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- ATTACHMENT 5 -

## PROCEDURE CHANGES, REASONS, AND REFERENCES

**PROCEDURE NO.:** __REP-12____ **TITLE:**__EOF ACCIDENT ASSESSMENT PROCEDURE_____

**DESCRIPTION OF NEW PROCEDURE OR CHANGE (detailed)**

Changed the FPB Assessment table of Attachment 6 to be consistent with EM-202 and EM-225 regarding prolonged steaming and OTSG with > 10 gpm leak to atmosphere. Changed header title information to reflect group name. Deleted reference to REP-02 as it is being cancelled. Changed REP-3 to REP-03A for setup from checklists. Changed name of EOF Accident Assessment Team to Technical Support Team to reflect function of team. Changed team member titles to reflect team name change. Deleted EOF Risk Analysis Representative from team and added responsibilities to EOF Technical Support Engineer, deleted EOF Risk Analysis Representative checklist and renumbered remaining attachments accordingly. Changed Manager, Radiological Emergency Planning to Supervisor, Emergency Preparedness. Changed Generating Complex in 4.1.4 to Energy Complex for consistency with other EP procedures. Changed REP abbreviation to EP and PRC to PNSC. Added Technical Support Team (TST) to abbreviations. Added computer points for containment pressure and AB ventilation flowrate to Attachment 8 per drill participant comment. Other editorial changes.

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**REASON AND REFERENCES FOR PROCEDURE CHANGE**

General housekeeping, address drill participant comments, delete cancelled reference, update organizational titles due to combined companies, and changed team name to reflect function.

| | | DATE |
|---|---|---|
| **ORIGINATED BY:** | S D Mousseau | 07/18/01 |
| **REVIEWED BY (IF APPLICABLE):** | | |
| **SEP APPROVAL:** | John D Stephenson | 7/18/01 |

# EMERGENCY OPERATIONS FACILITY TECHNICAL SUPPORT TEAM

## REQUIRED APPROVALS

John D Stephens

Supervisor, Emergency Preparedness

7/24/01

Date

## 1.0 PURPOSE

This procedure provides guidance for establishment and operation of the EOF Technical Support Team (TST).

## 2.0 SCOPE

2.1 The EOF Technical Support Team primarily functions as a technical resource for the EOF Technical Support Director and the EOF Director in development of public PARs. Duties also include assisting the TSC Accident Assessment Team as needed in development of mitigation strategies and in research of solutions to plant problems (however, the EOF TST personnel are NOT considered Severe Accident Management evaluators), interfacing with the Dose Assessment Team, and assisting in the development of recovery plans.

2.2 A Safety Assessment was performed for this procedure. A determination was made that this procedure is outside the scope of 10 CFR 50.59.

## 3.0 APPLICABLE REFERENCE DOCUMENTS

3.1.1 REP-03A, "Setup of the Emergency Operations Facility"

3.1.2 EM-225, "Duties of the Technical Support Center Accident Assessment Team"

3.1.3 EM-225A, "Post-Accident RB Hydrogen Control"

3.1.4 EM-225B, "Post-Accident Boron Concentration Management"

3.1.5 EM-225C, "Post-Accident Monitoring of Reactor Building Temperature"

3.1.6 EM-225D, "Guidance for Dry OTSG Tube to Shell Delta T Monitoring and Control"

3.1.7 EM-225E, "Guidelines for Long-Term Cooling"

3.1.8 Emergency Operating Procedures (EOPs)

3.1.9 MP-575, "Hydrogen Recombiner Installation"

3.1.10 OP-417B, "Operation of the Post-Accident Hydrogen Recombiners"

3.1.11 B&W Technical Bases Document

3.1.12 CR-3 Severe Accident Guideline

3.2.13 Response Technical Manual (RTM-96), USNRC, Vol. 1, Rev. 3

3.2.14 Radiological Emergency Response Plan

3.2.15 NUREG-1228, "Source Term Estimation During Incident Response to Severe Nuclear Power Plant Accidents"

3.2.16 FPC IOC CR97-0122, dated 12/23/97

3.2.17 CH-632A, "Post-Accident Sampling and Analysis of the Reactor Coolant System"

## 4.0 DEFINITIONS AND ABBREVIATIONS

### 4.1 DEFINITIONS

#### 4.1.1 Technical Support Team (TST)

Consists of EOF Technical Support Coordinator, EOF Technical Support Engineer, and EOF Technical Support Operations Representative.

#### 4.1.2 Candidate High Level Actions (CHLAs)

Actions described in the CR-3 Severe Accident Guideline that could be taken to mitigate a severe accident and are deemed appropriate based on PDCs.

#### 4.1.3 Critical Safety Functions (CSFs)

Those functions needed to ensure adequate core cooling and to preserve the integrity of the fission product barriers, thereby protecting the health and safety of the general public and plant personnel. They include reactivity control, coolant inventory control, decay heat removal capability, fission product barrier status, electrical power availability, and Control Room status.

#### 4.1.4 Emergency Action Levels (EALs)

Conditions or indications that may be used as thresholds for initiating specific emergency measures. (See EM-202, Enclosure 1)

4.1.5 **Protective Action Recommendations (PARs)**

Emergency measures recommended for purposes of preventing or minimizing radiological exposures to Energy Complex personnel or members of the public.

4.1.6 **Plant Damage Conditions (PDCs)**

Damage conditions used in the CR-3 Severe Accident Guideline to describe the status of the RCS, reactor core, and the containment during the progression of a severe accident.

4.1.7 **Severe Accident**

An accident (beyond that assumed in the CR-3 design and licensing basis) that results in catastrophic fuel rod failure, core degradation and fission product release into the reactor vessel, RB, or environment.

4.2 **ABBREVIATIONS**

AAT - Accident Assessment Team
ADV - Atmospheric Dump Valve
AP - Abnormal Procedure
BSP - Building Spray Pump
BWST - Borated Water Storage Tank
CFM - Cubic Feet per Minute
CFT - Core Flood Tank
CHLA - Candidate High Level Action
CR-3 - Crystal River Unit 3
CSF - Critical Safety Function
DCP - Decay Heat Closed Cycle Cooling System Pump
DHP - Decay Heat Pump
EAL - Emergency Action Level
ECCS - Emergency Core Cooling System
EDG or EGDG - Emergency Diesel Generator
EFIC - Emergency Feedwater Initiation and Control
EFP - Emergency Feedwater Pump
EFW - Emergency Feedwater
EOF - Emergency Operations Center
EOP - Emergency Operating Procedure
EP - Emergency Preparedness
FPC - Florida Power Corporation
HPI - High Pressure Injection
ICC - Inadequate Core Cooling
IOC - Inter-Office Communication

LOCA - Loss of Coolant Accident
MFW - Main Feedwater
MSSV - Main Steam Safety Valve
OTSG - Once-Through Steam Generator
PAR - Protective Action Recommendation
PASS - Post-Accident Sampling System
PDC - Plant Damage Condition
PORV - Pilot-Operated Relief Valve
PNSC - Plant Nuclear Safety Committee
PSIG - Pounds per Square Inch (Gauge)
QC - Quality Control
RB - Reactor Building
RCITS - Reactor Coolant Inventory Tracking System
RCP - Reactor Coolant Pump
RCS - Reactor Coolant System
RPS - Reactor Protection System
RWP - Raw Water Pump
Rx - Reactor
SPDS - Special Parameter Display System
SWP - Nuclear Services Closed Cycle Cooling System Pump
TSC - Technical Support Center
TST – Technical Support Team
USNRC - U.S. Nuclear Regulatory Commission
XFRM - Transformer

## 5.0 RESPONSIBILITIES AND ACTIONS

### 5.1 RESPONSIBILITIES

#### 5.1.1 EOF Technical Support Director

Notifies the EOF Technical Support Coordinator and the Corporate Health Physicist that the EOF has been activated and provides direction and control to the EOF functions associated with accident mitigation and assessment.

#### 5.1.2 EOF Technical Support Coordinator

a. Keeps the EOF Technical Support Director informed of TST activities and developments in plant status, especially those that may impact EALs and PARs.

b. Notifies the EOF Technical Support Operations Representative and EOF Technical Support Engineer that the EOF has been activated and provides coordination and support for the TST.

c. Ensures communication is established with the TSC on the TSC/EOF Ringdown until EOF Communicator arrives.

d. Assists in the setup of the EOF Technical Support Work Area.

e. Performs "plant conditions" portion of the EOF briefings using the briefing guidelines provided in Attachment 4.

f. Monitors the Accident Assessment Ringdown (communication between the Control Room and TSC AAT).

g. Provides support to the TSC AAT in determining the causes and consequences of the emergency.

h. Ensures interface is established with the EOF Dose Assessment Team using Section 5.1.5 and Attachment 7 as guidance.

i. Refers to attachments for additional accident assessment guidance and information.

j. Notifies Simulator support personnel when necessary (e.g., for testing mitigation strategies).

k. Monitors CSFs and provides status to EOF Technical Support Director and EOF personnel during briefings, as needed.

5.1.3 **EOF Technical Support Operations Representative**

a. Verifies that the SPDS computer is properly set up and operational.

b. Operates the SPDS computer.

c. Monitors plant parameters and provides status updates to the EOF Technical Support Coordinator.

d. Monitors communications between the Control Room and the TSC AAT via speaker in the EOF Technical Support Room.

e. Assists in the setup of the EOF Technical Support Work Area.

5.1.4 **EOF Technical Support Engineer**

a. Assesses plant conditions and provides Engineering support for developing accident mitigation strategies as needed.

b. Provides Engineering support to the TSC Technical Support Team.

c. Notifies additional Engineering resources when necessary.

d. Assists in the setup of the EOF Technical Support Work Area.

5.1.5 **Dose Assessment Team**

a. Supports the TSC Accident Assessment Team with on-site radiological data and with chemical and radiological analysis of samples as needed to assess the accident.

b. Provides plant radiation monitor readings and assessments.

c. Provides projected radiological data (on-site and off-site doses, dose rates, and deposition) (> 1 hour to obtain).

d. Provides RCS PASS data (> 2 hours to obtain results) on radionuclide composition, Chloride concentration, dissolved Hydrogen concentration, and Boron concentration.

e. Provides Reactor Building and/or Auxiliary Building atmosphere radionuclide composition (> 1 hour to obtain).

5.2 **INSTRUCTIONS**

5.2.1 The EOF Technical Support Coordinator or designee performs the duties of Attachment 1.

5.2.2 The EOF Technical Support Operations Representative performs the duties of Attachment 2.

5.2.3 The EOF Technical Support Engineer performs the duties of Attachment 3.

## 6.0 INTERPRETATION CONTACT

Supervisor Emergency Preparedness

## 7.0 REVISION HISTORY

| | | |
|---|---|---|
| Rev. 0 | Date: 01/26/99 | Initial Issue |
| Rev. 1 | Date: 10/09/00 | General revision to update attachments to match EM-225. |
| Rev. 2 | Date: 07/18/01 | NUPOST comment to make FPB Assessment table consistent with EM-202 & EM-225. Changed name of team to Technical Support Team. Other changes include organizational titles, deletion of cancelled reference and consistency-type changes. |

## - ATTACHMENT 1 -
## EOF TECHNICAL SUPPORT COORDINATOR CHECKLIST

**Check**

_____ 1. Perform telephone notifications (Refer to Emergency Response Personnel Roster):

___ EOF Technical Support Operations Representative
___ EOF Technical Support Engineer (3)

_____ 2. Check-in with EOF Director upon arrival and ensure name is posted on staffing board.

_____ 3. Ensure Accident Assessment Ringdown speaker is functional. (Refer to Attachment 9) (This is for monitoring only. Notify the TSC AAT by plant extension as necessary.)

_____ 4. Ensure CSFs are being assessed. (Refer to Attachment 4) For the initial assessment, this information may be readily available from the TSC AAT. For subsequent evaluations, discuss with the EOF Technical Support Director whether TSC information is sufficient or if independent evaluations are desirable.

_____ 5. Ensure PARs are appropriate per EM-202, Enclosure 8.

_____ 6. Ensure Room 124 SPDS computer is functional. (Refer to Attachment 9)

_____ 7. Ensure communication is established with TSC on TSC/EOF Ringdown, until EOF Communicator arrives. (Refer to Attachment 9)

_____ 8. Ensure EOP/AP procedure book is retrieved from Operations Briefing Room (A117) or Simulator Control Booth (A115).

_____ 9. Brief the EOF Technical Support Director on CSFs and plant status.

_____ 10. Provide CSFs and status reports to the EOF Staff during briefings. Use guidelines on Attachment 4 as necessary.

_____ 11. Establish interface with EOF Dose Assessment Team. Attachment 7 lists information needed by the Dose Assessment Team; Section 5.1.5 lists information available from the Dose Assessment Team.

_____ 12. Ensure the Status Board Coordinator is provided current CSFs information in the Main Conference Room.

_____ 13. Notify Simulator personnel for support, if needed (e.g., for testing mitigation strategies).

_____ 14. Provide assistance to EOF Director on recovery plan. (Refer to Attachment 8)

## - ATTACHMENT 2 –
## EOF TECHNICAL SUPPORT OPERATIONS REPRESENTATIVE CHECKLIST

**Check**

_____ 1. Confirm SPDS computer is operational. (Refer to Attachment 9)

_____ 2. Ensure work area is set up and functional. (Refer to REP-03A, Attachment 1)

_____ 3. Provide plant parameters/status to EOF Technical Support Coordinator.

_____ 4. Ensure Accident Assessment Ringdown speaker is functional. (Refer to Attachment 9) (This is for monitoring only. Notify the TSC AAT by plant extension as necessary.)

**- ATTACHMENT 3 -**

## EOF TECHNICAL SUPPORT ENGINEER CHECKLIST

**Check**

_____ 1. Ensure work area is set up and functional. (Refer to REP-03A, Attachment 1)

_____ 2. Establish communication with TSC AAT by plant phone as necessary

_____ 3. Obtain applicable drawings/documents, as needed.

_____ 4. Establish additional Engineering resources, if necessary.

_____ 5. Provide assistance for recovery plan development. (Refer to Attachment 8)

_____ 6. Obtain reactor core status. (Refer to Attachment 6)

_____ 7. Provide initial core assessment to EOF Technical Support Coordinator.

_____ 8. Provide core status updates to EOF Technical Support Coordinator.

_____ 9. Complete Attachment 7 for the Dose Assessment Team and provide to the EOF Technical Support Coordinator.

_____ 10. Notify additional resources (engineers, etc.) for core damage assessment, if needed.

_____ 11. If requested by the TSC to perform an assessment of core damage based on reactor coolant sample, use Enclosure 2 from CH-632A.

**- ATTACHMENT 4 -**

## BRIEFING GUIDELINE

Refer to Attachments 5 and 6 to aid in this evaluation.

I REACTOR SHUTDOWN — Yes ☐ No ☐

II. CORE ADEQUATELY COOLED — Yes ☐ No ☐

III. FISSION PRODUCT BARRIERS ASSESSMENT

| | | | | |
|---|---|---|---|---|
| Fuel Clad | ☐ Intact | ☐ Challenged | ☐ Lost | ☐ Regained |
| RCS | ☐ Intact | ☐ Challenged | ☐ Lost | ☐ Regained |
| Containment | ☐ Intact | ☐ Challenged | ☐ Lost | ☐ Regained |

IV. EMERGENCY ELECTRICAL POWER STATUS

| | | |
|---|---|---|
| Off-Site Power Available? | Yes ☐ | No ☐ |
| ES Buses Energized? | Yes ☐ | No ☐ |
| Emergency Diesel Generator's Available? | Yes ☐ | No ☐ |
| DC Power Available? | Yes ☐ | No ☐ |

V. CONTROL COMPLEX STATUS

| | | |
|---|---|---|
| Ventilation/Cooling Available? | Yes ☐ | No ☐ |
| Necessary Instrumentation Available?* | Yes ☐ | No ☐ |

VI. OTHER CONDITIONS/CHALLENGES

_____

_____

_____

_____

_____

_____

_____

_____

* Necessary refers to specific instruments and annunciators that are needed to identify, diagnose, and track the problems that are causing the emergency.

- ATTACHMENT 5 -

## CRITICAL SAFETY FUNCTION CHECKLIST

Monitor the parameters associated with the Critical Safety Functions. The parameter tables below are for reference only. It is NOT intended that the tables be completed during each evaluation. Plant computer point numbers or SPDS/RECALL point numbers are listed, if available.

Using pre-established RECALL Groups based on accident type in progress is recommended.

**Notify the Technical Support Coordinator immediately if any of the CSFs cannot be verified.**

# I. REACTOR SHUTDOWN STATUS:

### REACTIVITY CONTROL

| PARAMETER | COMPUTER POINT | RECALL POINT | | | |
|---|---|---|---|---|---|
| All Rods at in-limits Y/N | P057 | RECL-375 | | | |
| Intermediate Range detector NI-3 amps | P212 | RECL-150 | | | |
| Intermediate Range detector NI-4 amps | P213 | RECL-151 | | | |
| Source Range NI-1 cps | P202 | RECL-152 | | | |
| Source Range NI-2 cps | P203 | RECL-153 | | | |
| Low Range NI-14/15 | | RECL-102,103 | | | |
| Adequate Shutdown Margin | OP-103C Curve 18&19 | | | | |

- ATTACHMENT 5 (Continued) -

# II. CORE COOLING STATUS:

## ECCS/SUPPORT STATUS

| PARAMETER | COMPUTER POINT | RECALL POINT | | | |
|---|---|---|---|---|---|
| Subcooling Margin | M114 | | | | |
| A HPI Pump operating | | RECL-209 | | | |
| B HPI Pump operating | | RECL-210 | | | |
| C HPI Pump operating | | RECL-211 | | | |
| MUV-23 flow | W704 | RECL-52 | | | |
| MUV-24 flow | W706 | RECL-54 | | | |
| MUV-25 flow | W703 | RECL-51 | | | |
| MUV 26 flow | W705 | RECL-53 | | | |
| DHPs operating A/B (run/stop) | X063<br>X064 | RECL-207<br>RECL-208 | | | |
| DHP-1A flow | W409 | RECL-55 | | | |
| DHP-1B flow | W410 | RECL-56 | | | |
| CFT A level | P200 | | | | |
| CFT B level | P201 | | | | |
| CFT A press | | | | | |
| CFT B press | | | | | |
| BWST level (ft) | X335 | RECL-57 | | | |
| RWPs operating 1/2A/2B/3A/3B | | | | | |
| DCPs operating A/B (yes/no) | | | | | |
| SWPs operating A/B/C | | | | | |

## SECONDARY SYSTEM STATUS

| PARAMETER | COMPUTER POINT | RECALL POINT | | | |
|---|---|---|---|---|---|
| EFIC OTSG A press | W449 | RECL-252 | | | |
| EFIC OTSG B press | W452 | RECL-255 | | | |
| OTSG A level | S285 | RECL-92 | | | |
| OTSG B level | S286 | RECL-93 | | | |
| MFW flow A | S301 | RECL-100 | | | |
| MFW flow B | S302 | RECL-101 | | | |
| EFPs operating 1/2/3/7 | | | | | |
| EFW flow to A OTSG | S300 | RECL-245 | | | |
| EFW flow to B OTSG | S312 | RECL-247 | | | |

- ATTACHMENT 5 (Continued) –

## III. FISSION PRODUCT BARRIER ASSESSMENT:

| FUEL CLADDING BARRIER | | | |
|---|---|---|---|
| ☐ INTACT | ☐ CHALLENGED | ☐ LOST | ☐ REGAINED |
| • No indication of cladding damage | • RCS condition warrant entry into EOP-07<br>• Core Exit Thermocouples > 700 degrees F | • RCS conditions in (or previously in) Region 3 or Severe Accident Region<br>• PASS indicates increased RCS activity >300μCi/gr $I_{131}$ (refer to CH-632A)<br>• RM-G29/30 > 100 R/hr for ≥ 15 minutes<br>• Attachment 6 indicates failed fuel | • Cooling restored, no further degradation expected. |

| REACTOR COOLANT SYSTEM BARRIER | | | |
|---|---|---|---|
| ☐ INTACT | ☐ CHALLENGED | ☐ LOST | ☐ REGAINED |
| • Leakage is within normal makeup pump capacity | • RCS leak or OTSG tube leak requiring one or more injection valves to maintain adequate subcooling margin<br>• RCS pressure /Tincore relationship violates NDT limits<br>• RCS leak or OTSG tube leak results in ES actuation on low RCS pressure.<br>• HPI/PORV or HPI/Code Safety valve cooling is in progress | • RCS leak resulting in loss of adequate subcooling margin<br>• OTSG Tube Rupture resulting in loss of adequate subcooling margin<br>• RM-G29/30 >10R/hr for ≥ 15 minutes | • HPI/PORV or HPI Code Safety Cooling stopped<br>• Subcooling Margin restored and leak isolated |

| CONTAINMENT BARRIER | | | |
|---|---|---|---|
| ☐ INTACT | ☐ CHALLENGED | ☐ LOST | ☐ REGAINED |
| • No evidence of containment leakage<br>• Tube rupture release is only through condenser | • RB pressure > 54 psig<br>• RB hydrogen concentration > 4%<br>• RB pressure > 30 psig with no building spray available<br>• RMG-29 or 30 reading > 25,000 R/hr<br>• Core conditions in severe accident region of ICC curves for >15 min | • Containment isolation is incomplete and release path to environment exists.<br>• OTSG Tube Rupture > 10 gpm exists and prolonged steaming to atmosphere or an unisolable steam leak outside RB from affected OTSG.<br>• Containment pressure or sump level response not consistent with LOCA conditions<br>• Rapid unexplained RB pressure decrease following an initial increase | • Repair efforts have isolated leak<br>• Containment pressure has reduced to stop leakage |

Performed By: _____ Date: _____ Time: _____

- ATTACHMENT 5 (Continued) -

# IV. EMERGENCY ELECTRICAL POWER STATUS:

## OFF-SITE POWER

| PARAMETER | AVAILABLE | UNAVAILABLE |
|---|---|---|
| 500 KV SWITCHYARD | | |
| 230 KV SWITCHYARD | | |
| OFF-SITE POWER XFRM | | |
| BEST | | |

## ES BUSES

| PARAMETER | AVAILABLE | UNAVAILABLE |
|---|---|---|
| A-ES 4160V BUS | | |
| B-ES 4160V BUS | | |
| A-ES 480V BUS (Note 1) | | |
| B-ES 480V BUS (Note 1) | | |

## EMERGENCY DIESEL GENERATOR

| PARAMETER | RECALL PT. | LOADED | AVAILABLE | UNAVAILABLE |
|---|---|---|---|---|
| A-EDG | RECL-133,171 | | | |
| B-EDG | RECL-134,172 | | | |

## DC ELECTRICAL

| PARAMETER Note (1) | AVAILABLE | UNAVAILABLE |
|---|---|---|
| A-BATTERY | | |
| B-BATTERY | | |
| C-BATTERY | | |

Note (1) Battery failure will occur if associated battery chargers are de-energized.

- ATTACHMENT 5 (Continued) -

# V. CONTROL COMPLEX STATUS:

**CONTROL COMPLEX VENTILATION STATUS**

| PARAMETER | AVAILABLE | OPERATING | UNAVAILABLE |
|---|---|---|---|
| A-TRAIN EMERGENCY RECIRC | | | |
| B-TRAIN EMERGENCY RECIRC | | | |
| A-CHILLER | | | |
| B-CHILLER | | | |

**CONTROL ROOM INSTRUMENTATION STATUS**

| PARAMETER | AVAILABLE | UNAVAILABLE |
|---|---|---|
| NNI-X | | |
| NNI-Y | | |
| ICS | | |
| EFIC | | |
| RPS | | |
| ESAS | | |

COMMENTS: _____

## - ATTACHMENT 6 -
## CORE DAMAGE ASSESSMENT

Determine if core damage has occurred using one or more of the following methods. Estimate the extent of the damage. Evaluate the status of the fission product barriers. Report the results of the evaluation to the EOF Technical Support Operations Representative and the Dose Assessment Team Leader. Continue to reassess core and Fission Product Barrier status as conditions change.

☐ ESTIMATE CORE DAMAGE BASED ON RCS SAMPLES.

Core damage assessment based on Reactor Coolant samples is evaluated by the Dose Assessment Team using CH-632A Enclosure 2. The results are submitted to the TST. (May take >2 hours to obtain results)

☐ ESTIMATE CORE DAMAGE BASED ON RM-G29/30 RADIATION LEVELS.

**NOTE: (1) Use of RM-G29/30 for determining core status requires a failure of the RCS (i.e., LOCA or PORV open).**

**(2) Low monitor reading does NOT necessarily indicate lack of core damage. The release from the core may bypass the Containment, may be retained in the RCS, may be over a long period of time, or may NOT be uniformly mixed.**

**(3) Inconsistent readings may be due to the uneven mixing in the Containment (e.g., steam rising to the top).**
**IT MAY TAKE SEVERAL HOURS FOR UNIFORM MIXING.**

**ASSUMPTIONS:**

The table below assumes a short release. A long-term release cannot be characterized using these tables.

| TIME | ____:____ | ____:____ | ____:____ | ____:____ | ____:____ |
|---|---|---|---|---|---|
| RM-G29 | R/HR | R/HR | R/HR | R/HR | R/HR |
| RM-G30 | R/HR | R/HR | R/HR | R/HR | R/HR |

☐ NO CORE DAMAGE
< 100 R/HR

☐ POSSIBLE CLAD FAILURE AND GAS GAP RELEASE
100 - 25,000 R/HR WITH RB SPRAY
100 - 75,000 R/HR WITHOUT RB SPRAY

☐ POSSIBLE CORE MELTING
> 25,000 R/HR WITH RB SPRAY
> 75,000 R/HR WITHOUT RB SPRAY

- ATTACHMENT 6 (Continued) -

## CORE DAMAGE PROGRESSION ONCE UNCOVERED

☐ IF inadequate subcooling margin exists,
THEN determine if the core is uncovered.

Reactor Coolant Inventory Tracking System (RCITS) provides a continuous indication of reactor vessel head and hot leg coolant inventory trend with the reactor coolant pumps in operation or tripped. RCITS consists of an RCS Hot Leg Level Subsystem, Reactor Vessel Level Subsystem and RC Void Trending Subsystem.

The RCS Hot Leg Level Subsystem (RC-163A/B-LR1) can monitor the top of the hot leg to the bottom of the hot leg with zero flow conditions. The Reactor Vessel Level Subsystem (RC-164A/B-LR1) can monitor the top of the reactor vessel to the bottom of the hot leg with zero flow conditions. The bottom of the hot leg is approximately two feet above the top of the fuel. An off-scale low reading would indicate a high probability of loss of level below core level. Any flow (including natural circulation) in the RCS will result in a lower than actual reading. Thus, any indicated level will provide assurance that coolant level is above the core.

The Reactor Void Trend Subsystem (RC-169-XR) monitors void trends in the RCS when RCPs are running. RCP motor power and Tcold are used to infer average density of fluid passing through the pump (liquid or two-phase). A 0% reading infers NO voiding, while a 100% reading infers complete voiding.

Recorders are on the PSA panel in the Control Room and display on RECALL (points 62, 63, 64, 65, 70, 71).

| A-HOT LEG | B-HOT LEG | A-VESSEL | B-VESSEL | VOID TREND |
|---|---|---|---|---|
| RC-163A-LR1 | RC-163B-LR1 | RC-164A-LR1 | RC-164B-LR1 | RC-169-XR |
| RECALL PT 63 | RECALL PT 70 | RECALL PT 62 | RECALL PT 65 | RECALL PT 64,71 |
| | | | | |

☐ CORE REMAINS COVERED
TINCORE indicates saturated conditions
RCITS indicates any level

☐ UNCOVERED FOR 15 TO 45 MINUTES
Core temperature 1800-2400°F
Fuel cladding failure (occurred in 34 minutes at Three Mile Island)
Rapid hydrogen generation
Release of fission products out of fuel pin gap (gas gap failure)
Local fuel melt

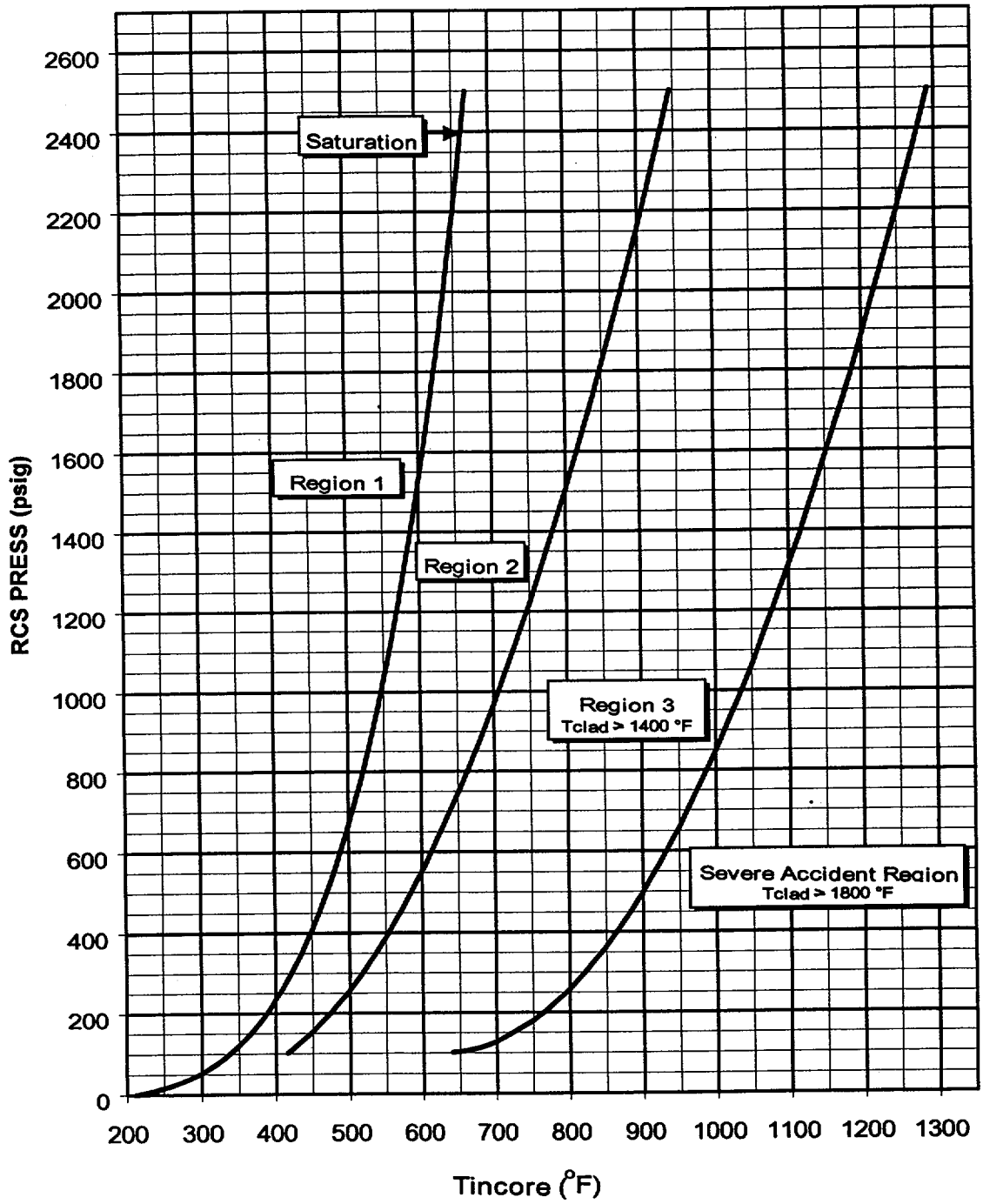
☐ UNCOVERED FOR 30 TO 90 MINUTES
Core temperature 2400-4200°F
Possible uncoolable core
Possible slump of molten core
Rapid release of volatile fission products (grain boundary release)

☐ UNCOVERED FOR 1 TO 3+ HOURS
Core temperature > 4200°F
Maximum core melt and hydrogen generation
Maximum in-vessel fission product release
Possible melt-through of vessel

- ATTACHMENT 6 (Continued) -

## CORE DAMAGE ASSESSMENT BASED ON ICC CURVE

☐ ASSESS CORE DAMAGE BY PLOTTING RCS PRESSURE/INCORE TEMPERATURE ON THE ICC CURVE BELOW.
- ☐ Regions 1 and 2 indicate no fuel damage (normal RCS activity).
- ☐ Region 3 indicates possible gas gap failure.
- ☐ Severe Accident Region indicates possible core melt.



- ATTACHMENT 7 -

## DOSE ASSESSMENT TEAM NOTIFICATION

1. The Technical Support Team is responsible for supplying the Dose Assessment Team with an evaluation of the accident type, the release pathway, and the release flow rate. The accident type affects the radionuclide distribution (i.e., percentage of each isotope) used by Dose Assessment to predict off-site doses.
2. The accident type is determined by physical parameters and instrument readings throughout the plant.
3. Complete the checklist below to the extent possible and give to the Dose Assessment Team Leader.

**ACCIDENT TYPE**

| | | |
|---|---|---|
| _____ **LOCA** | _____ **Waste Gas Decay Tank Rupture** | _____ **OTSG Tube Leak** |
| _____ **Fuel Handling** | _____ **Other:**_____ | |

☐ **LOSS-OF-COOLANT ACCIDENT** **TIME OF RX TRIP:**_____

a. Normal Activity ____ Clad Damage ____ Fuel melt ____ (from Attachment 7)
b. Release pathway information (leak from where to where) _____
c. Release path flow rate (estimated for unmonitored releases) _____
d. Estimated duration _____ Unknown _____
e. Reactor Building spray on/off times _____
f. Auxiliary Building Ventilation (W351): Flow rate _____ Charcoal banks in service A: B: C: D
g. Containment pressure (X147) _____ PSIG
h. Loose Parts Monitor indications No ___ Yes___ Location: _____

☐ **WASTE GAS DECAY TANK RUPTURE**

a. Release pathway: Tank rupture _____ Valve leakage _____ Other_____
b. Tank volume _____ pressure_____
c. Release rate: Unknown_____ Estimate _____ CFM
d. Estimated duration: Unknown_____ Time _____
e. Auxiliary Building Ventilation (W351): Flow rate _____ Charcoal banks in service A: B: C: D

☐ **STEAM GENERATOR TUBE RUPTURE** **TIME OF RX TRIP:** _____

a. Primary-to-secondary leak rate: _____ gpm
b. Core status: Cladding damage ____ Fuel melt ____ Normal ____
c. Leaking OTSG isolated: Yes _____ No _____
d. MSSV Open: Yes____ No____ ADV Open: Yes ____ No ____
e. Condenser vacuum: Yes ____ No ____ RM-A2 In Service?: Yes ____ No ____
f. Potential for change in status of leak: Yes _____ No _____
g. Estimated duration of leak: _____
h. Auxiliary Building Ventilation (W351): Flow rate _____ Charcoal banks in service A: B: C: D

☐ **FUEL HANDLING ACCIDENT**

a. Location of damaged fuel: Pool A ____ Pool B ____ Number of Elements _____
b. Damage caused by: Mechanical impact _____ Overheating _____ Unknown _____
c. Auxiliary Building Ventilation (W351): Flow rate _____ Charcoal banks in service A: B: C: D
d. Release pathway: _____ Unknown _____
e. Estimated duration _____ Unknown _____

Status as of _____ Date: _____ Completed By: _____

## - ATTACHMENT 8 -
## SHORT-TERM RECOVERY PLAN GENERIC OUTLINE

### PHASE I - INCIDENT STABILITY

1. Verify Security System integrity.
2. Assess integrity of systems required for long-term cooling by system walkdown:
   - Decay Heat
   - Spent Fuel
   - Ventilation
3. Continue cooldown using an appropriate heat removal method.
4. Verify termination of release.

### PHASE II - DATA GATHERING

1. Auxiliary Building Filter Changeout and Analysis
2. Plant and Off-Site Radiation Surveys and Dose Assessments
3. Primary System and RB Atmosphere Sampling
4. Debrief key personnel.
5. Equipment inspection/develop damage report:
   - Emergency Feedwater System (including electrical)
   - Makeup System (HPI Valve)
   - PORV and Block Valves
   - Fuel Handling Area
   - Diesel Generator
6. Community Reaction Survey
7. Develop detailed incident report.
8. Establish whole body counting capability for emergency workers.

- ATTACHMENT 8 (Continued) -

## SHORT-TERM RECOVERY PLAN GENERIC OUTLINE

### PHASE III - RESTORATION

Based on results of Phase II assessment:

1. Prepare procedures as required.
2. Begin repair efforts.
3. Establish team for system cleanup and waste disposal activities.
4. Establish community educational and public relations activities.
5. Establish Recovery Team organization and off-site support liaison.
6. Re-establish normal site operations.
7. Establish claim office.
8. Assure regulatory communication.
9. Establish technical assessment team (FPC, Framatome Technologies, other Architect/Engineer, etc.).
10. Develop long-term organizational recovery responsibilities and plant status objectives.

**NOTE:** The completed recovery plan and implementing procedures shall be submitted to the PNSC for approval before implementation.

**- ATTACHMENT 9 -**

## EQUIPMENT INSTRUCTIONS

### Accident Assessment Ringdown Speaker

1. Ensure the speaker is connected to phone jack 124-D5.
2. Turn on power using the knob on the front of the speaker.
3. Activate the speaker circuit by removing receiver from the Accident Assessment Ringdown phone in the Simulator Control Room. This phone must be off the hook for the Room 124 speaker to function.

### SPDS

1. Ensure both computers and monitors designated for SPDS in northwest corner of Room 124 are turned on.

2. In the "Access Control Client" dialog box, ensure "CR3 SPDS" is selected in the drop-down box (or "SIM SPDS" for drills).

3. Click "LOGON."

4. Press "ALT TAB" until "SPDS" is selected.

### TSC/EOF Ringdown Phone

The TSC/EOF Ringdown phone is located in southwest corner of Room 124. This is an automatic circuit that connects when either the TSC phone or EOF phone is taken off the hook.

- ATTACHMENT 5 -

## PROCEDURE CHANGES, REASONS, AND REFERENCES

**PROCEDURE NO.:** __REP-12____ **TITLE:**__EOF ACCIDENT ASSESSMENT PROCEDURE_____

**DESCRIPTION OF NEW PROCEDURE OR CHANGE (detailed)**

Changed the FPB Assessment table of Attachment 6 to be consistent with EM-202 and EM-225 regarding prolonged steaming and OTSG with > 10 gpm leak to atmosphere. Changed header title information to reflect group name. Deleted reference to REP-02 as it is being cancelled. Changed REP-3 to REP-03A for setup from checklists. Changed name of EOF Accident Assessment Team to Technical Support Team to reflect function of team. Changed team member titles to reflect team name change. Deleted EOF Risk Analysis Representative from team and added responsibilities to EOF Technical Support Engineer, deleted EOF Risk Analysis Representative checklist and renumbered remaining attachments accordingly. Changed Manager, Radiological Emergency Planning to Supervisor, Emergency Preparedness. Changed Generating Complex in 4.1.4 to Energy Complex for consistency with other EP procedures. Changed REP abbreviation to EP and PRC to PNSC. Added Technical Support Team (TST) to abbreviations. Added computer points for containment pressure and AB ventilation flowrate to Attachment 8 per drill participant comment. Other editorial changes.

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**REASON AND REFERENCES FOR PROCEDURE CHANGE**

General housekeeping, address drill participant comments, delete cancelled reference, update organizational titles due to combined companies, and changed team name to reflect function.

| | DATE |
|---|---|
| **ORIGINATED BY:** S D Mansfield | 07/18/01 |
| **REVIEWED BY (IF APPLICABLE):** | |
| **SEP APPROVAL:** John D Stephenson | 7/18/01 |

# EMERGENCY OPERATIONS FACILITY TECHNICAL SUPPORT TEAM

## REQUIRED APPROVALS

John D Stephenson

Supervisor, Emergency Preparedness

7/24/01

Date

## 1.0 PURPOSE

This procedure provides guidance for establishment and operation of the EOF Technical Support Team (TST).

## 2.0 SCOPE

2.1 The EOF Technical Support Team primarily functions as a technical resource for the EOF Technical Support Director and the EOF Director in development of public PARs. Duties also include assisting the TSC Accident Assessment Team as needed in development of mitigation strategies and in research of solutions to plant problems (however, the EOF TST personnel are NOT considered Severe Accident Management evaluators), interfacing with the Dose Assessment Team, and assisting in the development of recovery plans.

2.2 A Safety Assessment was performed for this procedure. A determination was made that this procedure is outside the scope of 10 CFR 50.59.

## 3.0 APPLICABLE REFERENCE DOCUMENTS

3.1.1 REP-03A, "Setup of the Emergency Operations Facility"

3.1.2 EM-225, "Duties of the Technical Support Center Accident Assessment Team"

3.1.3 EM-225A, "Post-Accident RB Hydrogen Control"

3.1.4 EM-225B, "Post-Accident Boron Concentration Management"

3.1.5 EM-225C, "Post-Accident Monitoring of Reactor Building Temperature"

3.1.6 EM-225D, "Guidance for Dry OTSG Tube to Shell Delta T Monitoring and Control"

3.1.7 EM-225E, "Guidelines for Long-Term Cooling"

3.1.8 Emergency Operating Procedures (EOPs)

3.1.9 MP-575, "Hydrogen Recombiner Installation"

3.1.10 OP-417B, "Operation of the Post-Accident Hydrogen Recombiners"

3.1.11 B&W Technical Bases Document

3.1.12 CR-3 Severe Accident Guideline

3.2.13 Response Technical Manual (RTM-96), USNRC, Vol. 1, Rev. 3

3.2.14 Radiological Emergency Response Plan

3.2.15 NUREG-1228, "Source Term Estimation During Incident Response to Severe Nuclear Power Plant Accidents"

3.2.16 FPC IOC CR97-0122, dated 12/23/97

3.2.17 CH-632A, "Post-Accident Sampling and Analysis of the Reactor Coolant System"

## 4.0 DEFINITIONS AND ABBREVIATIONS

### 4.1 DEFINITIONS

#### 4.1.1 Technical Support Team (TST)

Consists of EOF Technical Support Coordinator, EOF Technical Support Engineer, and EOF Technical Support Operations Representative.

#### 4.1.2 Candidate High Level Actions (CHLAs)

Actions described in the CR-3 Severe Accident Guideline that could be taken to mitigate a severe accident and are deemed appropriate based on PDCs.

#### 4.1.3 Critical Safety Functions (CSFs)

Those functions needed to ensure adequate core cooling and to preserve the integrity of the fission product barriers, thereby protecting the health and safety of the general public and plant personnel. They include reactivity control, coolant inventory control, decay heat removal capability, fission product barrier status, electrical power availability, and Control Room status.

#### 4.1.4 Emergency Action Levels (EALs)

Conditions or indications that may be used as thresholds for initiating specific emergency measures. (See EM-202, Enclosure 1)

4.1.5 **Protective Action Recommendations (PARs)**

Emergency measures recommended for purposes of preventing or minimizing radiological exposures to Energy Complex personnel or members of the public.

4.1.6 **Plant Damage Conditions (PDCs)**

Damage conditions used in the CR-3 Severe Accident Guideline to describe the status of the RCS, reactor core, and the containment during the progression of a severe accident.

4.1.7 **Severe Accident**

An accident (beyond that assumed in the CR-3 design and licensing basis) that results in catastrophic fuel rod failure, core degradation and fission product release into the reactor vessel, RB, or environment.

4.2 **ABBREVIATIONS**

AAT - Accident Assessment Team
ADV - Atmospheric Dump Valve
AP - Abnormal Procedure
BSP - Building Spray Pump
BWST - Borated Water Storage Tank
CFM - Cubic Feet per Minute
CFT - Core Flood Tank
CHLA - Candidate High Level Action
CR-3 - Crystal River Unit 3
CSF - Critical Safety Function
DCP - Decay Heat Closed Cycle Cooling System Pump
DHP - Decay Heat Pump
EAL - Emergency Action Level
ECCS - Emergency Core Cooling System
EDG or EGDG - Emergency Diesel Generator
EFIC - Emergency Feedwater Initiation and Control
EFP - Emergency Feedwater Pump
EFW - Emergency Feedwater
EOF - Emergency Operations Center
EOP - Emergency Operating Procedure
EP - Emergency Preparedness
FPC - Florida Power Corporation
HPI - High Pressure Injection
ICC - Inadequate Core Cooling
IOC - Inter-Office Communication

LOCA - Loss of Coolant Accident
MFW - Main Feedwater
MSSV - Main Steam Safety Valve
OTSG - Once-Through Steam Generator
PAR - Protective Action Recommendation
PASS - Post-Accident Sampling System
PDC - Plant Damage Condition
PORV - Pilot-Operated Relief Valve
PNSC - Plant Nuclear Safety Committee
PSIG - Pounds per Square Inch (Gauge)
QC - Quality Control
RB - Reactor Building
RCITS - Reactor Coolant Inventory Tracking System
RCP - Reactor Coolant Pump
RCS - Reactor Coolant System
RPS - Reactor Protection System
RWP - Raw Water Pump
Rx - Reactor
SPDS - Special Parameter Display System
SWP - Nuclear Services Closed Cycle Cooling System Pump
TSC - Technical Support Center
TST – Technical Support Team
USNRC - U.S. Nuclear Regulatory Commission
XFRM - Transformer

## 5.0 RESPONSIBILITIES AND ACTIONS

### 5.1 RESPONSIBILITIES

#### 5.1.1 EOF Technical Support Director

Notifies the EOF Technical Support Coordinator and the Corporate Health Physicist that the EOF has been activated and provides direction and control to the EOF functions associated with accident mitigation and assessment.

#### 5.1.2 EOF Technical Support Coordinator

a. Keeps the EOF Technical Support Director informed of TST activities and developments in plant status, especially those that may impact EALs and PARs.

b. Notifies the EOF Technical Support Operations Representative and EOF Technical Support Engineer that the EOF has been activated and provides coordination and support for the TST.

c. Ensures communication is established with the TSC on the TSC/EOF Ringdown until EOF Communicator arrives.

d. Assists in the setup of the EOF Technical Support Work Area.

e. Performs "plant conditions" portion of the EOF briefings using the briefing guidelines provided in Attachment 4.

f. Monitors the Accident Assessment Ringdown (communication between the Control Room and TSC AAT).

g. Provides support to the TSC AAT in determining the causes and consequences of the emergency.

h. Ensures interface is established with the EOF Dose Assessment Team using Section 5.1.5 and Attachment 7 as guidance.

i. Refers to attachments for additional accident assessment guidance and information.

j. Notifies Simulator support personnel when necessary (e.g., for testing mitigation strategies).

k. Monitors CSFs and provides status to EOF Technical Support Director and EOF personnel during briefings, as needed.

5.1.3 **EOF Technical Support Operations Representative**

a. Verifies that the SPDS computer is properly set up and operational.

b. Operates the SPDS computer.

c. Monitors plant parameters and provides status updates to the EOF Technical Support Coordinator.

d. Monitors communications between the Control Room and the TSC AAT via speaker in the EOF Technical Support Room.

e. Assists in the setup of the EOF Technical Support Work Area.

5.1.4 **EOF Technical Support Engineer**

a. Assesses plant conditions and provides Engineering support for developing accident mitigation strategies as needed.

b. Provides Engineering support to the TSC Technical Support Team.

c. Notifies additional Engineering resources when necessary.

d. Assists in the setup of the EOF Technical Support Work Area.

5.1.5 **Dose Assessment Team**

a. Supports the TSC Accident Assessment Team with on-site radiological data and with chemical and radiological analysis of samples as needed to assess the accident.

b. Provides plant radiation monitor readings and assessments.

c. Provides projected radiological data (on-site and off-site doses, dose rates, and deposition) (> 1 hour to obtain).

d. Provides RCS PASS data (> 2 hours to obtain results) on radionuclide composition, Chloride concentration, dissolved Hydrogen concentration, and Boron concentration.

e. Provides Reactor Building and/or Auxiliary Building atmosphere radionuclide composition (> 1 hour to obtain).

5.2 **INSTRUCTIONS**

5.2.1 The EOF Technical Support Coordinator or designee performs the duties of Attachment 1.

5.2.2 The EOF Technical Support Operations Representative performs the duties of Attachment 2.

5.2.3 The EOF Technical Support Engineer performs the duties of Attachment 3.

## 6.0 INTERPRETATION CONTACT

Supervisor Emergency Preparedness

## 7.0 REVISION HISTORY

| | | |
|---|---|---|
| Rev. 0 | Date: 01/26/99 | Initial Issue |
| Rev. 1 | Date: 10/09/00 | General revision to update attachments to match EM-225. |
| Rev. 2 | Date: 07/18/01 | NUPOST comment to make FPB Assessment table consistent with EM-202 & EM-225. Changed name of team to Technical Support Team. Other changes include organizational titles, deletion of cancelled reference and consistency-type changes. |

**- ATTACHMENT 1 -**

## EOF TECHNICAL SUPPORT COORDINATOR CHECKLIST

**Check**

_____ 1. Perform telephone notifications (Refer to Emergency Response Personnel Roster):

___ EOF Technical Support Operations Representative
___ EOF Technical Support Engineer (3)

_____ 2. Check-in with EOF Director upon arrival and ensure name is posted on staffing board.

_____ 3. Ensure Accident Assessment Ringdown speaker is functional. (Refer to Attachment 9) (This is for monitoring only. Notify the TSC AAT by plant extension as necessary.)

_____ 4. Ensure CSFs are being assessed. (Refer to Attachment 4) For the initial assessment, this information may be readily available from the TSC AAT. For subsequent evaluations, discuss with the EOF Technical Support Director whether TSC information is sufficient or if independent evaluations are desirable.

_____ 5. Ensure PARs are appropriate per EM-202, Enclosure 8.

_____ 6. Ensure Room 124 SPDS computer is functional. (Refer to Attachment 9)

_____ 7. Ensure communication is established with TSC on TSC/EOF Ringdown, until EOF Communicator arrives. (Refer to Attachment 9)

_____ 8. Ensure EOP/AP procedure book is retrieved from Operations Briefing Room (A117) or Simulator Control Booth (A115).

_____ 9. Brief the EOF Technical Support Director on CSFs and plant status.

_____ 10. Provide CSFs and status reports to the EOF Staff during briefings. Use guidelines on Attachment 4 as necessary.

_____ 11. Establish interface with EOF Dose Assessment Team. Attachment 7 lists information needed by the Dose Assessment Team; Section 5.1.5 lists information available from the Dose Assessment Team.

_____ 12. Ensure the Status Board Coordinator is provided current CSFs information in the Main Conference Room.

_____ 13. Notify Simulator personnel for support, if needed (e.g., for testing mitigation strategies).

_____ 14. Provide assistance to EOF Director on recovery plan. (Refer to Attachment 8)

- ATTACHMENT 2 -

## EOF TECHNICAL SUPPORT OPERATIONS REPRESENTATIVE CHECKLIST

**Check**

_____ 1. Confirm SPDS computer is operational. (Refer to Attachment 9)

_____ 2. Ensure work area is set up and functional. (Refer to REP-03A, Attachment 1)

_____ 3. Provide plant parameters/status to EOF Technical Support Coordinator.

_____ 4. Ensure Accident Assessment Ringdown speaker is functional. (Refer to Attachment 9) (This is for monitoring only. Notify the TSC AAT by plant extension as necessary.)

- ATTACHMENT 3 -

## EOF TECHNICAL SUPPORT ENGINEER CHECKLIST

**Check**

_____ 1. Ensure work area is set up and functional. (Refer to REP-03A, Attachment 1)

_____ 2. Establish communication with TSC AAT by plant phone as necessary

_____ 3. Obtain applicable drawings/documents, as needed.

_____ 4. Establish additional Engineering resources, if necessary.

_____ 5. Provide assistance for recovery plan development. (Refer to Attachment 8)

_____ 6. Obtain reactor core status. (Refer to Attachment 6)

_____ 7. Provide initial core assessment to EOF Technical Support Coordinator.

_____ 8. Provide core status updates to EOF Technical Support Coordinator.

_____ 9. Complete Attachment 7 for the Dose Assessment Team and provide to the EOF Technical Support Coordinator.

_____ 10. Notify additional resources (engineers, etc.) for core damage assessment, if needed.

_____ 11. If requested by the TSC to perform an assessment of core damage based on reactor coolant sample, use Enclosure 2 from CH-632A.

**- ATTACHMENT 4 -**
**BRIEFING GUIDELINE**

Refer to Attachments 5 and 6 to aid in this evaluation.

I REACTOR SHUTDOWN — Yes ☐ No ☐

II. CORE ADEQUATELY COOLED — Yes ☐ No ☐

III. FISSION PRODUCT BARRIERS ASSESSMENT

| | | | | |
|---|---|---|---|---|
| Fuel Clad | ☐ Intact | ☐ Challenged | ☐ Lost | ☐ Regained |
| RCS | ☐ Intact | ☐ Challenged | ☐ Lost | ☐ Regained |
| Containment | ☐ Intact | ☐ Challenged | ☐ Lost | ☐ Regained |

IV. EMERGENCY ELECTRICAL POWER STATUS

| | | |
|---|---|---|
| Off-Site Power Available? | Yes ☐ | No ☐ |
| ES Buses Energized? | Yes ☐ | No ☐ |
| Emergency Diesel Generator's Available? | Yes ☐ | No ☐ |
| DC Power Available? | Yes ☐ | No ☐ |

V. CONTROL COMPLEX STATUS

| | | |
|---|---|---|
| Ventilation/Cooling Available? | Yes ☐ | No ☐ |
| Necessary Instrumentation Available?* | Yes ☐ | No ☐ |

VI. OTHER CONDITIONS/CHALLENGES

_____

_____

_____

_____

_____

_____

_____

_____

* Necessary refers to specific instruments and annunciators that are needed to identify, diagnose, and track the problems that are causing the emergency.

- ATTACHMENT 5 -

## CRITICAL SAFETY FUNCTION CHECKLIST

Monitor the parameters associated with the Critical Safety Functions. The parameter tables below are for reference only. It is NOT intended that the tables be completed during each evaluation. Plant computer point numbers or SPDS/RECALL point numbers are listed, if available.

Using pre-established RECALL Groups based on accident type in progress is recommended.

**Notify the Technical Support Coordinator immediately if any of the CSFs cannot be verified.**

## I. REACTOR SHUTDOWN STATUS:

### REACTIVITY CONTROL

| PARAMETER | COMPUTER POINT | RECALL POINT | | | |
|---|---|---|---|---|---|
| All Rods at in-limits Y/N | P057 | RECL-375 | | | |
| Intermediate Range detector NI-3 amps | P212 | RECL-150 | | | |
| Intermediate Range detector NI-4 amps | P213 | RECL-151 | | | |
| Source Range NI-1 cps | P202 | RECL-152 | | | |
| Source Range NI-2 cps | P203 | RECL-153 | | | |
| Low Range NI-14/15 | | RECL-102,103 | | | |
| Adequate Shutdown Margin | OP-103C Curve 18&19 | | | | |

- ATTACHMENT 5 (Continued) -

# II. CORE COOLING STATUS:

## ECCS/SUPPORT STATUS

| PARAMETER | COMPUTER POINT | RECALL POINT | | | |
|---|---|---|---|---|---|
| Subcooling Margin | M114 | | | | |
| A HPI Pump operating | | RECL-209 | | | |
| B HPI Pump operating | | RECL-210 | | | |
| C HPI Pump operating | | RECL-211 | | | |
| MUV-23 flow | W704 | RECL-52 | | | |
| MUV-24 flow | W706 | RECL-54 | | | |
| MUV-25 flow | W703 | RECL-51 | | | |
| MUV 26 flow | W705 | RECL-53 | | | |
| DHPs operating A/B (run/stop) | X063<br>X064 | RECL-207<br>RECL-208 | | | |
| DHP-1A flow | W409 | RECL-55 | | | |
| DHP-1B flow | W410 | RECL-56 | | | |
| CFT A level | P200 | | | | |
| CFT B level | P201 | | | | |
| CFT A press | | | | | |
| CFT B press | | | | | |
| BWST level (ft) | X335 | RECL-57 | | | |
| RWPs operating 1/2A/2B/3A/3B | | | | | |
| DCPs operating A/B (yes/no) | | | | | |
| SWPs operating A/B/C | | | | | |

## SECONDARY SYSTEM STATUS

| PARAMETER | COMPUTER POINT | RECALL POINT | | | |
|---|---|---|---|---|---|
| EFIC OTSG A press | W449 | RECL-252 | | | |
| EFIC OTSG B press | W452 | RECL-255 | | | |
| OTSG A level | S285 | RECL-92 | | | |
| OTSG B level | S286 | RECL-93 | | | |
| MFW flow A | S301 | RECL-100 | | | |
| MFW flow B | S302 | RECL-101 | | | |
| EFPs operating 1/2/3/7 | | | | | |
| EFW flow to A OTSG | S300 | RECL-245 | | | |
| EFW flow to B OTSG | S312 | RECL-247 | | | |

- ATTACHMENT 5 (Continued) –

## III. FISSION PRODUCT BARRIER ASSESSMENT:

### FUEL CLADDING BARRIER

| ☐ INTACT | ☐ CHALLENGED | ☐ LOST | ☐ REGAINED |
|---|---|---|---|
| • No indication of cladding damage | • RCS condition warrant entry into EOP-07<br>• Core Exit Thermocouples > 700 degrees F | • RCS conditions in (or previously in) Region 3 or Severe Accident Region<br>• PASS indicates increased RCS activity >300μCi/gr $I_{131}$ (refer to CH-632A)<br>• RM-G29/30 > 100 R/hr for ≥ 15 minutes<br>• Attachment 6 indicates failed fuel | • Cooling restored, no further degradation expected. |

### REACTOR COOLANT SYSTEM BARRIER

| ☐ INTACT | ☐ CHALLENGED | ☐ LOST | ☐ REGAINED |
|---|---|---|---|
| • Leakage is within normal makeup pump capacity | • RCS leak or OTSG tube leak requiring one or more injection valves to maintain adequate subcooling margin<br>• RCS pressure /Tincore relationship violates NDT limits<br>• RCS leak or OTSG tube leak results in ES actuation on low RCS pressure.<br>• HPI/PORV or HPI/Code Safety valve cooling is in progress | • RCS leak resulting in loss of adequate subcooling margin<br>• OTSG Tube Rupture resulting in loss of adequate subcooling margin<br>• RM-G29/30 >10R/hr for ≥ 15 minutes | • HPI/PORV or HPI Code Safety Cooling stopped<br>• Subcooling Margin restored and leak isolated |

### CONTAINMENT BARRIER

| ☐ INTACT | ☐ CHALLENGED | ☐ LOST | ☐ REGAINED |
|---|---|---|---|
| • No evidence of containment leakage<br>• Tube rupture release is only through condenser | • RB pressure > 54 psig<br>• RB hydrogen concentration > 4%<br>• RB pressure > 30 psig with no building spray available<br>• RMG-29 or 30 reading > 25,000 R/hr<br>• Core conditions in severe accident region of ICC curves for >15 min | • Containment isolation is incomplete and release path to environment exists.<br>• OTSG Tube Rupture > 10 gpm exists and prolonged steaming to atmosphere or an unisolable steam leak outside RB from affected OTSG.<br>• Containment pressure or sump level response not consistent with LOCA conditions<br>• Rapid unexplained RB pressure decrease following an initial increase | • Repair efforts have isolated leak<br>• Containment pressure has reduced to stop leakage |

Performed By: _____ Date: _____ Time: _____

- ATTACHMENT 5 (Continued) -

# IV. EMERGENCY ELECTRICAL POWER STATUS:

## OFF-SITE POWER

| PARAMETER | AVAILABLE | UNAVAILABLE |
|---|---|---|
| 500 KV SWITCHYARD | | |
| 230 KV SWITCHYARD | | |
| OFF-SITE POWER XFRM | | |
| BEST | | |

## ES BUSES

| PARAMETER | AVAILABLE | UNAVAILABLE |
|---|---|---|
| A-ES 4160V BUS | | |
| B-ES 4160V BUS | | |
| A-ES 480V BUS (Note 1) | | |
| B-ES 480V BUS (Note 1) | | |

## EMERGENCY DIESEL GENERATOR

| PARAMETER | RECALL PT. | LOADED | AVAILABLE | UNAVAILABLE |
|---|---|---|---|---|
| A-EDG | RECL-133,171 | | | |
| B-EDG | RECL-134,172 | | | |

## DC ELECTRICAL

| PARAMETER Note (1) | AVAILABLE | UNAVAILABLE |
|---|---|---|
| A-BATTERY | | |
| B-BATTERY | | |
| C-BATTERY | | |

Note (1) Battery failure will occur if associated battery chargers are de-energized.

- ATTACHMENT 5 (Continued) -

# V. CONTROL COMPLEX STATUS:

**CONTROL COMPLEX VENTILATION STATUS**

| PARAMETER | AVAILABLE | OPERATING | UNAVAILABLE |
|---|---|---|---|
| A-TRAIN EMERGENCY RECIRC | | | |
| B-TRAIN EMERGENCY RECIRC | | | |
| A-CHILLER | | | |
| B-CHILLER | | | |

**CONTROL ROOM INSTRUMENTATION STATUS**

| PARAMETER | AVAILABLE | UNAVAILABLE |
|---|---|---|
| NNI-X | | |
| NNI-Y | | |
| ICS | | |
| EFIC | | |
| RPS | | |
| ESAS | | |

COMMENTS: _____

- ATTACHMENT 6 -

## CORE DAMAGE ASSESSMENT

Determine if core damage has occurred using one or more of the following methods. Estimate the extent of the damage. Evaluate the status of the fission product barriers. Report the results of the evaluation to the EOF Technical Support Operations Representative and the Dose Assessment Team Leader. Continue to reassess core and Fission Product Barrier status as conditions change.

☐ ESTIMATE CORE DAMAGE BASED ON RCS SAMPLES.

Core damage assessment based on Reactor Coolant samples is evaluated by the Dose Assessment Team using CH-632A Enclosure 2. The results are submitted to the TST. (May take >2 hours to obtain results)

☐ ESTIMATE CORE DAMAGE BASED ON RM-G29/30 RADIATION LEVELS.

**NOTE: (1) Use of RM-G29/30 for determining core status requires a failure of the RCS (i.e., LOCA or PORV open).**

**(2) Low monitor reading does NOT necessarily indicate lack of core damage. The release from the core may bypass the Containment, may be retained in the RCS, may be over a long period of time, or may NOT be uniformly mixed.**

**(3) Inconsistent readings may be due to the uneven mixing in the Containment (e.g., steam rising to the top).**
**IT MAY TAKE SEVERAL HOURS FOR UNIFORM MIXING.**

**ASSUMPTIONS:**

The table below assumes a short release. A long-term release cannot be characterized using these tables.

| TIME | ____:____ | ____:____ | ____:____ | ____:____ | ____:____ |
|---|---|---|---|---|---|
| RM-G29 | R/HR | R/HR | R/HR | R/HR | R/HR |
| RM-G30 | R/HR | R/HR | R/HR | R/HR | R/HR |

☐ NO CORE DAMAGE
< 100 R/HR

☐ POSSIBLE CLAD FAILURE AND GAS GAP RELEASE
100 - 25,000 R/HR WITH RB SPRAY
100 - 75,000 R/HR WITHOUT RB SPRAY

☐ POSSIBLE CORE MELTING
> 25,000 R/HR WITH RB SPRAY
> 75,000 R/HR WITHOUT RB SPRAY

- ATTACHMENT 6 (Continued) -

## CORE DAMAGE PROGRESSION ONCE UNCOVERED

☐ IF inadequate subcooling margin exists,
THEN determine if the core is uncovered.

Reactor Coolant Inventory Tracking System (RCITS) provides a continuous indication of reactor vessel head and hot leg coolant inventory trend with the reactor coolant pumps in operation or tripped. RCITS consists of an RCS Hot Leg Level Subsystem, Reactor Vessel Level Subsystem and RC Void Trending Subsystem.

The RCS Hot Leg Level Subsystem (RC-163A/B-LR1) can monitor the top of the hot leg to the bottom of the hot leg with zero flow conditions. The Reactor Vessel Level Subsystem (RC-164A/B-LR1) can monitor the top of the reactor vessel to the bottom of the hot leg with zero flow conditions. The bottom of the hot leg is approximately two feet above the top of the fuel. An off-scale low reading would indicate a high probability of loss of level below core level. Any flow (including natural circulation) in the RCS will result in a lower than actual reading. Thus, any indicated level will provide assurance that coolant level is above the core.

The Reactor Void Trend Subsystem (RC-169-XR) monitors void trends in the RCS when RCPs are running. RCP motor power and Tcold are used to infer average density of fluid passing through the pump (liquid or two-phase). A 0% reading infers NO voiding, while a 100% reading infers complete voiding.

Recorders are on the PSA panel in the Control Room and display on RECALL (points 62, 63, 64, 65, 70, 71).

| A-HOT LEG | B-HOT LEG | A-VESSEL | B-VESSEL | VOID TREND |
|---|---|---|---|---|
| RC-163A-LR1 | RC-163B-LR1 | RC-164A-LR1 | RC-164B-LR1 | RC-169-XR |
| RECALL PT 63 | RECALL PT 70 | RECALL PT 62 | RECALL PT 65 | RECALL PT 64,71 |
| | | | | |

☐ CORE REMAINS COVERED
TINCORE indicates saturated conditions
RCITS indicates any level

☐ UNCOVERED FOR 15 TO 45 MINUTES
Core temperature 1800-2400°F
Fuel cladding failure (occurred in 34 minutes at Three Mile Island)
Rapid hydrogen generation
Release of fission products out of fuel pin gap (gas gap failure)
Local fuel melt

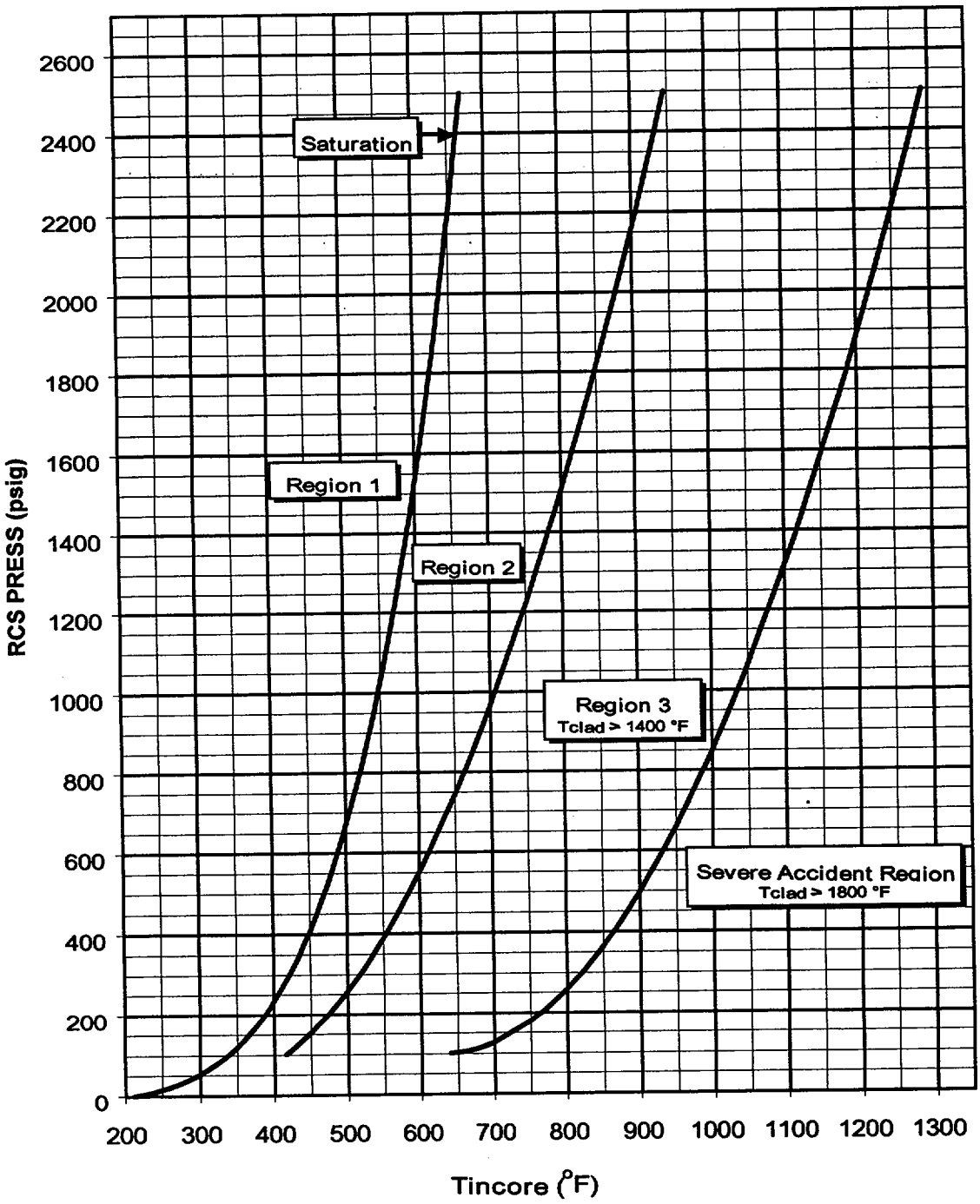
☐ UNCOVERED FOR 30 TO 90 MINUTES
Core temperature 2400-4200°F
Possible uncoolable core
Possible slump of molten core
Rapid release of volatile fission products (grain boundary release)

☐ UNCOVERED FOR 1 TO 3+ HOURS
Core temperature > 4200°F
Maximum core melt and hydrogen generation
Maximum in-vessel fission product release
Possible melt-through of vessel

- ATTACHMENT 6 (Continued) -

## CORE DAMAGE ASSESSMENT BASED ON ICC CURVE

☐ ASSESS CORE DAMAGE BY PLOTTING RCS PRESSURE/INCORE TEMPERATURE ON THE ICC CURVE BELOW.
- ☐ Regions 1 and 2 indicate no fuel damage (normal RCS activity).
- ☐ Region 3 indicates possible gas gap failure.
- ☐ Severe Accident Region indicates possible core melt.



## - ATTACHMENT 7 –
## DOSE ASSESSMENT TEAM NOTIFICATION

1. The Technical Support Team is responsible for supplying the Dose Assessment Team with an evaluation of the accident type, the release pathway, and the release flow rate. The accident type affects the radionuclide distribution (i.e., percentage of each isotope) used by Dose Assessment to predict off-site doses.
2. The accident type is determined by physical parameters and instrument readings throughout the plant.
3. Complete the checklist below to the extent possible and give to the Dose Assessment Team Leader.

**ACCIDENT TYPE**

| | | |
|---|---|---|
| _____ **LOCA** | _____ **Waste Gas Decay Tank Rupture** | _____ **OTSG Tube Leak** |
| _____ **Fuel Handling** | _____ **Other:**_____ | |

☐ **LOSS-OF-COOLANT ACCIDENT** **TIME OF RX TRIP:**_____

a. Normal Activity ____ Clad Damage ____ Fuel melt ____ (from Attachment 7)
b. Release pathway information (leak from where to where) _____
c. Release path flow rate (estimated for unmonitored releases) _____
d. Estimated duration _____ Unknown _____
e. Reactor Building spray on/off times _____
f. Auxiliary Building Ventilation (W351): Flow rate _____ Charcoal banks in service A: B: C: D
g. Containment pressure (X147) _____ PSIG
h. Loose Parts Monitor indications No ___ Yes___ Location: _____

☐ **WASTE GAS DECAY TANK RUPTURE**

a. Release pathway: Tank rupture _____ Valve leakage _____ Other_____
b. Tank volume _____ pressure_____
c. Release rate: Unknown_____ Estimate _____ CFM
d. Estimated duration: Unknown_____ Time _____
e. Auxiliary Building Ventilation (W351): Flow rate _____ Charcoal banks in service A: B: C: D

☐ **STEAM GENERATOR TUBE RUPTURE** **TIME OF RX TRIP:** _____

a. Primary-to-secondary leak rate: _____ gpm
b. Core status: Cladding damage ____ Fuel melt ____ Normal ____
c. Leaking OTSG isolated: Yes _____ No _____
d. MSSV Open: Yes____ No____ ADV Open: Yes ____ No ____
e. Condenser vacuum: Yes ____ No ____ RM-A2 In Service?: Yes ____ No ____
f. Potential for change in status of leak: Yes _____ No _____
g. Estimated duration of leak: _____
h. Auxiliary Building Ventilation (W351): Flow rate _____ Charcoal banks in service A: B: C: D

☐ **FUEL HANDLING ACCIDENT**

a. Location of damaged fuel: Pool A ____ Pool B ____ Number of Elements _____
b. Damage caused by: Mechanical impact _____ Overheating _____ Unknown _____
c. Auxiliary Building Ventilation (W351): Flow rate _____ Charcoal banks in service A: B: C: D
d. Release pathway: _____ Unknown _____
e. Estimated duration _____ Unknown _____

Status as of _____ Date: _____ Completed By: _____

- ATTACHMENT 8 -

## SHORT-TERM RECOVERY PLAN GENERIC OUTLINE

### PHASE I - INCIDENT STABILITY

1. Verify Security System integrity.
2. Assess integrity of systems required for long-term cooling by system walkdown:
   - Decay Heat
   - Spent Fuel
   - Ventilation
3. Continue cooldown using an appropriate heat removal method.
4. Verify termination of release.

### PHASE II - DATA GATHERING

1. Auxiliary Building Filter Changeout and Analysis
2. Plant and Off-Site Radiation Surveys and Dose Assessments
3. Primary System and RB Atmosphere Sampling
4. Debrief key personnel.
5. Equipment inspection/develop damage report:
   - Emergency Feedwater System (including electrical)
   - Makeup System (HPI Valve)
   - PORV and Block Valves
   - Fuel Handling Area
   - Diesel Generator
6. Community Reaction Survey
7. Develop detailed incident report.
8. Establish whole body counting capability for emergency workers.

- ATTACHMENT 8 (Continued) -

## SHORT-TERM RECOVERY PLAN GENERIC OUTLINE

### PHASE III - RESTORATION

Based on results of Phase II assessment:

1. Prepare procedures as required.
2. Begin repair efforts.
3. Establish team for system cleanup and waste disposal activities.
4. Establish community educational and public relations activities.
5. Establish Recovery Team organization and off-site support liaison.
6. Re-establish normal site operations.
7. Establish claim office.
8. Assure regulatory communication.
9. Establish technical assessment team (FPC, Framatome Technologies, other Architect/Engineer, etc.).
10. Develop long-term organizational recovery responsibilities and plant status objectives.

**NOTE:** The completed recovery plan and implementing procedures shall be submitted to the PNSC for approval before implementation.

- ATTACHMENT 9 -

## EQUIPMENT INSTRUCTIONS

### Accident Assessment Ringdown Speaker

1. Ensure the speaker is connected to phone jack 124-D5.
2. Turn on power using the knob on the front of the speaker.
3. Activate the speaker circuit by removing receiver from the Accident Assessment Ringdown phone in the Simulator Control Room. This phone must be off the hook for the Room 124 speaker to function.

### SPDS

1. Ensure both computers and monitors designated for SPDS in northwest corner of Room 124 are turned on.

2. In the "Access Control Client" dialog box, ensure "CR3 SPDS" is selected in the drop-down box (or "SIM SPDS" for drills).

3. Click "LOGON."

4. Press "ALT TAB" until "SPDS" is selected.

### TSC/EOF Ringdown Phone

The TSC/EOF Ringdown phone is located in southwest corner of Room 124. This is an automatic circuit that connects when either the TSC phone or EOF phone is taken off the hook.