograms made generic to all three sites; all pages issued.
15	10/30/96	Pg. 3, App. B, and App. D; Add reference to App. I of CECC EPIP-7, remove deleted pages, make correction to Nomogram Alignment Check Table.
16	5/30/97	Editorial changes, update manual dose assessment methodology, update preliminary assessment table, revise river miles on tables in Appendix G, annual review. All pages issued.
17	8/8/97	Revise default river flow rate for BFN, revise responsibilities of Norris Lab, add water intake tables. All pages issued.

REVISION LOG (Continued)

Rev. No.	Date	Revised Pages
18	6/9/98 ~~6/4/98~~ RR	Annual review. Organization title changes. In Appendix D clarify Type I and Type II formulas. Remove Tennessee River miles from tables. All pages issued.
19	10/27/98	Correct reference to CECC EPIP-1 Appendix on Appendix J.
20	5/20/99	Annual review. Editorial and clarification changes, revise public water use tables. All pages issued.
21	9/8/00	Annual review. Editorial changes. All pages issued.
22	3/30/01	Revised to incorporate the new source term methodology in the RED suite of codes revision
23	11/22/02	Revised all pages to reflect human factor improvements in REP codes and manual Included changes due to code revision necessary for H-3 project.
24	3/31/03	Added sections to Appendix F to provide instructions for manual method of calculating TEDE and thyroid CDE doses at Site Boundary (0.62 miles). All pages issued.
25	6/16/03	Make corrections to Appendix F. All pages issued.
26	07/01/03	Revision consistency in accordance with CHPER 03-000257-000. Revise location indicator in step C.4 on page 18. Correct the location indicator, in step 2.b on page 19. Correct spelling of Circle on page 22. Minor format alignment changes. All pages issued.

Table of Contents

DOSE ASSESSMENT STAFF ACTIVITIES DURING NUCLEAR PLANT RADIOLOGICAL EMERGENCIES

1.0 PURPOSE

To guide Dose Assessment in obtaining necessary information, calculating dose rates and doses, and communicating assessment results used in responding to radiological emergencies at nuclear power plants.

2.0 SCOPE

This procedure applies to activities of Dose Assessment in actual and hypothetical radiological emergency situations. While the activities of the Dose Assessment staff are expected to follow this procedure, it is expected that circumstances may arise during an event which will void portions of this procedure. Therefore, this procedure is a guide for the operation of the Dose Assessment staff under the ideal conditions.

3.0 STAFFING

3.1 Activation and Notification

The Initial notification of an event comes from the Operations Duty Specialist via the Emergency Paging System (EPS) or manual callout. Additional Dose Assessor support is contacted in accordance with Appendix A. The Dose Assessor is a position required for the CECC to make Protective Action Recommendations and to meet minimum staffing levels.

Upon reporting to the CECC, perform initial activities in accordance with the checklist provided as **Appendix A.**

3.2 Shift Change

Shift change notification and transition and transfer of responsibilities should be conducted in accordance with the Dose Assessment Shift Change and Termination Checklist (Appendix B).

3.3 Termination

Termination of an event should include the following actions and follow the Dose Assessment Shift Change and Event Termination Checklist (Appendix B).

4.0 DOSE ASSESSOR INTERFACES

4.1 Radiological Assessment Manager / Coordinator (RAM/RAC)

The Dose Assessor should interface directly with the Radiological Assessment Coordinator (RAC). In the absence of the RAC, communication is provided directly to the RAM. Requests for any special-case assessments should come to Dose Assessment through the RAC/RAM or be cleared by the RAC/RAM prior to their performance.

Dose Assessment is responsible for performing the offsite dose assessment activities of the CECC in order to determine Protective Action Recommendations using the appropriate appendix in CECC EPIP-1. Dose assessment results are also evaluated against criteria for declaration of Emergency Classification levels, and evaluations are communicated to the RAM/RAC.

Dose Assessment should provide results of all dose assessments and plume plots from FRED to the RAC/RAM, who will approve them and distribute them to CECC staffs. Initial dose assessments (those made at the start of an event or when the conditions have changed significantly as defined in this procedure) will receive the approval of the RAC/RAM and then are transmitted to the TSC and the State. Under most other conditions, the results are directly transmitted to the TSC and the State on the State Update form via computer spooling. However, if the computer spooling is unavailable, then the Dose Assessor shall prepare a State Update form manually as defined in EPIP-1. CECC Clerical staff have instructions for distribution.

Dose Assessment should provide to the RAC/RAM copies of plume plots from RED for ongoing releases or plots of estimated centerline location (if there is not a known release but potential exists for one to occur). This information should be transmitted by the RAC/RAM to the CECC, TSC, and State. Dose assessment will also support post event recovery efforts.

4.2 Meteorologist (MET)

The CECC Meteorologist is responsible for providing to Dose Assessment the real time and forecast meteorological data and associated advice on atmospheric dispersion and transport. If a meteorologist is not initially available for response to the CECC, support can be obtained from Muscle Shoals. Telephone and pager numbers for the Muscle Shoals response personnel are available in the REND.

Meteorological data is provided to the CECC by computer inputs and by the CECC meteorologist. In the event of a monitored airborne release, the 15-minute meteorological data is automatically accessed by the RED and FRED codes. This data should be verified against the distributed meteorological data or by the meteorologist. The meteorologist is also available to convert flow rates to exit velocities for use in the codes. The meteorologists will also provide forecast information for use in the FRED code.

4.3 Environs Assessor/Field Coordinator

Dose Assessors provide plume plots to the CECC Environs Assessor and to the Field Coordinator at the Radiological Monitoring Control Center (RMCC) via the RAC. These plume plots are used to assist with decisions on field team deployments. Real time plume plots from the RED code are to be distributed to the EA/FC and the State for that purpose.

Field data is also shared to assist with comparison of dose projections with field measurements. This comparison can assist with evaluations if field teams are at maximum centerline locations, or if reported plant release rates coincide with actual field measurements.

In the event of an unmonitored release from a site, field team data can be used in the BRED code to assist with determination of a release rate.

4.4 Core Damage Assessors

The CECC Core Damage group (in Plant Assessment) is responsible for supplying Dose Assessment with projections of potential, anticipated, and/or worst-case release rates and pathways.

4.5 Technical Support Center (TSC)

The TSC is a source of information for radioactivity release rates, pathways, flow rates, and information on plant status and prognosis. The primary point of contact is TSC Chemistry. Release information is also available via the Integrated Computer System (ISC) using the CECC computers.

4.6 River Operations

River Operations may assist in providing Dose Assessment information on water dispersion characteristics for releases to the river. This information may be used in running the WATERDOSE code, or for use of the manual methodology if the dose code is unavailable.

4.7 State (Radiological)

Dose Assessment shall ensure that communication with the State Dose Assessment Team is established and maintained. The State should be given hourly updates, as a minimum. These updates should include discussions of all technical information relative to dose assessments being made (incoming release rates, assumptions used, problems with information flow). The State should also be contacted if the conditions have changed significantly as defined in this procedure. DO NOT discuss protective action recommendations with the State.

5.0 PERFORMING DOSE ASSESSMENTS

5.1 Data Verification

All dose assessment results (computer generated or hand calculated) involving data input will be verified by a second party verifier. The verifier may be a Dose Assessor or the RAM/RAC. The verifier will verify the accuracy and appropriateness of data input and reasonableness of the results. Both preparer and verifier will initial and date the results page of the assessment (e.g., State Update Form for FRED assessments).

5.2 Preliminary Assessments

Dose Assessors should provide results of all preliminary assessments to the RAC/RAM. Preliminary Assessments are provided as part of a FRED run. Preliminary assessments will be performed at the start of an event or when the conditions have changed significantly as defined in this procedure.

5.3 Criteria for a Significant Change in Conditions

Criteria for a significant change which will require a new dose assessment run are:

- the release type / path has changed,
- the release rates have changed by a factor of 10
- the stability class has changed by 2 classes,
- or the wind speed has changed by a factor of 2.

5.4 FRED or RED Assessments - Collection of Data

Gather information as provided on Appendix C. Sources of information may include the Technical Support Center (Chemistry), ICS, CECC Meteorologist or CECC Core Damage Assessors. Refer to Appendix C for instructions on running the dose codes.

5.5 Preparing a Protective Action Recommendation (PAR)

TVA must satisfy regulatory requirements to provide State Authorities a PAR within 15 minutes of the declaration of a General Emergency. Therefore, Dose Assessors should anticipate and initiate development of a PAR to allow ample time for review, approval and transmittal to State Authorities.

A Protective Action Recommendation for airborne releases is determined based upon results of a FRED run. If the FRED program is unavailable, then the manual methodology should be utilized as provided in this procedure. A PAR form contained in CECC EPIP-1 should be completed, with attention to identification of affected sectors as page 2 of that document..

Dose Assessment should provide technical guidance to the RAC/RAM in the preparation of protective action recommendations based on dose assessments. The RAC is responsible for written preparation of recommendations to the RAM.

5.6 Changes in Conditions for a PAR

Changes to a PAR must be communicated to the State by the CECC Director within 15 minutes of determination. Criteria for a changes which will require evaluation a new PAR are:

- the release type / path has changed,
- the release rates have changed by a factor of 10
- the stability class has changed by 2 classes,
- or the wind speed has changed by a factor of 2.
- a wind direction change resulting in a change of an affected sector

5.7 BRED Assessment - Back Calculation of Release Rate from Measured Field Data

Measured field data (consisting of dose rates in mrem/hr and I-131/H-3 concentrations) are assessed in several ways. If there is a monitored release ongoing, the field data are compared to the results of the most applicable data produced by the RED or FRED computer models.

However, in cases where the release is unknown or questionable, the field data are then input into the BRED computer model to determine the applicable release rates. These calculated release rates are then input into the RED/FRED codes, as applicable, which can be used to perform dose assessments and any applicable Protective Action Recommendation (PAR).

5.8 Comparison of Measured Field Data to Dose Projections

Field data is compared with dose projections to assist with evaluations if field teams are at maximum centerline locations, or if reported plant release rates coincide with actual field measurements. Appendix G is provided as a reference to perform comparisons.

5.9 WATERDOSE Assessments

Liquid releases to the River are assessed using the WATER DOSE code as provided on Appendix H. If the WATER DOSE code is unavailable, a manual methodology is provided as Appendix I.

5.10 Manual Methodologies for Dose Assessments

In the event that the FRED, RED or WATERDOSE computer codes are unavailable, instructions are provided in the Appendixes of this procedure for manual calculation methods. In consideration that the computer programs also normally spool data outputs directly to the State, the Dose Assessor will need to ensure that the applicable pages of the State Update Form, contained in CECC EPIP-1, are also manually completed and transmitted accordingly.

6.0 REFERENCES

FRED User's Manual
RED/FRED/BRED Documentation
FRED User's Manual
WATERDOSE User's Manual
BRED User's Manual
Model Comparison
REP CODE Revision 2, Specifications and Documentation, August 2002, L61 020814 800

7.0 ABBREVIATIONS AND DEFINITIONS

CECC - Central Emergency Control Center
CTM - Containment building
SGTR (above) - Steam Generator Tube Rupture above the steam generator water level
SGTR (below) - Steam Generator Tube Rupture below the steam generator water level
MSLB - Main Steam Line Break
TSC - Technical Support Center
EPS - Emergency Paging System
RED - Radiological Emergency Dose Code
RO - River Operations
FRED - Forecast Radiological Emergency Dose Code
BRED - Back-calculation Radiological Emergency Dose Code
TRM - Tennessee River Mile
ICS - Integrated Computer System
WGDT - Waste Gas Decay Tank (as in rupture event)
RAM/RAC -Radiological Assessment Manager or Radiological Assessment Coordinator

APPENDIX A

Dose Assessor Initial Reporting Checklist

(steps do not need to be performed in sequential order)

1. **SIGN IN** on the CECC staffing board and don your CECC position tag.

2. **START** logkeeping of key activities and notifications in the position logbook.

3. **ENSURE** that the following support staffs are notified and/or staffed. Refer to the REND call out list for contact information.
   - Second Dose Assessor, if needed.
   - Muscle Shoals Meteorologist (if serving as CECC pager duty person).

4. **CONFIRM** position notebook procedures match revision levels in controlled copies.

5. **ESTABLISH** contact with the TSC Chemistry (programmed on phone and in REND section B). Ascertain if a release has been, or is occurring. **IF YES, INITIATE** a dose assessment as noted below.

6. Perform preliminary assessments and dose projections.

7. **ESTABLISH** initial contact with the State Radiological Dose Assessment staff (programmed on phone and in REND section B).

8. **OBTAIN** a briefing from the RAC/RAM and **INFORM** the RAC/RAM when the activities above are completed. Report/request if a radiological release has been, or is occurring.

NOTES: **COMPARE** dose assessment results against the levels for the declared REP class and advise the RAC/RAM to advise the TSC if an upgrade is indicated.

For Preliminary Assessments and Dose Projections use the **FRED** Code (Appendix C and D).

For Plume Plots to track actual releases in current time, use the **RED** Code (Appendix C and D).

When the plant release rate is unmonitored or questionable, use the **BRED** code to arrive at a plant release rate based upon Field Team data. (Appendix E).

For releases to the River, use the **WATERDOSE** Code (Appendix H)

If computer problems are encountered, Immediately contact Computer Support.

If the FRED computer code is inoperative, use the **MANUAL METHODOLOGY** to assess airborne radioactivity releases (Appendix F).

If the WATERDOSE computer code is inoperative, use the **MANUAL METHODOLOGY** to assess liquid releases to the river (Appendix I and J).

APPENDIX B

Dose Assessor Shift Change and Termination Checklist

1. The following should be discussed between staff for Shift Turnover.

- Current release data and projections.
- Current met data and projections.
- Current plant status and projections.
- Current environs data and projections.
- Pertinent historical data/plant conditions
- Status of any Protective Action Recommendations made and the rationale for these
- Status of any (incoming or outgoing) unfulfilled requests for information.
- Dose methodologies being used.
- Identification of problems in response capability.
- Identification of contacts at the TSC, State, Core Damage staff and RO
- Time for next periodic update to the State
- Time for next periodic update of the RED plume plot
- Identify individuals external to CECC who were activated or placed on standby

2. Transfer of Shift Change Responsibility

- Obtain approval from the RAC for the transfer of responsibility
- The on duty Dose Assessment Staff should remain available or at least respond in case transfer problems are identified

3. Termination

- Log off CECC computer system/turn off plotters.
- Notify all on-call staff of event termination, such as:
  - Meteorologist (if staffed in Muscle Shoals)
  - Additional Dose Assessment staff on standby
  - River Operations
- Collect and turn in all records to the EP staff

Appendix C Page 1 of 2

FRED / RED Data Inputs

FRED
RED
Data Inputs

NOTE: The source for this information may be the site Technical Support Center or from ICS.

1. Plant: ☐ BFN ☐ SQN ☐ WBN
2. Meteorological data will be: ☐ ACTUAL or ☐ EXERCISE (confirm with drill controller).
3. Release start time: ______________ ☐ Eastern ☐ Central
4. Elapsed Time from reactor shutdown to start of release: __________ (hours) (enter 0 if Rx under power)
5. Release Vent Type (this is used by the code to calculate effective plume height):

   SQN/WBN
   ☐ Shield Bldg
   ☐ Near ground

   BFN
   ☐ Stack
   ☐ Radwaste Zone (of Rx Bldg)
   ☐ Refueling Bldg zone (of Rx Bldg)
   ☐ Reactor Bldg zone (of Rx Bldg)
   ☐ Turbine Bldg zone (of Rx Bldg)
   ☐ Near ground

6. Effluent flow rate (exit speed) (if measured and available): ______________ cfm.

NOTE: Consult with the meteorologist as to whether the default Exit Velocity based on this flow rate should be over-ridden. Code defaults can be used for conservatism or if flow data is unavailable.

7. Release Type:

   ☐RCS ☐Core Damage ☐User Specified
   ☐Gap *(default)* ☐Fuel Melt (for noble gas and Tritium only)

NOTE: Initially, a GAP Release Type should be used unless otherwise specified by the Core Damage Assessment team. Alternately, particulate-to-I^{131} field team air concentration data can be used as follows:

$$\frac{\text{Field Team Data Particulate microCi/cc}}{\text{Field Team Data Iodine}^{131}\text{ microCi/cc}} = \text{Ratio}$$

Ratio = Release Type:	*Gap*	*Core Damage*	*Fuel Melt*
	≥ 0.18	≥ 2.0	≥ 3.5

FRED / RED Data Inputs

FRED
RED
Data Inputs

8. **Release Path:**

SQN/WBN
☐ Filtered via containment (CTM)
☐ Unfiltered via containment (CTM)
☐ SGTR with rupture located BELOW water level
☐ Steam Generator Tube Rupture with rupture located ABOVE water level
☐ Turbine Bldg
☐ Reactor Bldg
☐ Auxiliary Bldg.

BFN
☐ Stack (filtered)
☐ Stack (unfiltered)
☐ Turbine Bldg, Reactor Bldg
☐ Main Steam Line Break (MSLB)

9. **Release rates:**

Basis for rates: ☐ Monitor reading ☐ Plant personnel ☐ BRED estimate

________________ μCi/s Noble Gas

________________ μCi/s I-131 (pre-treatment value only, if available)

________________ μCi/s Total Particulate (pre-treatment value only, if available)

________________ μCi/s H-3 (if applicable see note below)

NOTE:
- For a TPBAR handling accident, the H-3 release can be estimated as:

  H-3 Release Rate ____________ = $\frac{\text{\_\_\_\_\_\_\_\_ μCi/cc H-3} \times \text{\_\_\_\_\_\_\_\_ cfm} \times \text{28320 cc/cf}}{\text{60 min/sec}}$

  { H-3 release (μCi/s) = #μCi/cc H-3 x building exhaust flow rate (cfm) x 28320 cc/cf x 1/60 min/s }

- For a WGDT Rupture accident, the default H-3 release is 2500 Ci over one hr or 6.94E+05 μCi/s for 1 hr

Appendix D Page 1 of 2

FRED RED Code Runs

FRED or RED Assessment of Airborne Releases

1. **DOUBLE CLICK** on the "CECC VAX" icon if the VAX User Window is not displayed on computer screen. Depress [RETURN] until prompted for the user name.

2. **ENTER** user name and password: ☐ RED and CECC or ☐ FRED and CECC

3. **FOLLOW** computer prompts to begin or exit program.

NOTE: TYPE CTRL Z any time to exit or re-start program.

When executing the RED code you will be asked whether this is a "new run." ANSWER "Y" and ENTER "NEW RUN," unless you desire to modify or append to a current run.

4. **INPUT** data as collected on Appendix C.

For a user-specified release (for noble gas and/or tritium releases only), **ENTER** the nuclide number below (as applicable) and the associated nuclide-specific release rates.

Nuclide #	Nuclide	Nuclide #	Nuclide
1	H-3	28	XE-131M
6	KR-85	29	XE-133
7	KR-85M	30	XE-133M
8	KR-87	31	XE-135
9	KR-88	32	XE-138

5. **CONFIRM** whether the release rate data is correct, (Y/N). Edit as necessary.

6 **CONFIRM** whether the calculated release rate data is correct, (Y/N). Edit as necessary.

7. **RUN** the code for the expected event duration:

- For FRED Preliminary Assessments use 1 hour;
- For FRED Dose Projections use a 4-hour duration unless known otherwise.
- For RED assessments run once per 15-min during ongoing releases.

FRED or RED Assessment of Airborne Releases

FRED
RED
Code Runs

8. **OBTAIN** code outputs by as follows:

   a. **ANSWER** "Y" to the prompt to "Print dose charts or plume plots."
   b. **SELECT** State Update Form (SUF) and plume plot as minimum outputs
   c. **SELECT** scale to be used:
      [1] for 10 mile, [2] for 50 mile, [3] to exit code or go to next time segment)
   d. For plume plot, **CLICK** print button at bottom of screen to perform a screen print of plot. Be sure that the pop-up dialog box has the Graphic Image set to "Swap Black/White."
   e. For Preliminary Assessments, **OBTAIN** the Protective Action Guide (PAG) release rates from the FRED output and the actual/projected release rates from the State Update Form.
   f. The Preparer and Verifier shall **INITIAL and DATE** the results.

9. **COMPARE** the declared REP class with that indicated in the FRED output. Notify the RAC/RAM (to advise the TSC) of the need for REP class changes based on radiological conditions.

10. **GIVE** PAG and actual/projected release rates to the Board Writer.

11. **GIVE** the FRED results (SUF, PAG Release Rates, plume plot, and REP class information) to the RAC for distribution. (The SUF may be sent directly through the computer to the State and the TSC.)

12. At the request of the RAC/RAM, **PREPARE** a PAR using the CECC Protective Action Logic Diagram and the PAR form found in EPIP-7 and give to the RAM with the results of the FRED run.

13. **REQUEST** that the RAC distribute the SUF, and any plume plots to all standard distribution locations, via CECC Clerical instructions.

14. Preferably once every 15-min (at least once per hour) during an actual release,

    a. **ENTER** the release data into the RED code for use in tracking the plume
    b. **COMPARE** the estimated impacts to measured field data.
    c. **GIVE** the results (plume plot only) to the RAC for distribution to the CECC, the State, and the TSC.

15. **TYPE** CTRL Z any time to exit or re-start program.

Appendix E Page 1 of 2
BRED Evaluation of Airborne Field Data

BRED Code Run

1. Log on to BRED. DOUBLE CLICK on the "CECC VAX" icon. PRESS return until prompted for username. ENTER username (BRED) and password (CECC).

2. OBTAIN the following field data from Environs Assessment.

   NOTE: As a minimum, only need one of the following measurements:
   Dose Rate OR Iodine-131 OR Tritium (H-3)

Distance (miles)	Direction (sector)	Time Taken	Dose Rate mrem/hr (1 meter w/c)

Iodine-131 µCi/cc	^{3}H Concentration µCi/cc

3. **Elapsed Time from reactor shutdown to time of field measurement:** __________ (hours) (enter 0 if Rx under power)

4. **DETERMINE the Release Path:**

   SQN/WBN
   - ☐ Filtered via containment (CTM)
   - ☐ Unfiltered via containment (CTM)
   - ☐ SGTR with rupture located BELOW water level
   - ☐ Steam Generator Tube Rupture with rupture located ABOVE water level
   - ☐ Turbine Bldg
   - ☐ Reactor Bldg
   - ☐ Auxiliary Bldg.

   BFN
   - ☐ Stack (filtered)
   - ☐ Stack (unfiltered)
   - ☐ Turbine Bldg, Reactor Bldg
   - ☐ Main Steam Line Break (MSLB)

5. **DETERMINE the Release Type:** ☐RCS ☐Gap ***(default)*** ☐Core Damage ☐Fuel Melt

NOTE: Initially, a GAP Release Type should be used unless otherwise specified by the Core Damage Assessment team. Alternately, particulate-to-I^{131} field team air concentration data can be used as follows:

$$\frac{\text{Field Team Data Particulate microCi/cc}}{\text{Field Team Data Iodine}^{131}\text{ microCi/cc}} = \text{Ratio}$$

	Gap	*Core Damage*	*Fuel Melt*
Ratio = Release Type:	≥ 0.18	≥ 2.0	≥ 3.5

Appendix E Page 2 of 2

BRED Code Run

BRED Evaluation of Airborne Field Data

6. **RUN** the BRED computer model and follow prompts using the information from sections 2-5. **TYPE** CTRL Z any time to exit or re-start program.

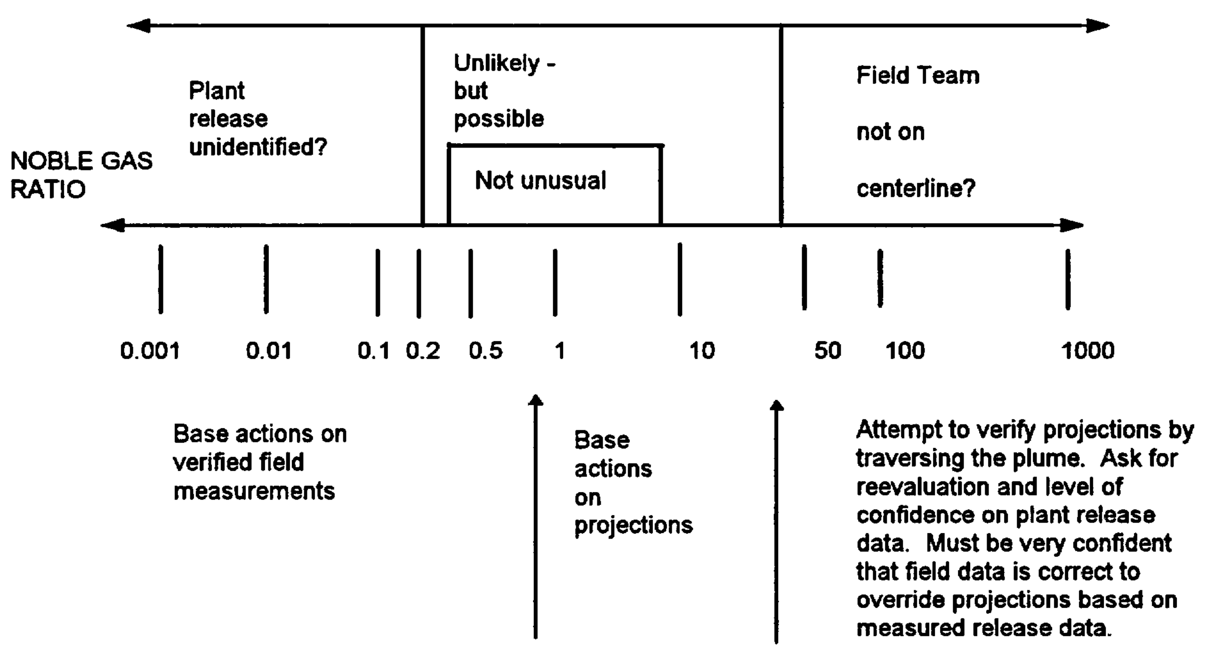
7. **RECORD** the printed Release Rate output (as applicable) and **INPUT into the FRED code.**

Noble Gas (µCi/s)	Iodine 131 (pre-treatment) µCi/s	Tritium (^{3}H) µCi/s

8. **COMPARE** the new FRED run dose rate output to the previous RED/FRED computer model by **CALCULATING** a data ratio as follows:

FRED or RED Centerline Dose Rate ***divided by*** the FIELD DATA Centerline Dose Rate

______________________ divided by __________________ = _________ RATIO



9. **PROVIDE** feedback to the Environs Assessor and RAC/RAM. **UPDATE** dose projections as necessary and give the results to the RAC for use in preparing Protective Action Recommendations (PAR), or prepare a PAR in accordance with CECC EPIP-1.

Manual Method for Assessing Airborne Releases

MANUAL AIR SB TEDE PAG

A. Calculating TEDE PAG Release Rate at SITE BOUNDARY (0.62 miles)

1. **DETERMINE:** Plant: ☐ BFN ☐ SQN ☐ WBN

Wind Speed: ____________ (m/s)

Stability Class: (circle) A B C D E F G

Release Type: ☐RCS ☐Gap *(default)* ☐Core Damage ☐Fuel Melt ☐User Specified (for noble gas and Tritium only)

Release Path:

SQN/WBN
- ☐ Filtered via CTM
- ☐ Unfiltered via CTM
- ☐ SGTR with rupture BELOW water
- ☐ SGTR with rupture ABOVE water
- ☐ Turbine, Reactor, Auxiliary Building

BFN
- ☐ filtered via stack
- ☐ unfiltered via stack
- ☐ Turbine, Reactor Bldg
- ☐ Main Steam Line Break

2. **CIRCLE** the **TEDE PAG FACTOR** (µCi/m) below, based on the stability class and release level.

NOTE: USE ground level for all cases except for BFN stack.

	A	B	C	D	E	F	G
Ground	1.7E+09	4.8E+08	2.2E+08	1.1E+08	7.4E+07	4.9E+07	2.9E+07
Stack	1.8E+09	9.1E+08	9.1E+08	8.3E+08	8.3E+08	8.0E+08	9.1E+08

3. **CIRCLE** the appropriate **TEDE Ratio** below, based on release type/path:

TEDE Ratio (for 0.62 mi)	BWR RCS	PWR RCS	Gap	Core Damage	Fuel Melt	User Spec
Stack (unfiltered	2.0	N/A	1.8	1.3	2.0	1.0
Stack (filtered)	1.9	N/A	1.0	1.0	1.0	1.0
CTM (unfiltered) or SGTR (below)	N/A	7.4	9.0	5.3	11	1.0
CTM (filtered)	N/A	3.7	1.0	0.9	1.0	1.0
SGTR (above water)	N/A	95	221	111	263	1.0
MSLB (BFN)	7.4	N/A	84	44	100	1.0
Turbine, Reactor or Aux Bldg	4.4	17	32	16	37	1.0

4. **CALCULATE** the **TEDE PAG** Release Rate (0.62 mi) as follows:

____________ X ____________ / ____________ = ____________

TEDE PAG FACTOR (µCi/m, item 2)	wind speed (m/s, item 1)	TEDE Ratio (item 3)	TEDE NGPAG Release Rate SB 0.62 mi (µCi/s)

Manual Method for Assessing Airborne Releases

MANUAL AIR
SB TEDE PAG

5. **OBTAIN** the actual/projected Noble Gas Release Rate ______________ µCi/s.

6. **IF** noble gas release rate (item 5) ≥ TEDE PAG Release Rate (item 4),
THEN radiological conditions indicate a **General Emergency.**

For Tritium Accidents (e.g., TPBAR handling or WGDT rupture),

7. **CIRCLE** the **Tritium PAG FACTOR** (µCi/m) below, based on the stability class.

A	B	C	D	E	F	G
4.0E+09	8.7E+08	2.9E+08	1.0E+08	5.9E+07	3.1E+07	1.4E+07

8. **CALCULATE** the **Tritium PAG Release Rate** as follows:

______________ X ______________ = ______________

Tritium PAG FACTOR (µCi/m, item 7) | wind speed (m/s item 1) | TEDE Tritium PAG Release Rate SB 0.62 MI (µCi/s)

9. **OBTAIN** the actual/projected Tritium Release Rate (see below) ______________ µCi/s.

NOTE:
- For a TPBAR handling accident, the H-3 release can be estimated as:

H-3 Release Rate ______________ = $\frac{\text{\_\_\_\_\_\_ µCi/cc H-3} \times \text{\_\_\_\_\_\_ cfm} \times 28320 \text{ cc/cf}}{60 \text{ min/sec}}$

{ H-3 release (µCi/s) = #µCi/cc H-3 x building exhaust flow rate (cfm) x 28320 cc/cf x 1/60 min/s }

- For a WGDT Rupture accident, the default H-3 release is 2500 Ci over one hr or 6.94E+05 µCi/s for 1 hr

10. **IF** tritium release rate (item 9) ≥ TEDE PAG Release Rate (item 8),
THEN radiological conditions indicate a **General Emergency.**

11. **IF tritium accident also involves noble gases, THEN perform the following calculation:**

$$\frac{\text{NG Release Rate}}{\text{TEDE NG PAG Release Rate}} + \frac{\text{Tritium Release Rate}}{\text{TEDE Tritium PAG Release Rate}}$$

$$\frac{\text{\_\_\_\_\_\_ (item 5)}}{\text{\_\_\_\_\_\_ (item 4)}} + \frac{\text{\_\_\_\_\_\_ (item 9)}}{\text{\_\_\_\_\_\_ (item 8)}} = \underset{\text{Ratio}}{\text{\_\_\_\_\_\_}}$$

12. **IF the value in item 11** ≥ **1.0, THEN** radiological conditions indicate a **General Emergency.**

B. Calculating THYROID CDE PAG Release Rate at SITE BOUNDARY (0.62 MILES)

1. **USE** the data from section A.1

2. **CIRCLE** the **CDE PAG FACTOR** (μCi/m), based on the stability class and release level.

NOTE: USE ground level for all cases except for BFN stack.

	A	B	C	D	E	F	G
Ground	4.1E+05	8.6E+04	3.0E+04	1.0E+04	6.0E+03	3.1E+03	1.4E+03
Stack	4.1E+05	2.3E+05	3.7E+05	6.5E+05	1.4E+06	2.1E+07	4.9E+11

3. **CALCULATE** the CDE PAG Release Rate as follows:

_______________ X _______________ = _______________

CDE PAG FACTOR (μCi/m, item 2) X wind speed (m/s) = CDE PAG Release Rate SB 0.62 mi (μCi/s)

4. a. If known, **RECORD** the actual/projected I-131 release rate __________ μCi/s and go to Step 5.

If unknown, **CIRCLE** the I-131 to NG ratio below, based on release type and path and continue with step 4 b.

I-131 to Noble Gas Ratio

	RCS	Gap	Core Damage	Fuel Melt
CTM filtered	1.7E-06	3.0E-05	1.2E-05	2.2E-05
Stack filtered	4.6E-07	3.0E-05	1.2E-05	2.2E-05
CTM (unfiltered) or SGTR (below)	1.7E-04	3.0E-03	1.2E-03	2.2E-03
Stack (unfiltered)	4.6E-05	3.0E-03	1.2E-03	2.2E-03
TB, AuxB, RxB	5.8E-04	1.0E-02	4.1E-03	7.7E-03
SGTR (above)	4.2E-03	8.0E-02	3.0E-02	5.5E-02
MSLB (BFN)	4.6E-04	3.0E-02	1.2E-02	2.2E-02

4.b. **CALCULATE** actual/projected iodine-131 release rate as follows:

_______________ X _______________ = _______________

Actual/Projected NG release rate (item A.5) X I-131 to NG ratio (item 4a) = Actual/proj. I-131 release rate (μCi/s)

5. **IF** I-131 release rate (item 4a or b) ≥ CDE PAG Release Rate (item 3), **THEN** radiological conditions indicate a **General Emergency**.

Appendix F Page 4 of 8
Manual Method for Assessing Airborne Releases

MANUAL AIR SB TEDE RATE

C. Calculating TEDE Dose Rate at SITE BOUNDARY (0.62 miles)

1. **DETERMINE:** Plant: ☐ BFN ☐ SQN ☐ WBN

Wind Speed: ____________ (m/s)

Stability Class: (circle) A B C D E F G

Release Type: ☐RCS ☐Core Damage ☐User Specified (for noble gas and Tritum only)
☐Gap *(default)* ☐Fuel Melt

Release Path: SQN/WBN BFN

☐ Filtered via CTM ☐ filtered via stack
☐ Unfiltered via CTM ☐ unfiltered via stack
☐ SGTR with rupture BELOW water ☐ Turbine, Reactor Bldg
☐ SGTR with rupture ABOVE water ☐ Main Steam Line Break
☐ Turbine, Reactor, Auxiliary Building

Noble Gas Release Rate : ______________ (µCi/s)

2. **CIRCLE** the **TEDE FACTOR** (rem/h per µCi/m) below, based on the stability class and release level.

NOTE: USE ground level for all cases except for BFN stack.

	A	B	C	D	E	F	G
Ground	6.0E-10	2.1E-09	4.6E-09	9.5E-09	1.4E-08	2.1E-08	3.5E-08
Stack	5.5E-10	1.1E-09	1.1E-09	1.2E-09	1.2E-09	1.3E-09	1.1E-09

3. **CIRCLE** the appropriate TEDE Ratio below, based on release type/path:

TEDE Ratio (at 0.62 mi)

	BWR RCS	PWR RCS	GAP	Core Damage	Fuel Melt	User Spec
Stack (unfiltered)	2.0	N/A	1.8	1.3	2.0	1.0
Stack (filtered)	1.9	N/A	1.0	1.0	1.0	1.0
CTM (unfiltered) or SGTR (below)	N/A	7.4	9	5.3	11	1.0
CTM (filtered)	N/A	3.7	1.0	0.9	1.0	1.0
SGTR (above water)	N/A	95	221	111	263	1.0
MSLB (BFN)	7.4	N/A	84	44	100	1.0
TB, RxB, AB	4.4	17	32	16	37	1.0

4. **CALCULATE** the **TEDE Dose** as follows:

______________ X	____________ X	____________ /	____________ =	____________
NG release rate (µCi/s) (item A-5)	TEDE FACTOR (item 2)	TEDE Ratio (item 3)	wind sp. (m/s) (item 1)	**TEDE (rem/h) 0.62 mile**

* For Tritium Accidents (e.g., TPBAR handling or WGDT rupture),

5. **CIRCLE** the **Tritium TEDE FACTOR** (rem/h per µCi/m) below, based on the stability class.

A	B	C	D	E	F	G
2.5E-10	1.2E-09	3.5E-09	1.0E-08	1.7E-08	3.3E-08	7.0E-08

*Revision

Appendix F Page 5 of 8
Manual Method for Assessing Airborne Releases

MANUAL AIR
5 mi TEDE RATE
5 mi THY CDE RATE

6. **CALCULATE** the **Tritium TEDE** as follows:

__________ X __________ / ________ = __________

Tritium Release Rate* (µCi/s) | Tritium TEDE FACTOR (item 5) | wind speed (m/s) (item 1) | Tritium TEDE (rem/h)

*NOTE:
- For a TPBAR handling accident, the H-3 release can be estimated as:

H-3 Release Rate ________ = ________ µCi/cc H-3 x ________ cfm x 28320 cc/cf / 60 min/sec

{ H-3 release (µCi/s) = #µCi/cc H-3 x building exhaust flow rate (cfm) x 28320 cc/cf x 1/60 min/s }

- For a WGDT Rupture accident, the default H-3 release is 2500 Ci over one hr or 6.94E+05 µCi/s for 1 hr

7. **IF tritium accident also involves noble gases, THEN CALCULATE Total TEDE rate as follows:**

________ + ________ = ________

TEDE (rem/h) | Tritium TEDE (rem/h) | **Total TEDE (rem/h) 0.62 mile**

D. Calculating SB THYROID CDE Dose Rate

1. **CIRCLE** the Thyroid **CDE FACTOR** (rem/h per µCi/m), based on the stability class and release level.

NOTE: USE ground level for all cases except for BFN stack.

	A	B	C	D	E	F	G
Ground	1.2E-05	5.8E-05	1.7E-04	4.8E-04	8.3E-04	1.6E-03	3.5E-03
Stack	1.2E-05	2.2E-05	1.4E-05	7.7E-06	3.5E-06	2.3E-07	1.0E-11

2 a. If known, **RECORD** the I-131 release rate ________ µCi/s and **go to Step 3.**

If unknown, **CIRCLE** the I-131 to NG ratio below, based on release type and path and continue with step 2b.

I-131 to Noble Gas Ratio

	RCS	Gap	Core Damage	Fuel Melt
CTM filtered	1.7E-06	3.0E-05	1.2E-05	2.2E-05
Stack filtered	4.6E-07	3.0E-05	1.2E-05	2.2E-05
CTM (unfiltered) or SGTR (below)	1.7E-04	3.0E-03	1.2E-03	2.2E-03
Stack (unfiltered)	4.6E-05	3.0E-03	1.2E-03	2.2E-03
TB, AuxB, RxB	5.8E-04	1.0E-02	4.1E-03	7.7E-03
SGTR (above)	4.2E-03	8.0E-02	3.0E-02	5.5E-02
MSLB (BFN)	4.6E-04	3.0E-02	1.2E-02	2.2E-02

2b. **CALCULATE** actual/projected Iodine-131 release rate as follows:

________ X ________ = ________

NG release rate (item A.5) | I-131 to NG ratio (item 2a) | I-131 release rate (µCi/s)

*

*Revision

Manual Method for Assessing Airborne Releases

3. **CALCULATE** Thyroid CDE Dose Rate as follows:

_______________ X _______________ / ________ = _______________

I-131 release rate (µCi/s) (item 2) X Thyroid CDE FACTOR (item 1) / wind speed (m/s) = **Thyroid CDE (rem/h) 0.62 mile**

E. Calculating 5 mile TEDE

USE the data from section A.1, **THEN**

1. **OBTAIN** an estimate of the release duration (t) ________ hours. Use 4 (four) hours unless known otherwise.

2. **CIRCLE** the **TEDE FACTOR** (rem/h per per µCi/m) below, based on the stability class and release level.

NOTE: USE ground level for all cases except for BFN stack.

	A	B	C	D	E	F	G
Ground	9.5E-11	1.5E-10	2.8E-10	9.5E-10	1.8E-09	3.5E-09	6.5E-09
Stack	9.0E-11	1.5E-10	2.6E-10	7.5E-10	1.1E-09	1.3E-09	1.1E-09

3. **CIRCLE** the appropriate TEDE Ratio below, based on release type/path:

TEDE Ratio (at 5 mi)

	BWR RCS	PWR RCS	GAP	Core Damage	Fuel Melt	User Spec
Stack (unfiltered)	2.1	N/A	2.8	1.9	3.1	1.0
Stack (filtered)	2.1	N/A	1.0	0.9	1.0	1.0
CTM (unfiltered) or SGTR (below)	N/A	3.5	4.9	2.9	5.8	1.0
CTM (filtered)	N/A	1.8	1.0	1.0	1.0	1.0
SGTR (above water)	N/A	43	100	51	116	1.0
MSLB (BFN)	4.5	N/A	40	21	47	1.0
TB, RxB, AB	3.1	7.4	15	7.9	17	1.0

4. **CALCULATE** the **TEDE Dose** as follows:

_______________ X _____________ X _________ X _________ / ________ = _____________

NG release rate (µCi/s) (item A.5) X TEDE FACTOR (item 2) X TEDE Ratio (item 3) X Duration (hrs) (item 1) / wind sp. (m/s) (item 1) = **TEDE (rem) 5 mile**

For Tritium Accidents (e.g., TPBAR handling or WGDT rupture),

5. **CIRCLE** the **Tritium TEDE FACTOR** (rem/h per µCi/m) below, based on the stability class.

A	B	C	D	E	F	G
4.0E-11	5.0E-11	1.1E-10	4.4E-10	9.5E-10	2.4E-09	5.5E-09

6. **CALCULATE** the Tritium TEDE as follows:

______ X ______ X ______ / ______ = ______

Tritium Release Rate (µCi/s) X Tritium TEDE FACTOR (item 5) X Duration (hrs) (item A.1) / wind speed (m/s) (item A.1) = Tritium TEDE (rem)

NOTE:
- For a TPBAR handling accident, the H-3 release can be estimated as:

  H-3 Release Rate ______ = $\frac{\text{\_\_\_\_ µCi/cc H-3} \times \text{\_\_\_\_ cfm} \times 28320 \text{ cc/cf}}{60 \text{ min/sec}}$

  { H-3 release (µCi/s) = #µCi/cc H-3 x building exhaust flow rate (cfm) x 28320 cc/cf x 1/60 min/s }

- For a WGDT Rupture accident, the default H-3 release is 2500 Ci over one hr or 6.94E+05 µCi/s for 1 hr

7) **IF** tritium accident also involves noble gases, **THEN CALCULATE** Total TEDE as follows:

______ + ______ = ______

TEDE (rem) + Tritium TEDE (rem) = Total TEDE (rem) 5 mile

F. Calculating 5 mi THYROID CDE Doses

1. **CIRCLE** the Thyroid CDE FACTOR (rem/h per µCi/m), based on the stability class and release level.

   NOTE: USE ground level for all cases except for BFN stack.

	A	B	C	D	E	F	G
Ground	2.0E-06	2.5E-06	5.0E-06	2.2E-05	4.7E-05	1.2E-04	2.7E-04
Stack	2.0E-06	2.5E-06	4.5E-06	7.7E-06	3.5E-06	2.3E-07	1.0E-11

2. **OBTAIN** the I-131 release rate from B.4 ______ (µCi/s).

3. **CALCULATE** Thyroid CDE Dose as follows:

______ X ______ X ______ / ______ = ______

I-131 release rate (µCi/s) (item 2) X Thyroid CDE FACTOR (item 1) X duration (hrs) (item C.1) / wind speed (m/s) (item A.1) = Thyroid CDE (rem) 5 mile

***G. Summary of Results**

Site Boundary TEDE Rates

1. Total TEDE Rate (item C.7) ____________ rem/h

***2.** Circle REP Emergency class based on TEDE rate above:

For 0.62 mi TEDE dose rate ≥	REP Emergency Class
1E-04 rem/h	NOUE
1E-02 rem/h	ALERT
1E-01 rem/h	SAE
1E+00 rem/h	GE

Site Boundary Thyroid CDE Dose Rate

3. CDE Dose Rate (section D.3) ____________ rem/h

***4.** Circle REP Emergency class based on CDE rate above:

For 0.62 mi CDE dose rate ≥	REP Emergency Class
NA	---
NA	---
0.5 rem/h	SAE
5 rem/h	GE

5 Mile TEDE

5. TEDE without Tritium (section E.4) ____________ rem.

6. Total TEDE with Tritium (section E.7) ____________ rem.

5 Mile Thyroid CDE

7. Thyroid CDE (section F.3) ____________ rem.

Emergency Class

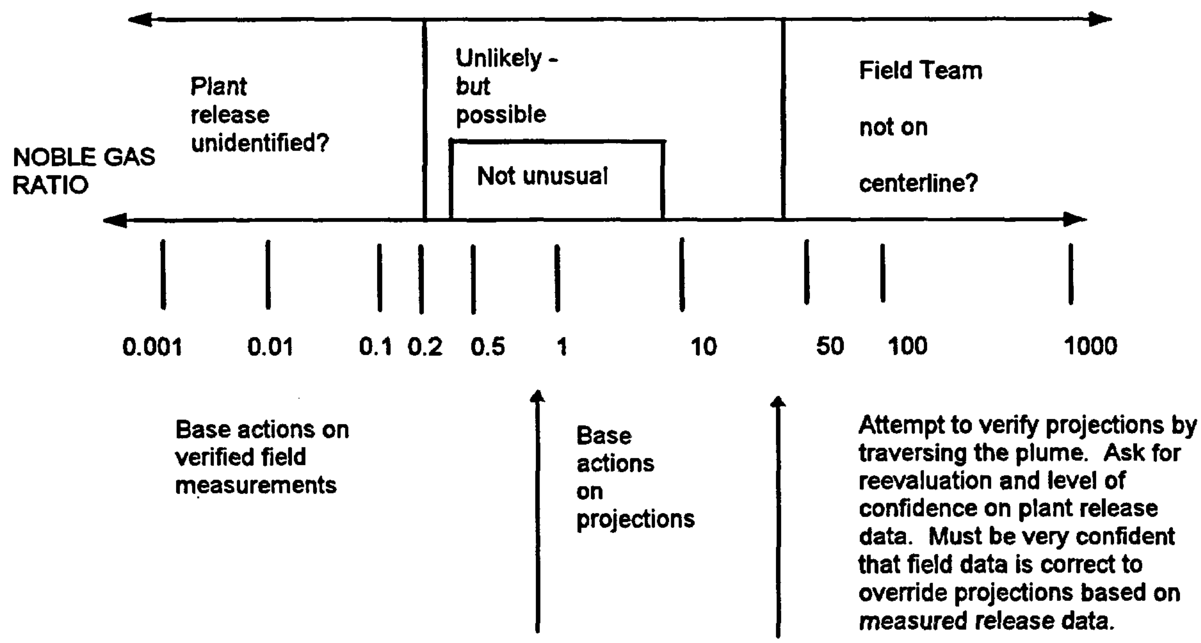
8. Circle the most restrictive REP class from items 2 and 4: NOUE Alert SAE GE

END OF MANUAL ASSESSMENT DATA VERIFICATION	Calculated by: (initial / date)	________/__________
	Verified by: (initial / date)	________/__________

*Revision

APPENDIX G
COMPARISON OF MEASURED FIELD DATA TO DOSE PROJECTION MODELS

$$\text{Ratio} = \frac{\text{PROJECTED centerline dose rate}}{\text{MEASURED centerline dose rate}}$$



WATERDOSE Code Run

APPENDIX H
WATERDOSE Evaluation of Liquid Release to the River

1. Log on to WATERDOSE. DOUBLE CLICK on the "CECC VAX" icon. PRESS return until prompted for username. ENTER username (WATERDOSE) and password (CECC).

2. OBTAIN the following information for input to WATERDOSE and FOLLOW code prompts.

NOTE: TYPE CTRL Z any time to exit or re-start program.

Plant: ☐ BFN ☐ SQN ☐ WBN

a. Determine Release Point:: ☐ Diffuser ☐ Shoreline

b. Length of release: ________________ Hours

c. Volume of release: ________________ (ft^3)

d. Release Mix (nuclides and concentrations)

Nuclide	Concentration (µCi/ml)

3. RUN the WATERDOSE code using the available (or default below) information to obtain an estimate of the dose impact. (If the computer code is not operational, the dose calculation methodology contained in Appendix I can be used.)
   BFN-33000 cfs SQN - 29000 cfs WBN - 2700 cfs

NOTE: TYPE CTRL Z any time to exit or re-start program.

4. OBTAIN the State Update Form (SUF). The Preparer and Verifier shall initial and date the results.

5. TRANSMIT (by spooling through the computer) the SUF to the TSC and State if approval to do so has been given by the RAC/RAM.

6. TYPE CTRL Z any time to exit or re-start program.

MANUAL RIVER

APPENDIX I Page 1 of 5

Manual Evaluation of Liquid Releases to the River

1. Plant: ☐ BFN ☐ SQN ☐ WBN

2. Release Point: ☐Diffuser ☐Shoreline

3. Release Time: Start ________ End __________ ☐Eastern ☐Central

4. Release Volume (V) _____ ft^3 (1 gal = 0.134 ft^3)

5. Calculation of Hazard Index (HI):

Nuclide	Concentration (µCi/ml)	Dose (rem/day per µCi/ml)	Hazard Index (rem/day)
	C	DF- Table 3	HI=C * DF
		Total Hazard Index	

6. Riverflow at the plant ____________ ft^3/s (cfs). This can be obtained from the ICS (for SQN and WBN) or River Operations. If flow data is not available use the following default values:
BFN-33000 cfs SQN - 29000 cfs WBN - 2700 cfs

7. Calculate the downstream dose rate to hypothetical individual at first downstream Public Water Supply and then other locations of interest. Refer to Appendix J.

Location TRM	(table 2) Arrival Time Hours	(table 1) Dilution Factor (1/ft^3) D	(item 5) Hazard Index (rem/day) HI	(item 4) Release Volume (ft^3) V	Dose Rate (rem/day) D*H*V

8. Record the applicable data on the State Update Form in CECC EPIP-1 and distribute.

Comments: ____

____

END OF MANUAL ASSESSMENT	Calculated by: (initial / date)	________/______
DATA VERIFICATION	Calculated by: (initial / date)	________/______

APPENDIX I **Page 2 of 5**

Manual Evaluation of Liquid Releases to the River

TABLE 1
DILUTION FACTOR
(D)

RELATIVE CONCENTRATION FACTORS-PER CUBIC FOOT RELEASED

BROWNS FERRY NUCLEAR PLANT

Tennessee River Mile (TRM)	SHORELINE RELEASE Plant Side Shoreline D	SHORELINE RELEASE Opposite Shoreline D	DIFFUSER PIPE RELEASE Centerline D	DIFFUSER PIPE RELEASE Shoreline D
294.00 Plant				
293.00	2.8E-08	.00E+00	1.4E-08	.00E+00
292.00	1.4E-08	.00E+00	6.9E-09	.00E+00
291.00	9.2E-09	.00E+00	4.6E-09	1.2E-36
290.00	6.9E-09	.00E+00	3.4E-09	8.2E-30
289.00	5.5E-09	.00E+00	2.8E-09	1.0E-25
288.00	4.6E-09	.00E+00	2.3E-09	5.1E-23
287.00	3.9E-09	.00E+00	2.0E-09	4.3E-21
286.00	3.4E-09	.00E+00	1.7E-09	1.2E-19
285.00	3.1E-09	.00E+00	1.5E-09	1.5E-18
284.00	2.8E-09	1.8E-42	1.4E-09	1.2E-17
283.00	2.5E-09	1.8E-39	1.3E-09	6.2E-17
282.00	2.3E-09	5.8E-37	1.1E-09	2.4E-16
281.00	2.1E-09	7.4E-35	1.1E-09	7.7E-16
280.00	2.0E-09	4.8E-33	9.8E-10	2.1E-15
279.00	1.8E-09	1.8E-31	9.2E-10	4.8E-15
278.00	1.7E-09	4.1E-30	8.6E-10	1.0E-14
274.90 Downstream Dam				

SEQUOYAH NUCLEAR PLANT

TRM	SHORELINE RELEASE Plant Side Shoreline D	SHORELINE RELEASE Opposite Shoreline D	DIFFUSER PIPE RELEASE Centerline D	DIFFUSER PIPE RELEASE Shoreline D
484.50 Plant				
484.00	3.5E-08	5.9E-34	1.8E-08	1.1E-14
483.00	1.4E-08	5.3E-19	7.2E-09	3.0E-11
482.00	9.1E-09	3.2E-15	4.6E-09	1.9E-10
481.00	6.6E-09	1.5E-13	3.3E-09	3.9E-10
480.00	5.2E-09	1.4E-12	2.6E-09	5.6E-10
479.00	4.3E-09	5.4E-12	2.2E-09	6.8E-10
478.00	3.7E-09	1.4E-11	1.8E-09	7.6E-10
477.00	3.2E-09	2.7E-11	1.6E-09	8.2E-10
476.00	2.8E-09	4.5E-11	1.4E-09	8.4E-10
475.00	2.5E-09	6.6E-11	1.3E-09	8.6E-10
474.00	2.3E-09	9.0E-11	1.2E-09	8.6E-10
473.00	2.1E-09	1.2E-10	1.1E-09	8.6E-10
472.00	1.9E-09	1.4E-10	1.0E-09	8.5E-10
471.00	1.8E-09	1.7E-10	9.8E-10	8.3E-10
471.00 Downstream Dam				

APPENDIX I Page 3 of 5
Manual Evaluation of Liquid Releases to the River

TABLE 1
DILUTION FACTOR
(D)

RELATIVE CONCENTRATION FACTORS-PER CUBIC FOOT RELEASED

WATTS BAR NUCLEAR PLANT

	SHORELINE RELEASE		DIFFUSER PIPE RELEASE	
TRM	Plant Side Shoreline D	Opposite Shoreline D	Centerline D	Shoreline D
528.00 Plant				
527.00	3.7E-08	1.2E-20	1.8E-08	2.3E-11
526.00	1.8E-08	1.5E-14	9.1E-09	4.1E-09
525.00	1.2E-08	1.3E-12	6.1E-09	1.0E-09
524.00	9.1E-09	1.2E-11	4.6E-09	1.4E-09
523.00	7.3E-09	4.0E-11	3.7E-09	1.7E-09
522.00	6.1E-09	9.0E-11	3.1E-09	1.8E-09
521.00	5.2E-09	1.6E-10	2.7E-09	1.8E-09
520.00	4.6E-09	2.3E-10	2.4E-09	1.8E-09
519.00	4.1E-09	3.1E-10	2.2E-09	1.8E-09
518.00	3.7E-09	3.8E-10	2.0E-09	1.8E-09
517.00	3.3E-09	4.6E-10	1.9E-09	1.7E-09
516.00	3.0E-09	5.2E-10	1.8E-09	1.7E-09
515.00	2.8E-09	5.8E-10	1.7E-09	1.6E-09
514.00	2.6E-09	6.4E-10	1.6E-09	1.6E-09
513.00	2.4E-09	6.8E-10	1.6E-09	1.5E-09
512.00	2.3E-09	7.2E-10	1.5E-09	1.5E-09
511.00	2.2E-09	7.6E-10	1.5E-09	1.4E-09
510.00	2.0E-09	7.9E-10	1.4E-09	1.4E-09
510.00	1.9E-09	8.2E-10	1.4E-09	1.4E-09
508.00	1.8E-09	8.4E-10	1.3E-09	1.3E-09
507.00	1.8E-09	8.6E-10	1.3E-09	1.3E-09
506.00	1.7E-09	8.7E-10	1.3E-09	1.3E-09
505.00	1.6E-09	8.8E-10	1.2E-09	1.2E-09
504.00	1.5E-09	8.9E-10	1.2E-09	1.2E-09
503.00	1.5E-09	9.0E-10	1.2E-09	1.2E-09
502.00	1.4E-09	9.1E-10	1.2E-09	1.2E-09
501.00	1.4E-09	9.1E-10	1.1E-09	1.1E-09
500.00	1.3E-09	9.1E-10	1.1E-09	1.1E-09
471.00 Downstream Dam				

APPENDIX I Page 4 of 5

Manual Evaluation of Liquid Releases to the River

TABLE 2 ARRIVAL TIME (HRS)

APPROXIMATE TRAVEL TIME TO MAXIMUM CONCENTRATION - HOURS

BROWNS FERRY NUCLEAR PLANT

TRM — RIVER FLOW IN CUBIC FEET/SECOND

	25000	30000	33000	35000	37000	39000
294.00 Plant						
292.00	25	21	19	18	17	16
290.00	49	41	37	35	33	31
288.00	74	62	56	53	50	47
286.00	99	82	75	71	67	63
284.00	124	103	92	88	84	80
282.00	148	124	112	106	100	94
280.00	173	144	131	124	117	110
278.00	198	165	150	141	134	128
276.00	222	185	169	159	150	142
274.90 Downstream Dam						

SEQUOYAH NUCLEAR PLANT

	21000	25000	29000	30000	33000
484.50 Plant					
483.00	5	4	4	3	3
481.00	12	10	8	8	7
479.00	18	15	13	13	12
477.00	25	21	18	17	16
475.00	32	26	23	22	20
473.00	38	32	28	27	24
471.00	45	38	32	31	28
471.00 Downstream Dam					

WATTS BAR NUCLEAR PLANT

	19000	20000	25000	30000	35000
528.00 Plant					
526.00	5	4	3	3	3
524.00	10	9	7	6	6
522.00	15	14	11	9	9
520.00	20	19	15	12	12
518.00	25	24	19	16	15
516.00	30	29	23	19	18
514.00	35	33	27	22	21
512.00	40	38	30	25	24
510.00	45	43	34	29	28
508.00	50	48	38	32	31
506.00	56	53	42	35	34
504.00	61	58	46	38	37
502.00	66	62	50	41	40
500.00	71	67	54	45	43
471.00 Downstream Dam					

TABLE 3
DOSE
FACTORS
(DF)

APPENDIX I   Page 5 of 5

Manual Evaluation of Liquid Releases to the River

Critical Ingestion Dose Rate Factors
(Derived from Regulatory Guide 1.109)
Rem/day per µCi/ml

Nuclide	Dose Factor	Organ1	Age2	Nuclide	Dose Factor	Organ1	Age2
H-3	0.28	TB	C	Ru-103	43.2	GIT	A
C-14	16.9	B	C	Ru-105	58.9	GIT	C
Na-24	8.12	TB	C	Ru-106	356	GIT	A
P-32	1155	B	C	Ag-110m	121	GIT	A
Cr-51	1.34	GIT	A	Te-125m	21.8	K	A
Mn-54	28	GIT	A	Te-127m	55	K	A
Mn-56	67.8	GIT	C	Te-127	25.8	GIT	C
Fe-55	16.1	B	C	Te-129m	96	K	C
Fe-59	68	GIT	A	Te-129	20.4	GIT	I
Co-58	30	GIT	A	Te-131	168	GIT	A
Co-60	80.4	GIT	A	Te-131	6.4	GIT	I
Ni-63	753	B	C	Te-132	154	GIT	A
Ni-65	35.8	GIT	C	I-130	1332	THY	I
Cu-64	14.2	GIT	A	I-131	12500	THY	I
Zn-65	51.1	L	C	I-132	142	THY	I
Zn-69	12.33	GIT	I	I-133	2980	THY	I
Br-83	0.24	TB	C	I-134	37.4	THY	I
Br-84	0.28	TB	C	I-135	584	THY	I
Br-85	0.013	TB	C	Cs-134	538	L	C
Rb-86	153	L	I	Cs-136	90.4	L	C
Rb-88	0.45	L	I	Cs-137	438	B	I
Rb-89	0.26	L	I	Cs-138	1.13	GIT	I
Sr-89	1850	B	C	Ba-139	50.2	GIT	I
Sr-90	23800	B	C	Ba-140	116	B	C
Sr-91	74.2	GIT	C	Ba-141	7.27	GIT	I
Sr-92	239	GIT	C	Ba-142	1.06	GIT	I
Y-90	204	GIT	A	La-140	185	GIT	A
Y-91m	2.43	GIT	I	La-142	46.3	GIT	C
Y-91	155	GIT	A	Ce-141	48.4	GIT	A
Y-92	146	GIT	C	Ce-143	91.2	GIT	A
Y-93	238	GIT	C	Ce-144	330	GIT	A
Zr-95	61.8	GIT	A	Pr-143	80.6	GIT	A
Zr-97	210	GIT	A	Pr-144	4.44	GIT	I
Nb-95	42	GIT	A	Nd-147	69.8	GIT	A
Mo-99	39.8	K	C	W-187	56.4	GIT	A
Tc-99m	1.44	GIT	C	Np-239	48	GIT	A

1. THY = thyroid, GIT = Gastrointestinal Tract, K = Kidney, L = Liver, TB = Total Body, B = Bone

2. A = Adult, C = Child, I = Infant

APPENDIX J Page 1 of 3

BFN - PUBLIC AND INDUSTRIAL SURFACE WATER SUPPLIES

County-State	Plant Name	Water Source	Type of Water Supply	Notification Advise State or Local Authorities listed in the REND
10-Mile Radius				
Limestone-Alabama	Browns Ferry Nuclear Plant	Tennessee River	Industrial	
Lawrence-Alabama	W. Morgan, E. Lawrence	Tennessee River	Municipal	
Lawrence-Alabama	Water Authority			
	Champion International (Courtland Plant)	Tennessee River	Industrial & Potable	
25-Mile Radius				
State of Alabama	Joe Wheeler State Park	Tennessee River	Municipal	
Lawrence-Alabama	TVA-Wheeler Dam[1]	Tennessee River	Industrial	
50-Mile Radius				
Lauderdale-Alabama	Florence City-Wilson Plant	Tennessee River	Municipal	
Colbert-Alabama	Reynolds Metals Company	Tennessee River	Industrial	
Colbert-Alabama	Muscle Shoals	Tennessee River Fleet Hollow Embayment	Municipal	
Colbert-Alabama	TVA ERL	Fleet Hollow Embayment	Industrial & Potable	
Colbert-Alabama	TVA-Wilson Dam	Tennessee River	Industrial	
Colbert-Alabama	Occidental Chemical Company	Tennessee River	Industrial	
Colbert-Alabama	Sheffield	Tennessee River	Municipal	
Colbert-Alabama	Sheffield Police			
Colbert-Alabama	TVA Colbert Fossil Plant	Tennessee River	Industrial	
Colbert-Alabama	Cherokee Water Works & Gas	Tennessee River	Municipal	
Colbert-Alabama	Cherokee Police (Day)			
Colbert-Alabama	Cherokee Police (Night)			
Colbert-Alabama	Laroche Industries	Tennessee River	Industrial	

[1]Potable water obtained from East Lauderdale County Water District.

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SQN - PUBLIC AND INDUSTRIAL SURFACE WATER SUPPLIES

County-State	Plant Name	Water Source	Type of Water Supply	Notification Advise State or Local Authorities listed in the REND
10-Mile Radius				
Hamilton-Tennessee	Sequoyah Nuclear Plant	Tennessee River	Industrial	
	Gold Point Marina	Tennessee River		
	East Side Utility	Tennessee River	Industrial	
		Tennessee River	Industrial	
	Chickamauga Dam (Power Service Center)	Tennessee River	Industrial	
	Chickamauga Dam	Tennessee River		
25-Mile Radius				
	E. I. Dupont Co.	Tennessee River	Industrial and Potable	
	Tennessee American Water Co.	Tennessee River	Municipal	
	Rock-Tennessee Mill[1]	Tennessee River	Industrial	
	Vulcan Sand & Gravel[1]	Tennessee River	Industrial	
	Signal Mountain Cement[1]	Tennessee River	Industrial	
	Medusa Cement Co.	Tennessee River	Industrial	
50-Mile Radius	Signal Mountain Cement (Plant)	Tennessee River	Industrial	
Marion-Tennessee	Signal Mountain Cement (Quarry)	Tennessee River	Industrial	
	South Pittsburg	Tennessee River	Municipal	
	Nickajack Dam	Tennessee River	Industrial	
Jackson-Alabama	Bridgeport	Tennessee River and Spring	Municipal	
Jackson-Alabama	Bridgeport Police	Tennessee River and Spring		
	*Widows Creek Fossil Plant[2]	Tennessee River	Industrial	
	Mead Corporation	Tennessee River	Industrial	

[1]Obtains potable water from Tennessee-American Water Company.
*[2]Obtains potable water supply from Bridgeport - physically removed potable water intake in November 1986.

APPENDIX J Page 3 of 3

WBN - PUBLIC AND INDUSTRIAL SURFACE WATER SUPPLIES ON THE TENNESSEE RIVER

County-State	Plant Name	Water Source	Type of Water Supply	Notification Advise State or Local Authorities listed in the REND
10-Mile Radius				
Rhea-Tennessee	Watts Bar Fossil & Hydro Plant[1]	Tennessee River	Industrial[2,3]	
	Watts Bar Nuclear Plant	Tennessee River	*Industrial[3,4]	
25-Mile Radius				
Rhea-Tennessee	City of Dayton	Tennessee River	Municipal	
	Dayton Police			
50-Mile Radius				
Hamilton-Tennessee	TVA Sequoyah Nuclear Plant	Tennessee River	Industrial	
	East Side Utility	Tennessee River	Industrial	
	E. I. Dupont	Tennessee River	Industrial and Potable	
	Chickamauga Dam	Tennessee River	Industrial	
	Tennessee American Water Co.	Tennessee River	Municipal	
	Rock-Tennessee Mill[5]	Tennessee River	Industrial	
	Vulcan Sand and Gravel[5]	Tennessee River	Industrial	
	Signal Mountain Cement[5]	Tennessee River	Industrial	

[1]On layby status - water use when activated is about 445 MGD.
[2]Cooling water.
[3]Potable water to nuclear plant, steam plant, hydro plant, and resort area, provided through Watts Bar Reservation System (wells).
[4]Cooling water and cooling tower makeup.
[5]Obtains potable water supply from Tennessee-American Water Company.

CECC EPIP Coversheet

Tennessee Valley Authority CENTRAL EMERGENCY CONTROL CENTER EMERGENCY PLAN IMPLEMENTING PROCEDURES	Title TERMINATION AND RECOVERY	CECC EPIP-13 REV. 10 Effective Date: 7/01/03

WRITTEN BY: Al Salatka (Signature) REVIEWED BY: (Signature) ~~7~~ 6 /19/03 (Date) 7/1/03

PLAN EFFECTIVENESS DETERMINATION: (Signature) ~~7~~ 6 /19/03 (Date) 7/1/03

CONCURRENCES

Concurrence Signature	Date
☒ Manager, EP Program Planning and Implementation	6/20/03
☒ Manager, Emergency Preparedness	6/27/03
☒ Manager, Radiological and Chemistry Services	07/01/03
☐	

APPROVAL

APPROVED BY: (Signature) VP Eng & Tech Svcs (Title / Organization) 7/01/03 (Date)

CECC-EPIP-13

TERMINATION AND RECOVERY

REVISION LOG

Rev. No.	Date	Revised Pages
0	3/22/88	All (Formerly IP-16; changed from IPD to EPIP)
1	7/8/88	Page 1
2	12/12/88	All
3	7/13/89	All
4	6/20/90	All--*formerly EPIP-23 (former EPIP-13 transferred to EPIP-14)
5	5/15/92	Pgs. 2 & 3 revised. New coversheet and rev. log added. All pages issued.
6	9/27/95	All pages revised.
7	10/30/96	Pg. 3 remove reference to Appendix C. Procedure put in new format. All pages issued.
8	7/10/00	Annual review and self-assessment items. All pages issued.
9	3/19/03	Annual review. Add information for NRC Administrative Letter 97-03. All pages issued.
10	7/1/03	The procedure was completely rewritten to provide for a graded transition from termination phase to recovery phase based on the severity of the event. Separate checklists were provided for the termination and recovery phases. A organization chart for the recovery organization was added. Procedure put in new EPIP format. All pages issued.

Table of Contents

1.0 PURPOSE

* This procedure provides guidance on termination and recovery from an incident for which onsite and offsite emergency centers were activated by the Site Emergency Director and transition from the Emergency Response Organization to the Recovery organization if necessary.

* Termination begins when personnel responsible for the response effort determine that conditions are sufficiently stabilized to begin comparing them to pre-established decisional criteria. The termination decision and subsequent notification that an event no longer constitutes an Operational Emergency establishes the beginning of recovery.

* Recovery is defined as those actions taken, after a plant has been brought to a stable or shutdown condition, to return the plant to normal operation. Recovery will begin when the emergency response is declared terminated. The level of recovery operations depends on the severity of the event. The recovery phase may be implemented in a graded approach from one of no recovery actions necessary to a fully implemented course of actions. When implemented, the recovery phase continues until the plant and any affected areas meet predetermined criteria for the resumption of normal operation or use.

* Types of activities conducted during the recovery phase may include (but are not limited to):

- Damage assessment
- Environmental consequence assessment
- Long-term protective action determinations
- Plant and/or environmental restoration
- Dissemination of information

2.0 SCOPE

* This procedure applies to the termination of a REP event which required activation of onsite and offsite emergency centers and actions for reentry and recovery activities required to restore the plant to normal operating condition and to provide assistance to state and local organizations.

*3.0 RESPONSIBILITIES

*3.1 The Senior Vice President, Nuclear Operations, or his designee will direct the overall recovery effort. If expected to be a long-term process, he may establish a recovery organization to be responsible for continuous direction and control of the recovery operation. This organizational structure would be contingent upon the emergency situation and required actions for recovery. Staffing of the CECC may remain in whole or in part as necessary. The LRC is also available to provide additional office space near the site to support the recovery operation.

*3.2 The CECC Director is responsible for coordinating with the Site Emergency Director, NRC, and appropriate offsite agencies in determining when to enter the recovery phase. Once that decision has been made, the CECC Director will notify the Senior Vice President, Nuclear Operations, or his designee.

* Revision

If the event was associated with an emergency off-site either natural or manmade which impacted the off-site (State and local) emergency response, the NRC regional administrator will inform the affected license when the condition of the off-site emergency preparedness infrastructure can support a safe reactor restart. NRC Administrative Letter 97-03 which provides information for plant restart discussions following natural disasters is provided as Appendix C.

*3.3 The CECC Public Information Manager (PIM) acts as an interface between TVA and the news media. The PIM assists the Senior Vice President, Nuclear Operations, CECC Director, or their designees with:

- drafting news releases concerning progress of the recovery operation
- coordinating all news releases with TVA management and State and Federal officials as required.
- coordinating all press briefings and interviews concerning the incident.

*3.4 Radiological Assessment Manager (RAM) provides radiological support as necessary.

*3.5 The Vice President, Engineering and Technical Services, will provide required technical support to the site.

*3.6 The Manager, Nuclear Fuels, will provide needed technical services to the site. Technical services available include fuel management and core analysis, core performance, nuclear fuel control and accountability, and startup support.

*4.0 PROCEDURES

*4.1 Termination

The decision to terminate an incident for which onsite and offsite emergency centers have been activated will be made by the Site Emergency Director after consultation with the plant technical and operations staffs and coordinated with the CECC Director. Proposals for termination of an emergency and entry into recovery will be coordinated with the State and NRC, if appropriate, through the CECC. Termination decisions should be based on site-specific EPIP-16 criteria and broad-based parameters such as:

Radiation or hazardous material exposure levels within the affected plant or area(s) are stable or decreasing with time.
The affected plant is in a stable condition, and there is a high probability that it can be maintained in that condition.
Releases of hazardous material to the environment have ceased or are controlled within permissible regulatory limits, and the potential for an uncontrolled release is low.
All emergency notifications have been completed.
The Site Emergency Director and CECC Director in consultation with the NRC and appropriate offsite agencies do not identify a valid reason to continue operating in the emergency response mode.
Initial recovery activities have been clearly identified and prioritized.
When applicable, a recovery staffing plan has been developed, approved and can be implemented.

* Revision

4.2 Recovery Operations

Recovery planning and implementation will start with assessment of plant, site, and environmental conditions. There are three general areas of recovery operations: accident assessment and investigation, recovery planning and scheduling, and repair and restoration.

4.2.1 Accident Assessment and Investigation

The following type of activities should be considered for accident assessment and investigation:

- Plant management in coordination with TVAN Corporate management, should establish an investigation board to determine the root cause of the event and prepare a formal accident report.
- All documents generated during the emergency response and useful to the accident investigation should be collected and organized.
- Plant technical, operations, and maintenance staffs should assess the condition of the plant including structural integrity, equipment status, hazardous material containment/confinement barriers, and safety systems.
- Provide support, when requested, to federal, state, and local government agencies for assistance with offsite dose assessment and related activities.

4.2.2 Recovery Planning and Scheduling

The following type of activities should be considered for recovery planning and scheduling:

- Notification to persons and agencies involved in the emergency response of the establishment of the Recovery Organization and the name of the person in charge.
- Evaluation of emergency plans to determine if adequate emergency preparedness status can be maintained during degraded plant conditions (e.g., inaccessibility of assembly areas, inoperative emergency/safety instrumentation and equipment, etc.)
- Establishment of specific criteria to be met prior to the resumption of normal operations or facility use.
- Contact with the affected State to coordinate any support required for assessment and recovery of affected offsite areas.
- Preparation of plans for the establishment of safe long-term conditions when the assessment indicates that a plant or affected area cannot be safely returned to normal operation or use.

Entire Page Revised

- Identification of required repair and restoration work based on the assessment results.
- Plan for the proper handling and disposal of all hazardous waste generated during recovery activities.
- Establishment of a tracking organization to monitor all assigned tasks, including developing work packages, scheduling activities, and estimating costs.
- Formation of a procedures review group to determine if specialized procedures are required and should be developed and to review and approve all special procedures.
- Continued evaluation of site or facility hazards and contamination levels during estimating exposure to workers.

4.2.3 Repair and Restoration Activities

The following type of activities should be considered for repair and restoration activities:

- Ensure that occupation exposure limits are followed in accordance with SPP-5.1, *Radiological Controls.*
- Ensure that any discharges form recovery activities are controlled within regulatory and environmental compliance limits. If discharges are necessary beyond these limits, ensure all documentation is prepared, approvals obtained, and notifications made.
- Conduct recovery activities through normal work organizations, practices, limitations, and procedures to the extent practical.
- Replenish, repair, or replace any emergency equipment or consumable materials used during the emergency response.
- Train applicable personnel on changes that occurred as a result of repair, restoration, and accident investigation.

*5.0 LOCAL RECOVERY CENTER (LRC)

*5.1 The purpose of the LRC is to provide a nearsite facility for TVA recovery management as well as NRC emergency response personnel and other emergency and/or recovery personnel.

*5.2 The LRC provides adequate space for TVA and others who may locate there to support the site should additional office space near the site become necessary during the recovery phase.

*5.3 The LRC will provide space for NRC personnel. Adequate supplies, communications, and data necessary for them to carry out appropriate functions is available.

*6.0 ENVIRONMENTAL SAMPLE COLLECTION AND ANALYSIS

*6.1 The TVA emergency field monitoring vans will be used to collect appropriate samples. This sample collection will be coordinated with the State. Samples will be divided and delivered to the State and the appropriate TVA laboratory.

*6.2 * Western Area Radiological Laboratory (WARL) will perform (or coordinate performance by approved testing facilities) environmental sample analysis. Information concerning the samples will be provided to the State and the RAM.

*7.0 REFERENCES

NP Radiological Emergency Plan
NRC Administrative Letter 97-03
CECC EPIP

*8.0 ABBREVIATIONS

WARL - Western Area Radiological Laboratory.
NP - Nuclear Power.
LRC - Local Recovery Center.
CECC - Central Emergency Control Center.
SED - Site Emergency Director.

*Revision

APPENDIX A Page 1 of 3

CECC DIRECTOR'S TERMINATION CHECKLIST

	Check box when action complete	Action	Concurrence
1	☐	• Radiation or hazardous material exposure levels within the affect plant or area(s) are stable or decreasing with time. ☐ YES ☐ NO	PAM: RAM: CECC Dir.: Date: Time:
2	☐	• The affected plant is in a stable condition, and is there a high probability that it can be maintained in that condition (site-specific EPIP-16 criteria verified by CECC Plant Assessment and Radiological Assessment staffs). ☐ YES ☐ NO	PAM: RAM: CECC Dir.: Date: Time:
3	☐	• Releases of hazardous material to the environment have ceased or are controlled within permissible regulatory limits, and the potential for an uncontrolled is release low. ☐ YES ☐ NO	PAM: RAM: CECC Dir.: Date: Time:
4		• All emergency notifications have been completed. ☐ YES ☐ NO	CECC Dir.: Date: Time:
5		• The Site Emergency Director and CECC Director, in consultation with the NRC and appropriate offsite agencies agree that no valid reason exists to continue operating in the emergency response mode. ☐ YES ☐ NO	CECC Dir.: Date: Time:

Entire Page Revised

APPENDIX A Page 2 of 3
CECC DIRECTOR'S TERMINATION CHECKLIST

	Check box when action complete	Action	Concurrence
6	☐	• Document event termination on CECC EPIP-13, Appendix A, page 3 of 3 and make appropriate notifications to the affected state. ☐ YES ☐ NO	CECC Dir.: Date: Time:
7	☐	• The Senior Vice President, Nuclear Operations has been notified of event termination. ☐ YES ☐ NO **Stop here if no recovery actions are necessary.** If recovery operations are necessary, continue with Steps 9 & 10 and continue to CECC EPIP-13, Appendix B.	CECC Dir.: Date: Time:
9	☐	• If applicable, Initial recovery activities have been clearly identified and prioritized. ☐ YES ☐ NO	PAM: RAM: CECC Dir.: Date: Time:
10	☐	• If applicable, a recovery staffing plan has been developed, approved, and can be implemented. ☐ YES ☐ NO	PAM: RAM: CECC Dir.: Date: Time:

Entire Page Revised

APPENDIX A Page 3 of 3
CECC DIRECTOR'S TERMINATION CHECKLIST

EVENT TERMINATION:

The: ☐ NOUE ☐ ALERT ☐ SITE AREA EMERGENCY ☐ GENERAL EMERGENCY

Affecting: BFN U2 ☐, U3 ☐ SQN U1 ☐, U2 ☐ WBN U1 ☐

EAL Designator: ______________________________

HAS BEEN TERMINATED

Event Termination Time: ____________________ **Date:** ____________________

Call affected State and provide this Information

State Notification Time: ____________________ **Date:** ____________________

CECC Director: ______________________________

APPENDIX B Page 1 of 1
CECC DIRECTOR'S RECOVERY CHECKLIST

	Check box when action complete	Action	Concurrence
1	☐	The recovery organization has been established. ☐ YES ☐ NO	CECC Dir.: Date: Time:
2	☐	Accident Assessment and Investigation activities have been considered and implemented as determined, based on the severity of the event, including the collection and organization of all documents generated during the emergency response. ☐ YES ☐ NO	CECC Dir.: Date: Time:
3	☐	The affected state agency has been contacted to coordinate any support required for assessment and recovery of affected offsite areas. ☐ YES ☐ NO	CECC Dir.: Date: Time:
4	☐	Appropriate Recovery Planning and Scheduling activities have been considered and implemented as determined, based on the severity of the event. ☐ YES ☐ NO	CECC Dir.: Date: Time:
5	☐	The NRC has been contacted as applicable to NRC Administrative Letter 97-03. Refer to Appendix C. ☐ YES ☐ NO	CECC Dir.: Date: Time:

Entire Page Revised

APPENDIX C
NRC ADMINISTRATIVE LETTER 97-03
Page 1 of 4

UNITED STATES
NUCLEAR REGULATORY COMMISSION
OFFICE OF NUCLEAR REACTOR REGULATION
WASHINGTON, D.C. 20555-0001

March 28, 1997

NRC ADMINISTRATIVE LETTER 97-03: PLANT RESTART DISCUSSIONS FOLLOWING NATURAL DISASTERS

Addressees

All holders of operating licenses or construction permits for nuclear power reactors.

Purpose

The U.S. Nuclear Regulatory Commission (NRC) is issuing this administrative letter to inform addressees about a recently adopted internal practice. This practice involves coordinating the assessment of offsite recovery and onsite restart activities following a natural disaster (hurricane, tornado, flood, storm, earthquake, etc.) where offsite damage may be substantial or undetermined. This administrative letter does not transmit or imply any new or changed requirements or staff positions. No specific action or written response is required.

Background

Numerous events have occurred in recent years in which natural disasters have affected power reactor facilities. Most notable of these is Hurricane Andrew and its impact on the Turkey Point Station. The licensee for the Turkey Point plant shut the reactors down in anticipation of the storm. Onsite damage from the hurricane was extensive. After that event, the licensee repaired the damage and was ready to restart the plant before the offsite emergency preparedness infrastructure was ready to support the restart. An assessment of offsite conditions and infrastructure prior to restart was necessary to assure emergency preparedness in the event of a subsequent reactor accident.

Events have also occurred in which plants have shut down in anticipation of hurricane damage, which turned out to be minimal. Despite the absence of onsite damage, either some offsite damage occurred that affected the state of offsite emergency preparedness, or some damage occurred offsite such that the state of offsite emergency preparedness could not be determined immediately. For these cases, the NRC coordinated with the Federal Emergency Management Agency (FEMA) and the licensees involved to ensure that the restarts occurred after the offsite emergency preparedness infrastructure could safely support them.

APPENDIX C
NRC ADMINISTRATIVE LETTER 97-03

Discussion

Although the overall responsibility for confirming the adequacy of radiological emergency preparedness of commercial nuclear power plants is vested with the NRC, it relies on FEMA's assessment of offsite emergency planning and response activities when carrying out this responsibility.

Section III of the Memorandum of Understanding (MOU) Between FEMA and the NRC, dated June 17, 1993, lists responsibilities for both agencies for cooperating in the recovery from a disaster that affects the offsite emergency preparedness infrastructure surrounding power reactors. FEMA's headquarters (HQ) in Washington, D.C., is responsible for providing findings and determinations to the NRC concerning the adequacy of offsite preparedness in the areas surrounding power reactor sites following a severe natural event. FEMA HQ bases its assessment on information from State and local governmental authorities, as well as from the affected FEMA regional office and the NRC.

In two recent instances (Hurricane Bertha, July 1996 and Hurricane Fran, September 1996), FEMA HQ chartered special evaluation teams to assess whether the offsite emergency preparedness infrastructure could support the restart of plants that had shut down in anticipation of hurricanes that affected the sites. These teams consisted of FEMA and NRC regional representatives, State and local emergency management representatives, and, in a limited capacity, power reactor licensee personnel. These teams provided assessments to FEMA HQ for its ultimate determinations that offsite emergency preparedness could support plant restart in both cases. The chartering of these special evaluation teams helped ensure a timely assessment of the condition of the offsite infrastructure and was based on experience gained with Hurricane Opal (October 1995) and the Quad Cities tornado (May 1996).

In some cases, a natural disaster may occur where onsite damage is minimal, but offsite damage may be substantial or undetermined. In these cases, the plant may be ready to start up shortly after the event. Communications in these cases between the licensee and NRC, the NRC and FEMA, and FEMA and offsite officials will be aggressive; however, stringent protocols will be observed to ensure that FEMA and the NRC operate within the guidelines of the MOU.

The NRC uses FEMA's determinations to inform power reactor licensees when the condition of the offsite emergency preparedness infrastructure can support a reactor restart. The Office of Nuclear Reactor Regulation (NRR), as well as NRC regional offices, have adopted a communication protocol that links key personnel in the two agencies and the affected licensee organization. An overview of this protocol is attached. Some of the key points of this protocol are:

1. NRC regional office personnel maintain close contact with the affected power reactor licensee to determine the state of onsite emergency preparedness and the plans for restart. The NRC regional office communicates this information rapidly to NRR.

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NRC ADMINISTRATIVE LETTER 97-03
Page 3 of 4

2. FEMA regional office personnel maintain close contact with their evaluators in the field, the affected State and local emergency management officials, and the affected NRC regional office to determine the state of offsite emergency preparedness. The FEMA regional office communicates this information rapidly to FEMA HQ.

3. The final assessment that offsite emergency preparedness can support a power reactor restart originates from FEMA HQ.

4. A single individual in NRR serves as the point of contact with FEMA HQ to receive this assessment. The individual communicates this information rapidly to NRR management and the cognizant NRC regional office.

5. After the assessment from FEMA is received and discussed with NRR management, the NRC regional administrator informs the affected licensee that the condition of the offsite emergency preparedness infrastructure can support a safe reactor restart.

The NRC has developed this protocol as a result of discussions with FEMA, as well as lessons learned from Hurricane Andrew and other events. The objective of this protocol is to ensure that aggressive and rapid information flow occurs between the involved organizations following natural disasters at power reactors. The NRC expects that the use of this protocol will ensure that the determination that the condition of the offsite emergency preparedness infrastructure can support a reactor restart will be made before the licensee is actually ready to restart the reactor plant(s). In the event that the determination is not made before the licensee is ready to restart the plant(s), the NRC will evaluate the need to delay the restart through the issuance of an order or confirmatory action letter. By accomplishing this protocol, the licensee, FEMA, and NRC can provide for safe and rapid restarts of power reactors in the wake of these disasters and assure that the offsite emergency preparedness infrastructure can function as expected if called upon in an emergency.

This administrative letter requires no specific action or written response. If you have any questions about this letter, please contact the contact listed below or the appropriate Office of Nuclear Reactor Regulation (NRR) project manager.

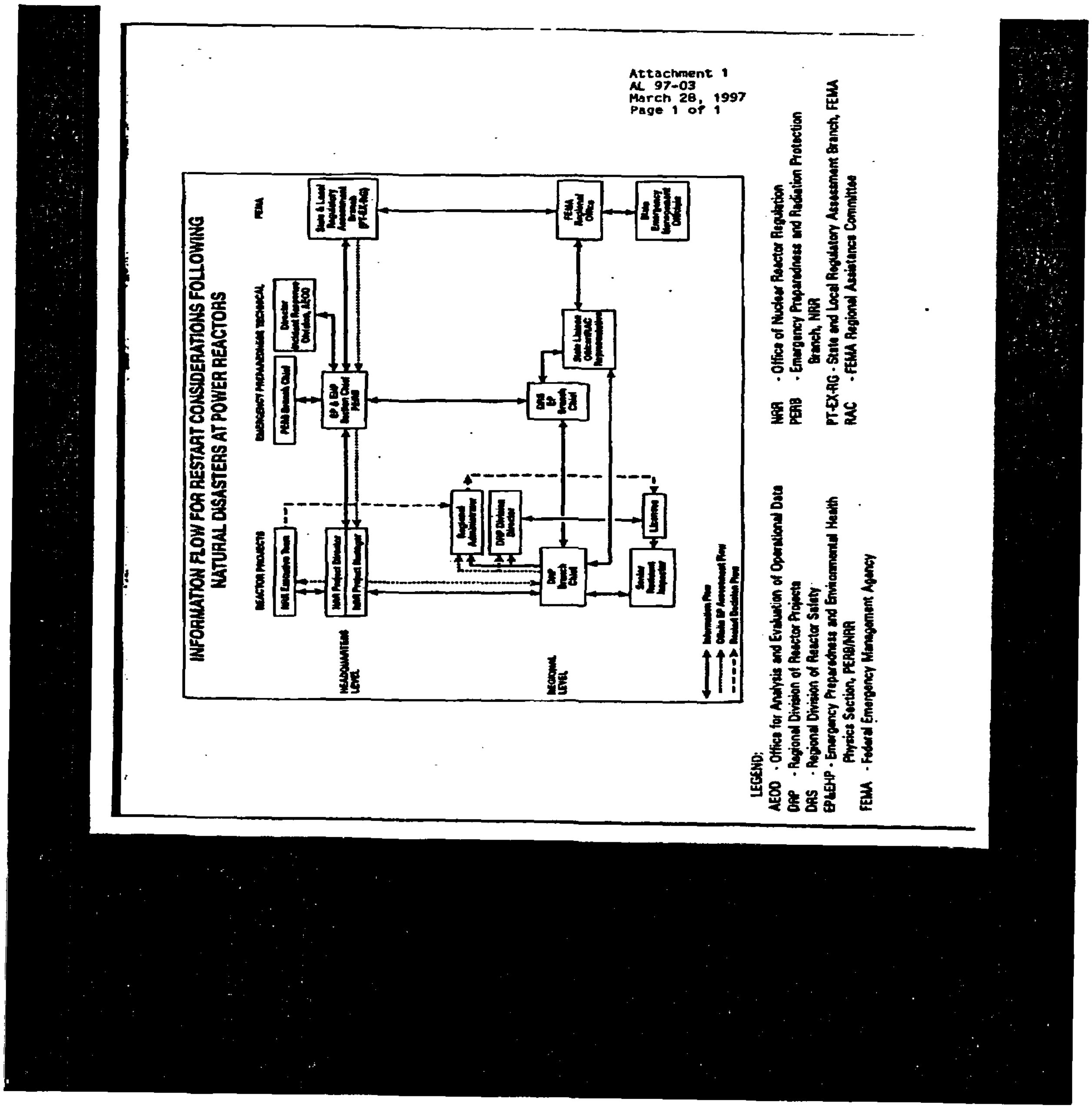
signed by

Thomas T. Martin, Director
Division of Reactor Program Management
Office of Nuclear Reactor Regulation

Contact: W. Maier, NRR
(301) 415-2926
E-mail: wam@nrc.gov

Attachments:
1. Information Flow for Restart Considerations Following Natural Disasters at Power Reactors

APPENDIX C
NRC ADMINISTRATIVE LETTER 97-03
Page 4 of 4



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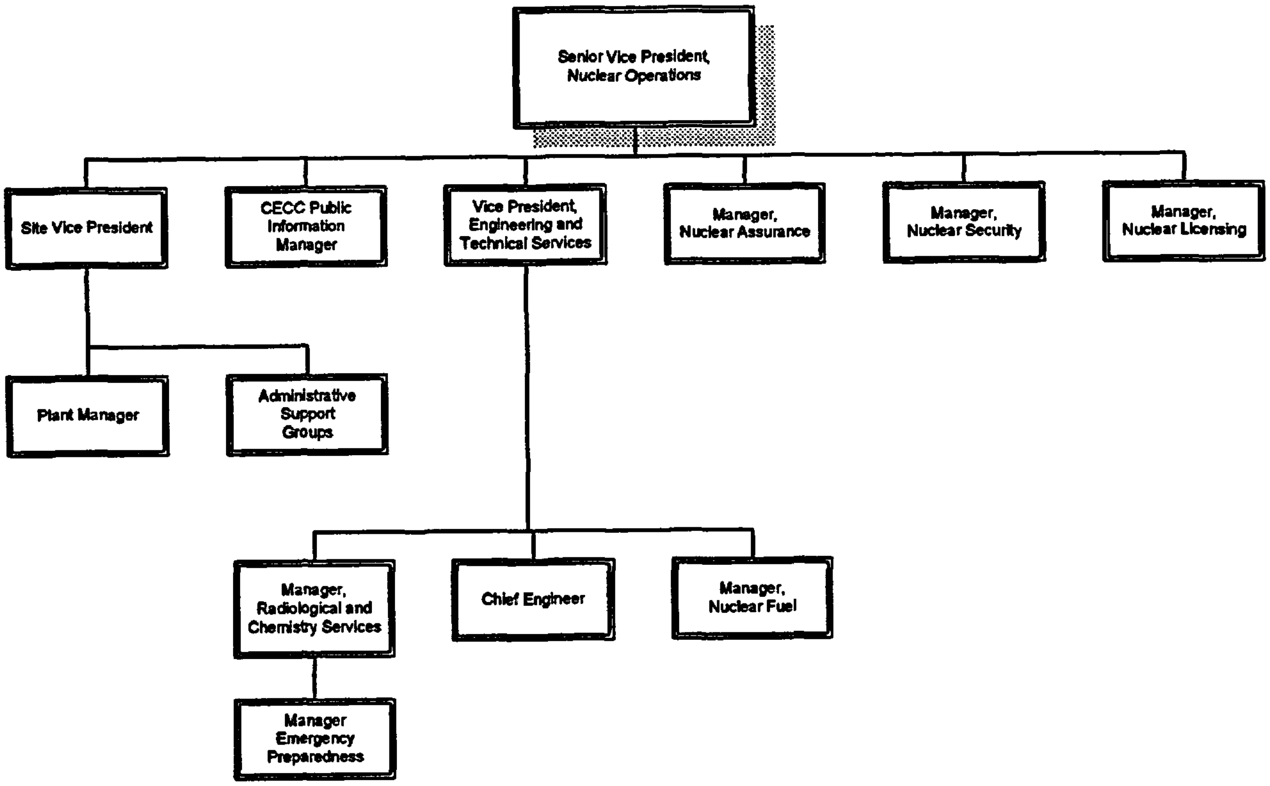
RECOVERY ORGANIZATION



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