Duke Energy®
Crystal River Nuclear Plant
Docket No. 50-302
Operating License No. DPR-72

Ref: 10 CFR 50.4(b)(5)(iii)
10 CFR 50.54(q)(5)
10 CFR 50, Appendix E, Section V

November 21, 2012
3F1112-04

U.S. Nuclear Regulatory Commission
Attn: Document Control Desk
Washington, DC 20555-0001

Subject: Crystal River Unit 3 – Revisions to the Radiological Emergency Response Plan Implementing Procedures

Dear Sir:

In accordance with 10 CFR 50.4(b)(5)(iii), 10 CFR 50.54(q)(5), and 10 CFR 50, Appendix E, Section V, Florida Power Corporation hereby submits revisions to the Radiological Emergency Response Plan implementing procedures for Crystal River Unit 3 (CR-3).

CR-3 has evaluated these revisions, in accordance with 10 CFR 50.54(q), and determined the revisions do not reduce the effectiveness of the CR-3 Radiological Emergency Response Plan and the Plan continues to meet the standards of 10 CFR 50.47(b) and the requirements of 10 CFR 50, Appendix E.

Enclosure 1 provides a list of the revised Radiological Emergency Response Plan implementing procedures. Enclosure 2 provides a 10 CFR 50.54(q)(5) analysis summary for the revised Radiological Emergency Response Plan implementing procedures. Enclosure 3 provides a copy of the revised Radiological Emergency Response Plan implementing procedures.

There are no new regulatory commitments made within this submittal.

If you have any questions regarding this submittal, please contact Mr. Dan Westcott, Superintendent, Licensing and Regulatory Programs, at (352) 563-4796.

Sincerely,

MD Rigsby

Mark D. Rigsby
Manager – Support Services – Nuclear
Crystal River Nuclear Plant

MDR/sam

xc: Regional Administrator, Region II
Senior Resident Inspector
NRR Project Manager

Enclosures:
1. List of Revisions to the Radiological Emergency Response Plan Implementing Procedures
2. 10 CFR 50.54(q)(5) Analysis Summary
3. Copy of Revised Radiological Emergency Response Plan Implementing Procedures

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FLORIDA POWER CORPORATION

CRYSTAL RIVER UNIT 3

DOCKET NUMBER 50-302 / LICENSE NUMBER DPR-72

ENCLOSURE 1

LIST OF REVISIONS TO THE RADIOLOGICAL EMERGENCY RESPONSE PLAN IMPLEMENTING PROCEDURES

LIST OF REVISIONS TO THE RADIOLOGICAL EMERGENCY RESPONSE PLAN IMPLEMENTING PROCEDURES

Title	Revision	Effective Date
CH-631, Post Accident Sampling and Analysis of Reactor Building Vent, Auxiliary Building Vent, and Reactor Building Atmosphere	7	10/25/2012
Emergency Action Level Bases Manual	15	10/24/2012
EM-202, Duties of the Emergency Coordinator	98	10/24/2012
EM-204A, Off-Site Dose Assessment During Radiological Emergencies (Control Room Method)	25	10/30/2012
EM-204B, Off-Site Dose Assessment During Radiological Emergencies For Monitored Releases – Mixtures (User Instructions For RASCAL)	42	10/30/2012
EM-219, Duties Of The Dose Assessment Team	21	10/30/2012
EM-225, Duties Of The Technical Support Center Accident Assessment Team	27	10/30/2012
EM-225A, Post Accident RB Hydrogen Control	11	10/30/2012
EM-402, Emergency Operations Facility Technical Support Team	6	10/30/2012
EM-500, Equipment Important to Emergency Preparedness and Response	0	10/30/2012

CH = Chemistry Emergency Response Plan Implementing Procedure
EM = Emergency Plan Implementing Procedure

FLORIDA POWER CORPORATION

CRYSTAL RIVER UNIT 3

DOCKET NUMBER 50-302 / LICENSE NUMBER DPR-72

ENCLOSURE 2

10 CFR 50.54(q)(5) ANALYSIS SUMMARY

10 CFR 50.54(q)(5) ANALYSIS SUMMARY

In accordance with 10 CFR 50.54(q)(5), Florida Power Corporation (FPC) is providing an analysis summary for the revised Radiological Emergency Response Plan (RERP) implementing procedures being submitted with this letter. The analysis summary for changes associated with program elements, administrative changes, and editorial corrections are described below.

The replacement of the Reactor Building (RB) Purge Exhaust Duct radiation area monitor (RM-A1) and the Auxiliary Building (AB) and Fuel Handling Area (FHA) Exhaust Duct radiation monitor (RM-A2), by Engineering Change (EC) 76363, resulted in changes to the Crystal River Unit 3 (CR-3) Emergency Action Level (EAL) scheme that required a 10 CFR 50.54(q) evaluation. Changes to RERP implementing procedures that established compensatory actions with the monitors out of service are described in the CR-3 to NRC letter, 3F0712-04, dated July 19, 2012, "Crystal River Unit 3 – Revisions to the Radiological Emergency Response Plan Implementing Procedures," and in the CR-3 to NRC letter, 3F0812-01, dated August 1, 2012, "Crystal River Unit 3 – Revisions to the Radiological Emergency Response Plan Implementing Procedure and Document." Changes resulting from the new radiation monitors, in addition to other changes that are not related to this equipment modification, are included in this enclosure.

The replacement of RM-A1 and RM-A2 removed the particulate, iodine, and noble gas sampling skids; the noble gas range skids, with their internal particulate/iodine sample filters previously located in the AB; and the associated control, indication, and alarm components previously in the Main Control Room (MCR). New sample skids have been installed at approximately the same location as the previous skids in the AB. The new Remote Display Units are installed on the front panel of the MCR Radiation Monitor Control Console at the location previously occupied by the RM-A1 modules. The new radiation monitoring systems do not have online particulate and iodine channels; however, those channels are not used in EALs or in emergency dose assessment. The new systems provide continuous real time monitoring and display channels for Normal and Accident range noble gas activity. The new ranges have the same units of measurement with one decade of overlap and an automatic transition point from the Normal Range to the Accident Range. For each RM-A1 and RM-A2 display unit, the primary output measured against alarm setpoints is scaled in micro Curies per cubic centimeter (μCi/cc), however an output in Curies per second (Ci/sec) is present also. The new systems also provide continuous filter sampling and grab sampling capability of Normal and Accident range particulate, iodine, and noble gas in the sample streams. All of the existing heating, ventilation, and air conditioning, waste disposal valve, containment purge valve (containment isolation), and hydrogen purge valve (containment isolation) interlock functions are supported by the new systems. The new monitors were evaluated to ensure compliance with the existing licensing basis requirements and are deemed acceptable.

The function to establish a standard scheme of emergency classification and action levels in the Emergency Classification standard, as defined by 10 CFR 50.47(b)(4), was affected by this equipment modification. The current EAL scheme is based on, "Methodology for Development of Emergency Action Levels," Nuclear Energy Institute (NEI) 97-03 (NUMARC/NESP-007), Revision 2, which includes the four impacted gaseous effluent EALs that contain effluent radiation monitor threshold values: EALs 1.1 Unusual Event (UE), 1.2 Alert, 1.3 Site Area Emergency (SAE), and 1.4 General Emergency (GE). Each of these EALs has at least one other

option, in addition to the RM-A1/A2 threshold value. Each threshold value has been updated with characteristics of the new RM-A1/A2 systems and the addition of current extended shutdown plant conditions in the development methodology.

The EAL descriptions with respect to the Emergency Classification Table of EM-202, "Duties of the Emergency Coordinator, ENCLOSURE 1, EMERGENCY CLASSIFICATION TABLE," are modified as follows:

- EAL 1.1, Item 1 - "Normal Range" replaces "Gas Channel." The new Normal Range monitor is comparable in range to the previous low-range Gas Channel monitor. The new systems have continuous gas monitors only, as described above, which eliminates the need to distinguish the monitor type. EAL 1.1, Item 1 will continue to list the high alarm setpoint as the threshold value.
- In EAL 1.1, Item 2 - the threshold value of 5.0E-4 μCi/cc on the Normal Range replaces the previous threshold defined as "2 times the ODCM noble gas release setpoint." The Off-Site Dose Calculation Manual (ODCM) has been revised to administratively reduce the high alarm setpoint to 5.0E-4 μCi/cc. The new high alarm setpoint is based on extended shutdown conditions and is conservative for online operations as the monitor measures the noble gas component of a release only. The ODCM setpoint was set arbitrarily to 10 percent of the Alert threshold value that results in a logical progression from UE to Alert and Alert to SAE. Lowering the high alarm setpoint and lowering EAL 1.1, Item 2 threshold are conservative changes that support plant control and actions to determine release status.
- In EAL 1.2, Item 1 - 5.0E-3 μCi/cc on the "Accident Range" replaces 30 mR/hour on the "Mid-Range." The new Accident Range monitor is comparable in range to the previous mid and high range monitors. The new threshold value of 5.0E-3 μCi/cc is significantly lower than the previous value, which is more conservative.
- In EAL 1.2, Item 2 - the threshold value of 5.0E-3 μCi/cc on the Accident Range replaces "200 times the ODCM noble gas release setpoint." This threshold value is based on extended shutdown conditions and is conservative for online operations. The threshold value was set arbitrarily to the monitor transition point from the Normal Range to the Accident Range, in order to establish an identifiable threshold and a logical progression from Alert to SAE.
- In EAL 1.3, Item 1 - "Accident Range" replaces "Mid-Range" based upon the new monitor design described above. The threshold values for stability class groups are replaced with threshold values for online operations and for current extended shutdown. The monitor measures the noble gas component of a release only. EAL 1.3 threshold values are 10 percent of EAL 1.4 values.
- In EAL 1.4, Item 1 - "Accident Range" replaces "Mid-Range" based upon the new monitor design described above. The development of the new EAL 1.4 threshold values applies engineering rigor with different assumptions than the previous process described in the former ATTACHMENT 2 of the EAL Bases Manual. Although the station is in an extended shutdown, EALs 1.3 and 1.4 will list threshold values for both online and shutdown conditions. This supports Licensed Operator Continuing Training and emergency drill scenarios that frequently assume online operations to maintain proficiency in this plant condition. In addition, the calculation that establishes the setpoints changes assumptions from the previous method to more appropriate parameters for a Loss of Coolant Accident. The iodine and noble gas source term assumes 100 percent gas gap release and 30-minute decay. This is consistent with the design basis accident modeling of Regulatory Guide 1.183, "Alternative Radiological Source Terms for Evaluating Design Basis Accidents at Nuclear Power Reactors," and assures a large iodine release relative to noble gases, thereby lowering the noble gas thresholds. RASCAL version 3.0.5, CR-3 dose assessment software, was used

to determine the isotopic mix for online operations and for extended shutdown conditions. With the new mix, the dose projection results identify that the Thyroid Committed Dose Equivalent (CDE) dose value exceeds the Protective Acton Guideline (PAG) level limiting output, whereas the Total Effective Dose Equivalent (TEDE) dose value exceeded the PAG level with the previous calculation results.

Reference: Emergency Response Regulatory Review Action Request (EREG AR) 520663

The following changes were also evaluated against the 10 CFR 50.47(b)(4) Emergency Classification standard for a standard scheme of emergency classification and action levels:

"Emergency Action Level Bases Manual," Revision 15

- EAL changes described above are incorporated into the EAL Bases Manual information.
- The previous radiation controller scale operation description is removed.
- The contingency for use of a derived air concentration value from a Health Physics air sample is removed from EAL 1.1, EAL 1.2, and ATTACHMENT 2. This contingency is no longer needed because CH-281, "Conduct of Environmental and Chemistry during Abnormal and Emergency Events," contains detailed instruction for determining when EAL 1.1 is met that is based on sampling data. ODCM limit information was modified as described above.
- ATTACHMENT 2, "DEVELOPMENT OF PARAMETERS AND VALUES USED IN SELECTED EALS," was updated to remove threshold values in EALs 1.1, 1.2, 1.3, and 1.4 with the information contained in CR-3 Calculation N12-0001, "Calculation of RM-A1 and RM-A2 Threshold Values for Emergency Action Levels."
- EAL 2.10, "Toxic or Flammable Gas," was updated to add the definition of normal plant operations that clarifies the applicability of the UE EAL, as described by NEI 99-01, Revision 5, "Methodology for Development of Emergency Action Levels."

All threshold value changes are more conservative than the previous values and the requirements of 10 CFR 50.47(b) (4) are still met. The replacement monitors provide adequate input for EAL decision-making. Changes to the procedures do not negate the capability to perform timely and accurate emergency classifications.

Reference: EREG AR 520663

The radiation monitor replacements also affected the 10 CFR 50.47(b)(9) Emergency Assessment Capability standard that ensures adequate methods, systems, and equipment for assessing and monitoring actual or potential offsite consequences of a radiological emergency condition are maintained. Emergency dose assessment is performed with several RERP implementing procedures. Each of these procedures has been changed to incorporate the characteristics of the new RM-A1/A2 systems and current extended shutdown plant conditions, in addition to other changes not related to this equipment modification.

Various procedures evaluated against the 10 CFR 50.47(b)(9) standard were updated as follows:

1. EM-202, "Duties of the Emergency Coordinator," Revision 98

- SECTION 9, "INSTRUCTIONS," was updated to remove a description of the previous system controller operation within a step that is taken after an event declaration. The new systems will switch automatically from the Normal Range to the Accident Range and the step has been simplified to remove the description of the previous monitor output and necessary controller adjustment to measure the release. This step also has been changed to increase the time permitted to complete the dose projection for plant conditions, but does not reduce the

effectiveness of this action. The portion of the steps that recommends completion of actions contained in the offsite dose assessment procedures within 15 minutes of the event declaration was moved to the "recommended within 30" sub-section. The new timeframe is more practical since this action is completed by the Dose Assessment Team and requires more time for data interpretation. The requirement to determine the Release Significance Category (a basic form of dose assessment) and to report this information to the State in 15 minutes is still maintained and no other requirements are affected by this change.

- ENCLOSURE 1, "EMERGENCY CLASSIFICATION TABLE," is updated with the modified EAL descriptions shown above.

2. EM-204A, "Off-Site Dose Assessment during Radiological Emergencies (Control Room Method)," Revision 25

- SECTION 3, "DEFINITIONS," added a new definition of extended shutdown. Spent Fuel Pool (SFP) decay heat load and the potential source term are significantly diminished. In extended shutdown, particulates dominate the dose contribution, lowering the proportional concentration of noble gas needed to reach PAG doses compared to online operations.
- SECTION 6, "PRECAUTIONS, LIMITATIONS, AND NOTES," was updated to clarify that due to the low probability of G stability class, F dispersion factor is used for the F and G category. In addition, a statement to specify a 30 minute duration for the decay time assumption was incorporated.
- SECTION 9, "INSTRUCTIONS," were modified as follows:
  a. Nomenclature changes, new controller features, and the conditional steps that reference Dose Rate Tables for the analysis of RM-A2 indication with respect to plant shutdown status, rather than monitor range, were incorporated. A separate table for each range is no longer necessary with the new monitor output units. Both the new Normal and Accident Ranges read in μCi/cc and are included in each table.
  b. OSI PI, the CR-3 plant monitoring and system trending computer program, is identified as a data source for the "Sigma Theta" parameter, defined as the standard deviation of a set of wind range measurements.
  c. Redundant information was removed that describes the "Sigma-Theta" basis and reading use.
  d. OSI PI was identified as an optional data source for obtaining delta temperature and wind direction data, in addition to the existing source.
  e. Internet sources are identified as optional data sources for obtaining wind direction data, in addition to the existing source.
  f. Wind speed recorder units of measurement are defined in meters per second. Also, a step to clarify when a conversion of wind speed units is necessary was added, based upon differences in OSI PI data units and the recorder units.
  g. A new instruction was added to the Florida Nuclear Plant Emergency Notification Form (ENF) to record the affected sectors.
  h. A reference previously used to correct for wind speed was removed as the dose tables currently assume the wind speed that produces the maximum dose.
- ENCLOSURE 1 was updated to reflect new monitor units (μCi/cc) and new table dose units (mR). Dose Rate Tables 1 and 2 in this enclosure previously contained units of measurement in counts per minute (cpm) and mR/hour. Table 1 is for reference in the current extended shutdown condition and Table 2 is for online operating conditions. The dose tables reflect the results of the calculation for determining EAL 1.4 radiation monitor threshold values.

Factors in the dose tables that convert noble gas concentration (in µCi/cc) to TEDE dose and Thyroid CDE dose have been revised to use the current version of RASCAL for the noble gas concentration required to yield PAG doses.

- ATTACHMENT 1, "DATA SHEET," incorporates the previous EM-204A, ENCLOSURE 1, "DATA SHEET," content. Both pages of the data sheet were significantly modified as a result of reformatting, characteristics of the new monitors, adding references to OSI PI, and changes to required data on the ENF.

3. **EM-204B, "Off-Site Dose Assessment During Radiological Emergencies For Monitored Releases – Mixtures (User Instructions For RASCAL)," Revision 42**

- SECTION 3, specifically Step 3.3, "LIMITS AND PRECAUTIONS," was updated with a description of the impact of a station blackout (SBO) event to the new RM-A1 and RM-A2 systems. The skids, pumps and detectors will be powered from the Engineered Safeguards Motor Control Center (ES MCC) 3A1 and will lose power in a SBO. With the previous system, the detectors and meters remained powered, but were not used for monitoring releases during a SBO. The new control room display units will be energized, but will not have process information. Like the previous system, the new system will lose the capability of monitoring releases in a SBO. Unlike the previous system, the new system will not be able to monitor area background levels around the skid. However, background levels around the skid have no impact on the intent of this procedure. In addition, instructions used with the previous system controller were removed and replaced.
- ENCLOSURE 1, "RELEASE RATE WORKSHEETS," is modified as follows:
  a. Worksheets 1 and 2 descriptions are updated to indicate that the worksheet is not necessary if Ci/sec is already known since the new monitors feed Ci/sec to OSI PI and may be entered directly into RASCAL. The worksheets previously required numerous conversions for data entry into RASCAL.
  b. Worksheets 1 and 2 are updated to add instruction that if Ci/sec is known, go to Worksheets 6 and 7. Worksheets 1 and 2 are used for converting µCi/cc to Ci/sec when Ci/sec is not directly available. New monitor range, units of measurement, and output guidelines were incorporated.
  c. Worksheet 6 calculates an iodine source term scaled to the noble gas with iodine/noble gas ratios and various reduction mechanisms. Instructions were added to use Worksheet 7 during extended shutdown conditions.
  d. Worksheet 7 calculates a particulate source term scaled to the noble gas with particulate/noble gas ratios and various reduction mechanisms. Base particulate/noble gas ratios for extended shutdown conditions with the pool drained and the release composition as filtered and unfiltered were incorporated. Parameters for a drained SFP were selected since only a release from a drained SFP will produce significant dose at the site boundary. RASCAL is used to establish particulate/noble gas ratios for filtered and unfiltered release conditions.

4. **EM-219, "Duties of the Dose Assessment Team," Revision 21**

- ENCLOSURE 2, "DATA FROM THE PLANT COMPUTER," instructions were updated to remove steps associated with the previous system and updated instructions to the new RM-A1 and RM-A2 nomenclature, monitor function, and the Safety Parameters Display System (SPDS) identified parameter.
- SECTION 4, "INSTRUCTIONS," step and CAUTION NOTE were removed that referenced the previous style radiation monitor controller.

5. **EM-225, "Duties of the Technical Support Center Accident Assessment Team," Revision 27**
- ATTACHMENT 9, "DOSE ASSESSMENT TEAM NOTIFICATION," instructions were removed to record the previous system controller mode setpoints. The setpoint was formerly calculated weekly per the ODCM and is currently a fixed value.

6. **EM-225A, "Post-Accident RB Hydrogen Control," Revision 11**
- ENCLOSURE 4, "PREREQUISITE FIELD ACTIONS," was modified to add description of the current RM-A1 operation and eliminated reference to the previous system controller mode of operation. Guidance for start-up of the current monitor was added also.

7. **CH-631, "Post Accident Sampling and Analysis of Reactor Building Vent, Auxiliary Building Vent, and Reactor Building Atmosphere," Revision 7**
- SECTIONS 3, 4, and 5 guidelines for sample collection, RM-A1/A2 specifications and operation, contingencies, limits and precautions, and nomenclature were updated for the new monitors and guidelines for the previous systems were removed. The new monitors provide comparable post-accident sampling capability from the perspective of effort and time.
- ENCLOSURES 7, 8, 9, 10, and 11 were removed since the information does not apply to the new RM-A1/RM-A2 systems.

All capabilities are maintained and the requirements of 10 CFR 50.47(b)(9) are met. Changes to the procedures do not challenge the capability to perform timely and accurate emergency classifications or timely and credible emergency dose assessments. EALs and emergency dose assessment are more versatile since EALs 1.3 and 1.4 list threshold values for both the current extended shutdown condition and online conditions. Online condition threshold values support Licensed Operator Continuing Training and emergency drill scenarios that assume online operations to maintain proficiency in this plant condition. The changes continue to meet the requirements of 10 CFR 50, Appendix E and 10 CFR 50.47(b).
Reference: EREG AR 520663 (Items 1-7 above)

The following changes do not require a 10 CFR 50.54 (q) Evaluation:

1. **"Emergency Action Level Bases Manual," Revision 15**
- EAL 5.1, "Loss of Fuel Clad," Item 2 for Reactor Coolant System Activity threshold, was modified to correct the distance at which dose is measured from the sample lines in the Nuclear Sample Room or in the PASS Sample Room, to resolve the error identified by CR 556007.
- EAL 2.17, "Control Room Evacuation," was updated to replace an obsolete position title, "Superintendent Shift Operations," with "Shift Manager."

2. **EM-202, "Duties of the Emergency Coordinator," Revision 98**
- SECTION 9, "INSTRUCTIONS," was updated to indicate that Citrus and Levy County officials, along with State Watch Office (SWO), must be contacted within 15 minutes of a declaration. This addition supports the change in notification timeliness made to the last revision, evaluated under EREG AR 542232.
- Summary of changes was updated to replace the term, "Procedure Revision Requests (PRRs)" with "Document Revision Requests (DRRs)," based upon an administrative change.
- ENCLOSURE 3, "GUIDELINES FOR PROTECTIVE ACTION RECOMMENDATIONS (PARs) FOR NON-ESSENTIAL ENERGY COMPLEX PERSONNEL AND GENERAL POPULATION," was modified to move notes to the PAR table for clarity.

- ATTACHMENT 2, "FLORIDA NUCLEAR PLANT EMERGENCY NOTIFICATION FORM ASSOCIATED INFORMATION AND PROTOCOL," was updated to include the reference to the CR-3 Nuclear Operations Commitment System (NOCS) identification number 100521 in guidelines for initial notification.

3. EM-204A, "Off-Site Dose Assessment During Radiological Emergencies (Control Room Method)," Revision 25

- NOCS identification numbers 1029 and 13140 were superseded by NOCS number 100442 and references were updated.
- SECTION 1, "PURPOSE," was updated to specify that the procedure is an Emergency Plan Implementing Procedure and that any revisions must be carefully considered for impact.
- SECTION 2, "REFERENCES," added a reference to CR-3 Calculation N12-0001. Also, a new "Implementing References" sub-section is added.
- SECTION 9, "INSTRUCTIONS," was modified as follows:
  a. "State Warning Point" replaced with "SWO," as a result of a terminology change by the State of Florida.
  b. The Current 33' Primary Tower for meteorological data is updated to "Meteorological Monitor Panel, MMP-5."
  c. Reference to Deep Dose Equivalent and dose rate is removed since this information is no longer required for completing the ENF.
  d. The step for recording Ci/sec is removed since this information is no longer required for completing the ENF.
- The previous ENCLOSURE 1 was converted to ATTACHMENT 2 that results in the previous ENCLOSURE 2 content transfer to ENCLOSURE 1 and previous ENCLOSURE 3 content transfer to ENCLOSURE 2.

4. EM-204B, "Off-Site Dose Assessment During Radiological Emergencies For Monitored Releases – Mixtures (User Instructions For RASCAL)," Revision 42

- SECTION 1, "PURPOSE," was updated to specify that the procedure is an Emergency Plan Implementing Procedure and that any revisions must be carefully considered for impact.

5. EM-225, "Duties of the Technical Support Center Accident Assessment Team," Revision 27

- SECTION 2, "REFERENCES," was updated to add CR-3 Calculation M89-0063, "Waste Gas Decay Tank Rupture Environmental Condition," that identifies the waste gas tank volume.
- ATTACHMENT 9, "DOSE ASSESSMENT TEAM NOTIFICATION," was reformatted to improve the appearance of the forms. In addition, the definition for spiking factor and the waste gas decay tank volume were added.

6. EM-225A, "Post Accident RB Hydrogen Control," Revision 11

- SECTION 2, "REFERENCES," added "EC 76363" to the Developmental References.

7. **CH-631, "Post Accident Sampling and Analysis of Reactor Building Vent, Auxiliary Building Vent, and Reactor Building Atmosphere," Revision 7**

- SECTION 2, "REFERENCES," added "EC 76363" to the Developmental References and CR-3 Equipment Database References was updated to identify new equipment.

Reference: EREG AR 520663 (Items 1-7 above)

8. **EM-402, "Emergency Operations Facility Technical Support Team," Revision 6**

- ENCLOSURE 2, "TECHNICAL SUPPORT OPERATIONS REPRESENTATIVE CHECKLIST," and ENCLOSURE 3, "TECHNICAL SUPPORT ENGINEER CHECKLIST," were updated to correct enclosure reference numbers.
- ENCLOSURE 7, "DOSE ASSESSMENT TEAM NOTIFICATION," was reformatted to improve the appearance of the forms, similar to EM-225, ATTACHMENT 9, reformatting.
- ENCLOSURE 9, "EQUIPMENT INSTRUCTIONS," was updated to correct nomenclature.

Reference: EREG AR 569492

9. **EM-500, "Equipment Important to Emergency Preparedness and Response," Revision 0**

- This is a new procedure that formalizes compensatory measures associated with equipment important to Emergency Preparedness, incorporates information from a Fleet procedure (EMG-NGGC-0007, "Equipment Important to Emergency Preparedness and Response"), and implements guidance found in Inititute of Nuclear Power Operations (INPO) Guideline 10-007, "Equipment Important to Emergency Response." The compensatory measures identified do not reduce the effectiveness of any of the Emergency Planning elements or functions. The implementation of this new procedure is an improvement to the Emergency Preparedness program.

Reference: EREG AR 565689

FLORIDA POWER CORPORATION

CRYSTAL RIVER UNIT 3

DOCKET NUMBER 50-302 / LICENSE NUMBER DPR-72

ENCLOSURE 3

COPY OF REVISED RADIOLOGICAL EMERGENCY RESPONSE PLAN
IMPLEMENTING PROCEDURES

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CRYSTAL RIVER UNIT 3

PLANT OPERATING MANUAL

CH-631

Post Accident Sampling and Analysis of Reactor Building Vent, Auxiliary Building Vent, and Reactor Building Atmosphere

TABLE OF CONTENTS

ENCLOSURES

1.0 **PURPOSE**

This procedure provides instructions for sampling the RB Vent, AB Vent, and RB atmosphere during accident conditions using PASS. This procedure is an Emergency Plan Implementing Procedure (EPIP). Any revisions must be carefully considered for emergency plan impact.

2.0 **REFERENCES**

2.1 **Developmental References**

2.1.1 RERP, Radiological Emergency Response Plan

2.1.2 Regulatory Guide 1.183, Alternative Radiological Source Terms for Evaluating Design Basis Accidents at Nuclear Power Reactors. July 2000.

2.1.3 NUREG 0737, Post-TMI Requirements

2.1.4 Regulatory Guide 1.97, Instrumentation For Light-Water Cooled Nuclear Power Plants to Assess Plant and Environs Conditions During and Following an Accident

2.1.5 ADM-NGGC-0105, ALARA Planning

2.1.6 Applied Physical Technology, Volumes A through C (Crystal River Installation PASS manuals)

2.1.7 Drawing M.D. 0211033.003

2.1.8 EOP-14, Enclosure 2, PPO Post Event Actions

2.1.9 EM-104, Operation of the Operational Support Center

2.1.10 FD-302-693, Containment Monitoring System

2.1.11 FD-302-694, PASS Containment Monitoring AIM Detection System

2.1.12 FD-302-695, Noble Gas Effluent Monitoring System

2.1.13 FD-302-766, Auxiliary Building Post Accident

2.1.14 EC 76363, Radiation Monitors RM-A1 and RM-A2 Replacement

2.1.15 EMG-NGGC-0002, Off-Site Dose Assessment

2.2 **Equipment Database References**

WSV-3	WSV-36	RM-A1A-V-12	WSV-63	WSSB-2	WSP-1
WSV-4	WSV-37	RM-A1A-V-13	WSV-64	WS-14-FI	RMV-11
WSV-5	WSV-53	RM-A1A-V-14	WSV-67	RM-A1A	RM-A1-FT
WSV-6	WSV-54	RM-A1A-V-15	WSV-70	RM-A2A	RM-A1-FI
WSV-32	WSV-57	RM-1A-SBA	WSV-71	CMP	RM-A2-FI
WSV-33	WSV-59	RM-1A-SBB	WSV-72	DPDP-5A	DPDP5B
WSV-34	WSV-60	RM-2A-SBA	RMV-23	DPDP-8A	DPDP-8B
WSV-35	WSV-61	RM-2A-SBB	ACDP-59	AHF-67	

3.0 LIMITS AND PRECAUTIONS

3.1.1 Any or all of this procedure is done by direction of the EC or designee.

3.1.2 Re-entry must have RMT preplanning, concurrence, and coverage as outlined in EM-104, Operation of the Operational Support Center. Controlled access areas will be defined by the RMT personnel.

3.1.3 Extremely high radiation dose rates may be present during post-accident sampling. These high dose rates could result in high radiation exposure. Performing this procedure requires ALARA pre-planning.

3.1.4 Emergency Sample Team will STOP and go to a low dose area (i.e. primary chemistry laboratory) if dose rates at re-entry work area exceeds limits specified in pre-job briefing.

3.1.5 All sampling actions are performed from the Main Control Board by Operations or from the Count Room unless specifically noted.

3.1.6 WSP-1 is a positive displacement pump and may be damaged if operated without complete or proper discharge valve line-up.

3.1.7 WSV-70 is interlocked with the following valves and will not open if any of these valves are open.

- WSV-33
- WSV-35
- WSV-36
- WSV-37

3.1.8 A maximum filter loading of 3 Ci total activity is recommended for particulate and iodine filter grab samples.

3.1.9 RB dome sampling via WSV-34 and WSV-35 is the preferred sample point for sampling the RB atmosphere.

3.1.10 RB emergency recirculation discharge duct sampling via WSV-32 and WSV-33 is the preferred alternate sample point for sampling the RB atmosphere.

3.1.11 RB normal recirculation duct sampling via WSV-3 and WSV-4 is the least preferred sample point for sampling the RB atmosphere because this ventilation duct is normally secured during accident conditions.

3.1.12 ES must be bypassed or reset by Operations before WSV-3, 4, 5 or 6 can be opened from the Control Room.

3.1.13 Sampling described in Section 4.0 of this procedure **CANNOT** be performed concurrently due to shared piping in the different sample streams.

3.1.14 The B 480 Volt ES BUS provides power to PASS Equipment used in this procedure and must be in service.

3.2 Description

3.2.1 PASS is an on-line system designed to sample various liquid and gaseous sample streams during accident conditions. The RANGE system samples the RB atmosphere and gaseous effluents from both the RB and AB Vents.

3.2.2 The RANGE system provides the ability to obtain gaseous grab samples to be shipped off-site for analysis.

3.2.3 When estimating total activity for gaseous grab sample shipment, the following assumptions were made:

- Core Nuclide Mix and Half-lives from RADTRAD Code Library
- 8 hours since reactor shutdown
- Microshield software was used to determine conversion factors for calculating total µCi from dose rate. Sample assumed to be small enough at a distance of 7 inches to represent point source. Pig is 17.75 inches tall with a radius of 7 inches. Weight is 725 pounds. This results in effective density of 7.4 g/cc.
- Release fractions from Regulatory Guide 1.183 for gap and early in-vessel melt
- No removal of iodines and particulates from RB air

3.2.4 When estimating total activity for particulate and iodine grab sample shipment the following assumptions are made:

- Core Nuclide Mix and Half-lives from RADTRAD Code Library
- 8 hours since reactor shutdown
- Microshield software was used to determine conversion factors for calculating total µCi from dose rate. Pig is 13.5 inches X 11 inches X 6.5 inches. The pig is composed of 1.750 inches of lead and 0.5 inches of iron. The filter canister is 1.375 inches tall with a diameter of 2.25 inches. The measurement distance is 3.0 inches from the center of the pig.
- Release fractions from Regulatory Guide 1.183 for gap and early in-vessel melt
- No removal of iodines and particulates from RB or AB air
- The RB or AB atmospheric mix is on the particulate and iodine filter. The particulate and iodine is filtered by the HEPA and charcoal banks, but the isotopic mix remains unchanged. Some of the Xe and Kr gas is retained in the iodine cartridge. The RB or AB particulate and iodine is reduced by ~ 99% through the filters, and approximately 1% of gas is retained in the cartridge.

3.3 Definitions

3.3.1	PASS	Post Accident Sampling System
3.3.2	RANGE	Reactor and Auxiliary Noble Gas Effluent monitoring system
3.3.3	RE-ENTRY	Return of personnel to an area evacuated by an emergency condition
3.3.4	RMT	Radiation Monitoring Team
3.3.5	TMI	Three Mile Island nuclear plant
3.3.6	SRP	Sample Routing Plate

3.4 **Responsibilities**

3.4.1 EC or designee shall authorize re-entry.

3.4.2 OSC Chemistry Coordinator or designee

- ensures EC approval for re-entry has been obtained
- determines which sections of procedure are to be performed during re-entry
- ensures re-entry prerequisites are complete

3.4.3 This procedure is performed by a qualified Emergency Sample Team member.

3.5 **Prerequisites**

3.5.1 ASSEMBLE sample team.

Sample Team Leader ______________________

Sample Team Members ______________________

____

____

NOTE
B 480V ES BUS provides power to PASS Equipment

3.5.2 ___ VERIFY B 480V ES BUS Operational.

3.5.3 DETERMINE sampling to be performed.

Section Number	Description
__________	______________________________
__________	______________________________
__________	______________________________
__________	______________________________

3.5.4 REVIEW procedures.

___ EM-104, Operation of the Operational Support Center

___ Emergency Team Member duties per Section 4.0

___ Team Briefing/Re-entry checklist

___ Sections of this procedure being performed

3.5.5 **IF** gas grab sampling via WSSB-2,**THEN** ENSURE the following:

[ ] Gas Grab sampler currently installed

___ AB elevator is operable to transport sampler

OR

[ ] Gas Grab sampler NOT currently installed

___ AB elevator is operable to transport sampler

___ New break-away type device available to attach transit cover and transit cover bolts to sampler

___ Replacement sample bomb and pig (Catalog ID 1400513) available to install on grab sampler transit cart

3.5.6 **IF** particulate and iodine sampling via RM-A1A-SBA/B or RM-A2A-SBA/B, **THEN** PERFORM the following:

[ ] ESTIMATE sample stream activity, (EMG-NGGC-0002)

Estimated sample stream activity _______________ µCi/cc

[ ] DETERMINE where filters will be stored

Filter storage location ____________________

[ ] ENSURE timing device available.

[ ] Particulate/Iodine sampler currently installed

___ AB elevator is operable to transport sampler

OR

[ ] Particulate/Iodine sampler is NOT currently installed

___ AB elevator is operable to transport sampler

___ Replacement filter canister and particulate/Iodine sampler available to install

3.5.7 **IF** sampling RB atmosphere, **THEN** ENSURE electrical breakers are closed.

[ ] Operations has performed EOP-14 Enclosure 2, PPO Post Event Actions

OR

[ ] Operations has NOT performed EOP-14 Enclosure 2, PPO Post Event Actions

1. ___ REQUEST operations CLOSE the following breakers
   - DPDP-5A, Breaker 2 (WSV-35)
   - DPDP-8A, Breaker 14 (WSV-34)
   - DPDP-5B, Breaker 27 (WSV-33)
   - DPDP-8B, Breaker 21 (WSV-32)

2. ___ Operations REPORTS breakers closed

3.5.8 PERFORM pre-job brief.

___ ENSURE RMT member is present for briefing; AND DISCUSS the following

- ___ access route
- ___ exit route
- ___ Communications

  Radio channel to be used _____

Telephone number(s) __________ __________ __________ __________

3.5.9 VERIFY ALL steps of this section are completed before sample team leaves OSC.

____/____
Section 3.5 complete Initial/Date

4.0 INSTRUCTIONS

> **NOTE**
>
> The sampling described in Section 4.0 of this procedure CANNOT be performed concurrently due to shared piping in the different sample streams.

4.1 RB Atmosphere Gas Grab Sample via WSSB-2

4.1.1 ___ **WHEN** sample team exits OSC,
THEN VERIFY radio communication with OSC Chemistry Coordinator or designee.

> **NOTE**
>
> WSSB-2 exhaust fan (AHF-67) provides ventilation for gas grab sampling. The switch is located on wall left of AHF-67.

4.1.2 ___ POSITION AHF-67 switch to ON

4.1.3 ENSURE gas grab sampler, WSSB-2, installed.

[ ] Gas grab sampler already installed

OR

[ ] REFER TO Enclosure 4, Guidelines For Gas Grab Sampler Installation And Removal for instructions to install.

4.1.4 ALIGN system for gas grab sample.

1. NOTIFY Operations to perform the following:
   a. ___ ENSURE ES actuations are reset or bypassed
   b. ___ OPEN WSV-5
   c. ___ OPEN WSV-6
   d. OPEN RB sample isolation valves
      [ ] RB dome (**preferred sample**)
      ___ WSV-34
      ___ WSV-35

      OR

      [ ] emergency recirculation ventilation discharge duct (alternate sample)
      ___ WSV-32
      ___ WSV-33

      OR

      [ ] normal recirculation ventilation duct sampling (not representative)
      ___ WSV-3
      ___ WSV-4

2. ___ Operations reports valve line-up complete

3. OPEN the following valves:
   [ ] RB dome or emergency recirculation ventilation discharge duct sample
   ___ WSV-59
   ___ WSV-60
   ___ WSV-37

   OR

   [ ] Normal recirculation ventilation duct sample
   ___ WSV-36
   ___ WSV-59
   ___ WSV-60
   ___ WSV-37

CAUTION

WSP-1 may be damaged if operated without a complete discharge valve line-up.

4.1.5 ALIGN for RB atmosphere gas grab sample.

1. ___ START WSP-1
2. ___ VERIFY flow at WS-14-FI
3. ___ PURGE at least 10 minutes

4.1.6 ISOLATE grab sample.

1. ___ CLOSE WSV-72
2. ___ CLOSE WSV-71
3. ___ RECORD sample time ____________________ Grab sample Date/Time

4.1.7 ALIGN for Instrument Air purge.

1. ___ OPEN WSV-53
2. NOTIFY Operations to ENSURE the following valves are closed:
   - ___ WSV-3
   - ___ WSV-4
   - ___ WSV-32
   - ___ WSV-33
   - ___ WSV-34
   - ___ WSV-35
3. ___ Operations REPORTS valves are closed
4. ___ PURGE at least 10 minutes.

4.1.8 RESTORE system line-up.

1. ___ OPEN WSV-61
2. ___ CLOSE WSV-59
3. ___ CLOSE WSV-60
4. ___ PURGE at least 1 minute
5. ___ STOP WSP-1
6. ENSURE CLOSED the following valves:
   - ___ WSV-53
   - ___ WSV-61
   - ___ WSV-37
   - ___ WSV-36
7. NOTIFY Operations to CLOSE the following valves:
   - ___ WSV-5
   - ___ WSV-6

4.1.9 REMOVE Gas Grab Sampler, WSSB-2.

1. ___ REMOVE gas grab sampler from sample station, REFER to Enclosure 4.
2. ___ INSTALL transit cover over quick connects
3. ___ TRANSPORT gas grab sampler to 95' TB Crane Well
4. ___ UNBOLT grab sampler from cart using ¾" wrench or equivalent as determined by Chemistry Technician
5. ___ MEASURE dose rates from grab sampler

Contact dose rate (side of pig) __________ mR/hr

Dose rate @ 3 feet __________ mR/hr

4.1.10 PREPARE for grab sample shipment.

___ REFER to Enclosure 5, Grab Sample Shipment And Notifications for off-site shipment and notifications

_____/____
Section 4.1 complete Initials/Date

4.2 **RB Vent Particulate and Iodine Grab Sample**

NOTE

1. If not actually performing equipment manipulations, wait in low dose area.

2. Flow must be established through RM-A1A accident-range gas monitor to perform this section.

4.2.1.1 ___ENSURE RM-A1A is in service. Remote indications may be used to accomplish this task.

4.2.1.2 ___ **WHEN** sample team exits OSC, **THEN** VERIFY radio communication with OSC Chemistry Coordinator or designee.

NOTE

Total activity loaded on filters is limited to ≤ 3 curies.

4.2.2 ENSURE particulate/iodine sampler, RM-A1A-SBB is installed

[ ] Particulate/Iodine Sampler already installed

OR

[ ] Install Particulate/Iodine Filter per Enclosure 7

4.2.3 DETERMINE sample collection time.

1. ___ DETERMINE RM-A1-FI flow rate

RM-A1-FI actual flow rate = [RM-A1-FI indicated flow rate] X 0.1

RM-A1-FI flow rate __________ cfm

2. ___ CALCULATE maximum sample collection duration

$$\text{sample collection time (minutes)} = \frac{3\text{E}6\ \mu\text{Ci}}{(\text{RM}-\text{A1}-\text{FI flow rate (cfm)})\left(\text{Estimated sample stream activity}\left(\frac{\mu\text{Ci}}{\text{cc}}\right)\right)\left(2.832\text{E}4\ \frac{\text{cc}}{\text{cf}}\right)}$$

$$\text{sample collection time (minutes)} = \frac{3\text{E}6\ \mu\text{Ci}}{(\_\_\_\_\_\_\_\_\_\_\ (\text{cfm}))\left(\_\_\_\_\_\_\_\_\_\_\left(\frac{\mu\text{Ci}}{\text{cc}}\right)\right)\left(2.832\text{E}4\ \frac{\text{cc}}{\text{cf}}\right)}$$

maximum sample collection duration __________ minutes

_____/_____
Initials/Date

_____/_____
Independent Verification Initials/Date

3. ___ REPORT maximum sample collection duration to OSC Chemistry Coordinator or designee.

4.2.4 OBTAIN equipment from Post Accident Sampling Kit. Kit is located 143' AB west side of SF Pool wall.

___ (1) Particulate/iodine Sampler Tee Handle

___ (1) Particulate/Iodine Filter Canister with particulate filter (CAT ID 1400352) and silver zeolite iodine cartridge (CAT ID 1400995) installed.

___ (2) ¾" open end wrenches or equivalent

4.2.5 ___NOTIFY the control room that the Reactor Building Vent Release rate will momentarily decrease when purging the particulate/iodine sample.

4.2.6 INSTALL Particulate/Iodine filter canister in RM-A1A-SBB, Particulate/Iodine Grab Sampler for RM-A1A

1. ___ OPEN RM-A1A-SBB shield door
2. ___ INSTALL Particulate/Iodine Filter Canister with particulate filter UP
3. ___ CLOSE RM-A1A-SBB shield door

4.2.7 START sample collection.

1. OPEN valves:

   ___RM-A1A-V-13

   ___RM-A1A-V-15

2. CLOSE valves:

   ___RM-A1A-V-12

   ___RM-A1A-V-14

3. RECORD start time sample start time ___________

CAUTION
Exceeding maximum sample collection time may result in higher than expected filter dose rates.

NOTE
1. RM-A1A NG Activity Release will significantly decrease when a purge is initiated.
2. Both the RM-A1 and RM-A2 SRP Control Junction Boxes are located under the RM-A2 Sample Routing Plate.

4.2.8 STOP sample collection.

1. ___At the RM-A1 SRP control junction box, PLACE the Purge Control Switch (RM-A1-SW2) in the PURGE position.
2. ___RECORD sample stop time   sample stop time ______________
3. ___PURGE at least 3 minutes.
4. OPEN valves:
   - ___ RM-A1A-V-12
   - ___ RM-A1A-V-14
5. CLOSE valves:
   - ___ RM-A1A-V-13
   - ___ RM-A1A-V-15
6. ___ At the RM-A1 SRP control junction box, PLACE the Purge Control Switch(RM-A1-SW2) in the AUTO position.

4.2.9 DISCONNECT particulate/iodine grab sampler RM-A1A-SBB

1. CLOSE valves:
   - ___ RM-A1A-V-18
   - ___ RM-A1A-V-19
2. DISCONNECT quick connects:
   - ___ RM-A1A-V-18 quick connect
   - ___ RM-A1A-V-19 quick connect
3. ___ REMOVE bolts holding grab sampler to the wall.

4.2.10 STORE RM-A1A-SBB

1. ___ Using the Tee Handle, TRANSPORT RM-A1A-SBB to pre-determined storage location
2. ___ MEASURE dose rates on RM-A1A-SBB

Contact dose rate __________ mR/hr
Dose rate @ 3 feet __________ mR/hr

4.2.11 PREPARE for grab sample shipment using Enclosure 5, Grab Sample Shipment And Notifications for off-site shipment and notifications.

_____/____
Section 4.2 complete Initials/Date

4.3 AB Vent Particulate and Iodine Grab Sample

NOTE
1. If not actually performing equipment manipulation, wait in the low dose area.
2. Flow must be established through RM-A2A accident range gaseous monitors to perform this section.

4.3.1 ___ENSURE RM-A2A is in service. Remote indications may be used to accomplish this task.

4.3.2 ___ **WHEN** sample team exits OSC, **THEN** VERIFY radio communication with OSC Chemistry Coordinator or designee.

NOTE
Total activity loaded on filters is limited to ≤ 3 curies.

4.3.3 ENSURE particulate/iodine sampler, RM-A2A-SBB, installed.

[ ] Particulate/Iodine sampler already installed

OR

[ ] REFER TO Enclosure 7, Guidelines For Particulate/Iodine Sampler Installation.

4.3.4 DETERMINE sample collection time.

1. ___ DETERMINE RM-A2-FI flow rate

RM-A2-FI actual flow rate = [RM-A2-FI indicated flow rate] X 0.1

RM-A2-FT actual flow rate __________ cfm

2. ___ CALCULATE maximum sample collection duration

$$\text{sample collection time (minutes)} = \frac{3\text{E}6 \ \mu\text{Ci}}{(\text{RM} - \text{A2} - \text{FI flow rate (cfm)})\left(\text{Estimated sample stream activity}\left(\frac{\mu\text{Ci}}{\text{cc}}\right)\right)\left(2.832\text{E}4\ \frac{\text{cc}}{\text{cf}}\right)}$$

$$\text{sample collection time (minutes)} = \frac{3\text{E}6 \ \mu\text{Ci}}{(\_\_\_\_\_\_\_\_\_\_\_\_\text{(cfm)})\left(\_\_\_\_\_\_\_\_\_\_\_\_\left(\frac{\mu\text{Ci}}{\text{cc}}\right)\right)\left(2.832\text{E}4\ \frac{\text{cc}}{\text{cf}}\right)}$$

maximum sample collection duration __________ minutes

_____/_____
Initials/Date

_____/_____
Independent Verification Initials/Date

3. ___ REPORT maximum sample collection duration to OSC Chemistry Coordinator or designee

Section 4.3 **AB Vent Particulate and Iodine Grab Sample** (continued)

4.3.5 OBTAIN equipment from Post Accident Sampling Kit. Kit is located 143' AB west side of SF Pool wall.

___ (1) Particulate/Iodine Sampler Tee Handle

___ (1) Particulate/Iodine Filter Canister with particulate filter (CAT ID 1400352) and silver zeolyte iodine cartridge (CAT ID 1400995) installed

___ (2) ¾" open end wrenches or equivalent

4.3.6 ___ NOTIFY the control room that the AB Vent Release rate will momentarily decrease when purging the particulate/iodine sample.

4.3.7 INSTALL filter canister in RM-A2A-SBB, , Particulate/Iodine Grab Sampler For RM-A2A.

1. ___ OPEN RM-A2A-SBB shield door
2. ___ INSTALL Particulate/Iodine Filter Canister with particulate filter UP
3. ___ CLOSE RM-A2A-SBB shield door

4.3.8 START sample collection.

1. OPEN valves:
   ___ RM-A2A-V-13
   ___ RM-A2A-V-15
2. CLOSE valves:
   ___ RM-A2A-V-12
   ___ RM-A2A-V-14
3. RECORD start time sample start time ______________

CAUTION

Exceeding maximum sample collection time may result in higher than expected filter dose rates.

NOTE
1. RM-A2A NG activity will significantly decrease when a purge is initiated.
2. Both the RM-A1 and RM-A2 SRP Control Junction Boxes are located under the RM-A2 Sample Routing Plate.

4.3.9 STOP sample collection.

1. ___ At the RM-A2 SRP control junction box PLACE the Purge Control Switch, RM-A2-SW2, in PURGE position.
2. ___RECORD sample stop time   sample stop time _______________
3. ___PURGE at least 3 minutes.
4. OPEN Valves:
   ___ RM-A2A-V-12
   ___ RM-A2A-V-14
5. CLOSE Valves:
   ___ RM-A2A-V-13
   ___ RM-A2A-V-15
6. ___ At the RM-A2 SRP junction control box PLACE the Purge Control switch, RM-A2-SW2, in AUTO position.

4.3.10 DISCONNECT Particulate/Iodine Grab sampler, RM-A2A-SBB.

1. CLOSE valves:
   ___ RM-A2A-V-18
   ___ RM-A2A-V-19
2. DISCONNECT quick connects:
   ___ RM-A2A-V-18 quick connect
   ___ RM-A2A-V-19 quick connect
3. ___ REMOVE bolts holding grab sampler to the wall.

Section 4.3 **AB Vent Particulate and Iodine Grab Sample** (continued)

4.3.11 STORE Particulate/Iodine Grab Sampler, RM-A2A-SBB.

1. ___ Using the Tee Handle, TRANSPORT RM-A2A-SBB to pre-determined storage location.
2. ___ MEASURE dose rates on RM-A2A-SBB

Contact dose rate __________ mR/hr

Dose rate @ 3 feet __________ mR/hr

4.3.12 PREPARE for grab sample shipment.

___REFER to Enclosure 5, Grab Sample Shipment And Notifications for off-site shipment and notifications.

_____/_____
Section 4.3 complete Initials/Date

4.4 RB Ventilation Duct Gas Grab Sample via WSSB-2

4.4.1 ___ENSURE RM-A1A is in service. Remote indications may be used to complete this task.

4.4.2 ___ After sample team exits OSC, VERIFY radio communication with OSC Chemistry Coordinator or designee.

> **NOTE**
>
> WSSB-2 exhaust fan (AHF-67) provides ventilation for gas grab sampling. The switch is located on wall left of AHF-67.

4.4.3 ESTABLISH ventilation for gas grab sampling.

___ POSITION AHF-67 switch to ON

4.4.4 ENSURE gas grab sampler, WSSB-2, installed.

[ ] Gas grab sampler already installed

OR

[ ] REFER to Enclosure 4, Guidelines For Gas Grab Sampler Installation And Removal instructions to install.

> **CAUTION**
>
> WSP-1 may be damaged if operated without a complete discharge valve line-up.

4.4.5 ALIGN system for gas grab sample.

1. OPEN the following valves:
   - ___ RMV-011
   - ___ WSV-59
   - ___ WSV-60
   - ___ WSV-70
2. ___ START WSP-1
3. ___ VERIFY flow at WS-14-FI
4. ___ PURGE at least 5 minutes

4.4.6 ISOLATE grab sample.

1. ___ CLOSE WSV-72
2. ___ CLOSE WSV-71
3. RECORD sample time Grab sample Date/Time__________________

4.4.7 ALIGN for Instrument Air purge.

1. ___ OPEN WSV-63
2. ___ CLOSE RMV-011
3. ___ PURGE at least 10 minutes

4.4.8 RESTORE system line-up.

1. ___ OPEN WSV-61
2. ___ CLOSE WSV-59
3. ___ CLOSE WSV-60
4. ___ PURGE at least 1 minute
5. ___ STOP WSP-1
6. CLOSE the following valves:
   ___ WSV-63
   ___ WSV-61
   ___ WSV-70

4.4.9 REMOVE Gas Grab Sampler, WSSB-2.

1. ___ REMOVE gas grab sampler from sample station, REFER to Enclosure 4, Guidelines For Gas Grab Sampler Installation And Removal
2. ___ INSTALL transit cover over quick connects
3. ___ TRANSPORT gas grab sampler to 95' TB Crane Well
4. ___ UNBOLT grab sampler from cart using ¾" wrench or equivalent as determined by Chemistry Technician
5. ___ MEASURE dose rates from grab sampler

Contact dose rate (side of pig) __________ mR/hr

Dose rate @ 3 feet __________ mR/hr

4.4.10 PREPARE for grab sample shipment.

___REFER to Enclosure 5, Grab Sample Shipment and Notifications for off-site shipment and notifications

_____/____
Section 4.4 complete Initials/Date

4.5 AB Ventilation Duct Gas Grab Sample via WSSB-2

4.5.1 ___ENSURE RM-A2A is in service. Remote indications may be used to perform this task.

4.5.2 ___After sample team exits OSC, VERIFY radio communication with OSC Chemistry Coordinator or designee.

> **NOTE**
>
> WSSB-2 exhaust fan (AHF-67) provides ventilation for gas grab sampling The switch is located on wall left of AHF-67.

4.5.3 ESTABLISH ventilation for gas grab sampling.

___POSITION AHF-67 switch to ON

4.5.4 ENSURE gas grab sampler, WSSB-2, installed.

[ ] Gas grab sampler already installed

<u>OR</u>

[ ] REFER to <u>Enclosure 4</u>, Guidelines For Gas Grab Sampler Installation And Removal to install.

> **CAUTION**
>
> WSP-1 may be damaged if operated without a complete discharge valve line-up.

4.5.5 ALIGN system for gas grab sample.

1. OPEN the following valves:
   ___ RMV-23
   ___ WSV-59
   ___ WSV-60
   ___ WSV-70
2. ___ START WSP-1
3. ___ VERIFY flow at WS-14-FI
4. ___ PURGE at least 5 minutes

4.5.6 ISOLATE grab sample.

1. ___ CLOSE WSV-72
2. ___ CLOSE WSV-71
3. RECORD sample time Grab Sample Date/Time___________/______

4.5.7 ALIGN for Instrument Air purge.

1. ___ OPEN WSV-63
2. ___ CLOSE RMV-23
3. ___ PURGE at least 10 minutes

4.5.8 RESTORE system line-up.

1. ___ OPEN WSV-61
2. ___ CLOSE WSV-59
3. ___ CLOSE WSV-60
4. ___ PURGE at least 1 minute
5. ___ STOP WSP-1
6. CLOSE the following valves:
   ___ WSV-63
   ___ WSV-61
   ___ WSV-70

4.5.9 REMOVE Gas Grab Sampler, WSSB-2.

1. ___ REMOVE gas grab sampler from sample station, REFER to Enclosure 4, Guidelines For Gas Grab Sampler Installation And Removal
2. ___ INSTALL transit cover over quick connects
3. ___ TRANSPORT gas grab sampler to 95' TB Crane Well
4. ___ UNBOLT grab sampler from cart using ¾" wrench or equivalent as determined by E&C Technician
5. MEASURE dose rates from grab sampler

Contact dose rate (side of pig) __________ mR/hr

Dose rate @ 3 feet __________ mR/hr

4.5.10 PREPARE for grab sample shipment.

___REFER to Enclosure 5, Grab Sample Shipment And Notifications for off-site shipment and notifications.

_____/_____
Section 4.5 complete Initials/Date

5.0 CONTINGENCIES

5.1 Estimating Grab Sample Shipment Curie Content

5.1.1 ESTIMATE curie content of grab sample.

___REFER to Enclosure 6.

_____/____
Section 5.1 complete Initials/Date

5.2 Changeout of RM-A1A Particulate/Iodine Sampler (RM-A1A-SBA)

5.2.1 NOTIFY CRS/SM of the following:

___RM-A1A particulate/iodine sampler will be changed out.
___The requirements of ODCM Specification 2.2 are applicable

CRS/SM __________ / __________ / __________
Initials Time Date

CAUTION

Extremely high radiation dose rates may be present from the particulate/iodine sampler. These high dose rates could result in high radiation exposure. Performing this section requires ALARA pre-planning.

5.2.2 ___DETERMINE where sampler will be stored

Sampler storage location ____________________

5.2.3 ___IF necessary, ENSURE the AB elevator is available.

5.2.4 ___REFER TO Enclosure 7, Guidelines For Particulate/Iodine Sampler Installation and prepare a sampler for installation.

5.2.5 ___NOTIFY the control room to place RM-A1A in Standby.

5.2.6 ___ENSURE RM-A1A is in standby. Remote indications may be used to complete this task.

5.2.6.1 ___ **WHEN** sample team exits OSC,**THEN** VERIFY radio communication with OSC Chemistry Coordinator or designee.

5.2.7 ___TRANSPORT the particulate/iodine sampler to the AB 143'.

5.2.8 CLOSE VALVES:

___ RM-A1A-V-12

___ RM-A1A-V-14

___ RM-A1A-V-16

___ RM-A1A-V-17

5.2.9 DISCONNECT quick connects:

___RM-A1A-V-16 quick connect

___RM-A1A-V-17 quick connect

5.2.10 ___ REMOVE bolts holding sampler to the wall.

5.2.11 ___Using the Tee Handle, TRANSPORT RM-A1A-SBA away from RM-A1A skid while installing spare sampler.

5.2.12 ___REFER TO Enclosure 7, Guidelines For Particulate/Iodine Sampler Installation and INSTALL the spare Particulate/Iodine Sampler for RM-A1A.

5.2.13 OPEN VALVES:

___ RM-A1A-V-12

___ RM-A1A-V-14

___ RM-A1A-V-16

___ RM-A1A-V-17

5.2.14 INSTALL bolts holding sampler to the wall.

5.2.15 ___Using the Tee Handle, TRANSPORT RM-A1A-SBA to pre-determined storage location

5.2.16 ___NOTIFY the control room to place RM-A1A in Normal.

5.2.17 NOTIFY CRS/SM

___RM-A1A sampler changeout is complete

___The lineup is restored to normal

CRS/SM Notification completed:_____/_____/_____

Initial / Date / Time

5.3 Changeout of RM-A2A Particulate/Iodine Sampler (RM-A2A-SBA)

5.3.1 NOTIFY CRS/SM of the following:

___RM-A2A particulate/iodine sampler will be changed out.

___The requirements of ODCM Specification 2.2 are applicable

CRS/SM __________ / __________ / __________
Initials Time Date

CAUTION
Extremely high radiation dose rates may be present from the particulate/iodine sampler. These high dose rates could result in high radiation exposure. Performing this section requires ALARA pre-planning.

5.3.2 ___DETERMINE where sampler will be stored

Sampler storage location ____________________

5.3.3 ___IF necessary, ENSURE the AB elevator is available.

5.3.4 ___REFER TO Enclosure 7, Guidelines For Particulate/Iodine Sampler Installation and prepare a sampler for installation.

5.3.5 ___NOTIFY the control room to place RM-A2A in Standby.

5.3.6 ___ENSURE RM-A2A is in standby. Remote indications may be used to complete this task.

5.3.6.1 ___ **WHEN** sample team exits OSC,**THEN** VERIFY radio communication with OSC Chemistry Coordinator or designee.

5.3.7 ___TRANSPORT the particulate/iodine sampler to the AB 143'.

5.3.8 CLOSE VALVES:

___ RM-A2A-V-12

___ RM-A2A-V-14

___ RM-A2A-V-16

___ RM-A2A-V-17

5.3.9 <u>DISCONNECT</u> quick connects:

___RM-A2A-V-16 quick connect

___RM-A2A-V-17 quick connect

5.3.10 ___ REMOVE bolts holding sampler to the wall.

5.3.11 ___Using the Tee Handle, TRANSPORT RM-A2A-SBA away from RM-A2A skid while installing spare sampler.

Subsection 5.3 **Changeout of RM-A2A Particulate/Iodine Sampler (RM-A2A-SBA)** cont'd.

5.3.12 ___REFER TO Enclosure 7, Guidelines For Particulate/Iodine Sampler Installation and INSTALL the spare Particulate/Iodine Sampler for RM-A2A.

5.3.13 OPEN VALVES:

___ RM-A2A-V-12

___ RM-A2A-V-14

___ RM-A2A-V-16

___ RM-A2A-V-17

5.3.14 INSTALL bolts holding sampler to the wall.

5.3.15 ___Using the Tee Handle, TRANSPORT RM-A2A-SBA to pre-determined storage location

5.3.16 ___NOTIFY the control room to place RM-A2A in Normal.

5.3.17 NOTIFY CRS/SM

___RM-A2A sampler changeout is complete

___The lineup is restored to normal

CRS/SM Notification completed:______/______/______

Initial / Date / Time

OPERATIONAL SUPPORT CENTER DATA SHEET

Sample Point

[ ] RB Atmosphere
[ ] RB Vent Duct
[ ] AB Vent Duct

Gamma Isotopic

Total Activity ______________________ μCi/cc

Major Contributing Isotopes

ISOTOPE	ACTIVITY
________	______________________ μCi/cc
________	______________________ μCi/cc
________	______________________ μCi/cc
________	______________________ μCi/cc
________	______________________ μCi/cc
________	______________________ μCi/cc
________	______________________ μCi/cc
________	______________________ μCi/cc
________	______________________ μCi/cc
________	______________________ μCi/cc
________	______________________ μCi/cc
________	______________________ μCi/cc

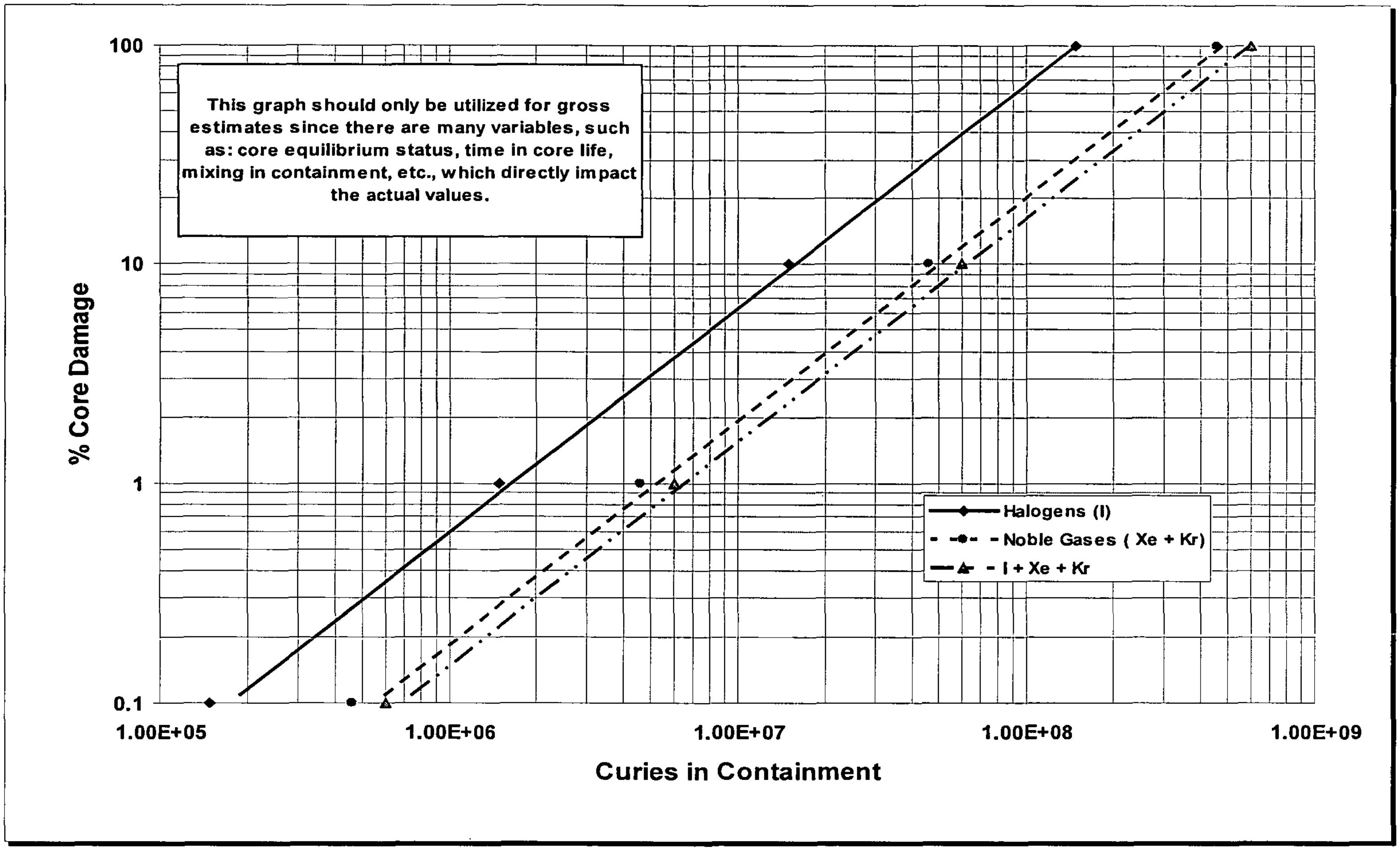
For RB atmosphere samples, calculate RB Total Activity as follows:

$$\text{RB TOTAL ACTIVITY (Ci)} = (2.0\text{E}6 \text{ cubic feet}) \times \left(\frac{28317 \text{ cc}}{\text{cubic foot}}\right) \times \left(\frac{1\text{E}-6 \text{ Ci}}{\mu\text{Ci}}\right) \times \left(\text{Total Activity} \frac{\mu\text{Ci}}{\text{cc}}\right)$$

RB Total Activity ______________________ Ci

____
Initial/Date/Time

ESTIMATE OF CORE DAMAGE FROM CURIES IN CONTAINMENT



POWER SUPPLIES

Component	Power Supply	Power Supply Location	Power Supply Normal Position	Component Operation Location
PASS Including CMP	B-ES ACDP-59	95'AB near RM-A7	Breakers Closed	NA
WSV-35	DPDP-5A Breaker 2	A-EFIC room	Locked OPEN	A-EFIC room
WSV-34	DPDP-8A Breaker 14	A-EFIC room	Locked OPEN	A-EFIC room
WSV-32	DPDP-8B Breaker 21	B-EFIC room	Locked OPEN	B-EFIC room
WSV-33	DPDP-5B Breaker 27	B-EFIC room	Locked OPEN	B-EFIC room

GUIDELINES FOR GAS GRAB SAMPLER INSTALLATION AND REMOVAL

INSTALLATION

NOTE
Grab sampler preparation is normally done in a low dose area.

1. PREPARE grab sampler
   a. ENSURE grab sampler bolted to grab sampler cart
   b. ENSURE transit cover removed from grab sampler
   c. STORE transit cover by attaching to lifting ring on grab sampler with break-away type device.
   d. OPEN WSV-72
   e. OPEN WSV-71

2. INSTALL grab sampler
   a. ENSURE ramp installed
   b. GUIDE grab sampler into sample station until sampler is within several inches of connection point
   c. CONTINUE to GENTLY guide grab sampler until fully inserted into sample station
   d. ENGAGE Cart to Station Lock
   e. GENTLY PULL Engagement Handle to connect quick connects
   f. DISENGAGE Cart to Station Lock
   g. ENSURE grab sampler moves when Engagement Handle is moved back and forth.
   h. ENGAGE Cart to Station Lock

REMOVAL

1. ENSURE ramp installed
2. SQUEEZE Engagement handle lever and PUSH engagement handle toward wall
3. DISENGAGE Cart to Station Lock
4. REMOVE grab sampler from sample station

GRAB SAMPLE SHIPMENT AND NOTIFICATIONS

NOTE
Notifications may be made in any order.

1. NOTIFY Superintendent, Nuclear Operations Materials Controls
   - A grab sample has been collected
   - Initiate acquisition process for shielded sample cask
2. NOTIFY RNP E&C Superintendent that a grab sample has been collected
3. The following information is needed:
   - Utility and plant name
   - Name and phone number of E&C Specialist to whom follow-up communication should be addressed
   - Number and type of samples being shipped
   - Measured radiation levels at surface and three feet from shipping container
   - Estimated shipping time
   - Mode of transportation
   - Carrier
   - Estimated time of arrival at RNP in Hartsville, SC
4. Use the following shipping address:

   Progress Energy Carolinas
   Robinson Nuclear Plant
   3581 West Entrance Road
   Hartsville, SC 29990
   Attn: E&C Superintendent
   Phone (Caronet) 450-1837

ESTIMATING GRAB SAMPLE CURIE CONTENT

1. DETERMINE which of the following best represents the sample. Emergency Response support personnel may be used to make this determination.
   [ ] Fuel Gap Release – use column A
   OR
   [ ] Fuel Melt Release – use column B

2. RECORD Contact Dose Rate (side of pig) from the grab sample in Table 1.

   Contact Dose Rate (side of pig) _____________ mR/hr

3. DETERMINE µCi per mR/hr
   For Gas Grab Sampler:
   [ ] Fuel Gap Release = 2.00E+4 µCi per mR/hr
   OR
   [ ] Fuel Melt Release = 2.50E+4 µCi per mR/hr

   For Particulate and Iodine Grab Sampler:
   [ ] Fuel Gap Release = 3.10E+1 µCi per mR/hr
   OR
   [ ] Fuel Melt Release = 4.70E+1 µCi per mR/hr

4. CALCULATE total activity.

   Total Activity = Contact Dose Rate (side of pig) **x** *µCi per mR/hr*

   Total Activity _____________ µCi

5. CALCULATE individual nuclide activity. RECORD results in Table 1.

   Individual Nuclide Activity = Total Activity **x** *nuclide fraction of total activity*

TABLE 1				
	Column A		Column B	
Nuclide	Nuclide Fraction of Total Activity	Individual Nuclide Activity (µCi)	Nuclide Fraction of Total Activity	Individual Nuclide Activity (µCi)
Co58			5.97E-06	
Co60			4.58E-06	
Kr85	1.22E-03		1.84E-03	
Kr85m	1.65E-02		2.49E-02	
Kr87	1.34E-03		2.02E-03	
Kr88	2.00E-02		3.01E-02	
Rb86	9.19E-05		4.16E-05	
Sr89			5.31E-03	
Sr90			2.88E-04	
Sr91			3.83E-03	
Sr92			9.24E-04	
Y90			2.83E-06	
Y91			6.48E-05	
Y92			1.49E-05	
Y93			4.69E-05	
Zr95			8.19E-05	
Zr97			6.17E-05	
Nb95			7.72E-05	
Mo99			1.04E-03	
Tc99m			3.90E-04	
Ru103			8.39E-04	
Ru105			1.58E-04	
Ru106			1.92E-04	
Rh105			3.25E-04	
Sb127			9.76E-04	
Sb129			1.02E-03	
Te127			5.53E-04	
Te127m			1.32E-04	
Te129			2.91E-05	
Te129m			9.02E-04	
Te131m			1.45E-03	
Te132			1.61E-02	
I131	1.54E-01		9.27E-02	
I132	2.09E-02		1.26E-02	
I133	2.56E-01		1.54E-01	
I134	6.63E-04		4.00E-04	
I135	1.36E-01		8.22E-02	
Xe133	3.20E-01		4.82E-01	
Xe135	3.41E-02		5.14E-02	
Cs134	2.13E-02		9.64E-03	

TABLE 1 (continued)				
	Column A		Column B	
Nuclide	Nuclide Fraction of Total Activity	Individual Nuclide Activity (µCi)	Nuclide Fraction of Total Activity	Individual Nuclide Activity (µCi)
Cs136	6.37E-03		2.88E-03	
Cs137	1.19E-02		5.39E-03	
Ba139			1.67E-04	
Ba140			9.08E-03	
La140			8.23E-05	
La141			2.10E-05	
La142			2.29E-06	
Ce141			2.09E-04	
Ce143			1.73E-04	
Ce144			1.26E-04	
Pr143			7.89E-05	
Nd147			3.51E-05	
Np239			2.18E-03	
Pu238			1.36E-07	
Pu239			3.07E-08	
Pu240			3.87E-08	
Pu241			6.52E-06	
Am241			1.72E-09	
Cm242			6.59E-07	
Cm244			3.86E-08	

SUMMARY OF CHANGES
PRR 423789

NOTE

1. Procedure Sponsor: Ensure that any changes to CH-631 that affect information contained in ERF posters, Enclosures, briefing cards, guidelines, etc. are made to those items as well.

2. Procedure Sponsor: Changes to certain parts of CH-631 may impact other EPIPs. Specifically, if any changes are made to activity calculations then ensure appropriate PRRs are initiated as needed.

SECTION	CHANGE
2.1.15	PRR 290368, 450454, Added reference to EMG-NGGC-0002 and note to section for the cross reference to EMG-NGGC-0002 when estimating sample stream activity.
1.0	PRR 321670, Added clarifying statement to purpose.
Rev Summary	PRR 411925, Added clarifying statement to the summary of changes for writers and reviewers.
2.1.5	PRR 423789, Corrected reference from RSP-600 to ADM-NGGC-0105.

Revision 15

Effective Date ________

EMERGENCY ACTION LEVEL BASES MANUAL

PROGRESS ENERGY

CRYSTAL RIVER UNIT 3

APPROVED BY: Manual Owner

(SIGNATURE ON FILE)

DATE: ____________

RESPONSIBLE UNIT:
Emergency Preparedness

TABLE OF CONTENTS

1.0 PURPOSE

This manual provides basis information for the Emergency Action Levels (EALs) contained in the Radiological Emergency Response Plan and implementing procedures. For each EAL, specific assumptions and background information are listed along with rationale explaining why the condition requires the declaration of an emergency.

Any revision to this manual must be carefully considered for impact on the Radiological Emergency Response Plan.

This manual also provides administrative control guidance for distribution and revision.

2.0 REFERENCES

2.1 DEVELOPMENTAL REFERENCES

2.1.1 NEI 97-03, Draft Final Revision 3, October 1998 (formerly NUMARC/NESP-007), Methodology for Development of Emergency Action Levels

2.2.2 Improved Technical Specifications

2.2.3 PRO-NGGC-204, Procedure Review and Approval

2.2.4 Radiological Emergency Response Plan (RERP)

2.2.5 REG-NGGC-0010, 10 CFR 50.59 Reviews

2.2.6 NRC Regulatory Issue Summary 2003-18, Use of NEI 99-01, "Methodology for Development of Emergency Action Levels,' Revision 4, Dated January 2003.

2.2.7 NEI 99-01, "Methodology for Development of Emergency Action Levels,' Revision 5, Dated February 2008

2.2.8 NCR 67029 assignment 24 documents calculation of Fission Product Barrier Matrix 5.1 sample line dose rate indication of coolant activity.

2.2.9 NRC Bulletin 2005-02, "Emergency Preparedness and Response Actions for Security-based Events".

2.2.10 EMG-NGGC-0010, Emergency Plan Change Screening and Evaluation 10 CFR 50.54(q)(3)

2.2.11 EMG-NGGC-1000, Fleet Conduct of Emergency Preparedness

2.2.12 Gilbert/Commonwealth Evaluation, "Internal Flooding of Power Plant Buildings", FCS-9852, October 12, 1988

2.2.13 Engineering Evaluation EC 86189

3.0 PERSONNEL INDOCTRINATION

3.1 DEFINITIONS

3.1.1 **AIRCRAFT:** Aircraft smaller than an AIRLINER.

3.1.1.1 **AIRLINER:** A large aircraft with the potential for causing significant damage to the Plant. (The NRC notification should designate aircraft vs. airliner.)

3.1.2 **BOMB:** An explosive device suspected of having sufficient force to damage Plant systems or structures. (See EXPLOSION.)

3.1.3 **CIVIL DISTURBANCE:** A group of persons violently protesting station operations or activities at the site. A civil disturbance is considered violent when force has been used in an attempt to injure site personnel or damage Plant property.

3.1.4 **COMMITTED DOSE EQUIVALENT (CDE):** Dose to an organ (e.g., thyroid) due to the intake of radioactive materials.

3.1.5 **CREDIBLE SITE-SPECIFIC SECURITY THREAT NOTIFICATION** – A threat specifically to CR3 confirmed and validated by Site Security or received over the Emergency Notification System (ENS) from the (Nuclear Regulatory Commission) NRC. Notification may be received from recognized law enforcement or governmental agencies (e.g. Federal Bureau of Investigation (FBI), Florida Department of Law Enforcement (FDLE), Division of Emergency Management (DEM), NRC.)

3.1.6 **EXPLOSION:** A rapid, violent, unconfined combustion, or a catastrophic failure of pressurized equipment that imparts energy of sufficient force to potentially damage permanent structures, systems, or components.

3.1.7 **EXTORTION:** An attempt to cause an action at CR3 by threat of force. Bomb threats that are unsubstantiated are <u>NOT</u> included in this definition.

3.1.8 **FIRE:** Combustion characterized by heat and light. Sources of smoke such as slipping drive belts or **overheated** electrical equipment do not constitute fires. Observation of flame is preferred but is <u>NOT</u> required if large quantities of smoke and heat are observed.

3.1.9 **HOSTAGE:** A person or object held as leverage against the station to ensure that demands will be met by CR3.

3.1.10 **HOSTILE ACTION:** An act toward a nuclear power Plant or its personnel that includes the use of violent force to destroy equipment, take hostages, and/or intimidate the licensee to achieve an end. This includes attack by air, land, or water using guns, explosives, projectiles, vehicles, or other devices used to deliver destructive force. Other acts that satisfy the destructive intent may be included. Hostile action should not be construed to include acts of civil disobedience or felonious acts that are not part of a concerted attack on the nuclear power Plant. Non-terrorism-based EALs should be used address such activities (e.g., violent acts between individuals in the owner controlled area).

3.1.11 **HOSTILE FORCE:** One or more individuals who are engaged in a determined assault, overtly or by stealth and deception, equipped with suitable weapons capable of killing, maiming, or causing destruction.

3.1.12 **IDLH LEVEL:** Level of toxic gas Immediately Dangerous to Life or Health.

3.1.13 **INTRUSION/INTRUDER:** Suspected hostile individual (outsider) present in the Protected Area without authorization. An intruder also includes a badged employee (insider) attempting to commit or providing assistance to others in committing sabotage. These activities may occur while the insider is either physically inside or outside the Protected Area. Upon identification, the insider's authorization is immediately revoked by Site Security.

3.1.14 **MODES:** The ITS based designator of Plant status based on Reactivity, Temperature and RCS status and includes operating modes 1 through 6 and defueled (no mode) as applicable. The term "MODES:ALL" applies to MODES 1-6 and defueled (no mode).

3.1.15 **OWNER-CONTROLLED AREA:** That area, including the PROTECTED AREA, that extends 4400 feet or 0.83 miles in a circle around the Reactor Building.

3.1.16 **PROTECTED AREA:** All areas within the CR3 security perimeter fence that require badged authorization for entry.

3.1.17 **RCS BARRIER:** The RCS primary side and its connections up to and including the pressurizer safety and relief valves, and other connections up to and including the primary isolation valves. An isolable leak in an interfacing or connecting system that contains reactor coolant (MU, DH, SF, WD, etc.) is <u>NOT</u> an "RCS leak."

3.1.18 **SABOTAGE:** Deliberate damage, mis-alignment, or mis-operation of Plant equipment with the intent to render the equipment unavailable. Equipment found tampered with or damaged due to malicious mischief may <u>NOT</u> meet the definition of SABOTAGE until this determination is made by Site Security.

3.1.19 **SAFE SHUTDOWN EQUIPMENT:** Equipment <u>necessary</u> to achieve and maintain the reactor subcritical with controlled decay heat removal to bring the Plant to the ITS applicable shutdown condition/mode.

3.1.20 **SECURITY CONDITION:** Any Security Event as listed in the approved security contingency plan that constitutes a threat/compromise to site security, threat/risk to site personnel, or a potential degradation to the level of safety of the plant. A SECURITY CONDITION does not involve a HOSTILE ACTION.

3.1.21 **SIGNIFICANT TRANSIENT:** An UNPLANNED event involving one or more of the following:

(1) Automatic turbine trip at >25% reactor thermal power

(2) Electrical load rejection >25% full electrical load

(3) Plant runback

(4) Reactor trip

(5) Safety injection system actuation

(6) >10% thermal power oscillations

(7) Loss of decay heat removal in Mode 4 ("Significant Transient" is <u>NOT</u> used in any Mode 5 or 6 EALs.)

3.1.22 **SITE BOUNDARY:** That area, including the PROTECTED AREA, that extends 4400 feet or 0.83 miles in a circle around the Reactor Building. Also referred to as the Owner Controlled Area.

3.1.23 **STRIKE ACTION:** Is a work stoppage within the PROTECTED AREA by a body of workers to enforce compliance with demands made. The strike actions must threaten to interrupt normal Plant operations.

3.1.24 **TOTAL EFFECTIVE DOSE EQUIVALENT (TEDE):** The sum of external dose (DDE) and the equivalent amount of whole body dose due to individual organ uptakes.

3.1.25 **UNPLANNED:** An event or action is UNPLANNED if it is NOT the expected result of normal operations, testing, or maintenance. Events that result in corrective or mitigative actions being taken in accordance with abnormal or emergency procedures are UNPLANNED.

NOTE: With specific regard to radioactive releases, a release of radioactivity is UNPLANNED if the release is NOT authorized by a Release Permit or exceeds the conditions (e.g., minimum dilution flow, maximum discharge flow, alarm setpoints, etc.) on the applicable permit.

3.1.26 **VALID:** An indication or report or condition is considered VALID when it is conclusively verified by (1) an instrument channel check, or (2) indications on related or redundant indicators, or (3) by direct observation by Plant personnel, such that doubt related to the indicator's operability, the condition's existence, or the report's accuracy is removed. Implicit in this definition is the need for timely assessment (e.g., within 15 minutes).

3.1.27 **VISIBLE DAMAGE:** Damage to equipment or structure that is readily observable without measurements, testing, or analyses. Damage is sufficient to cause concern regarding the continued operability or reliability of affected safety structure, system, or component. Example system/component damage includes: deformation due to heat or impact, denting, penetration, rupture, cracking, paint blistering due to fire. Example structure damage includes exposed and/or broken rebar, failed supports/pipe hangers, etc. Surface blemishing (e.g., paint chipping, scratches, concrete spalling) should NOT be included.

3.1.28 **VITAL AREA**: Any area, normally within the PROTECTED AREA, that contains equipment, systems, components, or material, the failure, destruction, or release of which could directly or indirectly endanger the public health and safety by exposure to radiation.

3.2 RESPONSIBILITIES

3.2.1 The Emergency Planning Coordinator (EPC) has the responsibility for interpretation and maintenance of this manual.

3.2.2 The Emergency Coordinator has the responsibility to use this manual as necessary to classify emergency conditions and identify to the EPC corrections, clarifications, or the need for additional information.

3.2.3 Document Services has the responsibility to control issue and distribution of this manual.

4.0 INSTRUCTIONS

4.1 USE OF THE MANUAL

4.1.1 LOCATE the desired EAL basis in Attachment 1 using the EAL number in the upper right corner or the title of the Fission Product Barrier.

4.1.2 IF a transient event condition is corrected before a declaration is made,
AND analyses of the event is NOT required to determine whether further Plant damage occurred while corrective actions were being taken,
THEN a declaration is NOT warranted but the event is reported and notification made to the NRC Operations Center via ENS within one hour of the event,
AND ENSURE the Emergency Preparedness staff is notified to NOTIFY the State and Local Governments on the next working day (e.g., the PORV (RCV-10) develops a leak or fails open with a leak rate of >25 gpm and the block valve (RCV-11) is closed and successfully isolates the leak to less than the EAL threshold)

4.2 MAINTENANCE OF THE MANUAL

4.2.1 MAINTAIN controlled copies of this manual in the Main Control Room, Technical Support Center (TSC), Emergency Operations Facility (EOF), and the Simulator Control Room.

4.2.2 IDENTIFY potential revisions to the manual to the EPC and document in a Document Revision Request (DRR).

4.2.3 DETERMINE if the changes decrease the effectiveness of the RERP by completing REG-NGGC-0010 (10 CFR 50.54(q)).

4.2.4 ENSURE changes comply with the requirements of EMG-NGGC-0010, Emergency Plan Change Screening and Evaluation 10 CFR 50.54(q)(3) and EMG-NGGC-1000, Fleet Conduct of Emergency Preparedness.

4.2.5 REVISE EM-202 as necessary and issue concurrently with the EAL Bases Manual.

ATTACHMENT 1

PROGRESS ENERGY

CRYSTAL RIVER UNIT 3

EMERGENCY ACTION LEVEL BASES

TABLE OF CONTENTS

FISSION PRODUCT BARRIER MATRIX BASIS

The Fission Product Barrier Matrix determines an emergency classification by assessing the status of fuel cladding, the Reactor Coolant System, and Containment. For each barrier, the matrix provides a list of symptoms of loss and a list of symptoms of potential loss. If one or more of the loss symptoms exists, the barrier is considered lost. If <u>NO</u> loss symptoms are present, but one or more of the potential loss symptoms exists, the barrier is considered potentially lost. Emergency classification based on barrier status is shown below.

GENERAL EMERGENCY:

- Loss of all three barriers
- Loss of two of three barriers with a potential loss of the third

SITE AREA EMERGENCY:

- Loss of any two barriers
- Loss of any barrier with potential loss of the other two
- Loss of any barrier with potential loss of another
- Potential loss of any two barriers

ALERT:

- Loss or potential loss of fuel clad
- Loss or potential loss of RCS

UNUSUAL EVENT:

- Loss or potential loss of containment

The Matrix simplifies the determination of the emergency classification by assigning points based on the loss or potential loss of each barrier. The sum of the points for all three barriers corresponds to an emergency classification.

Fuel Clad:	Loss = 4	Potential Loss = 3
RCS:	Loss = 4	Potential Loss = 3
Containment:	Loss = 2	Potential Loss = 1.5

>0 BUT ≤ 2 Unusual Event

>2 BUT ≤ 4 Alert

>4 BUT ≤ 8.5 Site Area Emergency

>8.5 General Emergency

FISSION PRODUCT BARRIER MATRIX BASIS (Continued)

FUEL CLAD LOSS INDICATIONS

5.1 LOSS OF FUEL CLAD If any item is checked, barrier is lost.	
1. CORE CONDITIONS IN REGION 3 OR SEVERE ACCIDENT REGION OF ICC CURVES	
2. RCS ACTIVITY >300 μCi/gm I-131 DOSE EQUIVALENT	
3. RM-G29 OR 30 >100 R/hr FOR 15 MINUTES OR LONGER	
4. EC DEEMS FUEL CLAD BARRIER IS LOST	

1. ***CORE CONDITIONS IN REGION THREE OR SEVERE ACCIDENT REGION OF INADEQUATE CORE COOLING CURVES (REFER TO EOP-07)***

   The initial core damage assessment curve is used to relate the observable parameters of incore temperature and RCS pressure to clad temperature. In region three or the severe accident region, elevated clad temperatures may exceed temperatures that will lead to zirc/water reactions and rapid failure of the clad will occur if NOT halted.

2. ***RCS ACTIVITY >300 μCi/gm I^{131} DOSE EQUIVALENT***

   This amount of coolant activity is well above that expected for iodine spikes and corresponds to about 2% to 5% fuel clad damage. This amount of clad damage indicates significant clad heating and thus the Fuel Clad Barrier is considered lost. In the absence of sample results, this coolant activity may be determined indirectly by measuring dose rates on sample lines. 100 mR/hr as measured by RM-G3 or portable instrument at two feet from the sample lines in the Nuclear Sample Room or 50 mR/hr measured by portable instrument two feet from sample lines in the PASS sample room are a conservative indication of this coolant activity.

3. ***RM-G29 OR 30 >100 R/HR FOR 15 MINUTES OR LONGER***

   Monitor readings have increased and are sustained, NOT spikes. Readings of >100 R/hr(1) on these monitors indicate activity in the Reactor Building above what would be expected for normal reactor coolant. The 15 minutes will aid in accounting for spikes and uneven mixing that occurs in the initial phases of an RCS leak in the RB. High initial concentrations that accumulate in the upper portion of the RB may lead to erroneous fuel damage assumptions.

4. ***EC DEEMS FUEL CLAD BARRIER IS LOST***

   Based on Emergency Coordinator judgment.

NOTE (1): Information on the selection of this value is in Attachment 2.

FUEL CLAD POTENTIAL LOSS INDICATIONS

5.2 POTENTIAL LOSS OF FUEL CLAD If any item is checked, barrier is potentially lost.	
1. ENTRY INTO EOP-07 BY PROCEDURAL DIRECTION	
2. CORE EXIT THERMOCOUPLES >700°F	
3. EC DEEMS FUEL CLAD BARRIER IN JEOPARDY	

1. *ENTRY INTO EOP-07 BY PROCEDURAL DIRECTION*

EOP-07 is the "Inadequate Core Cooling" procedure which indicates that there are superheated conditions in the core which may lead to clad degradation.

2. *CORE EXIT THERMOCOUPLES >700°F*

700°F is a good indicator of an extreme challenge to the ability to cool the core. Temperatures are determined using guidance in EOP-07.

3. *EC DEEMS FUEL CLAD BARRIER IN JEOPARDY*

Based on Emergency Coordinator judgment.

FISSION PRODUCT BARRIER MATRIX BASIS (Continued)

RCS LOSS INDICATIONS

6.1 LOSS OF REACTOR COOLANT SYSTEM If any item is checked, barrier is lost	
1. RCS LEAK OR OTSG TUBE LEAK RESULTING IN LOSS OF ADEQUATE SUBCOOLING MARGIN	
2. RM-G29 OR 30 > 10 R/hr FOR 15 MINUTES OR LONGER	
3. EC DEEMS RCS BARRIER IS LOST	

1. ***RCS LEAK OR OTSG TUBE LEAK RESULTING IN LOSS OF ADEQUATE SUBCOOLING MARGIN (SCM)***

A loss of adequate SCM resulting from RCS leakage would indicate that the rate of leakage from the RCS is exceeding the rate of addition from the injection system. In addition, with a loss of SCM, accurate RCS inventory cannot be determined. Therefore, the RCS boundary should be considered lost any time adequate SCM is lost due to leakage, including HPI/PORV or HPI/safety valve cooling. If SCM is regained during HPI/PORV or HPI/safety valve cooling, refer to RCS Potential Loss Factor #4.

NOTE: The momentary loss of subcooling margin that occurs with some trips from reduced pressure does NOT meet the intent of the loss of SCM and loss of RCS.

2. ***RM-G29 OR 30 > 10 R/hr FOR 15 MINUTES OR LONGER***

The reading of > 10 R/hr(1) is a value, which indicates the release of reactor coolant to the containment. The reading is based on RCS activity in normal operation concentrations.

3. ***EC DEEMS RCS BARRIER IS LOST***

Based on Emergency Coordinator judgment.

NOTE (1): Information on the selection of this value is in Attachment 2.

RCS POTENTIAL LOSS INDICATIONS

6.2 POTENTIAL LOSS OF REACTOR COOLANT SYSTEM If any item is checked, barrier is potentially lost.	
1. RCS LEAK OR OTSG TUBE LEAK REQUIRING ONE OR MORE INJECTION VALVES	
2. RCS LEAK OR OTSG TUBE LEAK RESULTS IN ES ACTUATION ON LOW RCS PRESSURE	
3. RCS PRESSURE/TEMPERATURE RELATIONSHIP VIOLATES NDT LIMITS	
4. HPI/PORV OR HPI/SAFETY VALVE COOLING IS IN PROGRESS	
5. EC DEEMS RCS BARRIER IN JEOPARDY	

1. ***RCS LEAK REQUIRING ONE OR MORE INJECTION VALVES***

   By procedure, the HPI injection valves will be used to increase RCS inventory if pressurizer level CANNOT be maintained greater than 50 inches with letdown isolated. Thus, the use of one or more injection valves would indicate leakage in excess of the normal makeup capability and therefore a potential loss of the RCS barrier. If an injection valve is being used for normal makeup, then the use of a second valve would constitute an RCS potential loss.

   OR

   OTSG TUBE LEAK REQUIRING ONE OR MORE INJECTION VALVES

   By procedure (EOP-06), the HPI injection valves will be used to increase RCS inventory if pressurizer level CANNOT be maintained at 200 inches during a tube leak event. Thus, the use of one or more injection valves would indicate leakage in excess of the normal makeup capability and therefore a potential loss of the RCS barrier. If an injection valve is being used for normal makeup, then the use of a second valve would constitute an RCS potential loss.

2. ***RCS LEAK OR OTSG TUBE LEAK RESULTS IN ES ACTUATION ON LOW RCS PRESSURE***

   Should the injection system fail or the operator fail to open the injection valves upon a failure of the Makeup system to maintain RCS inventory, RCS pressure will decrease to the ES actuation setpoint. This potential loss factor in addition to number one (above) will ensure that the RCS barrier will be considered potentially lost for any inability of the makeup system to maintain adequate inventory during a loss of coolant or OTSG tube leak event.

3 ***RCS PRESSURE TEMPERATURE RELATIONSHIP VIOLATES NDT LIMITS.***

   RCS conditions of high pressure accompanied by low temperature increase the potential for Reactor Coolant System brittle failure. This potential loss factor will ensure that the RCS barrier is considered potentially lost whenever the system is at risk of a non-ductile failure.

4. ***HPI/PORV OR HPI/SAFETY VALVE COOLING IN PROGRESS.***

   This method of cooling represents a failure of the OTSGs to remove heat from the core. The PORV must be opened to initiate cooling though the high pressure injection system. In effect, a self-imposed loss of coolant is established. The magnitude of this Plant condition is appropriately classified as an ALERT.

5. ***EC DEEMS RCS BARRIER IN JEOPARDY***

   Based on Emergency Coordinator judgment.

CONTAINMENT LOSS INDICATIONS

7.1 LOSS OF CONTAINMENT If any item is checked, barrier is lost.	
1. RAPID UNEXPLAINED RB PRESSURE DECREASE FOLLOWING INITIAL INCREASE	
2. CONTAINMENT PRESSURE OR SUMP LEVEL RESPONSE NOT CONSISTENT WITH LOCA CONDITIONS	
3. AN OTSG HAS > 10 GPM TUBE RUPTURE WITH PROLONGED STEAMING TO THE ATMOSPHERE FROM THE AFFECTED OTSG OR AN UNISOLABLE STEAM LEAK OUTSIDE RB FROM THE AFFECTED OTSG	
4. CONTAINMENT ISOLATION IS INCOMPLETE AND RELEASE PATH TO THE ENVIRONMENT EXISTS	
5. EC DEEMS CONTAINMENT BARRIER IS LOST	

1. *RAPID UNEXPLAINED RB PRESSURE DECREASE FOLLOWING INITIAL INCREASE*

During a loss of coolant event, RB pressure should rise to some value determined by the size of the leak and the response of the RB cooling systems. Following the initial peak, RB pressure should exhibit a steady decreasing trend. Any deviation from this should be the result of a known change in Plant status. A rapid decrease of unknown cause is therefore indicative of possible containment failure.

2. *CONTAINMENT PRESSURE OR SUMP LEVEL NOT CONSISTENT WITH LOCA CONDITIONS*

Sump level or containment pressure NOT increasing indicates containment bypass and a loss of containment integrity.

3. *AN OTSG HAS >10 GPM TUBE RUPTURE WITH PROLONGED STEAMING TO THE ATMOSPHERE FROM AFFECTED OTSG OR AN UNISOLABLE STEAM LEAK OUTSIDE RB FROM THE AFFECTED OTSG*

This condition is met by any of the following:

a) Intermittent or continuous use of the Atmospheric Dump Valve (ADV) (such as during a Loss of Off-Site Power) on the OTSG with the > 10 gpm tube rupture.

b) Open Main Steam Safety Valve (MSSV) on the OTSG with the > 10 gpm tube rupture that is not reseated within 15 minutes.

c) Failure of a pipe, valve, etc. on the OTSG with the > 10 gpm tube rupture that results in a direct steam path to the environment and is not isolated within 15 minutes.

NOTE:

- Lifting of an MSSV during a Plant Trip is NOT prolonged steaming if it is reseated within 15 minutes.
- If an OTSG has been successfully isolated in accordance with EOPs, then prolonged steaming NO longer exists.
- Steaming the faulted OTSG to the condenser is NOT considered prolonged steaming to the atmosphere even though there may be minor unmonitored release pathways through vents and other normal flow paths.
- If EFP-2 is running, it is NOT considered prolonged steaming if the associated steam supply valve (MSV-55 or -56) from the faulted OTSG is closed in accordance with EOPs. There is no time frame associated with the closing of the steam supply valve. If the valve cannot be closed in accordance with EOPs, then prolonged steaming exists.

CONTAINMENT LOSS INDICATIONS (Continued)

4. ***CONTAINMENT ISOLATION IS INCOMPLETE AND RELEASE PATH TO THE ENVIRONMENT EXISTS***

   This factor should be used any time an incomplete RB isolation results in a direct path from the RB atmosphere to the environment. The conditions expected for this EAL would be a known path or a visual indication of the failure or path. Confirmation may be from elevated radiation readings in areas adjacent to the RB (e.g. Aux. Bldg., Intermediate Bldg., Berm). Entry into this EAL is NOT intended to be made solely due to the Plant's inability to meet the acceptance criteria for penetration surveillances.

5. ***EC DEEMS CONTAINMENT BARRIER IS LOST***

   Based on Emergency Coordinator judgment. Entry into this EAL is NOT intended to be made solely due to the Plant's inability to meet the acceptance criteria for penetration surveillances.

CONTAINMENT POTENTIAL LOSS INDICATIONS

7.2 POTENTIAL LOSS OF CONTAINMENT If any item is checked, barrier is potentially lost.	
1. RB PRESSURE >54 psig	
2. RB HYDROGEN CONCENTRATION >4%	
3. RB PRESSURE >30 psig WITH NO BUILDING SPRAY AVAILABLE	
4. RMG-29 OR 30 READINGS >5000 R/hr	
5. CORE CONDITIONS IN SEVERE ACCIDENT REGION OF ICC CURVES FOR >15 MINUTES	
6. EC DEEMS CONTAINMENT BARRIER IN JEOPARDY	

1. ***RB PRESSURE >54 psig***

   RB design pressure is 54.4 psig. Internal pressure greater than this value has the potential to exceed design leakage values.

2. ***RB HYDROGEN CONCENTRATION >4%***

   Hydrogen concentrations > 4% are above the lower explosive limit.

3. ***RB PRESSURE >30 psig WITH NO BUILDING SPRAY AVAILABLE***

   The RB spray actuation setpoint is 30 psig. With RB pressure above this value and NO spray available, the potential exists to exceed the RB design values.

4. ***RMG-29 OR 30 READINGS >5000 R/hr (1)***

   This monitor reading is indicative of severe core damage conditions. Monitor readings have increased and are sustained, NOT spikes. Regardless of whether containment is challenged, this amount of activity in containment, if released, could have such severe consequences that it is prudent to treat this as a potential loss of containment, such that a General Emergency declaration is warranted.

5. ***CORE CONDITIONS IN SEVERE ACCIDENT REGION OF ICC CURVES FOR GREATER THAN 15 MINUTES***

   Core conditions in the Severe Accident Region represent imminent melt sequence which, if NOT corrected within 15 minutes, could lead to vessel failure and an increased potential for containment failure. The Emergency Coordinator should make the declaration as soon as it is determined that the restoration procedures have been, or will be ineffective.

6. ***EC DEEMS CONTAINMENT BARRIER IN JEOPARDY***

   Based on Emergency Coordinator judgment. Entry into this EAL is NOT intended to be made solely due to the Plant's inability to meet the acceptance criteria for penetration surveillances.

NOTE (1): Information on the selection of this value is in Attachment 2.

EAL 1.1

Gaseous Effluents

Initiating Condition:

An UNPLANNED release of gaseous radioactivity to the environment that exceeds 2 times the ODCM noble gas release setpoint for 60 minutes or longer

Emergency Action Level:

UNUSUAL EVENT
1.1 MODES: ALL
(1 or 2)
1. A VALID reading on RM-A1 or RM-A2 Normal Range monitor exceeds the high alarm setpoint for 60 minutes or longer
OR
2. Sample analysis confirms gaseous effluent being released exceeds 5.0E-4 µCi/cc for 60 minutes or longer

VALID: An indication or report or condition is considered VALID when it is conclusively verified by (1) an instrument channel check, or (2) indications on related or redundant indicators, or (3) by direct observation by Plant personnel, such that doubt related to the indicator's operability, the condition's existence, or the report's accuracy is removed. Implicit in this definition is the need for timely assessment (e.g., within 15 minutes).

Basis:

Releases in excess of the high alarm setpoint or sample results in excess of 5.0E-4 µCi/cc continuing for 60 minutes or longer represent an uncontrolled situation and hence, a potential degradation in the level of safety. The final integrated dose is NOT the primary concern here; it is the degradation in Plant control implied by the fact the release was NOT isolated within 60 minutes. Therefore, it is NOT intended for the release to be averaged over 60 minutes. For example, a release of 1.0E-3 µCi/cc for 30 minutes does NOT exceed this initiating condition. Further, the Emergency Coordinator should NOT wait until 60 minutes elapses, but declare the event as soon as it is determined the release duration will likely exceed 60 minutes. This is identified by an increasing trend in monitor readings. This does NOT include spikes or other erroneous instrument readouts.

The high alarm setpoint is set at 5.0E-4 µCi/cc on the Normal Range. It is based on extended shutdown conditions and is conservative for on-line operations. The monitor measures the Noble Gas component of a release only. In extended shutdown, particulates dominate the dose contribution, lowering the proportional concentration of Noble Gas needed to reach Protective Action Guideline doses. Since the ODCM deals exclusively with Noble Gases, the standard ODCM methodology using an annual limit of 500 mRem per year would yield an ODCM setpoint for a Noble Gas concentration that when multiplied by 200 for the Alert, would be greater than the Site Area Emergency threshold value. This necessitates administratively lowering ODCM setpoint. The ODCM setpoint was set arbitrarily to 10% of the Alert threshold value to produce a logical progression from Unusual Event to Alert to Site Area Emergency.

EALs 1.1-1.4 represent increasingly significant degradation in Plant conditions. The high alarm setpoint is conservative compared to two times the ODCM limit, but was chosen for the UNUSUAL EVENT EAL to create a logical, easily discernible progression.

CR3 Matrix Reference Number: 1.1

NEI 97-03 Reference: AU1

Gaseous Effluents

Initiating Condition:

An UNPLANNED release of gaseous radioactivity to the environment that exceeds 200 times the ODCM noble gas release setpoint for 15 minutes or longer

Emergency Action Level:

ALERT
1.2 MODES: ALL
(1 or 2)
1. A VALID reading on RM-A1 or RM-A2 Accident Range exceeds 5.0E-3 μCi/cc for 15 minutes or longer.
OR
2. Sample analysis confirms gaseous effluent being released exceeds 5.0E-3 μCi/cc for 15 minutes or longer

VALID: An indication or report or condition is considered VALID when it is conclusively verified by (1) an instrument channel check, or (2) indications on related or redundant indicators, or (3) by direct observation by Plant personnel, such that doubt related to the indicator's operability, the condition's existence, or the report's accuracy is removed. Implicit in this definition is the need for timely assessment (e.g., within 15 minutes).

Basis:

Unplanned releases in excess of 5.0E-3 μCi/cc on the Accident continuing for 15 minutes or longer represent an uncontrolled situation and hence, a potential substantial degradation in the level of safety. The primary concern for the time factor here is the loss of control of radioactive material allowing the release to continue. The Emergency Coordinator should NOT wait until 15 minutes elapses, but declare the event as soon as it is determined the release duration will likely exceed 15 minutes.

The single threshold value of 5.0E-3 μCi/cc on the Accident Range is based on extended shutdown conditions and is conservative for on-line operations. The monitor measures the Noble Gas component of a release only. In extended shutdown, particulates dominate the dose contribution, lowering the proportional concentration of Noble Gas needed to reach Protective Action Guideline doses. Since the ODCM deals exclusively with Noble Gases, the standard ODCM methodology using an annual limit of 500 mRem per year would yield an ODCM setpoint for a Noble Gas concentration that when multiplied by 200 for the Alert, would be greater than the Site Area Emergency threshold value. This necessitates administratively lowering ODCM setpoint and the threshold value. The threshold value is less than 200 times the ODCM setpoint. The threshold value was set arbitrarily to the transition point from the Normal Range to the Accident Range to produce an easily identifiable threshold and logical progression from Alert to Site Area Emergency.

EALs 1.1-1.4 represent increasingly significant degradation in Plant conditions.

CR3 Matrix Reference Number: 1.2

NEI 97-03 Reference: AA1

EAL 1.3

Gaseous Effluents

Initiating Condition:

SITE BOUNDARY dose resulting from an actual or projected release of airborne radioactivity exceeding 100 mR TEDE or 500 mR Thyroid CDE

Emergency Action Level:

SITE AREA EMERGENCY

1.3 **MODES: ALL**

(1 or 2 or 3)

1. VALID RM-A1 or RM-A2 Accident Range monitor reading exceeds the values on the following Table for the current Stability Class for 15 minutes or longer:

Stability Class	On-Line Operations (µCi/cc)	Extended Shutdown or SF Pools (µCi/cc)
A, B, or C	5.1E-1	7.5E-2
D or E	3.3E-1	5.4E-2
F or G	3.0E-1	4.5E-2

OR

2. Dose Assessment results indicate SITE BOUNDARY dose >100 mR TEDE or >500 mR thyroid CDE for the actual or projected duration of the release

OR

3. Field survey results indicate closed windows dose rates >100mR/hr expected to continue for more than one hour; or analyses of field survey samples indicate thyroid CDE of 500mR for one hour of inhalation, at or beyond SITE BOUNDARY

VALID: An indication or report or condition is considered VALID when it is conclusively verified by (1) an instrument channel check, or (2) indications on related or redundant indicators, or (3) by direct observation by Plant personnel, such that doubt related to the indicator's operability, the condition's existence, or the report's accuracy is removed. Implicit in this definition is the need for timely assessment (e.g., within 15 minutes).

SITE BOUNDARY: That area, including the PROTECTED AREA, that extends 4400 feet or 0.83 miles in a circle around the Reactor Building. Also referred to as the Owner Controlled Area.

PROTECTED AREA: All areas within the CR3 security perimeter fence that require badged authorization for entry.

Basis:

TEDE - Total Effective Dose Equivalent (external dose + equivalent amount of whole body dose due to individual organ uptakes)
CDE - Committed Dose Equivalent (dose to an organ due to the intake of radioactive materials)

Threshold values in item 1 above are provided for On-line Operations and Extended Shutdown/Spent Fuel Pools. **IF** the source of the release is unknown, **THEN** USE Extended Shutdown/Spent Fuel Pools.

"Extended Shutdown" as used in the right-hand column in item 1 above refers to all releases that occur in the unique period that began 9/26/09 (Refuel 16) and continuing through plant restart.

"SF Pools" as used in the right-hand column in item 1 above refers to releases that occur from the Spent Fuel Pools during the fuel cycle beginning at restart from the extended shutdown and continuing until additional irradiated fuel is off-loaded into the Spent Fuel Pools. When additional irradiated fuel is off-loaded, this EAL will be revised to revert back to a single column and no plant condition label will be needed.

(continued)

Gaseous Effluents, continued

Basis, continued

The threshold values in item 1 above are described in Calculation N12-0001. The monitor measures the Noble Gas component of a release only. In extended shutdown, particulates dominate the dose contribution, lowering the proportional concentration of Noble Gas needed to reach Protective Action Guideline doses.

Classification for items 2 & 3 above result from emergency response team input. For example, the Environmental Survey Team provides actual dose rates used to determine dose for the projected duration of the release. The Dose Assessment Team provides projected dose.

For Item 1 above, Stability Class groupings assume the most stable class in Groups "A, B, or C" and "D or E." The "F or G" group was calculated using "F" Stability Class due to the very low percentage of time the "G" Stability Class exists (< 0.4% based on FSAR Table 12.2).

EALs 1.1-1.4 represent increasingly significant degradation in Plant conditions.

The 100 mR integrated dose in this initiating condition is based on the 10 CFR 20 annual average population exposure. It is deemed exposures less than this are NOT consistent with the Site Area Emergency class description. These values are 10% of the EPA 400 Protective Action Guidelines (PAG).

CR3 Matrix Reference Number: **1.3**

NEI 97-03 Reference: AS1

EAL 1.4

Gaseous Effluents

Initiating Condition:

SITE BOUNDARY dose resulting from an actual or projected release of gaseous radioactivity exceeding 1000 mR TEDE or 5000 mR Thyroid CDE

Emergency Action Level:

GENERAL EMERGENCY

1.4 **MODES:** **ALL**

(1 or 2 or 3)

1. VALID RM-A1 or RM-A2 Accident Range monitor reading exceeds the values in the following table for 15 minutes or longer:

On-Line Operations (µCi/cc)	Extended Shutdown or SF Pools (µCi/cc)
3.0E+0	4.5E-1

OR

2. Dose Assessment results indicate SITE BOUNDARY dose >1000 mR TEDE or >5000 mR thyroid CDE for the actual or projected duration of the release AND core damage is suspected or has occurred

OR

3. Field survey results indicate closed windows dose rates >1000mR/hr expected to continue for more than one hour; or analyses of field survey samples indicate thyroid CDE of 5000 mR for one hour of inhalation, at or beyond SITE BOUNDARY

VALID: An indication or report or condition is considered VALID when it is conclusively verified by (1) an instrument channel check, or (2) indications on related or redundant indicators, or (3) by direct observation by Plant personnel, such that doubt related to the indicator's operability, the condition's existence, or the report's accuracy is removed. Implicit in this definition is the need for timely assessment (e.g., within 15 minutes).

SITE BOUNDARY: That area, including the PROTECTED AREA, that extends 4400 feet or 0.83 miles in a circle around the Reactor Building. Also referred to as the Owner Controlled Area.

PROTECTED AREA: All areas within the CR3 security perimeter fence that require badged authorization for entry.

Basis:

TEDE - Total Effective Dose Equivalent (external dose + equivalent amount of whole body dose due to individual organ uptakes)

CDE - Committed Dose Equivalent (dose to an organ due to the intake of radioactive materials)

Threshold values in item 1 above are provided for On-line Operations and Extended Shutdown/Spent Fuel Pools. **IF** the source of the release is unknown, **THEN** USE Extended Shutdown/Spent Fuel Pools.

"Extended Shutdown" as used in the right-hand column in item 1 above refers to all releases that occur in the unique period that began 9/26/09 (Refuel 16) and continuing through plant restart.

"SF Pools" as used in the right-hand column in item 1 above refers to releases that occur from the Spent Fuel Pools during the fuel cycle beginning at restart from the extended shutdown and continuing until additional irradiated fuel is off-loaded into the Spent Fuel Pools. When additional irradiated fuel is off-loaded, this EAL will be revised to revert back to a single column and no plant condition label will be needed.

(continued)

Gaseous Effluents, continued

Basis, continued

The threshold values in item 1 above are described in Calculation N12-0001. The monitor measures the Noble Gas component of a release only. In extended shutdown, particulates dominate the dose contribution, lowering the proportional concentration of Noble Gas needed to reach Protective Action Guideline doses.

To achieve the dose for this initiating condition, core damage with a failure of all the fission product barriers is necessary. Protective Action Guideline limits cannot be reached without some amount of fuel damage. In classifying this event, verifying that core damage is suspected or has occurred precludes erroneous protective action recommendations based on incorrect or default dose assessments when Plant conditions clearly do NOT support the magnitude of the release.

Classification for items 2 & 3 above result from emergency response team input. For example, the Environmental Survey Team provides actual dose rates used to determine dose for the projected duration of the release. The Dose Assessment Team provides projected dose.

EALs 1.1-1.4 represent increasingly significant degradation in Plant conditions.

The 1000 mR TEDE and the 5000 mR Thyroid CDE are based on the EPA protective action guidance, which indicates that public protective actions are indicated if the dose exceeds 1000 mRem TEDE or 5000 mRem Thyroid CDE. This is consistent with the emergency class description for a General Emergency. Actual meteorology (including forecasts) should be used whenever possible.

CR3 Matrix Reference Number: **1.4**

NEI 97-03 Reference: AG1

ABNORMAL RAD LEVELS / RADIOLOGICAL EFFLUENT

EAL 1.5

Liquid Effluents

Initiating Condition:

An UNPLANNED release of liquid radioactivity to the environment exceeding 2 times the ODCM release setpoint for 60 minutes or longer

Emergency Action Level:

UNUSUAL EVENT
1.5 MODES: ALL (1 or 2) 1. A VALID reading on RM-L2, RM-L7, or sample analysis confirms the release exceeds 2 times the ODCM release setpoint for 60 minutes or longer OR 2. Release continued for 60 minutes or longer with no dilution flow

VALID: An indication or report or condition is considered VALID when it is conclusively verified by (1) an instrument channel check, or (2) indications on related or redundant indicators, or (3) by direct observation by Plant personnel, such that doubt related to the indicator's operability, the condition's existence, or the report's accuracy is removed. Implicit in this definition is the need for timely assessment (e.g., within 15 minutes).

Basis:

This EAL is based on failure of the monitor interlock to perform its function or loss of dilution flow. "No dilution flow" would indicate that NO raw water flow is available. If the interlock failed, a factor of 2 times the release setpoint as compared to actual readings, can be used to judge if the EAL is exceeded. For other conditions, an evaluation of liquid effluent radioactivity must be performed and compared against the ODCM release setpoint to determine entry conditions.

Releases in excess of 2 times the ODCM limits continuing for 60 minutes or longer represent an uncontrolled situation and hence, a potential degradation in the level of safety. The final integrated dose is NOT the primary concern here; it is the degradation in Plant control implied by the fact the release was NOT isolated within 60 minutes. Therefore, it is NOT intended for the release to be averaged over 60 minutes. For example, a release of 4 times the ODCM limits for 30 minutes does NOT exceed this initiating condition. Further, the Emergency Coordinator should NOT wait until 60 minutes elapses, but declare the event as soon as it is determined the release duration will likely exceed 60 minutes. An evaluation is necessary to compare monitor setpoint against the EAL limit.

CR3 Matrix Reference Number: 1.5

NEI 97-03 Reference: AU1

EAL 1.6

Liquid Effluents

Initiating Condition:

An UNPLANNED release of liquid radioactivity to the environment exceeding 200 times the ODCM release setpoint for 15 minutes or longer

Emergency Action Level:

ALERT
1.6 **MODES: ALL**
A VALID reading on RM-L2, RM-L7, or sample analysis confirms the release exceeds 200 times the ODCM release setpoint for 15 minutes or longer

VALID: An indication or report or condition is considered VALID when it is conclusively verified by (1) an instrument channel check, or (2) indications on related or redundant indicators, or (3) by direct observation by Plant personnel, such that doubt related to the indicator's operability, the condition's existence, or the report's accuracy is removed. Implicit in this definition is the need for timely assessment (e.g., within 15 minutes).

Basis:

This EAL is based on failure of the monitor interlock to perform its function. If the interlock failed, a factor of 200 times the release setpoint as compared to actual readings, can be used to judge if the EAL is exceeded. For other conditions, an evaluation of liquid effluent radioactivity must be performed and compared against the ODCM release setpoint to determine entry conditions.

CR3 Matrix Reference Number: 1.6

NEI 97-03 Reference: AA1

EAL 1.7

Unexpected Radiation Levels

Initiating Condition:

An unexpected increase in radiation levels within the Plant

Emergency Action Level:

UNUSUAL EVENT
1.7 MODES: ALL
One or more VALID radiation monitor readings unexpectedly exceed the values below for 15 minutes or longer:
RM-G3 = 400 mR/hr RM-G4 = 600 mR/hr RM-G5 = 3,000 mR/hr RM-G9 = 100 mR/hr RM-G10 = 800 mR/hr RM-G14 = 1,000 mR/hr RM-G17 = 800 mR/hr

VALID: An indication or report or condition is considered VALID when it is conclusively verified by (1) an instrument channel check, or (2) indications on related or redundant indicators, or (3) by direct observation by Plant personnel, such that doubt related to the indicator's operability, the condition's existence, or the report's accuracy is removed. Implicit in this definition is the need for timely assessment (e.g., within 15 minutes).

Basis:

This EAL addresses unexpected increases in in-Plant radiation levels representing a degradation in the control of radioactive material, and a potential degradation in the level of safety of the Plant.

The values above represent approximately 1000 times normal monitor levels based on nominal historical data of the monitors during normal Plant operation. Portable surveys may be substituted for in Plant radiation monitors. The specific area radiation monitors were chosen as they represent potential release areas within the Plant and/or access corridors to the Plant.

Assessment should be completed such that after the 15 minutes elapsed time of the monitor exceeding the values on the table, a classification decision should be made. This Initiating Condition is NOT intended to apply to anticipated temporary increases due to planned events (e.g., incore detector movement, radwaste container movement, depleted resin transfers, etc.)

Monitor Locations:

RM-G3	(Primary Sample Room)
RM-G4	(Auxiliary Building entrance corridor)
RM-G5	(Waste Gas Decay Tank Area)
RM-G9	(Intermediate Building outside Reactor Building (RB) personnel airlock)
RM-G10	(Makeup Pump area)
RM-G14	(Spent Fuel Pool Storage Area – 143' elev. Aux. Bldg. general area)
RM-G17	(inside RB at personnel hatch)

CR3 Matrix Reference Number: **1.7**

NEI 97-03 Reference: **AU2**

EAL 1.8

Unexpected Radiation Levels

Initiating Condition:

An unexpected increase in radiation levels within the Plant impeding operation of systems required to maintain safe operations or to establish or maintain cold shutdown

Emergency Action Level:

ALERT
1.8 MODES: ALL (1 or 2) 1. VALID radiation reading greater than 15 mR/hr for 15 minutes or longer in the Control Room (RM-G1) or the Central Alarm Station (CAS) OR 2. One or more VALID radiation monitor readings unexpectedly exceed the values below for 15 minutes or longer: RM-G3 = 5,000 mR/hr RM-G4 = 5,000 mR/hr RM-G9 = 5,000 mR/hr RM-G10 = 5,000 mR/hr RM-G17 = 5,000 mR/hr

VALID: An indication or report or condition is considered VALID when it is conclusively verified by (1) an instrument channel check, or (2) indications on related or redundant indicators, or (3) by direct observation by Plant personnel, such that doubt related to the indicator's operability, the condition's existence, or the report's accuracy is removed. Implicit in this definition is the need for timely assessment (e.g., within 15 minutes).

Basis:

This addresses increased radiation levels impeding necessary access to operating stations, or other areas containing equipment operated manually, in order to maintain safe operation or perform a safe shutdown. The specific area radiation monitors were chosen as they represent access corridors to the Plant. These monitors cover general areas that would require access to maintain safe operations or to establish and maintain safe shutdown. It is this impaired ability to operate the Plant that results in the actual or potential substantial degradation of the level of safety of the Plant. The cause and/or magnitude of the increase in radiation levels is NOT a concern of this initiating condition. The Emergency Coordinator must consider the source or cause of the increased radiation levels and determine if any other Initiating Condition is involved. For example, a dose rate of 15 mR/hr in the control room may be a problem in itself. However, the increase may also be indicative of high dose rates in the containment due to a LOCA. In this latter case, a Site Area Emergency or General Emergency may be indicated by the Fission Product Barrier Matrix Initiating Conditions.

Portable surveys may be substituted for in-Plant radiation monitors. A generic emergency action level at greater than 5,000 mR/hr has been chosen for those areas in the Plant that would need to be accessed for safe operation or safe shutdown of the unit.

Monitor Locations:

RM-G3 (Primary Sample Room)

RM-G4 (Auxiliary Building entrance corridor)

RM-G9 (Intermediate Building outside RB personnel airlock)

RM-G10 (Makeup Pump area)

RM-G17 (inside RB at personnel hatch)

(continued)

Unexpected Radiation Levels, continued

Basis, continued

Assessment should be completed such that after the 15 minutes elapsed time of the monitor exceeding the values on the table, a classification decision should be made. This Initiating Condition is NOT intended to apply to anticipated temporary increases due to planned events (e.g., incore detector movement, radwaste container movement, depleted resin transfers, etc.)

CAS dose rates are determined by portable monitors.

Areas requiring continuous occupancy include the control room and any other control stations that are manned continuously, such as the Central Alarm Station. The value of 15 mR/hr is derived from the GDC 19 value of 5 Rem in 30 days with adjustment for expected occupancy times. Although Section III.D.3 of NUREG-0737, "Clarification of TMI Action Plan Requirements," provides that the 15 mR/hr value can be averaged over the 30 days, the value is used here without averaging, as a 30 day duration implies an event potentially more significant than an Alert.

CR3 Matrix Reference Number: 1.8

NEI 97-03 Reference: AA3

EAL 1.9

Fuel Handling
Spent Fuel Pool or Transfer Canal Water Level

Initiating Condition:
An uncontrolled water level decrease in spent fuel pool or transfer canal with fuel remaining covered

Emergency Action Level:

UNUSUAL EVENT
1.9 MODES: ALL
(1 and 2)
1. (a or b)
a. Uncontrolled level decrease resulting in indications of -2.5 feet in spent fuel pool
OR
b. Confirmed Plant personnel report of uncontrolled significant water level drop in spent fuel pool or transfer canal when Spent Fuel transfer tubes are open
AND
2. Fuel remains covered with water

Basis:
The "-2.5 feet" indication is relative to the normal "zero" reading for spent fuel pool level and represents the minimum 23 feet of water (156 feet Plant datum) over the top of the fuel as described in Improved Technical Specifications.

A level decrease that cannot be readily isolated is considered uncontrolled.

CR3 Matrix Reference Number: **1.9**
NEI 97-03 Reference: AU2

EAL 1.10

Fuel Handling/Fuel Handling Pool Water Level

Initiating Condition:

Damage to irradiated fuel or loss of water level has or will uncover irradiated fuel outside the reactor vessel

Emergency Action Level:

ALERT
1.10 MODES: ALL
(1 or 2)
1. (a and b)
a. Plant personnel report damage of irradiated fuel
AND
b. VALID high alarm as indicated on RM-G15 or RM-G16
OR
2. Plant personnel report spent fuel pool or transfer canal water level drop has or will exceed makeup capacity such that irradiated fuel will be uncovered

VALID: An indication or report or condition is considered VALID when it is conclusively verified by (1) an instrument channel check, or (2) indications on related or redundant indicators, or (3) by direct observation by Plant personnel, such that doubt related to the indicator's operability, the condition's existence, or the report's accuracy is removed. Implicit in this definition is the need for timely assessment (e.g., within 15 minutes).

Basis:

There is time available to take corrective actions, and there is little potential for substantial fuel damage if corrective actions are effective. Thus, an Alert classification for this event is appropriate. Escalation, if appropriate, would occur via other Abnormal Rad Levels/Radiological Effluents Initiating Conditions or Emergency Coordinator judgment.

Monitor Locations:

RM-G15 (Auxiliary Building Fuel Handling Bridge)
RM-G16 (Reactor Building Fuel Handling Bridge)

CR3 Matrix Reference Number: 1.10

NEI 97-03 Reference: AA2

EAL 2.1

Earthquake Experienced

Initiating Condition:
Earthquake detected by seismic instrumentation and sensed by Control Room personnel

Emergency Action Level:

UNUSUAL EVENT
2.1 MODES: ALL
(1 and 2)
1. Ground motion sensed by Plant personnel **AND**
2. Confirmed earthquake causing Annunciator C-3-14 "Seismic System Trouble" alarm

Basis:
Damage may be caused to some portions of the site, but should NOT affect ability of safe shutdown equipment to operate. Method of detection is based on instrumentation, validated by a reliable source, or operator assessment. As defined in the EPRI-sponsored, "Guidelines for Nuclear Plant Response to an Earthquake," dated October 1989, a "felt earthquake" is:

> "An earthquake of sufficient intensity such that: (a) the ground motion is felt at the nuclear Plant site and recognized as an earthquake based on a consensus of control room operators on duty at the time, and (b) for Plants with operable seismic instrumentation, the seismic switches of the Plant are activated."

CR3 Matrix Reference Number: 2.1
NEI 97-03 Reference: HU1

EAL 2.2

Earthquake Experienced

Initiating Condition:

Earthquake detected by seismic instrumentation and sensed by Control Room personnel greater than Operating Basis Earthquake

Emergency Action Levels:

ALERT
2.2 MODES: ALL
(1 and 2)
1. Ground motion sensed by Plant personnel or confirmed Annunciator C-3-14 "Seismic System Trouble" alarm AND
2. (a or b)
a. Analysis confirms the earthquake at >0.05g OR
b. Indications show degraded SAFE SHUTDOWN EQUIPMENT performance due to the earthquake

SAFE SHUTDOWN EQUIPMENT: Equipment necessary to achieve and maintain the reactor sub critical with controlled decay heat removal to bring the Plant to the ITS applicable shutdown condition/mode.

Basis:

Seismic events of this magnitude can cause damage to safety functions.

Analysis of earthquakes is completed using AP-961 and its supporting procedures. The analysis to determine the magnitude of an earthquake may take an extended period of time. If it is determined even after several hours that the earthquake was >0.05g, the event should be classified.

This EAL is intended to address an earthquake resulting in a Plant vital area being subjected to forces beyond design limits, and thus damage is assumed to have occurred to Plant safe shutdown equipment. Assessing SAFE SHUTDOWN EQUIPMENT performance is NOT interpreted as mandating a lengthy damage assessment before classification and NO attempt is made to assess the actual magnitude of the damage.

Additional information on the earthquake (confirmation and magnitude) can be obtained from the U. S. Geological Survey - Golden, Colorado at (303) 273-8500.

CR3 Matrix Reference Number: **2.2**

NEI 97-03 Reference: HA1

External Flooding

Initiating Condition:

Flood being experienced

Emergency Action Level:

UNUSUAL EVENT
2.3 MODES: ALL
Intake canal level or visual observation indicates flood water level ≥ 98 feet

Basis:

This EAL covers flooding due to natural phenomena. This EAL can be a precursor of more serious events. In particular, since CR3 may be subject to severe weather as defined in the NUMARC station blackout initiatives, this includes action based on activation of the severe weather mitigation procedures for flooding (e.g., precautionary shutdowns, diesel testing, staff call-outs, etc.).

Ninety-eight (98) feet is contained within the discharge and intake canal banks. The top of the concrete wall at the intake structure is 99 feet.

The highest water level recorded at CR3 was 99.5 feet during the 03/13/93 "No Name Storm."

At 98 feet, there is NO immediate impact on Plant equipment but heightened awareness is appropriate should the level increase.

CR3 Matrix Reference Number: 2.3

NEI 97-03 Reference: HU1

EAL 2.4

External Flooding

Initiating Condition:

Flood being experienced

Emergency Action Level:

ALERT
2.4 MODES: ALL (1 and 2) 1. Intake canal level or visual observation indicates flood water level ≥ 98 feet AND 2. Indications show degraded SAFE SHUTDOWN EQUIPMENT performance due to the flooding

SAFE SHUTDOWN EQUIPMENT: Equipment necessary to achieve and maintain the reactor sub critical with controlled decay heat removal to bring the Plant to the ITS applicable shutdown condition/mode.

Basis:

This EAL covers flooding due to natural phenomena.

This EAL is intended to address flooding that may have resulted in a Plant vital area being subjected to forces beyond design limits, and thus damage may be assumed to have occurred to Plant safety systems. Assessing SAFE SHUTDOWN EQUIPMENT performance is NOT interpreted as mandating a lengthy damage assessment before classification and NO attempt is made to assess the actual magnitude of the damage.

If damage from the flooding is clearly contained and localized to one train, and safe shutdown capability exists, then item 2 of the EAL is NOT met. If the extent of the damage is uncertain in terms of loss of safe shutdown capability, then entry into this EAL is required.

CR3 Matrix Reference Number: 2.4

NEI 97-03 Reference: HA1

EAL 2.5

Hurricane

Initiating Condition:
Hurricane Warning

Emergency Action Level:

UNUSUAL EVENT
2.5 MODES: ALL
The Plant is within a Hurricane Warning area

Basis:
This EAL can be a precursor of more serious events. In particular, since CR3 may be subject to severe weather as defined in the NUMARC station blackout initiatives.

This should include a notification from the National Hurricane Center via the State Warning Point.

CR3 Matrix Reference Number: 2.5
NEI 97-03 Reference: HU1

EAL 2.6

Tornado

Initiating Condition:

Tornado within the PROTECTED AREA

Emergency Action Level:

UNUSUAL EVENT
2.6 **MODES: ALL**
Report by Plant personnel of a Tornado striking within the PROTECTED AREA

PROTECTED AREA: All areas within the CR3 security perimeter fence that require badged authorization for entry.

Basis:

This EAL is based on the assumption a tornado strikes (touches down) within the protected area boundary and may have damaged Plant structures containing functions or systems required for safe shutdown of the Plant. If such damage is confirmed visually or by other in-Plant indications, the event may be escalated to an Alert.

Waterspouts remaining intact after coming onshore/land are classified as tornadoes.

CR3 Matrix Reference Number: 2.6

NEI 97-03 Reference: HU1

EAL 2.7

Tornado/High Winds

Initiating Condition:

Tornado or High Winds or windborne object(s) strike structures and results in significant VISIBLE DAMAGE

Emergency Action Level:

ALERT
2.7 MODES: ALL (1 or 2) 1. Tornado or High Winds or windborne object(s) cause significant VISIBLE DAMAGE to any of the following structures: - Auxiliary Building, - BWST, - Control Complex, - Diesel Generator Building (EGDG-1A/!B) - EFT-2 Building, - Intermediate Building, - Reactor Building - EFP-3 Building OR 2. Indications show degraded SAFE SHUTDOWN EQUIPMENT performance due to the tornado or high winds or windborne objects

VISIBLE DAMAGE: Damage to equipment or structure that is readily observable without measurements, testing, or analyses. Damage is sufficient to cause concern regarding the continued operability or reliability of affected safety structure, system, or component. Example system/component damage includes: deformation due to heat or impact, denting, penetration, rupture, cracking, paint blistering due to fire. Example structure damage includes exposed and/or broken rebar, failed supports/pipe hangers, etc. Surface blemishing (e.g., paint chipping, scratches, concrete spalling) should NOT be included.

SAFE SHUTDOWN EQUIPMENT: Equipment necessary to achieve and maintain the reactor sub critical with controlled decay heat removal to bring the Plant to the ITS applicable shutdown condition/mode.

Basis:

This EAL addresses events that may have resulted in a Plant vital area being subjected to forces beyond design limits, and thus damage may be assumed to have occurred to Plant safety systems. Assessing SAFE SHUTDOWN EQUIPMENT performance is NOT interpreted as mandating a lengthy damage assessment before classification and NO attempt is made to assess the actual magnitude of the damage.

As an example, the highest recorded sustained wind speed at CR3 during the 03/13/93 "No Name Storm" was 56 mph and NO VISIBLE DAMAGE resulted.

Sheet metal damage to the Spent Fuel Floor walls or roof does NOT constitute significant damage to the Auxiliary Building.

Waterspouts remaining intact after coming onshore/land are classified as tornadoes.

CR3 Matrix Reference Number: 2.7

NEI 97-03 Reference: HA1

EAL 2.8

Accidental Aircraft/Vehicle Crash

Initiating Condition:

Aircraft or Vehicle crash within the Protected Area potentially damaging Plant structures containing functions and systems required for safe shutdown of the Plant

Emergency Action Level:

UNUSUAL EVENT
2.8 MODES: ALL Report by Plant personnel of Aircraft or Vehicle Crash involving the following structures: - Auxiliary Building, - BWST - Control Complex - Diesel Generator Building (EGDG-1A/IB) - EFT-2 Building - Intermediate Building - Reactor Building - EFP-3 Building

Basis:

This EAL is intended to address the accidental crash of a plane, helicopter, or vehicle potentially damaging Plant structures containing functions and systems required for safe shutdown of the Plant. Automobiles, trucks, and forklifts are vehicles within the context of this EAL. The intent is to address any vehicle large enough that can cause significant damage to Plant structures. This EAL is NOT intended to include cosmetic damage because of light contact between vehicles and listed structures. This EAL does NOT include purposeful attacks to these structures (refer to Security EALs).

CR3 Matrix Reference Number: **2.8**

NEI 97-03 Reference: HU1

EAL 2.9

Accidental Aircraft / Vehicle Crash

Initiating Condition:

Aircraft or Vehicle strikes vital structures and results in significant VISIBLE DAMAGE

Emergency Action Level:

ALERT
2.9 MODES: ALL (1 or 2) 1. Confirmed report of significant VISIBLE DAMAGE to any of the following structures: - Auxiliary Building - BWST - Control Complex - Diesel Generator Building (EGDG-1A/!B) - EFT-2 Building - Intermediate Building - Reactor Building - EFP-3 Building OR 2. Indications show degraded SAFE SHUTDOWN EQUIPMENT performance due to the Aircraft or Vehicle Crash

VISIBLE DAMAGE: Damage to equipment or structure that is readily observable without measurements, testing, or analyses. Damage is sufficient to cause concern regarding the continued operability or reliability of affected safety structure, system, or component. Example system/component damage includes: deformation due to heat or impact, denting, penetration, rupture, cracking, paint blistering due to fire. Example structure damage includes exposed and/or broken rebar, failed supports/pipe hangers, etc. Surface blemishing (e.g., paint chipping, scratches, concrete spalling) should NOT be included.

SAFE SHUTDOWN EQUIPMENT: Equipment necessary to achieve and maintain the reactor sub critical with controlled decay heat removal to bring the Plant to the ITS applicable shutdown condition/mode.

Basis:

This EAL is intended to address an accidental crash of a plane, helicopter, vehicle crash damaging Plant structures containing functions and systems required for safe shutdown of the Plant. Automobiles, trucks, and forklifts are also vehicles within the context of this EAL. This EAL does NOT include purposeful attacks to these structures (refer to Security EALs).

This EAL is intended to address events that may have resulted in a Plant vital area being subjected to forces beyond design limits, and thus damage may be assumed to have occurred to Plant safety systems. Assessing SAFE SHUTDOWN EQUIPMENT performance is NOT interpreted as mandating a lengthy damage assessment before classification and NO attempt is made to assess the actual magnitude of the damage.

If damage from the vehicle or aircraft crash is clearly contained and localized to one train, and safe shutdown capability exists, then the EAL is NOT met. If the extent of the damage is uncertain in terms of loss of safe shutdown capability, then entry into this EAL is required.

CR3 Matrix Reference Number: **2.9**

NEI 97-03 Reference: HA1

EAL 2.10

Toxic or Flammable Gas

Initiating Condition:

Release of Toxic or Flammable Gas within, or potentially affecting the Protected Area

Emergency Action Level:

UNUSUAL EVENT
2.10 MODES: ALL (1 or 2) 1. Report or detection of Toxic or Flammable Gas within the SITE BOUNDARY that could enter the Protected Area at levels > IDLH or > 25% Lower Explosive Limits affecting NORMAL OPERATION OF THE PLANT. OR 2. Confirmed notification by PE, County, or State personnel to evacuate or shelter site personnel based on an offsite event

SITE BOUNDARY: That area, including the PROTECTED AREA, that extends 4400 feet or 0.83 miles in a circle around the Reactor Building. Also referred to as the Owner Controlled Area.

IDLH: Immediately Dangerous to Life or Health

NORMAL OPERATION OF THE PLANT: Activities at the plant site Associated with routine testing, maintenance, or equipment operations, in accordance with normal operating or administrative procedures. Entry into Abnormal or Emergency Operating Procedures, or deviation from normal security or radiological controls posture, is a departure from normal plant operations.

Basis:

This Initiating Condition is based on releases in concentrations within the Site Boundary that could; (1) affect the health and safety of Plant personnel; (2) affect the safe operation of the Plant; or (3) potentially put the Plant within an evacuation or sheltering area due to an offsite event.

Gases within the Site Boundary that are below life-threatening (< Immediately Dangerous to Life or Health [IDLH]) or flammable concentrations are NOT applicable to this Initiating Condition. Concentrations at these levels would NOT affect Plant personnel or the safe operation of the Plant. Gases at the Site Boundary that are above life-threatening or flammable concentrations, yet have NOT exceeded those concentrations within a facility structure, would satisfy the first EAL and would require the declaration of an Unusual Event.

Toxic or Flammable gases which are released offsite (e.g., transportation accident) confirmed by Progress Energy, County, Local, or State personnel have the potential for requiring the evacuation or sheltering of the Owner Controlled Area (Site Boundary).

A localized/small-scale event within the Site Boundary that may involve gases at life-threatening or flammable concentrations do NOT meet the intent of this Initiating Condition.

CR3 Matrix Reference Number: 2.10

NEI 97-03 Reference: HU3

EAL 2.11

Toxic or Flammable Gas

Initiating Condition:
Release of toxic or flammable gases within a facility structure which jeopardizes operation of systems required to maintain safe operations or to establish or maintain Cold Shutdown

Emergency Action Level:

ALERT
2.11 MODES: ALL
(1 or 2 or 3)
1. Flammable Gas levels > 25% Lower Explosive Limit in areas required to maintain safe operations or establish and maintain cold shutdown
OR
2. Toxic Gas levels ≥ IDLH levels in areas that require continuous occupancy to maintain safe operation or establish or maintain cold shutdown
OR
3. Toxic Gas levels ≥ IDLH levels within the PROTECTED AREA such that Plant personnel are unable to perform actions necessary to maintain safe operations or establish and maintain cold shutdown using protective equipment

PROTECTED AREA: All areas within the CR3 security perimeter fence that require badged authorization for entry.

Basis:
This Initiating Condition is based on gases that have entered a Plant structure affecting the safe operation of the Plant. This Initiating Condition applies to buildings and areas contiguous to Plant vital areas or other significant buildings or areas.

Concentrations at these amounts will restrict or prevent normal actions from being taken to operate the Plant. This EAL is NOT intended to include precautionary general evacuation of personnel.

If personnel can safely enter areas NOT required to be continuously occupied using protective equipment, this Initiating Condition/EAL is NOT met.

IDLH - Immediately Dangerous to Life or Health

CR3 Matrix Reference Number: 2.11
NEI 97-03 Reference: HA3

EAL 2.12

Explosions/Catastrophic Pressurized Equipment Failure

Initiating Condition:

UNPLANNED EXPLOSION or catastrophic failure of pressurized equipment within the PROTECTED AREA

Emergency Action Level:

UNUSUAL EVENT
2.12 **MODES: ALL** Report by Plant personnel of VISIBLE DAMAGE to permanent structures or equipment within the PROTECTED AREA due to an EXPLOSION or catastrophic failure of pressurized equipment *Refer to Security Event*

VISIBLE DAMAGE: Damage to equipment or structure that is readily observable without measurements, testing, or analyses. Damage is sufficient to cause concern regarding the continued operability or reliability of affected safety structure, system, or component. Example system/component damage includes: deformation due to heat or impact, denting, penetration, rupture, cracking, paint blistering due to fire. Example structure damage includes exposed and/or broken rebar, failed supports/pipe hangers, etc. Surface blemishing (e.g., paint chipping, scratches, concrete spalling) should NOT be included.

PROTECTED AREA: All areas within the CR3 security perimeter fence that require badged authorization for entry.

EXPLOSION: A rapid, violent, unconfined combustion, or a catastrophic failure of pressurized equipment that imparts energy of sufficient force to potentially damage permanent structures, systems, or components.

Basis:

For this EAL, only those explosions or catastrophic failure of pressurized equipment of sufficient force to damage permanent structures or equipment within the PROTECTED AREA should be considered. NO attempt is made in this EAL to assess the actual magnitude of the damage. The occurrence of the explosion or catastrophic failure of pressurized equipment with reports of evidence of damage (e.g., deformation, scorching) is sufficient for declaration. The concern is NOT with the pressurized equipment that catastrophically failed, but with the damage to the structures or equipment in the area caused by the release of energy.

This EAL is NOT intended to cover small pipe cracks or small steam/feedwater leaks.

The Emergency Coordinator also needs to consider security aspects of the explosion and, if applicable, refer to the Security EALs.

CR3 Matrix Reference Number: 2.12

NEI 97-03 Reference: HU1

EAL 2.13

Explosions/Catastrophic Pressurized Equipment Failure

Initiating Condition:

EXPLOSION or catastrophic failure of pressurized equipment within the Plant affecting the operability of SAFE SHUTDOWN EQUIPMENT

Emergency Action Level:

ALERT
2.13 MODES: ALL **(1 or 2)** 1. EXPLOSION or catastrophic failure of pressurized equipment causes significant VISIBLE DAMAGE to any of the following structures: - Auxiliary Building - BWST - Control Complex - Diesel Generator Building (EGDG-1A/!B) - EFT-2 Building, - Intermediate Building - Reactor Building - EFP-3 Building OR 2. Indications show degraded SAFE SHUTDOWN EQUIPMENT performance due to the EXPLOSION or pressurized equipment failure

EXPLOSION: A rapid, violent, unconfined combustion, or a catastrophic failure of pressurized equipment that imparts energy of sufficient force to potentially damage permanent structures, systems, or components.

VISIBLE DAMAGE: Damage to equipment or structure that is readily observable without measurements, testing, or analyses. Damage is sufficient to cause concern regarding the continued operability or reliability of affected safety structure, system, or component. Example system/component damage includes: deformation due to heat or impact, denting, penetration, rupture, cracking, paint blistering due to fire. Example structure damage includes exposed and/or broken rebar, failed supports/pipe hangers, etc. Surface blemishing (e.g., paint chipping, scratches, concrete spalling) should NOT be included.

SAFE SHUTDOWN EQUIPMENT: Equipment necessary to achieve and maintain the reactor sub critical with controlled decay heat removal to bring the Plant to the ITS applicable shutdown condition/mode.

Basis:

This EAL is intended to address events that may have resulted in a Plant vital area being subjected to forces beyond design limits, and thus damage may be assumed to have occurred to Plant safe shutdown equipment. Assessing SAFE SHUTDOWN EQUIPMENT performance is NOT interpreted as mandating a lengthy damage assessment before classification. NO attempt is made in this EAL to assess the actual magnitude of the damage. The observation of damage to a structure is sufficient to make a declaration.

The concern is NOT with the pressurized equipment that catastrophically failed, but with the actual or potential damage to safe shutdown equipment caused by the release of energy.

A catastrophic failure of pressurized equipment does not include small lines or equipment that may cause only localized damage. A catastrophic failure of a steam line is of sufficient size that would be characterized by an uncontrolled depressurization of the secondary side.

CR3 Matrix Reference Number: 2.13

NEI 97-03 Reference: HA2

EAL 2.14

Fire

Initiating Condition

FIRE within the PROTECTED AREA that could affect SAFE SHUTDOWN EQUIPMENT

Emergency Action Level:

UNUSUAL EVENT
2.14 MODES: ALL (1 and 2) 1. FIRE in or threatening one of the following structures: - Auxiliary Building - BWST - Control Complex, - Diesel Generator Building (EGDG-1A/!B) - EFT-2 Building - Intermediate Building - Reactor Building - EFP-3 Building AND 2. FIRE not extinguished within 15 minutes from either Control Room notification or receipt of a VALID fire alarm in the Control Room

FIRE: Combustion characterized by heat and light. Sources of smoke such as slipping drive belts or **overheated** electrical equipment do not constitute fires. Observation of flame is preferred but is NOT required if large quantities of smoke and heat are observed.

VALID: An indication or report or condition is considered VALID when it is conclusively verified by (1) an instrument channel check, or (2) indications on related or redundant indicators, or (3) by direct observation by Plant personnel, such that doubt related to the indicator's operability, the condition's existence, or the report's accuracy is removed. Implicit in this definition is the need for timely assessment (e.g., within 15 minutes).

Basis:

This EAL is to address the magnitude and extent of fires that may be potentially significant precursors to damage to safety systems. This excludes such items as fires within administration buildings, wastebasket fires, and other small fires of NO safety consequence. This Initiating Condition applies to buildings and areas contiguous to Plant vital areas or other significant buildings or areas.

Validation of the alarm in this context means those actions taken in the control room or other location to determine the control room alarm is NOT spurious. Fire in other areas adjacent to vital areas may warrant classification if the fire is of a magnitude that threatens vital areas.

The 15-minute time period begins with the time when a credible notification that a fire is occurring or the time a VALID fire detection system alarm is received. The intent of the 15-minute duration is to discriminate against small fires that are readily extinguished.

OP-880A, Appendix "R" Post-Fire Safe Shutdown Information contains additional information on fire damage assessment

CR3 Matrix Reference Number: **2.14**

NEI 97-03 Reference: HU2

EAL 2.15

Fire

Initiating Condition:

FIRE affecting the operability of SAFE SHUTDOWN EQUIPMENT

Emergency Action Level:

ALERT
2.15 MODES: ALL **(1 or 2)** 1. Report by Plant personnel of VISIBLE DAMAGE to SAFE SHUTDOWN EQUIPMENT due to the FIRE **OR** 2. Indications show degraded SAFE SHUTDOWN EQUIPMENT performance due to the FIRE

VISIBLE DAMAGE: Damage to equipment or structure that is readily observable without measurements, testing, or analyses. Damage is sufficient to cause concern regarding the continued operability or reliability of affected safety structure, system, or component. Example system/component damage includes: deformation due to heat or impact, denting, penetration, rupture, cracking, paint blistering due to fire. Example structure damage includes exposed and/or broken rebar, failed supports/pipe hangers, etc. Surface blemishing (e.g., paint chipping, scratches, concrete spalling) should NOT be included.

SAFE SHUTDOWN EQUIPMENT: Equipment necessary to achieve and maintain the reactor sub critical with controlled decay heat removal to bring the Plant to the ITS applicable shutdown condition/mode.

FIRE: Combustion characterized by heat and light. Sources of smoke such as slipping drive belts or **overheated** electrical equipment do not constitute fires. Observation of flame is preferred but is NOT required if large quantities of smoke and heat are observed.

Basis:

The key to classifying fires as an Alert is the damage because of the incident. The fact that the equipment required for safe shutdown of the unit has been affected or damaged because of the fire is the driving force for declaring the Alert.

If damage from the fire is clearly contained and localized to one train, and safe shutdown capability exists, then the EAL is NOT met. If the extent of the damage is uncertain in terms of loss of safe shutdown capability, then entry into this EAL is required.

CR3 Matrix Reference Number: 2.15

NEI 97-03 Reference: HA2

EAL 2.16

Control Room Evacuation

Initiating Condition:
Evacuation of Control Room is Required

Emergency Action Level:

ALERT
2.16 MODES: ALL
Control Room evacuation is required per AP-990, "Shutdown Outside of the Control Room"

Basis:
With the control room evacuated, additional support, monitoring and direction through the Technical Support Center and/or the Emergency Operations Facility is necessary.

Declaration of an Alert may be delayed until the transfer to remote shutdown is completed. This is appropriate since establishing control of the Plant takes precedence.

CR3 Matrix Reference Number: 2.16
NEI 97-03 Reference: HA5

EAL 2.17

Control Room Evacuation

Initiating Condition:
Evacuation of Control Room is Initiated and Plant Control cannot be established

Emergency Action Level:

SITE AREA EMERGENCY
2.17 MODES: ALL (1 and 2) 1. Control Room evacuation is required per AP-990, "Shutdown Outside of the Control Room" AND 2. Control of the necessary equipment not established per AP-990 within 15 minutes

Basis:
The 15 minutes begins at the first attempt to turn the transfer switch to transfer control from the Main Control Room to the Remote Shutdown Panel.

The timely transfer of control to alternate control areas has NOT been accomplished. The failure to transfer control would be evidenced by deteriorating reactor coolant system or steam generator parameters.

The determination of whether control is established at the Remote Shutdown Panel is based upon the judgment of the Shift Manager. The Shift Manager is expected to make a reasonable, informed judgment within fifteen minutes of the transfer from the Control Room that the operating crew has control of the Plant from the Remote Shutdown Panel.

CR3 Matrix Reference Number: 2.17

NEI 97-03 Reference: HS2

EAL 2.18

Security Event

Initiating Conditions:

Confirmed SECURITY CONDITION or threat which indicates a potential degradation in the level of safety of the Plant

Emergency Action Level:

UNUSUAL EVENT
2.18 **MODES: ALL** **(1 or 2 or 3)** Report by Security Shift Supervisor or NRC of one or more of the following events: 1. A validated notification from NRC providing information of an AIRCRAFT or AIRLINER threat. OR 2. A CREDIBLE SITE-SPECIFIC SECURITY THREAT NOTIFICATION OR 3. A SECURITY CONDITION that does NOT involve a HOSTILE ACTION as reported by the Security Shift Supervisor.

AIRCRAFT – Aircraft smaller than an AIRLINER.

AIRLINER: A large aircraft with the potential for causing significant damage to the Plant. (The NRC notification should designate aircraft vs. airliner.)

CREDIBLE SITE-SPECIFIC SECURITY THREAT NOTIFICATION – A threat specifically to CR3 confirmed and validated by Site Security or received over the Emergency Notification System (ENS) from the Nuclear Regulatory Commission (NRC). Notification may be received from recognized law enforcement or governmental agencies (e.g. Federal Bureau of Investigation (FBI), Florida Department of Law Enforcement (FDLE), Division of Emergency Management (DEM), NRC.)

HOSTILE ACTION: An act toward a nuclear power Plant or its personnel that includes the use of violent force to destroy equipment, take hostages, and/or intimidate the licensee to achieve an end. This includes attack by air, land, or water using guns, explosives, projectiles, vehicles, or other devices used to deliver destructive force. Other acts that satisfy the destructive intent may be included. Hostile action should not be construed to include acts of civil disobedience or felonious acts that are not part of a concerted attack on the nuclear power Plant. Non-terrorism-based EALs should be used address such activities.

SECURITY CONDITION: Any Security Event as listed in the approved security contingency plan that constitutes a threat/compromise to site security, threat/risk to site personnel, or a potential degradation to the level of safety of the plant. A SECURITY CONDITION does not involve a HOSTILE ACTION.

Basis:

Note: Timely and accurate communication between Security Shift Supervision and the Control Room is crucial for the implementation of effective Security EALs.

Security events which do not represent a potential degradation in the level of safety of the plant are reported under 10 CFR 73.71 or in some cases under 10 CFR 50.72. Security events assessed as HOSTILE ACTIONS are classifiable under EALs 2.19, 2.20, and 2.21.

A higher initial classification could be made based upon the nature and timing of the security threat and potential consequences. The licensee shall consider upgrading the emergency response status and emergency classification level in accordance with the site's Safeguards Contingency Plan and Emergency Plan.

(continued)

NATURAL / MAN-MADE HAZARDS AND EC JUDGMENT

EAL 2.18

EAL #1

The intent of this EAL is to ensure that notifications for the AIRCRAFT or AIRLINER threat are made in a timely manner and that offsite response organizations and CR3 personnel are at a state of heightened awareness regarding the credible threat.

This EAL is met when CR3 receives information regarding an AIRCRAFT or AIRLINER threat from NRC. Validation is performed by calling the NRC or by other approved methods of authentication.

The NRC Headquarters Operations Officer (HOO) will communicate to CR3 if the threat involves an AIRLINER. The status and size of the plane may be provided by NORAD through the NRC.

Escalation to Alert emergency classification level via EAL 2.19 would be appropriate if the threat involves an AIRLINER within 30 minutes of the plant.

EAL #2

This threshold is included to ensure that appropriate notifications for the security threat are made in a timely manner. This includes information of a credible threat. Only the plant to which the specific threat is made need declare the Notification of an Unusual Event.

The determination of "credible" is made through use of information found in the Safeguards Contingency Plan.

EAL #3

Reference is made to security shift supervision because these individuals are the designated personnel on-site qualified and trained to confirm that a security event is occurring or has occurred. Training on security event classification confirmation is closely controlled due to the strict secrecy controls placed on the plant Safeguards Contingency Plan.

This threshold is based on site specific security plans. Site specific Safeguards Contingency Plans are based on guidance provided by NEI 03-12.

CR3 Matrix Reference Number: 2.18

NEI 99-01 Reference: HU4

EAL 2.19

Security Event

Initiating Condition:

HOSTILE ACTION in the OWNER CONTROLLED AREA or airborne attack threat.

Emergency Action Level:

ALERT
2.19 MODES: ALL (1 or 2) 1. A validated notification from NRC of an AIRLINER attack threat less than 30 minutes away OR 2. A HOSTILE ACTION is occurring or has occurred within the OWNER CONTROLLED AREA as reported by the Security Shift Supervisor.

AIRLINER: A large aircraft with the potential for causing significant damage to the Plant. (The NRC notification should designate aircraft vs. airliner.)

HOSTILE ACTION: An act toward a nuclear power Plant or its personnel that includes the use of violent force to destroy equipment, take hostages, and/or intimidate the licensee to achieve an end. This includes attack by air, land, or water using guns, explosives, projectiles, vehicles, or other devices used to deliver destructive force. Other acts that satisfy the destructive intent may be included. Hostile action should not be construed to include acts of civil disobedience or felonious acts that are not part of a concerted attack on the nuclear power Plant. Non-terrorism-based EALs should be used address such activities.

OWNER CONTROLLED AREA: That area (excluding the PROTECTED AREA in this EAL) that encompasses the entire Crystal River Energy Complex.

PROTECTED AREA: All areas within the CR3 security perimeter fence that require badged authorization for entry.

Basis:

Note: Timely and accurate communication between Security Shift Supervision and the Control Room is crucial for the implementation of effective Security EALs.

These EALs address the contingency for a very rapid progression of events, such as that experienced on September 11, 2001. They are not premised solely on the potential for a radiological release. Rather the issue includes the need for rapid assistance due to the possibility for significant and indeterminate damage from additional air, land or water attack elements.

The fact that the site is under serious attack or is an identified attack target with minimal time available for further preparation or additional assistance to arrive requires a heightened state of readiness and implementation of protective measures that can be effective (such as on-site evacuation, dispersal or sheltering).

EAL #1

This EAL addresses the immediacy of an expected threat arrival or impact on the site within a relatively short time.

The intent of this EAL is to ensure that notifications for the AIRLINER attack threat are made in a timely manner and that offsite response organizations and plant personnel are at a state of heightened awareness regarding the credible threat.

(continued)

NATURAL / MAN-MADE HAZARDS AND EC JUDGMENT

EAL 2.19

This EAL is met when CR3 receives information regarding an AIRLINER attack threat from NRC and the AIRLINER is within 30 minutes of the plant.
The NRC Headquarters Operations Officer (HOO) will communicate to the licensee if the threat involves an AIRLINER. The status and size of the plane may be provided by NORAD through the NRC.

EAL #2

This EAL addresses the potential for a very rapid progression of events due to a HOSTILE ACTION. It is not intended to address incidents that are accidental events or acts of civil disobedience, such as small aircraft impact, hunters, or physical disputes between employees within the OWNER CONTROLLED AREA. Those events are adequately addressed by other EALs.

Note that this EAL is applicable for any HOSTILE ACTION occurring, or that has occurred, in the OWNER CONTROLLED AREA.

Although nuclear plant security officers are well trained and prepared to protect against HOSTILE ACTION, it is appropriate for offsite response organizations to be notified and encouraged to begin activation to be better prepared should it be necessary to consider further actions.

If not previously notified by the NRC that the airborne HOSTILE ACTION was intentional, then it would be expected, although not certain, that notification by an appropriate Federal agency would follow. In this case, appropriate federal agency is intended to be NORAD, FBI, FAA or NRC. However, the declaration should not be unduly delayed awaiting Federal notification.

CR3 Matrix Reference Number: **2.19**

NEI 99-01 Reference: HA4

EAL 2.20

Security Event

Initiating Condition:
HOSTILE ACTION within the PROTECTED AREA

Emergency Action Level:

SITE AREA EMERGENCY
2.20 MODES: ALL
1. A HOSTILE ACTION is occurring or has occurred within the PROTECTED AREA as reported by the Security Shift Supervisor.

HOSTILE ACTION: An act toward a nuclear power Plant or its personnel that includes the use of violent force to destroy equipment, take hostages, and/or intimidate the licensee to achieve an end. This includes attack by air, land, or water using guns, explosives, projectiles, vehicles, or other devices used to deliver destructive force. Other acts that satisfy the destructive intent may be included. Hostile action should not be construed to include acts of civil disobedience or felonious acts that are not part of a concerted attack on the nuclear power Plant. Non-terrorism-based EALs should be used address such activities.

HOSTILE FORCE: One or more individuals who are engaged in a determined assault, overtly or by stealth and deception, equipped with suitable weapons capable of killing, maiming, or causing destruction.

OWNER CONTROLLED AREA: That area (excluding the PROTECTED AREA in this EAL) that encompasses the entire Crystal River Energy Complex.

PROTECTED AREA: All areas within the CR3 security perimeter fence that require badged authorization for entry.

Basis:
This condition represents an escalated threat to plant safety above that contained in the Alert in that a HOSTILE FORCE has progressed from the OWNER CONTROLLED AREA to the PROTECTED AREA.

This EAL addresses the contingency for a very rapid progression of events, such as that experienced on September 11, 2001. It is not premised solely on the potential for a radiological release. Rather the issue includes the need for rapid assistance due to the possibility for significant and indeterminate damage from additional air, land or water attack elements.

The fact that the site is under serious attack with minimal time available for further preparation or additional assistance to arrive requires offsite response organization readiness and preparation for the implementation of protective measures.

This EAL addresses the potential for a very rapid progression of events due to a HOSTILE ACTION. It is not intended to address incidents that are accidental events or acts of civil disobedience, such as small aircraft impact, hunters, or physical disputes between employees within the PROTECTED AREA. Those events are adequately addressed by other EALs.

Although nuclear plant security officers are well trained and prepared to protect against HOSTILE ACTION, it is appropriate for offsite response organizations to be notified and encouraged to begin preparations for public protective actions to be better prepared should it be necessary to consider further actions.

(continued)

If not previously notified by NRC that the airborne HOSTILE ACTION was intentional, then it would be expected, although not certain, that notification by an appropriate Federal agency would follow. In this case, appropriate federal agency is intended to be NORAD, FBI, FAA or NRC. However, the declaration should not be unduly delayed awaiting Federal notification.

Escalation of this emergency classification level, if appropriate, would be based on actual plant status after impact or progression of attack.

CR3 Matrix Reference Number: **2.20**

NEI 99-01 Reference: **HS4**

EAL 2.21

Security Event

Initiating Condition:
HOSTILE ACTION resulting in loss of physical control of the facility

Emergency Action Level:

GENERAL EMERGENCY
2.21 MODES: ALL (1 or 2) 1. A HOSTILE ACTION has occurred such that plant personnel are unable to operate equipment required to maintain safety functions . OR 2. A HOSTILE ACTION has caused failure of Spent Fuel Cooling Systems and imminent fuel damage is likely.

HOSTILE ACTION: An act toward a nuclear power Plant or its personnel that includes the use of violent force to destroy equipment, take hostages, and/or intimidate the licensee to achieve an end. This includes attack by air, land, or water using guns, explosives, projectiles, vehicles, or other devices used to deliver destructive force. Other acts that satisfy the destructive intent may be included. Hostile action should not be construed to include acts of civil disobedience or felonious acts that are not part of a concerted attack on the nuclear power Plant. Non-terrorism-based EALs should be used address such activities.

VITAL AREA: Any area, normally within the PROTECTED AREA, that contains equipment, systems, components, or material, the failure, destruction, or release of which could directly or indirectly endanger the public health and safety by exposure to radiation.

Basis:

EAL #1

This EAL encompasses conditions under which a HOSTILE ACTION has resulted in a loss of physical control of a VITAL AREA or equipment required to maintain safety functions (reactivity control, reactor coolant system inventory, and secondary heat removal) and control of that equipment cannot be transferred to and operated from another location. If control of the plant equipment necessary to maintain safety functions can be transferred to another location, then the threshold is not met.

EAL #2

This EAL addresses failure of spent fuel cooling systems as a result of HOSTILE ACTION if imminent fuel damage is likely, such as when a freshly off-loaded reactor core is in the spent fuel pool. For the purposes of EAL 2.21, "imminent" means that mitigation actions have been ineffective, additional actions are not likely to be successful, and trended information indicates that fuel damage will occur.

CR3 Matrix Reference Number: **2.21**

NEI 99-01 Reference: HG1

EAL 2.22

Internal Flooding

Initiating Condition:

Internal flooding affecting areas containing SAFE SHUTDOWN EQUIPMENT

Emergency Action Level:

UNUSUAL EVENT
2.22 MODES: ALL (1 and 2) 1. Indication of uncontrolled flooding in the Auxiliary Building or Intermediate Building AND 2. Water level/flooding has the potential to affect or immerse SAFE SHUTDOWN EQUIPMENT

SAFE SHUTDOWN EQUIPMENT: Equipment necessary to achieve and maintain the reactor sub critical with controlled decay heat removal to bring the Plant to the ITS applicable shutdown condition/mode.

Basis:

This addresses the possible effects of flooding from system malfunctions, component failures, or repair activity mishaps that could threaten the safe operation of the Plant. The flooding could affect equipment NOT designed to be submerged.

CR3 Matrix Reference Number: 2.22

NEI 97-03 Reference: HU1

EAL 2.23

Internal Flooding

Initiating Condition:

Internal flooding affecting SAFE SHUTDOWN EQUIPMENT

Emergency Action Level:

ALERT
2.23 **MODES: ALL**
(1 and 2)
1. Water level exceed 5 inches in the Auxiliary Building or Intermediate Building
AND
2. **(a or b)**
a. Indications show degraded SAFE SHUTDOWN EQUIPMENT due to the flooding
OR
b. Electrical hazards prevent Plant personnel normal access to areas of Plant containing SAFE SHUTDOWN EQUIPMENT

SAFE SHUTDOWN EQUIPMENT: Equipment necessary to achieve and maintain the reactor sub critical with controlled decay heat removal to bring the Plant to the ITS applicable shutdown condition/mode.

Basis:

This addresses the possible effects of flooding from system malfunctions, component failures, or repair activity mishaps that has either threatened the safe operation of the Plant or resulted in a complete loss of function required for cold shutdown.

The water value was selected to be consistent with the site's flooding analysis and mitigative strategy of abnormal procedures. A flooding hazard evaluation established 7 inches as the level in the Auxiliary Building which would begin to affect equipment (ref: Gilbert/Commonwealth FCS-9852, 10/12/88). The 5-inch value was selected for conservatism.

If damage from the internal flooding is clearly contained and localized to one train, and safe shutdown capability exists, then item 2a of the EAL is NOT met. If the extent of the damage is uncertain in terms of loss of safe shutdown capability, then entry into this EAL is required. If all Auxiliary Building 95-ft. elevation motor control centers are de-energized in accordance with abnormal procedures as a result of flooding, then SAFE SHUTDOWN EQUIPMENT is deemed to be degraded and entry into this EAL is required.

CR3 Matrix Reference Number: 2.23

NEI 97-03 Reference: HA1

EAL 2.24

Emergency Coordinator Judgment

Initiating Conditions:

Other conditions existing, which in the judgment of the Emergency Coordinator, warrant declaration of an Unusual Event

Emergency Action Level:

UNUSUAL EVENT
2.24 MODES: ALL
Other conditions exist which indicate a potential degradation of the level of safety of the Plant

Basis:

This EAL addresses unanticipated conditions NOT addressed explicitly elsewhere but warrant declaration of an emergency because conditions exist which are believed by the Emergency Coordinator to fall under the Unusual Event emergency class.

This EAL should also be referenced if, in the judgment of the Emergency Coordinator, an Unusual Event should be classified if Plant symptoms are less than the threshold of an existing EAL.

CR3 Matrix Reference Number: 2.24

NEI 97-03 Reference: HU5

EAL 2.25

Emergency Coordinator Judgment

Initiating Conditions:

Other conditions exist, which in the judgment of the Emergency Coordinator, warrant declaration of an Alert

Emergency Action Level:

ALERT
2.25 MODES: ALL Other conditions exist which indicate that events are in process or have occurred which involve potential or actual substantial degradation of the level of safety of the Plant

Basis:

This EAL is intended to address unanticipated conditions NOT addressed explicitly elsewhere but warrant declaration of an emergency because conditions exist which are believed by the Emergency Coordinator to fall under the Alert emergency class.

Any release is expected to be limited to small fractions of the EPA plume Protective Action Guideline Exposure Levels.

This EAL should also be referenced if, in the judgment of the Emergency Coordinator, an Alert should be classified if Plant symptoms are less than the threshold of an existing EAL.

It is NOT necessary to declare an ALERT if an Initiating Condition/EAL is NOT met and it is desirable to have TSC support, however, if support of the TSC staff is vital to mitigate an event, an Alert declaration should be considered

CR3 Matrix Reference Number: 2.25

NEI-97-03 Reference: HA6

EAL 2.26

Emergency Coordinator Judgment

Initiating Conditions:

Other conditions exist, which in the judgment of the Emergency Coordinator, warrant declaration of a Site Area Emergency

Emergency Action Level:

SITE AREA EMERGENCY
2.26 MODES: ALL
Other conditions exist which indicate actual or likely major failures of Plant functions needed for the protection of the public

Basis:

This EAL is intended to address unanticipated conditions NOT addressed explicitly elsewhere but that warrant declaration of an emergency because conditions exist which are believed by the Emergency Coordinator to fall under the emergency class description for Site Area Emergency.

A release is NOT expected to result in exposure levels exceeding EPA plume Protective Action Guideline Exposure Levels beyond the SITE BOUNDARY (1 Rem TEDE or 5 Rem Thyroid CDE).

TEDE - Total Effective Dose Equivalent

CDE - Committed Dose Equivalent

This EAL should also be referenced if, in the judgment of the Emergency Coordinator, a Site Area Emergency should be classified if Plant symptoms are less than the threshold of an existing EAL.

CR3 Matrix Reference Number: 2.26

NEI 97-03 Reference: HS3

EAL 2.27

Emergency Coordinator Judgment

Initiating Condition:

Other conditions exist, which in the judgment of the Emergency Coordinator, warrant declaration of a General Emergency

Emergency Action Level:

GENERAL EMERGENCY
2.27 **MODES: ALL** **(1 or 2)** Other conditions exist which indicate: 1. Actual or imminent substantial core degradation with potential loss of containment integrity OR 2. The potential for uncontrolled radionuclide releases that can be expected to exceed EPA Protective Action Guidelines Plume Exposure Levels beyond the SITE BOUNDARY (see EAL 1.4)

SITE BOUNDARY: That area, including the PROTECTED AREA, that extends 4400 feet or 0.83 miles in a circle around the Reactor Building. Also referred to as the Owner Controlled Area.

PROTECTED AREA: All areas within the CR3 security perimeter fence that require badged authorization for entry.

Basis:

This EAL is intended to address unanticipated conditions NOT addressed explicitly elsewhere but that warrant declaration of an emergency because conditions exist which are believed by the Emergency Coordinator to fall under the General Emergency class.

Releases can reasonably be expected to exceed EPA Protective Action Guidelines Plume Exposure Levels beyond the Site Boundary (1 Rem TEDE or 5 Rem Thyroid CDE).

CR3 Matrix Reference Number: **2.27**

NEI 97-03 Reference: HG2

EAL 3.1

Loss of Communication

Initiating Condition:

Unplanned loss of all In-Plant or all offsite Communication capability

Emergency Action Level:

UNUSUAL EVENT
3.1 MODES: ALL (1 or 2) 1. Loss of all the following in-Plant communications capability: a. PE Internal Telephone System b. PAX c. Portable UHF Radios OR 2. Loss of all of the following Offsite Communication capability: a. PE Telephone System b. State Hot Ringdown (SHRD) c. All FTS 2001 NRC phones (ENS, HPN, etc.) d. State-Wide Emergency Management Network (EMnet) e. Cellular Telephones

Basis:

The purpose of this Initiating Condition and its associated EALs is to recognize a loss of communications capability either defeating the Plant operations staff ability to perform routine tasks necessary for Plant operations or the ability to communicate problems with offsite authorities. The loss of offsite communications ability is expected to be significantly more comprehensive than the condition addressed by 10 CFR 50.72.

The onsite or offsite communications loss must encompass the loss of all means of routine direct communications with intended parties. This includes the ENS, Commercial lines, Cellular Phones, Microwave, and FAX transmissions. This EAL is used only when extraordinary means are used to make communications possible (relaying of information from radio transmissions, individuals being sent to offsite locations, etc.). Credit is NOT taken for portable satellite phones located in the Technical Support Center due to the time it takes to establish a communications link. Once a link is established with a portable satellite phone, the event may be terminated.

CR3 Matrix Reference Number: 3.1

NEI 97-03 Reference: SU6

EAL 3.2

Failure of Reactor Protection

Initiating Condition:

Failure of Reactor Protection System (RPS) instrumentation to complete or initiate an automatic reactor trip once an RPS setpoint has been exceeded and manual trip was successful

Emergency Action Level:

ALERT
3.2 MODES: 1,2,3 (1 and 2) 1. RPS Trip setpoint exceeded and no Reactor trip occurred AND 2. Manual Reactor trip from Control Room was successful and reactor is shutdown

Basis:

This condition indicates failure of the Reactor Protection System to trip the reactor. This condition is more than a potential degradation of a safety system in that a front line automatic protection system did NOT function in response to a Plant transient and thus the Plant safety has been compromised, and design limits of the fuel may have been exceeded.

An Alert is indicated because conditions exist that lead to potential loss of fuel clad or RCS. Reactor protection system setpoint being exceeded is specified here because failure of the automatic protection system is the issue. A manual trip is any set of actions by the reactor operator(s) in the Control Room which causes sufficient control rods to be rapidly inserted into the core and brings the reactor subcritical (e.g., reactor trip button, de-energizing control rod power from the control room). Operator actions to drive rods or other actions taken or occurring outside the control room does NOT constitute a reactor trip because it does NOT meet the rapid insertion criterion.

An automatic reactor trip is considered as the RPS tripping the reactor.

Failure of the manual trip pushbutton when pressed in anticipation of an automatic trip does NOT constitute a failure of the RPS if it is certain that NO other trip setpoints have been exceeded AND de-energizing control rod power from the Control Room results in a subcritical reactor. An Alert declaration would NOT be required in this instance since design limits would NOT have been exceeded.

CR3 Matrix Reference Number: 3.2

NEI 97-03 Reference: SA2

EAL 3.3

Failure of Reactor Protection

Initiating Condition:

Failure of Reactor Protection System (RPS) instrumentation to complete or initiate an automatic reactor trip once an RPS setpoint has been exceeded and manual trip was not successful

Emergency Action Level:

SITE AREA EMERGENCY
3.3 MODES: 1,2 (1 and 2) 1. RPS Trip setpoint exceeded and no Reactor trip occurred AND 2. Manual Reactor trip from Control Room was not successful in shutting down the reactor

Basis:

Automatic and manual trips are NOT considered successful if action away from the Control Room was required to trip the reactor. Manual trip is successful if the trip push button or de-energizing control rod power in the Control Room results in shutting down the reactor.

An automatic reactor trip is considered as the RPS tripping the reactor.

The trip is considered successful when Control Room actions have inserted enough control rods to cause the reactor power to fall below that percent power associated with the ability of the safety systems to remove heat and continue to decrease. Subsequent actions necessary for the reactor to be prepared for a cooldown and depressurization are NOT to be considered.

Under these conditions, the reactor is producing more heat than the maximum decay heat load for which the safety systems are designed. A Site Area Emergency is indicated because conditions exist that lead to imminent loss or potential loss of both fuel clad and RCS. Although this Initiating Condition may be viewed as redundant to the Fission Product Barrier Matrix, its inclusion is necessary to better assure timely recognition and emergency response.

CR3 Matrix Reference Number: **3.3**

NEI 97-03 Reference: SS2

EAL 3.4

Failure of Reactor Protection

Initiating Condition:

Failure of the Reactor Protection System to complete an automatic trip and manual trip was NOT successful and there is indication of extreme challenge to the ability to cool the core

Emergency Action Level:

GENERAL EMERGENCY
3.4 MODES: 1,2 (1 and 2 and 3) 1. RPS Trip setpoint exceeded and no Reactor trip occurred AND 2. Manual Reactor trip from Control Room was not successful in shutting down the reactor AND (a or b) a. Core exit thermocouple temperatures > 700°F, as indicated on SPDS. OR b. Adequate Secondary Cooling not available

Basis:

Under the conditions of this Initiating Condition and its associated EALs, the efforts to bring the reactor subcritical have been unsuccessful and, as a result, the reactor is producing more heat than the maximum decay heat load for which the safety systems were designed. Although there are capabilities away from the reactor control console, such as emergency boration, the continuing temperature rise indicates that these capabilities are NOT effective. This situation could be a precursor for a core melt sequence.

700°F is a good indicator of an extreme challenge to the ability to cool the core and is consistent with the "potential loss" factor in the Fission Product Barrier Matrix.

Another consideration is the inability to initially remove heat during the early stages of this sequence. If emergency feedwater flow is insufficient to remove the amount of heat required by design from at least one steam generator, an extreme challenge should be considered to exist.

In the event either of these challenges exist at a time the reactor has NOT been brought below the power associated with the safety system design a core melt sequence exists. In this situation, core degradation can occur rapidly. For this reason, the General Emergency declaration is intended to be anticipatory of the Fission Product Barrier Matrix declaration to permit maximum offsite intervention time.

CR3 Matrix Reference Number: **3.4**

NEI 97-03 Reference: SG2

EAL 3.5

Inability to Reach Required Mode Within Improved Technical Specification Time Limits

Initiating Condition:

Inability to reach required operating mode within Improved Technical Specification limits

Emergency Action Level:

UNUSUAL EVENT
3.5 MODES: 1,2,3,4
(1 and 2)
1. Entry into an Improved Technical Specification LCO statement requiring a mode reduction
AND
2. The Plant is not in the required operating mode within the time prescribed by the LCO required action

Basis:

Limiting Conditions for Operation (LCOs) require the Plant to be brought to a required shutdown mode when the Improved Technical Specification required configuration cannot be restored. The Plant is within its safety envelope when being shut down within the allowable required action time in the Improved Technical Specifications. An immediate Notification of an Unusual Event is required when the Plant is NOT brought to the required operating mode within the allowable required action time in the Improved Technical Specifications.

Declaration of an Unusual Event is based on the time at which the LCO-specified required action time period elapses under the Improved Technical Specifications and is NOT related to how long a condition may have existed.

CR3 Matrix Reference Number: **3.5**

NEI 97-03 Reference: SU2

EAL 3.6

Loss of Alarms/Indications

Initiating Condition:

UNPLANNED loss of most or all Control Room Annunciators for 15 minutes or longer

Emergency Action Level:

UNUSUAL EVENT
3.6 MODES: 1,2,3,4
(1 or 2)
1. UNPLANNED loss of Annunciator panels A-L and Annunciator printer for 15 minutes or longer
OR
2. UNPLANNED loss of NNI-X and NNI-Y for 15 minutes or longer

UNPLANNED: An event or action is UNPLANNED if it is NOT the expected result of normal operations, testing, or maintenance. Events that result in corrective or mitigative actions being taken in accordance with abnormal or emergency procedures are UNPLANNED.

Basis:

This Initiating Condition and its associated EAL are intended to recognize the difficulty associated with monitoring changing Plant conditions without the use of a major portion of the annunciation or indication equipment. Recognition of the availability of computer-based indication equipment is considered (SPDS, Plant computer, etc.). The Annunciator printer includes: 1) the far left overhead Annunciator CRT display; 2) the printer in the cabinet labeled "Sequential Events Recorder;" and 3) the computer behind the main Control Board labeled "Annunciator Monitor."

A loss of Annunciators is considered a loss of the visual, as opposed to a loss of the audible portion of the Annunciator. Annunciator panels A-L contain the major control systems (RPS, ES, ICS, etc.).

Loss of NNI-X and NNI-Y will cause the loss of most or all safety system indication.

Fifteen minutes was selected as a threshold to exclude transient or momentary power losses.

Due to the limited number of safety systems in operation during cold shutdown, refueling, and defueled modes, NO IC is indicated during these modes of operation.

CR3 Matrix Reference Number: **3.6**

NEI 97-03 Reference: **SU3**

SYSTEM MALFUNCTION

EAL 3.7

Loss of Alarms/Indications

Initiating Condition:

UNPLANNED loss of most or all Control Room Annunciators for 15 minutes or longer with either a SIGNIFICANT TRANSIENT in progress or Plant Computer and SPDS unavailable

Emergency Action Level:

ALERT	
3.7 MODES: 1,2,3,4	
(1 and 2)	
1.	(a or b)
a.	UNPLANNED loss of Annunciator panels A-L and Annunciator printer for 15 minutes or longer
OR	
b.	UNPLANNED loss of NNI-X and NNI-Y for 15 minutes or longer
AND	
2.	(a or b)
a.	SIGNIFICANT TRANSIENT in progress
OR	
b.	Loss of Plant Computer and SPDS

UNPLANNED: An event or action is UNPLANNED if it is NOT the expected result of normal operations, testing, or maintenance. Events that result in corrective or mitigative actions being taken in accordance with abnormal or emergency procedures are UNPLANNED.

SIGNIFICANT TRANSIENT: An UNPLANNED event involving one or more of the following:

(1) Automatic turbine trip at >25% reactor thermal power
(2) Electrical load rejection >25% full electrical load
(3) Plant runback
(4) Reactor trip
(5) Safety injection system actuation
(6) >10% thermal power oscillations
(7) Loss of decay heat removal in Mode 4 ("Significant Transient" is NOT used in any Mode 5 or 6 EALs.)

Basis:

This Initiating Condition and its associated EAL are intended to recognize the difficulty associated with monitoring changing Plant conditions without the use of a major portion of the annunciation or indication equipment during a transient. Recognition of the availability of computer based indication equipment is considered (SPDS, Plant computer, etc.).

The Annunciator printer includes: 1) the far left overhead Annunciator CRT display; 2) the Rochester printer ("Sequential Events Recorder") in cabinet ANN/EVENT RCDC. CAB. #1; and 3) the computer behind the main Control Board labeled "Annunciator Monitor." A loss of both the Annuciator printer and computer-based indication is required to meet the IC.

A loss of Annunciators is considered a loss of the visual, as opposed to a loss of the audible portion of the Annunciator.

Annunciator panels A-L contain the major control systems (RPS, ES, ICS, etc.)

Due to the limited number of safety systems in operation during cold shutdown, refueling and defueled modes NO IC is indicated during these modes of operation.

CR3 Matrix Reference Number: **3.7**

NEI 97-03 Reference: SA4

SYSTEM MALFUNCTION

EAL 3.8

Loss of Alarms/Indications

Initiating Condition:

Inability to monitor a SIGNIFICANT TRANSIENT in progress

Emergency Action Level:

SITE AREA EMERGENCY
3.8 MODES: 1,2,3,4
(1 and 2 and 3 and 4)
1. (a or b)
a. Loss of Annunciator panels A-L and Annunciator printer for 15 minutes or longer OR
b. Loss of NNI-X and NNI-Y for 15 minutes or longer AND
2. SIGNIFICANT TRANSIENT in progress AND
3. Loss of Plant Computer and SPDS AND
4. Inability to directly monitor any one of the following: Subcriticality Core Cooling Containment RCS Inventory

SIGNIFICANT TRANSIENT: An UNPLANNED event involving one or more of the following:

(1) Automatic turbine trip at >25% reactor thermal power

(2) Electrical load rejection >25% full electrical load

(3) Plant runback

(4) Reactor trip

(5) Safety injection system actuation

(6) >10% thermal power oscillations

(7) Loss of decay heat removal in Mode 4 ("Significant Transient" is NOT used in any Mode 5 or 6 EALs.)

Basis:

This Initiating Condition and its associated EAL are intended to recognize the inability of the control room staff to monitor the Plant response to a transient.

The Annunciator printer includes: 1) the far left overhead Annunciator CRT display; 2) the Rochester printer ("Sequential Events Recorder") in cabinet ANN/EVENT RCDC. CAB. #1; and 3) the computer behind the main Control Board labeled "Annunciator Monitor." A loss of both the Annuciator printer and computer-based indication is required to meet the IC.

A loss of Annunciators is considered a loss of the visual, as opposed to a loss of the audible portion of the Annunciator.

Indications needed to monitor safety functions necessary for protection of the public must include control room indications, computer generated indications and dedicated annunciation capability. The specific indications should be those used to determine such functions as the ability to shut down the reactor, maintain the core cooled and in a coolable geometry, to remove heat from the core, to maintain the reactor coolant system intact, and to maintain containment intact.

Planned and unplanned actions are NOT differentiated in this EAL since the loss of instrumentation of this magnitude is of such significance during a transient, that the cause of the loss does NOT make the condition more tolerable.

CR3 Matrix Reference Number: **3.8**

NEI 97-03 Reference: SS6

SYSTEM MALFUNCTION

EAL 3.9

Fuel Clad Degradation

Initiating Condition:

Fuel Clad Degradation

Emergency Action Level:

UNUSUAL EVENT
3.9 MODES: 1,2,3,4,5
(a or b)
Radiochemistry analysis indicates:
a. Dose Equivalent Iodine (I-131) > 1.0 μCi/gm for 48 hours or longer
OR
b. Specific activity >100/E-bar for 48 hours or longer

Basis:

This Initiating Condition is included as an Unusual Event because it is considered a potential degradation in the level of safety of the Plant and a potential precursor of more serious problems. This EAL addresses RCS samples exceeding Improved Technical Specifications for radioactivity levels in the RCS.

RCS purification will provide for Iodine and crud cleanup in the reactor coolant system and reduce activity to < 1.0 μCi/gm within 48 hours.

The EAL values are based on Improved Technical Specification Limits.

E-bar is the weighted average energy of RCS isotopes.

CR3 Matrix Reference Number: **3.9**

NEI 97-03 Reference: SU4

EAL 3.10

Turbine Failure

Initiating Condition:

Turbine failure results in casing penetration

Emergency Action Level:

UNUSUAL EVENT
3.10 MODES: 1,2,3
Report by Plant personnel of main turbine failure causing penetration of the turbine casing or damage to main generator seals

Basis:

This EAL is intended to address main turbine rotating component failures of sufficient magnitude to cause observable damage to the turbine casing or to the seals of the turbine generator. Of major concern is the potential for leakage of combustible fluids (lubricating oils) and gases (hydrogen) to the Plant environs. Actual fires and flammable gas build up are appropriately classified via Fire and Flammable Gas EALs. This EAL is consistent with the definition of an Unusual Event while maintaining the anticipatory nature desired and recognizing the risk to non-safety related equipment.

Escalation of the emergency classification is based on the potential damage done by missiles generated by the failure. It is NOT the intent of this Initiating Condition to declare an event based on damage discovered in a maintenance evolution. Generator seal damage observed after generator purge does NOT meet the intent of this EAL because it did NOT impact normal operation of the Plant.

CR3 Matrix Reference Number: 3.10

NEI 97-03 Reference: HU1

EAL 3.11

Turbine Failure

Initiating Condition:

Turbine failure generated projectiles cause significant VISIBLE DAMAGE to SAFE SHUTDOWN EQUIPMENT

Emergency Action Level:

ALERT

3.11 **MODES: 1,2,3**

(1 or 2)

1. Report by Plant personnel of projectiles generated by a main turbine failure causing significant VISIBLE DAMAGE any of the following structures:
   - Auxiliary Building
   - BWST
   - Control Complex
   - Diesel Generator Building (EGDG-1A/!B)
   - EFT-2 Building
   - Intermediate Building
   - Reactor Building
   - EFP-3 Building

OR

2. Indications show degraded SAFE SHUTDOWN EQUIPMENT performance due to turbine generated projectiles

VISIBLE DAMAGE: Damage to equipment or structure that is readily observable without measurements, testing, or analyses. Damage is sufficient to cause concern regarding the continued operability or reliability of affected safety structure, system, or component. Example system/component damage includes: deformation due to heat or impact, denting, penetration, rupture, cracking, paint blistering due to fire. Example structure damage includes exposed and/or broken rebar, failed supports/pipe hangers, etc. Surface blemishing (e.g., paint chipping, scratches, concrete spalling) should NOT be included.

SAFE SHUTDOWN EQUIPMENT: Equipment necessary to achieve and maintain the reactor sub critical with controlled decay heat removal to bring the Plant to the ITS applicable shutdown condition/mode.

Basis:

This EAL is intended to address the threat to safe shutdown equipment imposed by missiles generated by main turbine rotating component failures. The list of areas includes all areas containing safe shutdown equipment, their controls, and their power supplies. This EAL is, therefore, consistent with the definition of an Alert in that if missiles have damaged or penetrated areas containing safety-related equipment the potential exists for substantial degradation of the level of safety of the Plant.

This EAL is intended to address events that may have resulted in a Plant vital area being subjected to forces beyond design limits, and thus damage may be assumed to have occurred to Plant safety systems. Assessing SAFE SHUTDOWN EQUIPMENT performance is NOT interpreted as mandating a lengthy damage assessment before classification and NO attempt is made to assess the actual magnitude of the damage. This EAL is NOT intended to be used for temporary loss of Control Complex habitability where timely repairs can be affected.

If damage from the turbine failure is clearly contained and localized to one train, then safe shutdown equipment is NOT affected and the EAL is NOT met. If the extent of the damage is uncertain in terms of loss of safe shutdown equipment, then entry into this EAL is required.

CR3 Matrix Reference Number: 3.11

NEI 97-03 Reference: HA1

EAL 3.12

RCS Leakage

Initiating Condition:
RCS leakage

Emergency Action Level:

UNUSUAL EVENT
3.12 MODES: 1,2,3,4 (1 or 2) 1. Unidentified Leakage ≥ 10 gpm or Pressure Boundary Leakage ≥ 10 gpm OR 2. Identified leakage ≥ 25 gpm

Basis:
The terms "identified," "unidentified," and "pressure boundary" leakage are as defined in Improved Technical Specifications. The intent of this EAL is that the loss of RCS inventory is due to a failure of equipment.

The Reactor Coolant System (RCS) barrier is the RCS primary side and its connections up to and including the pressurizer safety and relief valves, and other connections up to and including the primary isolation valves. Any leakage in an RCS interfacing system containing reactor coolant (MU, DH, SF, WD, etc.) that can be readily located and isolated within 15 minutes of detection does NOT require entry into this EAL.

OTSG Tube Leaks are considered as part of "identified" RCS leaks and apply to this EAL.

NOTE: See section 4.1.2 of this Manual for Transient Event Classification (i.e. PORV Failed open, block valve closed).

This Initiating Condition is included as an Unusual Event because it may be a precursor of more serious conditions and, as result, is considered a potential degradation of the level of safety of the Plant. The 10 gpm value for the unidentified and pressure boundary leakage was selected as it is observable with normal control room indications. Lesser values must generally be determined through time-consuming surveillance tests (e.g., mass balances). The EAL for identified leakage is set at a higher value due to the lesser significance of identified leakage in comparison to unidentified or pressure boundary leakage.

CR3 Matrix Reference Number: 3.12

NEI 97-03 Reference: SU5

EAL 3.13

Inability to Maintain Hot Shutdown

Initiating Condition:
Complete loss of core heat removal capability

Emergency Action Level:

SITE AREA EMERGENCY
3.13 MODES: 1,2,3,4
(1 and 2)
1. Complete loss of Main, Emergency, and Auxiliary Feedwater and unable to establish HPI cooling
AND
2. Loss of subcooling margin

Basis:
This EAL addresses complete loss of functions, including loss of heat removal capability, required for hot shutdown. Under these conditions, there is an actual major failure of a system intended for protection of the public. Thus, declaration of a Site Area Emergency is warranted.

CR3 Matrix Reference Number: 3.13
NEI 97-03 Reference: SS4

SYSTEM MALFUNCTION

EAL 3.14

Inadvertent Criticality

Initiating Condition:
Inadvertent criticality

Emergency Action Level:

UNUSUAL EVENT	
3.14 MODES:	2,3,4,5,6
An extended and unplanned sustained positive startup rate monitored by nuclear instrumentation	

Basis:
This condition can be identified using the startup rate monitor. The term "extended" is used to allow for exclusion of expected short term positive startup rates from planned fuel bundle or control rod movements during core alterations. The short term startup rates are the result of the increase in neutron population due to subcritical multiplication.

This Initiating Condition/EAL is NOT intended to classify an early criticality during reactor startup. This type event is indicative of errors in reactivity data/calculations and/or mis-operation. The loss of the required shutdown margin can be quickly restored by manual actions or automatic reactor trip.

CR3 Matrix Reference Number: 3.14
NEI 97-03 Reference: SU8

SYSTEM MALFUNCTION

EAL 3.15

Inability To Maintain Plant In Cold Shutdown

Initiating Condition:

Complete loss of functions required for core cooling during refueling and cold shutdown modes

Emergency Action Level:

ALERT
3.15 **MODES: 5,6** (1 or 2) 1. Inability to maintain reactor coolant temperature below 200°F OR 2. Uncontrolled reactor coolant temperature approaching 200°F

Basis:

For PWRs, this Initiating Condition and its associated EAL are based on concerns raised by Generic Letter 88-17 "Loss Of Decay Heat Removal." A number of phenomena, such as pressurization, vortexing, steam generator draining, RCS level differences when operating at a mid-loop condition, decay heat removal system design and level instrumentation problems can lead to conditions where decay heat removal is lost and core uncovery can occur. NRC analyses show that sequences can cause core uncovery in 15 to 20 minutes, and severe core damage within an hour after decay heat removal is lost. The site-specific indicators for these EALs are those methods used by the Plant in response to Generic Letter 88-17, which include core exit temperature monitoring and RCS water level monitoring. In addition, radiation monitor readings may also be appropriate as an indicator of this condition.

"Uncontrolled" means that system temperature increase is NOT the result of planned actions by the Plant staff. The EAL guidance related to uncontrolled temperature rise is necessary to preserve the anticipatory philosophy of NUREG-0654 for events starting from temperatures much lower than the cold shutdown temperature limit.

A momentary UNPLANNED excursion above 200 °F when the heat removal function is available is NOT intended to constitute an ALERT. For example, if the on line DH pump trips and if in the process of starting the alternate pump RCS temperature briefly exceeds 200 °F, an ALERT declaration is NOT required.

CR3 Matrix Reference Number: 3.15

NEI 97-03 Reference: NEI-SA3

SYSTEM MALFUNCTION

EAL 3.16

Loss of Water Level in Reactor Vessel That Has or Will Uncover Fuel

Initiating Condition:
Loss of water level in the reactor vessel that has or will uncover fuel.

Emergency Action Level:

SITE AREA EMERGENCY
3.16 MODES 5,6
(1 and 2)
1. Loss of decay heat removal per AP-404
AND
2. (a or b)
a. Incores indicating superheated conditions
OR
b. Incores unavailable and time to uncovery exceeded as specified in OP-103H

Basis:
Under the conditions specified by this Initiating Condition, severe core damage can occur and reactor coolant system pressure boundary integrity may NOT be assured. OP-103H, " Reactor Coolant System And Spent Fuel Pool Decay Heat Tables And Figures," contains time to core uncovery without decay heat removal curves.

This Initiating Condition covers sequences such as prolonged boiling following loss of decay heat removal. Thus, declaration of a Site Area Emergency is warranted under the conditions specified by the Initiating Condition.

Incore indication is sufficient for this EAL since NO means of water level indication exist in the active fuel region.

CR3 Matrix Reference Number: 3.16

NEI 97-03 Reference: SS5

EAL 4.1

Loss of AC Power

Initiating Condition:

Loss of All Offsite Power for 15 minutes or longer

Emergency Action Level:

UNUSUAL EVENT
4.1 MODES: ALL
(1 and 2)
1. Offsite Power Transformer (OPT) and Backup ES Transformer (BEST) and Auxiliary Transformer not available for 15 minutes or longer
AND
2. EDGs supplying power to 4160V ES Busses

Basis:

Prolonged loss of AC power reduces required redundancy and potentially degrades the level of safety of the Plant by rendering the Plant more vulnerable to a complete Loss of AC Power (Station Blackout). Fifteen minutes is used as a threshold to exclude transient or momentary power losses.

Available indicates transformers are capable of energizing ES busses.

CR3 Matrix Reference Number: 4.1

NEI 97-03 Reference: SU1

EAL 4.2

Loss of AC Power

Initiating Condition:

AC power capability to required 4160V ES busses reduced to a single source for 15 minutes or longer such that an additional failure would result in station blackout

Emergency Action Level:

ALERT
4.2 **MODES**: **1,2,3,4**
AC power capability to the 4160V ES busses reduced to a single power source for 15 minutes or longer such that **only** one of the following is available: - "A" EDG - "B" EDG - Offsite Power Transformer (OPT) - Backup ES Transformer (BEST)

Basis:

This Initiating Condition and the associated EALs are intended to provide an escalation from "Loss of Offsite Power for Greater Than 15 Minutes." The condition indicated by this Initiating Condition is the degradation of the offsite and onsite power systems such that any additional single failure would result in a station blackout.

Available indicates transformers are capable of energizing required busses.

EDG = Emergency Diesel Generator

CR3 Matrix Reference Number: **4.2**

NEI 97-03 Reference: **SA5**

LOSS OF POWER

EAL 4.3

Loss of AC Power

Initiating Condition:

Loss of All Offsite and required Onsite AC Power for 15 minutes or longer

Emergency Action Level:

SITE AREA EMERGENCY		
4.3	MODES:	1,2,3,4
Neither 4160V ES bus is capable of being energized within 15 minutes		

Basis:

Loss of all AC power compromises all Plant safety systems requiring electric power including ECCS, Containment Heat Removal and the Ultimate Heat Sink. Prolonged loss of all AC power will cause core uncovering and may challenge containment integrity. The fifteen-minute time duration is to exclude transient or momentary power losses and begins at the time power is lost to the ES busses (NOT when repair efforts begin).

NOTE: In Modes 5 and 6, the same initiating condition/EAL is an ALERT classification.

CR3 Matrix Reference Number: 4.3

NEI 97-03 Reference: SS1

EAL 4.4

Loss of AC Power

Initiating Condition:
Prolonged Loss of All Offsite and Onsite AC power

Emergency Action Level:

GENERAL EMERGENCY
4.4 MODES: 1,2,3,4
(1 and 2)
1. Neither 4160V ES bus is capable of being energized
AND
2. (a or b)
a. Restoration of 4160V ES Bus A or 4160V ES Bus B is not likely within 4 hours
OR
b. Core exit thermocouples > 700°F as indicated on SPDS

Basis:
Loss of all AC power compromises all Plant safety systems requiring electric power including ECCS and the Ultimate Heat Sink. Prolonged loss of all AC power will lead to loss of fuel clad, RCS, and may challenge containment integrity. The four hours to restore AC power is based on the CR3 station blackout coping analysis performed in conformance with 10 CFR 50.63 and Regulatory Guide 1.155, "Station Blackout." The four-hour time limit begins at the time power is lost to the ES busses (NOT when repair efforts begin).

Although this Initiating Condition may be viewed as redundant to the Fission Product Barrier Matrix, its inclusion is necessary to better assure timely recognition and emergency response.

700°F is a good indicator of an extreme challenge to the ability to cool the core and is consistent with the "potential loss" factor in the Fission Product Barrier Matrix.

This Initiating Condition is specified to assure that in the unlikely event of a prolonged station blackout, timely recognition of the seriousness of the event occurs and that declaration of a General Emergency occurs as early as is appropriate, based on a reasonable assessment of the event trajectory.

The likelihood of restoring at least one emergency bus should be based on a realistic appraisal of the situation since a delay in an upgrade decision based on only a chance of mitigating the event could result in a loss of valuable time in preparing and implementing public protective actions.

CR3 Matrix Reference Number: 4.4

NEI 97-03 Reference: SG1

EAL 4.5

Loss of AC Power (Shutdown)

Initiating Condition:

Loss of All Offsite and Onsite AC Power to Required Busses During Cold Shutdown or Refueling Mode for 15 minutes or longer

Emergency Action Level:

ALERT
4.5 **MODES: 5,6, No Mode**
Neither 4160V ES bus is capable of being energized within 15 minutes

Basis:

Loss of all AC power compromises all Plant safety systems requiring electric power including ECCS, Containment Heat Removal and the Ultimate Heat Sink. When in cold shutdown, refueling, or defueled mode the event can be classified as an Alert, because of the significantly reduced decay heat, lower temperature and pressure, increasing the time to restore one of the emergency busses, relative to that specified for the Site Area Emergency EAL. Fifteen minutes was selected as a threshold to exclude transient or momentary power losses and begins at the time power is lost to the ES busses (NOT when repair efforts begin).

CR3 Matrix Reference Number: 4.5

NEI 97-03 Reference: SA1

LOSS OF POWER

EAL 4.6

Loss of Vital DC Power

Initiating Condition:
Loss of all Vital DC Power for 15 minutes or longer

Emergency Action Level:

SITE AREA EMERGENCY
4.6 MODES: 1,2,3,4
Standby Power Status Lights for BUS A1, A2, and BUS B1, B2 on the Main Control Board (SSF Panel) are out for 15 minutes or longer

Basis:
Loss of all DC power compromises ability to monitor and control Plant safety functions. Prolonged loss of all DC power could cause core uncovery and loss of containment integrity when there is significant decay heat and sensible heat in the reactor system. Fifteen minutes is used to exclude transient or momentary power losses and begins at the time power is lost to the DC busses (NOT when repair efforts begin).

NOTE: In Modes 5 and 6, the same Initiating Condition/EAL is an UNUSUAL EVENT classification.

CR3 Matrix Reference Number: 4.6
NEI 97-03 Reference: SS3

EAL 4.7

Loss of Vital DC Power (Shutdown)

Initiating Condition:

Loss of all Vital DC Power for 15 minutes or longer

Emergency Action Level:

UNUSUAL EVENT
4.7 MODES: 5,6, No Mode
Standby Power Status Lights for BUS A1, A2, and BUS B1, B2 on the Main Control Board (SSF Panel) are out for 15 minutes or longer

Basis:

Loss of required DC power compromises ability to monitor and control Plant safety functions. Prolonged loss of all DC power could cause core uncovery and loss of containment integrity when there is significant decay heat and sensible heat in the reactor system. When in cold shutdown, refueling, or defueled mode the event can be classified as an Unusual Event, because of the significantly reduced decay heat, lower temperature and pressure, increasing the time to restore one of the emergency busses, relative to that specified for the Site Area Emergency EAL. Fifteen minutes is used to exclude transient or momentary power losses.

CR3 Matrix Reference Number: 4.7

NEI 97-03 Reference: SU7

ATTACHMENT 2

Development of Parameters and Values Used in Selected EALs

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Basis for RM-G29 and RM-G30 values used FPBM 5.1, 6.1, 7.2

FPBM 5.1: The 100 R/hr value listed in Fission Product Barrier Matrix (FPBM) 5.1 is based on the response of the containment monitors in the 2/26/80 event during which approximately 43,000 gallons of reactor coolant were released to the containment building. Response Technical Manual RTM-91 Workbook page 25 shows the CR3 containment monitor response graph peaking at about 80 R/hr. RTM-91 page 120 notes that due to uneven mixing, various containment locations may differ by several orders of magnitude and RTM-93 page B-8 notes that it may take several hours for uniform mixing. MicroShield calculations indicated that all normal coolant curies released to the containment and uniformly mixed would result in less than 1 R/hr. Since the 2/26/80 event did not involve gas gap damage, the peak monitor reading was due to uneven mixing. An arbitrary value of 100 R/hr was assumed as a conservative indication of the threshold of gas gap damage.

FPBM 6.1: As noted in the Emergency Action Level Bases Manual, the 10 R/hr value listed in FPBM 6.1 is a value which indicates the release of reactor coolant to containment. RM-G29 and RM-G30 typically read about 1 to 2 R/hr in normal operation due mostly to insensitivity at the extreme low end of the seven decade high-range monitors scales. 10 R/hr represents the beginning of the second decade on the monitor scale and a reading that can fairly confidently be attributed to a release of reactor coolant, while fuel barrier remains intact.

FPBM 7.2: The development of the 5,000 R/hr value in FPBM 7.2 is documented in EC 86189 and represents 20% clad damage. It is based on the discussion of RM-G29 and RM-G30 indication of 100% fuel clad damage in Engineering Evaluation EEF-00-009. NEI 97-03 recommends this potential loss threshold correspond to 20% gap release. Per Engineering Evaluation EC 86189, Vendor Manual 00506-000 states in section 2.2 that the monitor response is proportional (linear) over the range of concern. Therefore, 5000 R/hr corresponds to 20% gap release.

ATTACHMENT 3

EC 76363 RM-A1 & RM-A2 REPLACEMENT

INTERIM COMPENSATORY ACTIONS

EC 76363 RM-A1 & RM-A2 Replacement Compensatory Actions

Engineering Change 76363 replaces the RM-A1 and RM-A2. The new equipment will be installed in the same locations as the existing equipment. There will be an interim period of approximately 90 days when no exhaust duct (effluent) monitors will be operational. RM-A1 is not required in the current extended shutdown conditions and needs no compensatory actions.
Gaseous Effluent EALs 1.1 Unusual Event (UE), 1.2 Alert, 1.3 Site Area Emergency (SAE), and 1.4 General Emergency (GE) and emergency dose assessment are impacted.
All significant radioactive sources are in the Spent Fuel Pools and decay heat and source term are significantly diminished. Only a release from a drained Spent Fuel Pool will reach Alert, SAE or GE dose levels (reference 520663-08). Compensatory actions for these emergency levels focus on Spent Fuel Pool level with the intent to make emergency declarations proactively, before a significant release is in progress.
Compensatory actions focus on the Gaseous Effluent EALs only. There are many other EALs that could require an Unusual Event or Alert declaration that are not related to a release and are unaffected by this EC. The Emergency Coordinator will need to evaluate those EALs normally.

EAL Compensatory Actions:

1. RM-A4 or RM-A8 high setpoint will be used as a conservative indicator for declaration of an Unusual Event by Emergency Coordinator Judgment.

2. Two temporary redundant dose rate instruments have been installed on the Auxiliary Building effluent duct on the 143' elevation downstream of the filters (reference Engineering Evaluation 87296). These instruments (DMC-2000S) have specified range of 0.01 mRem/hr to 1000 Rem/hr. They do NOT replace RM-A2 EAL threshold values. They will provide additional information on release trending for the Emergency Coordinator to use for implementation of the Emergency Coordinator Judgment EALs. Dose rates from these instruments are transmitted to the GEDDS system, which can be accessed on any business computer. Access: NGG OSI-PI Displays, CR3 General, O&S tab, All GEDDS Tags on PI – Trend 1, scroll to 143FT AUXILIARY BUILDING EXH DUCT LOCATION 1 and 2 (U3GRP_OCB 101 and U3GRP_OCB 102).

3. A close-circuit TV (CCTV) camera is also available on the Spent Fuel floor, which may assist the Emergency Coordinator in monitoring Spent Fuel Pool level. A CCTV monitor is available in the Control Room.

4. For EALs 1.2, and 1.3 below, RM-G15 is included among the contingencies. It will not be available for up to 72 hours at the beginning of the project. **IF** EAL 1.9 conditions are met during this period, **THEN** consider arranging frequent monitoring of pool level. (See "Other Support Actions" item 5 below for why RM-G15 and other RM-Gs are initially not available).

5. EAL 1.1 Unusual Event:
   - As a conservative contingency for EAL 1.1 item 1, use exceeding RM-A4 or RM-A8 high setpoint for 60 minutes or longer as an indicator of a potential degradation of the level of safety of the plant (EAL 2.24 Emergency Coordinator Judgment).
   - Use EAL 1.7 (Unexpected Radiation Level) available general area radiation monitors (RM-Gs) threshold values.
   - Use EAL 1.1 item 2 sample analysis to determine two times the ODCM limit in accordance with CH-281.
   - Use EAL 1.9 Spent Fuel Pool level.

6. EAL 1.2 Alert: (Spent Fuel Pool release is the only source that can reach this level.)
   - Use EAL 1.10 Spent Fuel Pool damage/level (only credible Spent Fuel Pool Alert).
   - Use EAL 1.8 (Unexpected Radiation Level) available general area radiation monitors (RM-Gs) threshold values.

- Use EAL 1.2 item 2 sample analysis to determine 200 times the ODCM limit in accordance with CH-281. NOTE: If fuel remains covered, this level release is NOT credible.)
- As a conservative contingency, use factor of 100 increase on the temporary Auxiliary Building effluent duct dose rate instrument (from the point RM-A4 or RM-A8 is offscale) as an indicator of potential or actual substantial degradation of the level of safety of the plant (EAL 2.25 Emergency Coordinator Judgment). NOTE: If fuel remains covered, this level release is NOT credible.)

7. EAL 1.3 Site Area Emergency: (Spent Fuel Pool release with pool drained is the only source that can reach this level.)
   - Use a continuing uncontrolled loss of Spent Fuel Pool level approaching uncovery of fuel assemblies as indicated by extremely high Spent Fuel Pool area dose rates (e.g., RM-G15 offscale) due to direct shine from unshielded fuel assemblies or use images from CCTV as indicators of actual or likely major failures of plant systems needed for protection of the public (EAL 2.26 Emergency Coordinator Judgment).
   - **IF** Spent Fuel Pool level CANNOT be estimated, **THEN** use factor of 1000 increase on the temporary Auxiliary Building effluent duct dose rate instrument (from the point RM-A4 or RM-A8 is offscale) as a further indicator of actual or likely major failures of plant systems needed for protection of the public (EAL 2.26 Emergency Coordinator Judgment). NOTE: If fuel remains covered, this level release is NOT credible.)
   - Use EAL 1.3 option 2 dose assessment.
   - Use EAL 1.3 option 3 field team readings.

8. EAL 1.4 General Emergency: (Spent Fuel Pool release with pool drained is the only source that can reach this level.)
   - Use a continuing uncontrolled loss of Spent Fuel Pool level approaching draining of the pools as indicator of the potential for uncontrolled radionuclide releases that can be expected to exceed EPA Protective Action Guideline plume exposure levels beyond the site boundary (EAL 2.27 item 2 Emergency Coordinator Judgment). This may be indicated by increasing dose rates on RM-G14, estimated leak rates or images from CCTV.
   - **IF** Spent Fuel Pool level CANNOT be estimated, **THEN** use factor of 10,000 increase on the temporary Auxiliary Building effluent duct dose rate instrument (from the point RM-A4 or RM-A8 is offscale) as a further indicator of the potential for uncontrolled radionuclide releases that can be expected to exceed EPA Protective Action Guideline plume exposure levels beyond the site boundary (EAL 2.27 item 2 Emergency Coordinator Judgment). NOTE: If fuel remains covered, this level release is NOT credible.)
   - Use option 2 dose assessment
   - Use option 3 field team readings

Dose Assessment Compensatory Actions:

In the current plant condition, dose assessment is not expected to drive any emergency classifications with the possible exception of "what if" projections (before a release is in progress) contributing to Emergency Coordinator Judgment decisions.

Release Definition - The release definition in EM-202 has been revised to add that if RM-A2 is out of service, RM-A4 or RM-A8 reaching their warning setpoints (as a direct result of the event that has initiated an emergency declaration) also constitutes a release.

Temporary Instruments - Two temporary redundant dose rate instruments have been installed on the Auxiliary Building effluent duct on the 143' elevation downstream of the filters. Dose rates from these instruments are transmitted to the GEDDS system, which can be accessed on any business computer (reference Engineering Evaluation 87296).

RASCAL - Use of the RASCAL Spent Fuel method requires no compensatory actions. As a backup so the monitored release method remains available, an EM-204B worksheet will be developed for using RM-A4, RM-A8 or the temporary Auxiliary Building effluent duct dose rate instrument to estimate source terms for entry into RASCAL. RM-A4 and RM-A8 ranges are limited and could not be used for Alert, SAE or GE dose levels.

Other Support Actions:

1. The ODCM has been revised to allow the use of RM-A4 and RM-A8 as compensatory actions for RM-A2 not in service.
2. CH-281, "Conduct Of Environmental And Chemistry During Abnormal And Emergency Events," has been revised to provide enhanced guidance for EAL 1.1 (UE) and 1.2 (Alert) sampling options and the methodology of how Chemistry is to obtain these samples from the alternate RM-A4 and RM-A8. These options address sample analysis indicating the radioactive effluent exceeding the ODCM limits by a factor of two or 200 times respectively.
3. RM-A4 and RM-A8 will not be voluntarily removed from service except for scheduled preventive maintenance.
4. Also as part of maintaining RM-A4 and RM-A8 available, no work will be allowed that removes VBDP-4 or ES MCC 3B1 from service.
5. The control room instrumentation associated with RM-A1 and RM-A2 is primary powered from VBDP-3, breaker 30. In addition to powering RM-A1 and RM-A2, this breaker supplies power to RM-L1,RM-L3, RM-L5, RM-A7, RM-A12, RMG-1, RMG-3, RMG-5, RMG-7, RMG-9, RMG-11, RMG-13, RMG-15, and RMG-17. While not all of these monitors are applicable to our current no-mode condition, e.g. RM-L1, RCS Activity, it is appropriate to maximize redundant and diverse indication while RM-A2 is out of service. As a defense-in-depth action, as soon as the clearance is hung on VBDP-3, electricians will separate RM-A1/A2 from the 'daisy-chain' power supply (Engineering Change 86900) allowing Operations to return the remaining monitors to service. OP-700D will be changed in the interim to reflect this change. This evolution is expected to take 24 hours, but may take up to 72 hours.
6. RM-A2 interlocks associated with various Auxiliary Building supply fans will be defeated. AP-250, "Radiation Monitor Actuation," will be changed for the interim period to address manually performing the RM-A2 trip actions in the event of an RM-A4 or RM-A8 High Alarm. This helps to ensure that all effluents are passed through the charcoal filtration system prior to release.
7. As defense in depth, the following limitations on plant evolutions will be enforced while RM-A2 is inoperable:
    a. No fuel assembly or component manipulations in the Spent Fuel Pools will be performed.
    b. No major Waste Gas maintenance activities or WGDT releases will be performed.
    c. No primary resin transfer operations will be performed.

Revision 15 Change Summary:
DRR 513759

NOTE

Writers and Reviewers: Changes to certain parts of this document may impact other EPIPs.

EAL Bases Manual	EM-202
Section 3.1	Section 3.0
Attachment 1	Enclosure 1

Ensure appropriate PRRs are initiated as needed.

EALBM Section	CHANGE, REASON, REFERENCES
Throughout	Revised footer to reflect Revision 15. **Editorial change.**
EAL 5.1	Changed sample line measuring distance from 1 foot to 2 feet. (CR 556007)
EAL 1.1	Replaced current threshold values with new values resulting from replacement of RM-A1/-A2 and impact of extended shutdown isotopic mix. Revised discussion accordingly. (PRR 513740)
EAL 1.2	Replaced current threshold values with new values resulting from replacement of RM-A1/-A2 and impact of extended shutdown isotopic mix. Revised discussion accordingly. (PRR 513740)
EAL 1.3	Replaced current threshold values with new values resulting from replacement of RM-A1/-A2 and impact of extended shutdown isotopic mix. (PRR 513740)
EAL 1.4	Replaced current threshold values with new values resulting from replacement of RM-A1/-A2 and impact of extended shutdown isotopic mix. Deleted discussion on LMHVC operation. (PRR 513740)
EAL 2.10	Added the NEI 99-01 rev 5 definition of normal plant operations to clarify when this EAL is applicable. (CR 506400-15)
EAL 2.17	Replaced "Superintendent Shift Operations" with "Shift Manager" to delete obsolete title. **Editorial change.** (DRR 493226)

EALBM Section	CHANGE, REASON, REFERENCES
Attachment 2	Deleted discussion of DAC usage in determining ODCM limits for EALs 1.1 and 1.2. Reference to ODCM limits have been eliminated in these EALs. Deleted discussion of RM-A1/-A2 mid and high range monitor values for EALs 1.3 and 1.4 since these monitors have been removed. (PRR 513740)

Progress Energy

R
Reference
Use

CRYSTAL RIVER UNIT 3

PLANT OPERATING MANUAL

EMERGENCY PLAN IMPLEMENTING PROCEDURE

EM-202

DUTIES OF THE EMERGENCY COORDINATOR

REVISION 98

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1.0 PURPOSE

1. This procedure provides instructions and guidelines used by the Emergency Coordinator during initiation of the Radiological Emergency Response Plan. Specific guidelines include emergency classification, reporting and notification requirements, and protective action recommendations for non-essential Energy Complex personnel and the public. Portions of this procedure are also used by the Emergency Operations Facility staff for offsite notifications, protective action recommendations, and Emergency Action Level determinations.
2. This procedure is an Emergency Plan Implementing Procedure. Any revisions must be carefully considered for Emergency Plan impact.

2.0 REFERENCES

2.1 Developmental References

1. 10 CFR 50.47, Emergency Plans
2. 10 CFR 50, Appendix E, Emergency Planning and Preparedness for Production and Utilization Facilities
3. 10 CFR 50.72, Immediate Notification Requirements for Operating Nuclear Power Reactors
4. CR3 Severe Accident Guideline
5. Emergency Action Level Bases Manual
6. Letter FCS-9852, Oct 12, 1988, Gilbert Engineering Study "Internal Flooding of Power Plant Building."
7. Manual of Protective Action Guides and Protective Actions for Nuclear Incidents, EPA-400-R-92-001, Environmental Protection Agency (October, 1991)
8. NEI 91-04, Revision 1, Severe Accident Issue Closure Guidelines
9. NEI 97-03, Methodology for Development of Emergency Action Levels
10. NUREG-0654, Criteria for Preparation and Evaluation of Radiological Emergency Response Plans and Preparedness in Support of Nuclear Power Plants
11. Off-Site Dose Calculation Manual
12. Radiological Emergency Response Plan
13. Safety Evaluation of FPC proposed EAL changes for CR3 (TAC No. MA2231), NRC to FPC letter 3N0299-02
14. NRC Order for Interim Safeguards and Security Compensatory Measures, Dated 02/25/02
15. NRC RIS 2003-12, Clarification of NRC Guidance for Modifying Protective Actions
16. NRC Bulletin 2005-02, "Emergency Preparedness and Response Actions for Security-based Events"
17. NEI 99-01, Rev. 5, Methodology for Development of Emergency Action Levels
18. EMG-NGGC-0005, Activation of the Emergency Response Organization Notification System

19. NRC RIS 2009-10, Communications Between the NRC and Reactor Licensees During Emergencies and Significant Events
20. EM-102, Operation of the Technical Support Center (TSC)
21. EM-103, Operation and Staffing of the CR-3 Control Room During Emergency Classifications
22. EM-400. Operation of the Emergency Operations Facility (EOF)

3.0 DEFINITIONS

1. **Aircraft:** Aircraft smaller than an Airliner.
2. **Airliner:** A large aircraft with the potential for causing significant damage to the Plant. (The NRC notification should designate aircraft vs. airliner.)
3. **Bomb:** An explosive device suspected of having sufficient force to damage Plant systems or structures. (See EXPLOSION.)
4. **Civil Disturbance:** A group of persons violently protesting station operations or activities at the site. A civil disturbance is considered violent when force has been used in an attempt to injure site personnel or damage Plant property.
5. **Committed Dose Equivalent (CDE):** Dose to an organ due to the intake of radioactive materials.
6. **Credible Site-Specific Security Threat Notification:** A threat specifically to CR3 confirmed and validated by Nuclear Security or received over the Emergency Notification System (ENS) from the NRC. Notification may be received from recognized law enforcement or governmental agencies (e.g. Federal Bureau of Investigation (FBI), Florida Department of Law Enforcement (FDLE), Division of Emergency Management (DEM), Nuclear Regulatory Commission NRC.)
7. **Deep Dose Equivalent (DDE):** External whole body dose.
8. **Emergency Action Level (EAL):** A pre-determined, observable threshold for Plant conditions that places the Plant in a given emergency classification.
9. **Emergency Classification:** A system of classification in which emergency occurrences are categorized according to specific protective action levels. The four emergency classifications are:
    a. **Unusual Event:** This classification refers to any event(s), in process or having occurred, indicating a potential degradation of the level of safety of the Plant **OR** indicate a security threat to facility protection. NO releases of radioactive material requiring offsite response or monitoring are expected unless further degradation of safety occurs. This classification brings the operating staff to a state of readiness if escalation to a more severe action level classification occurs.
    b. **Alert:** This classification refers to event(s) that are in process, or have occurred, involving an actual or potentially substantial degradation of the level of safety of the Plant OR a security event that involves probable life threatening risk to site personnel or damage to site equipment because of HOSTILE ACTION. Any releases are expected to be limited to small fractions of the EPA Protective Action Guideline exposure levels. The Technical Support Center (TSC) and Emergency Operations Facility (EOF) are staffed and assembly and accountability are performed at local assembly areas.

c. **Site Area Emergency:** This classification refers to event(s) that are in process, or have occurred, involving actual or likely major failures of Plant functions needed for the protection of the public **OR** HOSTILE ACTION that results in intentional damage or malicious acts; (1) toward site personnel or equipment that could lead to the likely failure of or; (2) prevents effective access to equipment needed for the protection of the public. Any releases are **NOT** expected to result in exposure levels, which exceed EPA Protective Action Guideline exposure levels at the SITE BOUNDARY. The TSC and the Emergency Operations Facility (EOF) are staffed and radiation monitoring teams may be dispatched. Protected Area evacuation and accountability is performed at CR3. Assembly and accountability is performed at Units 1/2 & 4/5.

d. **General Emergency:** This classification refers to event(s) that are in process, or have occurred, involving actual or imminent substantial core degradation or nuclear fuel melting with the potential for loss of containment integrity **OR** HOSTILE ACTION that results in an actual loss of physical control of the facility. Releases can be reasonably expected to exceed EPA Protective Action Guidelines exposure levels for more than the immediate site area. This classification initiates predetermined protective actions for the public, provides continuous assessment of information from on-site and off-site measurements, initiates additional measures indicated by the event, and provides current information and consultation with off-site authorities and the public. The Emergency Coordinator will probably decide to evacuate the Energy Complex.

10. **Emergency Coordinator (EC):** This position is the highest level of authority for the CR3 Emergency Organization and on-site emergency activities. This position is held by the Plant General Manager or designated alternate. The Shift Manager assumes the position until the Plant General Manager or designated alternate arrives to assume Emergency Coordinator responsibilities.

11. **Emergency Response Data System (ERDS):** NRC requirement {10 CFR 50.72(a)(4)} to have the ability to acquire data from nuclear power Plants in the event of an emergency at the Plant. ERDS is a direct real-time transfer of data from CR3 to NRC. Once initiated, ERDS operates automatically.

12. **Explosion:** A rapid, violent, unconfined combustion, or a catastrophic failure of pressurized equipment that imparts energy of sufficient force to potentially damage permanent structures, systems, or components.

13. **Extortion:** An attempt to cause an action at CR3 by threat of force. Bomb threats that are unsubstantiated are **NOT** included in this definition.

14. **Fire:** Combustion characterized by heat and light. Sources of smoke such as slipping drive belts or overheated electrical equipment do **NOT** constitute fires. Observation of flame is preferred but is **NOT** required if large quantities of smoke and heat are observed.

15. **Hostage:** A person or object held as leverage against the station to ensure that demands will be met by CR3.

3.0 Definitions (Cont'd)

16. **Hostile Action:** An act toward a nuclear power Plant or its personnel that includes the use of violent force to destroy equipment, take hostages, and/or intimidate the licensee to achieve an end. This includes attack by air, land, or water using guns, explosives, projectiles, vehicles, or other devices used to deliver destructive force. Other acts that satisfy the overall intent may be included. Hostile action should **NOT** be construed to include acts of civil disobedience or felonious acts that are **NOT** part of a concerted attack on the nuclear power Plant. Non-terrorism-based EALs should be used address such activities (e.g., violent acts between individuals in the Owner Controlled Area).

17. **Hostile Force:** One or more individuals who are engaged in a determined assault, overtly or by stealth and deception, equipped with suitable weapons capable of killing, maiming, or causing destruction.

18. **IDLH Level:** Level of toxic gas Immediately Dangerous to Life or Health

19. **Incident Report:** A report of the actual scenario of the emergency, the identified cause(s) of the emergency, and the radiological history of the emergency, including released quantities, existing radiological activity, abnormal doses for emergency worker and population doses.

20. **Intrusion/Intruder:** Suspected hostile individual (outsider) present in the Protected Area without authorization. An intruder also includes a badged employee (insider) attempting to commit or providing assistance to others in committing sabotage. These activities may occur while the insider is either physically inside or outside the Protected Area. Upon identification, the insider's authorization is immediately revoked by Nuclear Security.

21. **Local Assembly Area:** A pre-designated area personnel report for organization, roll call, and supervision following an "Alert" emergency classification.

22. **Main Assembly Area (MAA):** The place personnel report for organization and supervision following an evacuation of the CR3 Protected Area. The Main Assembly Area is the Site Administration Building Auditorium.

23. **MODES:** The ITS based designator of Plant status based on Reactivity, Temperature and RCS status and includes operating modes 1 through 6 and defueled (no mode) as applicable. The term "MODES:ALL" applies to MODES 1-6 and defueled (no mode).

24. **Owner-Controlled Area:** That area, including the PROTECTED AREA, that extends 4400 feet or 0.83 miles in a circle around the Reactor Building.

25. **Protected Area:** All areas within the CR3 security perimeter fence that require badged authorization for entry.

26. **Protective Action Recommendations:** Emergency measures recommended for purposes of preventing or minimizing radiological exposures to Energy Complex personnel or members of the public. Protective Action Recommendations are made using all available data, primarily Plant conditions. Off-site dose projections and/or field survey results can also be factored in to Protective Action Recommendations if confidence in their accuracy is high (monitored release, confirmed field survey results).

3.0 Definitions (Cont'd)

27. **RCS Barrier:** The RCS primary side and its connections up to and including the Pressurizer safety and relief valves, and other connections up to and including the primary isolation valves. An isolable leak in an interfacing or connecting system that contains reactor coolant (MU, DH, SF, WD, etc.) is **NOT** an "RCS leak."

28. **Release** (Florida Nuclear Plant Emergency Notification Form)**:** Any of the following:

NOTE:	If RM-A2 is out of service and normal Auxiliary Building ventilation is in service, RM-A4 and/or RM-A8 exceeding its warning setpoint may be used to determine a release in progress.

- Exceeding the warning setpoint in count rate on an effluent monitor that is a direct result of an event that has initiated an emergency declaration

  OR
- Radioactivity detected by environmental monitoring

  OR
- OTSG tube rupture > 10 gpm with either of the following:
  - Prolonged steaming to the atmosphere from the affected OTSG

    OR
  - an unisolable steam leak outside RB from the affected OTSG

  OR

NOTE:	Design Basis Leakage or other suspected leakage should **NOT** be categorized as a release until confirmed by environmental monitoring.

- Radioactivity escaping unmonitored from the Plant.

29. **Release, Unplanned** (Reactor Plant Event Notification Worksheet)**:** Release is **NOT** authorized by a Release Permit or exceeds the conditions (e.g., minimum dilution flow, maximum discharge flow, alarm setpoints, etc.) on the applicable permit.

30. **Sabotage:** Deliberate damage, mis-alignment, or mis-operation of Plant equipment with the intent to render the equipment unavailable. Equipment found tampered with or damaged due to malicious mischief may **NOT** meet the definition of SABOTAGE until this determination is made by Nuclear Security.

31. **Safe Shutdown Equipment:** Equipment necessary to achieve and maintain the reactor subcritical with controlled decay heat removal to bring the Plant to the ITS applicable shutdown condition / mode.

32. **Security Condition:** Any Security Event as listed in the approved security contingency plan that constitutes a threat/compromise to site security, threat/risk to site personnel, or a potential degradation to the level of safety of the plant.
A SECURITY CONDITION does not involve a HOSTILE ACTION.

3.0 Definitions (Cont'd)

33. **Severe Accident:** An accident beyond that assumed in the CR3 design and licensing basis that results in catastrophic fuel rod failure, core degradation, and fission product release into the Reactor vessel, Reactor Building, or the environment.

34. **Significant Transient:** An UNPLANNED event involving one or more of the following:

    a. Automatic turbine trip at greater than 25% reactor thermal power

    b. Electrical load rejection greater than 25% full electrical load

    c. Plant runback

    d. Reactor trip

    e. Safety injection system actuation

    f. Greater than 10% thermal power oscillations

    g. Loss of decay heat removal in Mode 4 ("Significant Transient" is **NOT** used in any Mode 5 or 6 EAL)

35. **Site Boundary:** That area, including the PROTECTED AREA that extends 4400 feet or 0.83 miles in a circle around the Reactor Building. Also referred to as the Owner Controlled Area.

36. **Strike Action:** Is a work stoppage within the PROTECTED AREA by a body of workers to enforce compliance with demands made. The strike actions must threaten to interrupt normal Plant operations.

37. **Thyroid CDE Dose:** Dose to the thyroid due to intake of radioactive iodine.

38. **Total Effective Dose Equivalent (TEDE):** The sum of external dose (DDE) and the equivalent amount of whole body dose due to individual organ uptakes.

39. **Unplanned:** An event or action is UNPLANNED if it is **NOT** the expected result of normal operations, testing, or maintenance. Events that result in corrective or mitigative actions being taken in accordance with abnormal or emergency procedures are UNPLANNED.

40. **Valid:** An indication or report or condition is considered VALID when it is conclusively verified by (1) an instrument channel check, or (2) indications on related or redundant indicators, or (3) by direct observation by Plant personnel, such that doubt related to the indicator's operability, the condition's existence, or the report's accuracy is removed. Implicit in this definition is the need for timely assessment (e.g., within 15 minutes).

41. **Visible Damage:** Damage to equipment or structure that is readily observable without measurements, testing, or analyses. Damage is sufficient to cause concern regarding the continued operability or reliability of affected safety structure, system, or component. Example system/component damage includes: deformation due to heat or impact, denting, penetration, rupture, cracking, paint blistering due to fire. Example structure damage includes exposed and/or broken rebar, failed supports/pipe hangers, etc. Surface blemishing (e.g., paint chipping, scratches, concrete spalling) should **NOT** be included.

4.0 RESPONSIBILITIES

1. The Emergency Coordinator controls all activities at CR3 during activation of the Radiological Emergency Response Plan.
2. The Emergency Coordinator shall **NOT** delegate the decisions related to classification of the emergency condition.
3. The Emergency Coordinator shall **NOT** delegate the decisions related to notification and protective action recommendations to State and Local authorities who implement off-site emergency measures, until the EOF Director communicates to the Emergency Coordinator the EOF accepts the State notification and Protective Action Recommendations (PARs) responsibilities. At this time, the EOF completes the Florida Nuclear Plant Emergency Notification Form.
4. Upon arrival on-site, the Plant General Manager (PGM) or designated alternate contacts the Control Room Emergency Coordinator or goes to the Control Room and receives a briefing about the status of the emergency condition and the implementation of the Radiological Emergency Response Plan. When ready to assume responsibility as the Emergency Coordinator and declare the TSC operational, inform the Control Room Emergency Coordinator and Technical Support Center staff.
5. The Emergency Coordinator provides the Emergency Operations Facility Director an Incident Report when a sustained Site Area Emergency or General Emergency involves a Recovery Plan. This documents the emergency and serves as a basis for recovery phase operations.
6. During declared emergency conditions, the Emergency Coordinator is the sole contact for emergency regulatory directives and evaluates these directives for possible response to the emergency condition.
7. The Emergency Coordinator responsibilities in other Emergency Plan Implementing Procedures are implemented when Plant conditions warrant.

8. Based on the evaluation of the emergency condition, the Emergency Coordinator has the authority to implement the following actions:
   - Direct personnel to shelter or evacuate the Energy Complex.
   - Order Energy Complex Plants placed in a safe shutdown condition.
   - Notify all applicable agencies of the Plant status.
   - Suspend security safeguards as appropriate. {10 CFR 50.54(x) (y)} or Section 24, Temporary Suspension of Security Measures of the CR3 Physical Security Plan.
   - Request outside assistance, if necessary.
   - Make the necessary personnel assignments to provide continuing response for long-term activities.
   - Approve media releases until the EOF is operational and assumes responsibility.
   - Approve re-entries into the Plant by emergency response teams
   - Approve emergency exposure dose during re-entries. Refer to Enclosure 3, Guidelines for Protective Action Recommendations for Non-Essential Energy Complex Personnel and General Population for emergency worker exposure limits.
   - Provide support for the Incident Commander in performance of EM-913.
9. The Emergency Coordinator reports to the EOF Director, once the EOF is operational.
10. The EOF Director provides for the direction and control of all emergency phase activities once the EOF is declared operational. The EOF Director has authority and responsibility for management of emergency response resources, coordination of radiological and environmental assessment, recommendations for public protective actions, and coordination of emergency response activities with Federal, State, and local agencies.
11. The Licensing / Regulatory Programs Unit prepares a written summary of any Alert, Site Area Emergency or General Emergency for the NRC and the State of Florida within twenty-four hours (or the next working day) from termination of the event.
12. The TSC Emergency Coordinator and/or the EOF Director may be requested to participate in conference calls with the NRC during certain emergencies and significant events. The purpose of these calls is to assist the NRC in their understanding of the nature of the emergency or significant event in a timely fashion. Depending on the nature of the event, participants from the NRC may include the NRC Executive Team Director, NRC Headquarters Safeguards Team personnel, and/or NRC regional responders. These calls may be conducted over existing telecommunication networks (i.e. the FTS-2001 system). Other conference / bridge lines may also be established by the NRC. Refer to Enclosure 6, Communication with NRC Management During an Event, for typical topics likely to be discussed during these conference calls.

4.0 Responsibilities (Cont'd)

13. During Severe Accident conditions, the Emergency Coordinator reviews and provides final approval of all mitigation strategies developed by the Accident Assessment Team before implementation. [NOCS 100056]

14. Nuclear Security activates the Emergency Response Organization and implements evacuation of the Crystal River Energy Complex based on requests from the Emergency Coordinator. If Nuclear Security is unable to activate the ERO due to the nature of the event, ERO activation will be performed by the Control Room staff.

15. During certain emergencies (e.g., security-related events, large area fire), the Crystal River Energy Complex Emergency Response Coordinator may establish an Incident Command Post (ICP). In the event an ICP is established, the Emergency Coordinator may assign Operations personnel to staff the ICP to support its function and to provide liaison between CR3 Operations and off-site response agencies (e.g., local law enforcement, fire/rescue, emergency medical, etc.)

5.0 PREREQUISITES

None

6.0 PRECAUTIONS, LIMITATIONS, AND NOTES

1. Upon declaration of a General Emergency, the minimum protective action recommendation is EVACUATE ZONE 1.
2. Some EALs allow an off-normal condition to exist for a period of time before the EAL threshold is met. This time period is intended to be used for validation and assessment of the off-normal condition. For example, EAL 2.14 (Fire – Unusual Event) allows 15 minutes for a fire to be extinguished before the EAL threshold is met. However, the emergency assessment and declaration phases should occur concurrently in order to ensure that emergencies are declared in a timely manner. The Emergency Coordinator should not delay declaration of an event when it is likely that the event will meet an EAL threshold even if the specified assessment time period has not expired.
3. During the initial phase of an emergency condition, the lack of information may prevent the Emergency Coordinator from completing the Florida Nuclear Plant Emergency Notification Form. If information is **NOT** available, do **NOT** delay notification to State Watch Office. Indicate additional information will follow when it becomes available.
4. The Reactor Plant Event Notification Worksheet is used as a guideline to provide adequate detail to the NRC Headquarters Operations Officer to understand the event and its significance. The initial NRC notification may be performed using the information from Items 4 through 7 and Item 11 of the FLORIDA NUCLEAR PLANT EMERGENCY NOTIFICATION FORM, in order to expedite notification from the Main Control Room. Since the NRC is **NOT** familiar with the EAL numbers from Item 6, Enclosure 4 should be used to provide the paraphrased EAL. If an open communications channel is established, routine use of the form is **NOT** required, if verified changes in Plant / equipment status are communicated to the NRC verbally and a summary of the communications with the NRC is maintained in the log. All the information regarding an event may **NOT** be available at the time of notification, but at a minimum must provide the event classification and description as soon as possible after the State of Florida notification, within the required time.
5. For all radiological, hazardous material spills, toxic gas releases or violent weather conditions, the Emergency Coordinator determines the safe actions for Plant personnel, which may include delaying the staffing of the TSC and EOF until it is safe to do so.
6. The Emergency Coordinator directly notifies the Plant General Manager or EC On-Call and EOF Director to ensure the rationale of the emergency classification is understood. It is acceptable, if the EC requests the PGM or EC On-Call to notify the EOF Director or the EC may establish a conference call.
7. Individuals assigned to make notifications are trained on how to make notifications and are familiar with communication systems. [NOCS 21207]

6.0 Precautions, Limitations, and Notes (Cont'd)

8. The Technical Support Center (TSC) continues to complete items on the Florida Nuclear Plant Emergency Notification Form and transmits to the EOF until the EOF Director declares the EOF operational, and informs the Emergency Coordinator the EOF accepts responsibility for State notifications and Protective Action Recommendations. At this time, the EOF Director assumes full responsibility for completing the Florida Nuclear Plant Emergency Notification Form. Any exceptions to the transfer of these responsibilities (delay in transfer, etc) must be clearly communicated during the facility turnover briefing.

9. Telephone notifications to the Nuclear Regulatory Commission (NRC), State of Florida, Citrus and Levy Counties are complete when direct voice contacts are made with the responsible representatives of the agencies notified. The leaving of a message with an agency's telephone operator, secretary, answering service, or message recording device is **NOT** a completed notification.

10. The Emergency Action Levels are **NOT** intended for maintenance and/or testing situations where abnormal instrument readings, alarms, and observations are expected. Some maintenance evolutions may require compensatory actions.

11. A security threat or event presents unique challenges to protecting the health and safety of the public and Plant staff. Normal emergency response procedure steps may be hindered due to events that are occurring. EM-911 provides operational activities and considerations to protect Plant personnel for a security threat. All actions of EM-911 should still be completed from the Control Room even when the TSC/EOF are operational.

12. Once Protective Action Recommendations are made to the State of Florida and Risk Counties, do **NOT** relax / reduce the recommendations until the threat is clearly under control or the emergency is terminated.

7.0 SPECIAL TOOLS AND EQUIPMENT

None

8.0 ACCEPTANCE CRITERIA

None

9.0 INSTRUCTIONS

1. RECORD significant information, events, and actions taken during the emergency condition **AND** RETAIN for later evaluation. Information substantiating the sequence of events is compiled from procedures, communication logs, tape recordings, flip charts, message copies, photographs (if available) and other pertinent documentation
2. DETERMINE the emergency classification using Enclosure 1, Emergency Classification Table.
   - Page 2 FISSION PRODUCT BARRIER MATRIX
   - Page 3 ABNORMAL RAD LEVELS / RADIOLOGICAL EFFLUENT
   - Page 5 NATURAL / MANMADE HAZARDS AND EC JUDGEMENT
   - Page 12 SYSTEM MALFUNCTION
   - Page 17 LOSS OF POWER
3. PERFORM steps from the Emergency Coordinator Guide for each emergency classification as indicated in the following Subsections:
   - 9.1 UNUSUAL EVENT
   - 9.2 ALERT
   - 9.3 SITE AREA EMERGENCY
   - 9.4 GENERAL EMERGENCY
4. USE the time blocks in Subsections 9.1, 9.2, 9.3 and 9.4 to provide a reference of actions taken during the emergency condition. All actions, with the exception of decisions relating to classification and notification and Protective Action Recommendations made to State and Local authorities, can be performed in parallel by delegation from the Emergency Coordinator.
5. **IF** an emergency classification is upgraded before the first notification is made, **THEN** ENSURE SWO notification is made within 15 minutes of original classification.
6. **IF** it is discovered after the fact (review of routine log entries, etc.) that a condition previously existed that should have resulted in an emergency declaration, AND the condition NO longer exists, **THEN** make notifications to the NRC Operations Center via ENS within one hour of discovering the undeclared event, **AND** NOTIFY the Emergency Preparedness staff to NOTIFY the State and Local Governments on the next working day. An emergency declaration is **NOT** required.

7. **IF** a transient event condition is corrected before a declaration is made, **AND** analyses of the event is **NOT** required to determine whether further Plant damage occurred while corrective actions were being taken, **THEN** a declaration is **NOT** warranted but the event is reported and notification made to the NRC Operations Center via ENS within one hour of the event, **AND** ENSURE the Emergency Preparedness staff is notified to NOTIFY the State and Local Governments on the next working day. (e.g., the PORV (RCV-10) develops a leak or fails open with a leak rate of greater than 25 gpm and the block valve (RCV-11) is closed and successfully isolates the leak to less than the EAL threshold).

8. Information requested for TSC turnover is contained in Attachment 1 of EM-102, Operation of the Technical Support Center. CONSIDER establishing a conference call with the EC On-Call and EOF Director for this turnover.

9. REFER to EM-103 for additional Control Room activities during a declared emergency including dispatch of Operators outside of the Control Complex.

10. In most situations, events are terminated rather than downgraded. However, there may be conditions where downgrading is appropriate. For downgrading the emergency classification level, if the current Plant conditions have improved to satisfy a lower classification Emergency Action Level, NOTIFY the Emergency Coordinator On-Call and EOF Director for concurrence to downgrade. For Alerts or higher, unless the conditions are resolved within 30 minutes, downgrading should **NOT** occur until after the TSC and EOF (as appropriate) are operational and the event sufficiently evaluated by the Emergency Response Organization.

11. For Emergency Phase termination and transition to the Recovery Phase, from an Unusual Event or Alert, DETERMINE the need for a Recovery Plan and a support organization. For a Site Area Emergency or General Emergency, ASSIST the EOF Director with the completion of the Termination Checklist from EM-400. **IF** the Site Area Emergency event is of short duration (approximately 30 minutes or less), and the EOF is **NOT** operational, **THEN** TERMINATE the event. If conditions will allow for termination of the Emergency Phase, ENTER the Recovery Phase. If conditions do **NOT** support termination of the emergency and entry into the Recovery Phase, CONTINUE the Emergency Phase.

12. REFER to EM-913 for EC/EC Designee responsibilities in response to a large area fire.

9.1 **Emergency Coordinator's Guide for Unusual Event [NOCS 1129, 96042]**

TIME

1. **UNUSUAL EVENT DECLARED** .... DATE ______ / ______
   (If event is transitory in nature, refer to item 9.0.7 before declaring event)

2. RECOMMENDED WITHIN 5 MINUTES

   a. NOTIFY Control Room staff of: .... ______

      1) Emergency declaration

      2) Upgrade criteria (if any)

      3) Release status

   b. **IF** the emergency is due to a Security Event, **THEN** REFER TO EM-911 before proceeding with the following steps. .... ______

   c. NOTIFY Plant Personnel using information from Step 9.1.7. .... ______

3. **REQUIRED WITHIN 15 MINUTES**

   a. NOTIFY SWO, Citrus County, and Levy County within 15 minutes of declaration using Attachment 1, Florida Nuclear Plant Emergency Notification Form, **AND** FAX after notification is complete. [NOCS 1129, 96042] .... ______

4. RECOMMENDED WITHIN 30 MINUTES

   a. NOTIFY PGM or EC On-Call <u>and</u> the EOF Director. . ... _____ / _____

   b. NOTIFY Nuclear Security to place the Emergency Response Organization on standby using Scenario 2 (Enclosure 5, Emergency Response Facility Activation Scenarios)............................. _____

   c. **IF** Emergency Response Organization support is desired, **THEN** NOTIFY Nuclear Security to activate:

      - TSC/OSC using Scenario 3 (Enclosure 5, Emergency Response Facility Activation Scenarios). .... _____

      OR

      - TSC/OSC/EOF/ENC using Scenario 4 (Enclosure 5, Emergency Response Facility Activation Scenarios). .... _____

   d. NOTIFY CR3 NRC Resident Inspector... _____

   e. NOTIFY Units 1/2 & 4/5 Control Rooms per Attachment 4......... _____ / _____

TIME

f. REVIEW Enclosure 2, Evacuation Planning Guide for applicability to this event. ______

g. **IF** a release is occurring as a result of this event, **AND** RM-A2 is needed for evaluation, **THEN** COMPLETE EM-204A or EMG-NGGC-0002, as time permits. ______

h. NOTIFY NRC via ENS as soon as practicable after the State using information from Items 4 – 7 and Item 11 of the Florida Nuclear Plant Emergency Notification Form or Attachment 3, Reactor Plant Event Notification Worksheet. **REQUIRED WITHIN 60 MINUTES**. [NOCS 96042] ______

i. NOTIFY CR3 Emergency Preparedness. ______

5. **UNUSUAL EVENT UPDATES**

a. PROVIDE periodic Plant status updates to:

- SWO (every 60 minutes or as agreed upon) per Attachment 1, Florida Nuclear Plant Emergency Notification Form ☐
- NRC per Attachment 3, Reactor Plant Event Notification Worksheet (after State of Florida update, unless continuous communication is established) ☐
- Units 1/2 & 4/5 Control Rooms per Attachment 4, Emergency Notification for Units 1/2 & 4/5 ☐
- CR3 Plant Personnel via PA announcements ☐

6. **UNUSUAL EVENT TERMINATION**

a. Upon the decision to terminate, NOTIFY: DATE _____/_____

- Emergency Coordinator On-Call and EOF Director _____, _____
- SWO and document on Attachment 1, Florida Nuclear Plant Emergency Notification Form ______
- Nuclear Security to inform the Emergency Response Organization of event termination using Scenario 13 (Enclosure 5, Emergency Response Facility Activation Scenarios). ______
- NRC within one hour of termination with verbal summary ______
- Unit 1/2 & 4/5 Control Rooms per Attachment 4 ______
- CR3 Plant Personnel via PA announcement ______

7. **PA Announcement for an Unusual Event** (a. **OR** b.)

   a. ANNOUNCE **OR** PERFORM the following:

   Time: ______

   1) ACTUATE the appropriate local evacuation alarm if required. .... ☐

   2) "ATTENTION ALL PERSONNEL, CRYSTAL RIVER 3 IS IN AN UNUSUAL EVENT BASED ON

   ____

   ____

   3) "THERE (IS **OR** IS **NOT**) A RADIOLOGICAL RELEASE TO THE ENVIRONMENT IN PROGRESS." .... ☐

   4) STATE any appropriate special instructions (areas to be avoided or evacuated, etc.).. (**IF** conditions warrant personnel accountability, **THEN** REQUEST personnel to report to Local Assembly Areas).

   ____

   ____

   5) REPEAT the announcement. .... ☐

   6) ESTABLISH continuous monitoring on PL-1. .... ☐

OR

   b. USE this announcement for a Credible Site-Specific Security Threat where time is available and a decision has been made to use the Remote TSC (during normal hours) .... ☐

   1) ACTUATE the appropriate local evacuation alarm if required. .... ☐

   "ATTENTION ALL PERSONNEL, CRYSTAL RIVER 3 IS IN AN UNUSUAL EVENT BASED ON CREDIBLE SITE-SPECIFIC SECURITY THREAT. TSC / OSC STAFF PERSONNEL ARE TO REPORT TO THE EOF. HEALTH PHYSICS PERSONNEL ARE TO RELOCATE THE ESV AND EMERGENCY KITS TO THE EOF. FIRE BRIGADE MUSTER AT THE______________________________."

   2) REPEAT the announcement. .... ☐

   3) ESTABLISH continuous monitoring on PL-1 .... ☐

9.2 **Emergency Coordinator's Guide for an Alert** [NOCS 1129, 96042]

TIME

1. **ALERT DECLARED** ....DATE _____/_____
(If event is transitory in nature, refer to item 9.0.7 before declaring event)

2. RECOMMENDED WITHIN 5 MINUTES

   a. NOTIFY Control Room: .... _____

      1) Emergency declaration

      2) Upgrade criteria (if any)

      3) Release status

   b. **IF** the emergency is due to a Security Event, **THEN** REFER TO EM-911 before proceeding with the following steps. .... _____

   c. **IF** safe conditions exist, **THEN** NOTIFY Nuclear Security to activate the Emergency Response Organization using Scenario 5 (Enclosure 5, Emergency Response Facility Activation Scenarios)... _____

   d. **IF** conditions (security event, violent weather, natural disaster, etc.) require activation of remote emergency facilities, **THEN** NOTIFY Nuclear Security to activate the Emergency Response Organization using Scenario 8 (Enclosure 5, Emergency Response Facility Activation Scenarios). .... _____

   e. NOTIFY Plant Personnel using information from Step 9.2.9. .... _____

3. **REQUIRED WITHIN 15 MINUTES**

   a. NOTIFY SWO, Citrus County, and Levy County within 15 minutes of declaration per Attachment 1, Florida Nuclear Plant Emergency Notification Form, **AND** FAX after notification is complete. [NOCS 1129, 96042] .... _____

4. RECOMMENDED WITHIN 30 MINUTES **TIME**

   a. NOTIFY PGM or EC On-Call and the EOF Director .... _____, _____

   b. NOTIFY CR3 NRC Resident Inspector.. _____

   c. NOTIFY Units 1/2 & 4/5 Control Rooms per Attachment 4.......... _____, _____

   d. REVIEW Enclosure 2, Evacuation Planning Guide for applicability to this event.. _____

   e. **IF** a release is occurring as a result of this event, **AND** RM-A2 is needed for evaluation, **THEN** COMPLETE EM-204A or EMG-NGGC-0002, as time permits.................................. _____

   f. NOTIFY NRC via ENS as soon as practicable after the State using information from Items 4 – 7 and Item 11 of the Florida Nuclear Plant Emergency Notification Form or Attachment 3, Reactor Plant Event Notification Worksheet. **REQUIRED WITHIN 60 MINUTES.** [NOCS 96042] .... _____

   g. ENSURE ERDS is activated per Attachment 5, Initiation of the Emergency Response Data System (ERDS). **REQUIRED WITHIN 60 MINUTES** [NOCS 40730]............................. _____LL

   h. REVIEW EM-103 for operator dispatch requirements. .... _____

5. **ONCE TSC OPERATIONAL**

   a. NOTIFY ANI insurance that CR3 is in an emergency declaration. (Off-Site Support Phone Directory) .... _____

   b. NOTIFY Risk Management to notify NEIL insurance that CR3 is in an emergency declaration. (Off-Site Support Phone Directory) .... _____

   c. NOTIFY INPO that CR3 has declared an Alert (Off-Site Support Phone Directory)... _____

6. **ALERT UPDATES**

   a. PROVIDE periodic Plant status updates to:

      - SWO (every 60 minutes or as agreed upon) per Attachment 1, Florida Nuclear Plant Emergency Notification Form including Items 12, 13, and 14 ....☐
      - Units 1/2 & 4/5 Control Rooms per Attachment 4 ....☐
      - CR3 Plant Personnel via PA announcements..................................☐

7. **ALERT DOWNGRADING**

   a. CONSULT with the EC and EOF Director for concurrence before downgrading occurs .... _____/_____
   Date / Time

8. **ALERT TERMINATION**

TIME

a. Upon the decision to terminate, NOTIFY: .... Date: _______, ______

- PGM and EOF Director .... ______
- SWO and document on Attachment 1, Florida Nuclear Plant Emergency Notification Form .... ______
- Nuclear Security to inform the Emergency Response Organization of event termination using Scenario 13 (Enclosure 5, Emergency Response Facility Activation Scenarios). .... ______
- NRC within one hour of termination with verbal summary .... ______
- Unit 1/2 & 4/5 Control Rooms per Attachment 4 .... ______
- CR3 Plant Personnel via PA announcement .... ______
- American Nuclear Insurers (ANI) (Off-Site Support Phone Directory) .... ______
- Risk Management (Off-Site Support Phone Directory) .... ______
- INPO (Off-Site Support Phone Directory) .... ______

b. REQUEST the Licensing/Regulatory Programs Unit to prepare a written summary within twenty-four hours (or next working day) of termination to SWO and NRC. .... ______

9. **PA ANNOUNCEMENT FOR AN ALERT**

a. CONSIDER the safety of Plant personnel and then ANNOUNCE **OR** PERFORM the following:

Time: ______

1) ACTUATE the appropriate local evacuation alarm if required. .... ☐

2) "ATTENTION ALL PERSONNEL, CRYSTAL RIVER 3 IS IN AN ALERT BASED ON

____"

3) "THERE (IS **OR** IS **NOT**) A RADIOLOGICAL RELEASE TO THE ENVIRONMENT IN PROGRESS." .... ☐

4) "ACTIVATE THE TSC/OSC. REPORT TO YOUR SHOP OR LOCAL ASSEMBLY AREA FOR ACCOUNTABILITY." .... ☐

5) STATE any appropriate special instructions (areas to be avoided or evacuated, remaining at critical jobs, etc.).

____

____

6) "ALL EOF PERSONNEL, REPORT TO THE EOF." .... ☐

7) REPEAT the announcement. .... ☐

8) ESTABLISH continuous monitoring on PL-1 .... ☐

9.3 **Emergency Coordinator's Guide for Site Area Emergency** [NOCS 1129, 96042]

TIME

1. **SITE AREA EMERGENCY DECLARED** .... DATE: ______, ______

2. RECOMMENDED WITHIN 5 MINUTES

   a. NOTIFY Control Room staff: .... ______

      1) Emergency declaration

      2) Upgrade criteria (if any)

      3) Release status

   b. **IF** the emergency is due to a Security Event, **THEN** REFER TO EM-911 before proceeding with the following steps. .... ______

   c. **IF** safe conditions exist, **THEN** NOTIFY Nuclear Security to activate the Emergency Response Organization (if **NOT** already activated) using Scenario 6 (Enclosure 5, Emergency Response Facility Activation Scenarios)............................ ______

   d. **IF** conditions (security event, violent weather, natural disaster, etc.) require activation of remote emergency facilities, **THEN** NOTIFY Nuclear Security to activate the Emergency Response Organization (if **NOT** already activated) using Scenario 9 (Enclosure 5, Emergency Response Facility Activation Scenarios)... ______

   e. **IF** personnel can evacuate safely, **THEN** NOTIFY Plant Personnel using information from Step 9.3.10 **AND** ACTUATE Site Evacuation Alarm. REVIEW Enclosure 2, Evacuation Planning Guide for applicability.................... ______

3. **REQUIRED WITHIN 15 MINUTES**

   a. NOTIFY SWO, Citrus County, and Levy County within 15 minutes of declaration per Attachment 1, Florida Nuclear Plant Emergency Notification Form, **AND** FAX after notification is complete. (Also REFER to Step 9.3.0.c) [NOCS 1129, 96042] .... ______

4. RECOMMENDED WITHIN 15 MINUTES

   a. DETERMINE protective actions for Energy Complex using Enclosure 2, Evacuation Planning Guide. NOTIFY Nuclear Security to coordinate protective action instructions for all areas of the Energy Complex. .... ______

   b. NOTIFY Units 1/2 & 4/5 Control Rooms per Attachment 4...... ______, ______

TIME

5. RECOMMENDED WITHIN 30 MINUTES

   a. NOTIFY PGM or EC On-Call and the EOF Director. .... ______, ______

   b. NOTIFY CR3 NRC Resident Inspector. .... ______

   c. **IF** a release is occurring as a result of this event, **AND** RM-A2 is needed for evaluation, **THEN** COMPLETE EM-204A or EMG-NGGC-0002, as time permits. .... ______

   d. NOTIFY NRC via ENS as soon as practicable after the State using information from Items 4 - 7 and Item 11 of the Florida Nuclear Plant Emergency Notification Form or Attachment 3, Reactor Plant Event Notification Worksheet. **REQUIRED WITHIN 60 MINUTES.** [NOCS 96042] .... ______

   e. ENSURE ERDS is activated per Attachment 5, Initiation of the Emergency Response Data System (ERDS). **REQUIRED WITHIN 60 MINUTES**. [NOCS 40730]............................ ______

   f. REVIEW EM-103 for operator dispatch requirements. .... ______

6. **ONCE TSC OPERATIONAL**

   a. VERIFY Protected Area accountability is completed by Nuclear Security within 30 minutes of an evacuation of the Protected Area. .... ______

   b. NOTIFY ANI insurance that CR3 is in an emergency declaration. (Off-Site Support Phone Directory) .... ______

   c. NOTIFY Risk Management to notify NEIL insurance that CR3 is in an emergency declaration. (Off-Site Support Phone Directory) .... ______

   d. NOTIFY INPO that CR3 has declared a Site Area Emergency. (Off-Site Support Phone Directory)................................. ______

7. **SITE AREA EMERGENCY UPDATES**

   a. PROVIDE periodic Plant status updates to:

      - SWO (every 60 minutes or as agreed upon) per Attachment 1, Florida Nuclear Plant Emergency Notification Form including Items 12,13, and 14 .... ☐
      - Units 1/2 & 4/5 Control Rooms per Attachment 4 .... ☐
      - CR3 Plant Personnel via PA announcements................................... ☐

8. **SITE AREA EMERGENCY DOWNGRADING**

   a. **IF** the EC and EOF Director were notified, **THEN** CONSULT with them for concurrence before downgrading occurs. ....DATE _____/_____

9. **SITE AREA EMERGENCY TERMINATION** **TIME**

a. **IF** the EOF is operational, **THEN** ASSIST with the completion of the Termination Checklist from EM-400 .... _____

b. **IF** the event is of short duration (approximately 30 minutes or less) and the EOF is **NOT** operational, **THEN** TERMINATE the event .... _____

c. Upon the decision to terminate, NOTIFY: ....DATE _____/_____

- SWO and document on Attachment 1, Florida Nuclear Plant Emergency Notification Form .... _____
- NRC within one hour of termination with verbal summary .... _____
- Nuclear Security to inform the Emergency Response Organization of event termination using Scenario 13 (Enclosure 5, Emergency Response Facility Activation Scenarios). .... _____
- Units 1/2 & 4/5 Control Rooms per Attachment 4 .... _____
- CR3 Plant Personnel via PA announcement .... _____
- American Nuclear Insurers (ANI) (Off-Site Support Phone Directory) .... _____
- Risk Management (Off-Site Support Phone Directory) .... _____
- INPO (Off-Site Support Phone Directory) .... _____

d. REQUEST the Licensing/Regulatory Programs Unit to prepare a written summary within twenty-four hours (or next working day) of termination to SWO and NRC. .... _____

10. **PA Announcement for a Site Area Emergency** [NOCS 7455]

a. CONSIDER the safety of Plant personnel and then ANNOUNCE **OR** PERFORM the following:

Time: _____

1) ACTUATE the Site Evacuation alarm. .... ☐

2) "ATTENTION ALL PERSONNEL, CRYSTAL RIVER 3 IS IN A SITE AREA EMERGENCY BASED ON ____."

3) "THERE (IS **OR** IS **NOT**) A RADIOLOGICAL RELEASE TO THE ENVIRONMENT IN PROGRESS." .... ☐

4) **IF** the TSC/OSC is **NOT** activated, **THEN** ANNOUNCE: "ACTIVATE THE TSC/OSC." .... N/A ☐ ☐

5) "PERSONNEL ARE TO IMMEDIATELY EVACUATE THE PROTECTED AREA AND REPORT TO THE SITE ADMINISTRATION BUILDING AUDITORIUM." .... ☐

6) "ALL EOF PERSONNEL, REPORT TO THE EOF." .... ☐

7) STATE any appropriate special instructions (areas to be avoided or evacuated, etc.).

____

____

8) REPEAT the announcement. .... ☐

9) ESTABLISH continuous monitoring on PL-1 .... ☐

9.4 Emergency Coordinator's Guide for General Emergency [NOCS 1129, 96042]

1. **GENERAL EMERGENCY DECLARED.**............................ DATE _____ TIME _____

2. RECOMMENDED WITHIN 5 MINUTES

TIME

   a. **IF** the EOF is operational, **THEN** NOTIFY the EOF Director of the classification change. .... ______

   b. NOTIFY Control Room staff of: .... ______

      1) Emergency declaration

      2) Release status

   c. **IF** the emergency is due to a Security Event, **THEN** REFER TO EM-911 before proceeding with the following steps. .... ______

   d. **IF** safe conditions exist, **THEN** NOTIFY Nuclear Security to activate the Emergency Response Organization (if **NOT** already activated) using Scenario 7 (Enclosure 5, Emergency Response Facility Activation Scenarios)............................ ______

   e. **IF** conditions (security event, violent weather, natural disaster, etc.) require activation of remote emergency facilities, **THEN** NOTIFY Nuclear Security to activate the Emergency Response Organization (if **NOT** already activated) using Scenario 10 (Enclosure 5, Emergency Response Facility Activation Scenarios).. _____

   f. **IF** personnel can evacuate safely, **THEN** NOTIFY Plant Personnel using information from Step 9.4.9 **AND** ACTUATE Site Evacuation Alarm if Protected Area **NOT** already evacuated. REVIEW Enclosure 2, Evacuation Planning Guide for applicability. .... ______

3. **REQUIRED WITHIN 15 MINUTES**

   a. DETERMINE Protective Action Recommendations per Enclosure 3 .... ______
      (Minimum Protective Action Recommendation is to evacuate Zone 1.)

   b. **IF** the EOF is **NOT** operational, **THEN** NOTIFY SWO, Citrus County, and Levy County within 15 minutes of declaration per Attachment 1, Florida Nuclear Plant Emergency Notification Form, **AND** FAX after notification is complete. (Also REFER to Step 9.4.5.b) [NOCS 1129, 96042]............................ ______

4. RECOMMENDED WITHIN 15 MINUTES **TIME**

   a. DETERMINE Energy Complex protective actions per Enclosure 2, Evacuation Planning Guide, **AND** NOTIFY Nuclear Security to coordinate evacuation instructions for all areas of the Energy Complex. .... ______

   b. NOTIFY Units 1/2 & 4/5 Control Rooms per Attachment 4. .... _____, _____

5. RECOMMENDED WITHIN 30 MINUTES (**NOT** necessary if TSC and EOF operational) **TIME**

   a. NOTIFY CR3 NRC Resident Inspector. .... _____

   b. **IF** a release is occurring as a result of this event, **AND** RM-A2 is needed for evaluation, **THEN** COMPLETE EM-204A or EMG-NGGC-0002, as time permits. .... ______

   c. NOTIFY NRC via ENS as soon as practicable after the State using information from Items 4 - 7 and Item 11 of the Florida Nuclear Plant Emergency Notification Form or Attachment 3, Reactor Plant Event Notification Worksheet. **REQUIRED WITHIN 60 MINUTES.** [NOCS 96042] .... _____

   d. ENSURE ERDS is activated per Attachment 5, Initiation of the Emergency Response Data System (ERDS). **REQUIRED WITHIN 60 MINUTES.** [NOCS 40730] .... _____

   e. REVIEW EM-103 for operator dispatch requirements. .... _____

6. **ONCE TSC IS OPERATIONAL** **TIME**

   a. VERIFY Protected Area accountability is completed by Security within 30 minutes of an evacuation of the Protected Area. .... _____

   b. NOTIFY ANI insurance that CR3 is in an emergency declaration. (Off-Site Support Phone Directory) .... _____

   c. NOTIFY Risk Management to notify NEIL insurance that CR3 is in an emergency declaration. (Off-Site Support Phone Directory) .... _____

   d. NOTIFY INPO that CR3 that CR3 has declared a General Emergency.(Off-Site Support Phone Directory) .... _____

7. **GENERAL EMERGENCY UPDATES**

a. PROVIDE periodic Plant status updates to:

- SWO (every 60 minutes or as agreed upon) per Attachment 1, Florida Nuclear Plant Emergency Notification Form including Items 12, 13, and 14 ☐
- Units 1/2 & 4/5 Control Rooms per Attachment 4, Emergency Notification for Units 1/2 & 4/5 ☐
- CR3 Plant Personnel via PA announcements ☐

8. **GENERAL EMERGENCY TERMINATION**

a. **IF** the EOF is **NOT** operational, **THEN** WAIT until the EOF is operational before terminating. _____

b. **IF** the EOF is operational, **THEN** ASSIST with the completion of the Termination Checklist from EM-400 _____

c. Upon the decision to terminate, NOTIFY: DATE _____/_____

- NRC within one hour of termination with verbal summary _____
- Nuclear Security to inform the Emergency Response Organization of event termination using Scenario 13 (Enclosure 5, Emergency Response Facility Activation Scenarios). _____
- Unit 1/2 & 4/5 Control Rooms per Attachment 4, Emergency Notification for Units 1/2 & 4/5 _____
- CR3 Plant Personnel via PA announcement _____
- American Nuclear Insurers (ANI) (Off-Site Support Phone Directory) _____
- Risk Management (Off-Site Support Phone Directory) _____
- INPO (Off-Site Support Phone Directory) _____

d. REQUEST the Licensing/Regulatory Programs Unit to prepare a written summary within twenty-four hours (or next working day) of termination to SWO and NRC. _____

9. **PA ANNOUNCEMENT FOR A GENERAL EMERGENCY**
[NOCS 7455]

a. CONSIDER the safety of Plant personnel and then ANNOUNCE or PERFORM the following:

Time: ______

1) **IF** the Protected Area has **NOT** been evacuated,
THEN ACTUATE the Site Evacuation alarm........................ N/A ☐ ☐

2) "ATTENTION ALL PERSONNEL, CRYSTAL RIVER 3 IS IN A GENERAL EMERGENCY BASED ON ______________________"

3) "THERE (IS **OR** IS **NOT**) A RADIOLOGICAL RELEASE TO THE ENVIRONMENT IN PROGRESS." .... ☐

4) **IF** the TSC/OSC is **NOT** activated,
THEN ANNOUNCE: "ACTIVATE THE TSC/OSC." .... N/A ☐ ☐

5) **IF** the Protected Area has **NOT** been evacuated,
THEN ANNOUNCE: "ALL NON-ESSENTIAL PERSONNEL, IMMEDIATELY EVACUATE THE PROTECTED AREA **AND** FOLLOW INSTRUCTIONS FROM SECURITY." .... N/A ☐ ☐

6) **IF** the EOF is **NOT** activated, **THEN** ANNOUNCE:
"ALL EOF PERSONNEL, REPORT TO THE EOF." .... N/A ☐ ☐

7) STATE any appropriate special instructions (areas to be avoided or evacuated, etc.). ____

____

8) REPEAT the announcement. .... ☐

9) ESTABLISH continuous monitoring on PL-1 .... ☐

10.0 RECORDS

Subsection 9.1 – Emergency Coordinator's Guide for Unusual Event
Subsection 9.2 – Emergency Coordinator's Guide for an Alert
Subsection 9.3 – Emergency Coordinator's Guide for Site Area Emergency
Subsection 9.4 – Emergency Coordinator's Guide for General Emergency
Attachment 1 – Florida Nuclear Plant Emergency Notification Form

EMERGENCY CLASSIFICATION TABLE

EMERGENCY ACTION LEVEL INDEX

ABNORMAL RAD LEVELS / RADIOLOGICAL EFFLUENT				
CATEGORY	UE	ALERT	SAE	GE
Gaseous Effluents	1.1	1.2	1.3	1.4
Liquid Effluents	1.5	1.6		
Unexpected Radiation Levels	1.7	1.8		
Irradiated Fuel Damage Due to Mechanical Damage or Uncontrolled Loss of Water Level Outside the Reactor Vessel	1.9	1.10		

NATURAL / MANMADE HAZARDS AND EC JUDGEMENT				
CATEGORY	UE	ALERT	SAE	GE
Earthquake Experienced	2.1	2.2		
External Flooding	2.3	2.4		
Hurricane	2.5			
Tornado/High Winds	2.6	2.7		
Aircraft/Vehicle Crash	2.8	2.9		
Toxic or Flammable Gases	2.10	2.11		
Explosions/Catastrophic Pressurized Equipment Failure	2.12	2.13		
Fire	2.14	2.15		
Control Room Evacuation		2.16	2.17	
Security Event	2.18	2.19	2.20	2.21
Internal Flooding	2.22	2.23		
Emergency Coordinator Judgment	2.24	2.25	2.26	2.27

SYSTEM MALFUNCTION				
CATEGORY	UE	ALERT	SAE	GE
Loss of Communications	3.1			
Failure of Reactor Protection		3.2	3.3	3.4
Inability to Reach ITS Time Limits	3.5			
Loss of Alarms/Indications	3.6	3.7	3.8	
Fuel Clad Degradation	3.9			
Turbine Failure	3.10	3.11		
RCS Leakage	3.12			
Inability to Maintain Hot Shutdown			3.13	
Inadvertent Criticality	3.14			
Inability to Maintain Plant in Cold Shutdown		3.15		
Loss of Water Level in Reactor Vessel that has Uncovered or Will Uncover Fuel			3.16	

LOSS OF POWER				
CATEGORY	UE	ALERT	SAE	GE
Loss of AC Power	4.1	4.2	4.3	4.4
Loss of AC Power (Shutdown)		4.5		
Loss of Vital DC Power			4.6	
Loss of Vital DC Power (Shutdown)	4.7			

MODES: ALL = Modes 1-6 and Defueled/No Mode

EMERGENCY CLASSIFICATION TABLE
FISSION PRODUCT BARRIER MATRIX
Applicable Modes: 1 - 4 Complete For All Barriers

5.1 LOSS OF FUEL CLAD If any item is checked, barrier is lost. Enter 4 for FUEL CLAD in classification table below.		6.1 LOSS OF REACTOR COOLANT SYSTEM If any item is checked, barrier is lost Enter 4 for RCS in classification table below.		7.1 LOSS OF CONTAINMENT If any item is checked, barrier is lost. Enter 2 for CONTAINMENT in classification table below.	
1. CORE CONDITIONS IN REGION 3 OR SEVERE ACCIDENT REGION OF ICC CURVES		1. RCS LEAK OR OTSG TUBE LEAK RESULTING IN LOSS OF ADEQUATE SUBCOOLING MARGIN		1. RAPID UNEXPLAINED RB PRESSURE DECREASE FOLLOWING INITIAL INCREASE	
2. RCS ACTIVITY >300 µCi/gm I-131 DOSE EQUIVALENT [NOCS 100441]		2. RM-G29 OR 30 > 10 R/hr FOR 15 MINUTES OR LONGER		2. CONTAINMENT PRESSURE OR SUMP LEVEL RESPONSE NOT CONSISTENT WITH LOCA CONDITIONS	
3. RM-G29 OR 30 >100 R/hr FOR 15 MINUTES OR LONGER		3. EC DEEMS RCS BARRIER IS LOST		3. AN OTSG HAS > 10 GPM TUBE RUPTURE WITH PROLONGED STEAMING TO THE ATMOSPHERE FROM THE AFFECTED OTSG OR AN UNISOLABLE STEAM LEAK OUTSIDE RB FROM THE AFFECTED OTSG	
4. EC DEEMS FUEL CLAD BARRIER IS LOST				4. CONTAINMENT ISOLATION IS INCOMPLETE AND RELEASE PATH TO THE ENVIRONMENT EXISTS	
				5. EC DEEMS CONTAINMENT BARRIER IS LOST	
5.2 POTENTIAL LOSS OF FUEL CLAD If any item is checked, barrier is potentially lost. Enter 3 for FUEL CLAD in classification table below.		**6.2 POTENTIAL LOSS OF REACTOR COOLANT SYSTEM** If any item is checked, barrier is potentially lost. Enter 3 for RCS in classification table below.		**7.2 POTENTIAL LOSS OF CONTAINMENT** If any item is checked, barrier is potentially lost. Enter 1.5 for CONTAINMENT in classification table below.	
1. ENTRY INTO EOP-07 BY PROCEDURAL DIRECTION		1. RCS LEAK OR OTSG TUBE LEAK REQUIRING ONE OR MORE INJECTION VALVES		1. RB PRESSURE >54 psig	
2. CORE EXIT THERMOCOUPLES >700°F		2. RCS LEAK OR OTSG TUBE LEAK RESULTS IN ES ACTUATION ON LOW RCS PRESSURE		2. RB HYDROGEN CONCENTRATION >4%	
3. EC DEEMS FUEL CLAD BARRIER IN JEOPARDY		3. RCS PRESSURE/TEMPERATURE RELATIONSHIP VIOLATES NDT LIMITS		3. RB PRESSURE >30 psig WITH NO BUILDING SPRAY AVAILABLE	
		4. HPI/PORV OR HPI/SAFETY VALVE COOLING IS IN PROGRESS		4. RMG-29 OR 30 READINGS >5,000 R/hr	
		5. EC DEEMS RCS BARRIER IN JEOPARDY		5. CORE CONDITIONS IN SEVERE ACCIDENT REGION OF ICC CURVES FOR >15 MINUTES	
				6. EC DEEMS CONTAINMENT BARRIER IN JEOPARDY	

CLASSIFICATION TABLE

ENTER LOSS OR POTENTIAL LOSS OR ZERO FOR EACH BARRIER THEN TOTAL AND DETERMINE CLASS BELOW

FUEL CLAD ______ + RCS ______ + CONTAINMENT _____ = ______

IF TOTAL IS:	RECOMMENDED EVENT CLASSIFICATION IS:
> 0 BUT ≤ 2	UNUSUAL EVENT
> 2 BUT ≤ 4	ALERT
> 4 BUT ≤ 8.5	SITE AREA EMERGENCY
> 8.5	GENERAL EMERGENCY

EMERGENCY CLASSIFICATION TABLE
ACCIDENT CONDITION:
ABNORMAL RAD LEVELS / RADIOLOGICAL EFFLUENT

<table>
<tr><th>CATEGORY</th><th>UNUSUAL EVENT</th><th>ALERT</th><th>SITE AREA EMERGENCY</th><th>GENERAL EMERGENCY</th></tr>
<tr>
<td>Gaseous Effluents
MODES: ALL</td>
<td>1.1 MODES: ALL
(1 or 2)
1. A VALID reading on RM-A1 or RM-A2 Normal Range monitor exceeds the high alarm setpoint for 60 minutes or longer
<u>OR</u>
2. Sample analysis confirms gaseous effluent being released exceeds 5.0E-4 μCi/ for 60 minutes or longer</td>
<td>1.2 MODES: ALL
(1 or 2)
1. A VALID reading on RM-A1 or RM-A2 Accident Range monitor exceeds 5.0E-3 μCi/cc for 15 minutes or longer
<u>OR</u>
2. Sample analysis confirms gaseous effluent being released exceeds 5.0E-3 μCi/cc for 15 minutes or longer</td>
<td>1.3 MODES: ALL
(1 or 2 or 3)
1. VALID RM-A1 or RM-A2 Accident Range monitor reading exceeds the values on the following Table for the current Stability Class for 15 minutes or longer:
<table>
<tr><th>Stability Class</th><th>On-Line Operations (μCi/cc)</th><th>Extended Shutdown or SF Pools (μCi/cc)</th></tr>
<tr><td>A, B, or C</td><td>5.1E-1</td><td>7.5E-2</td></tr>
<tr><td>D or E</td><td>3.3E-1</td><td>5.4E-2</td></tr>
<tr><td>F or G</td><td>3.0E-1</td><td>4.5E-2</td></tr>
</table>
<u>OR</u>
2. Dose Assessment results indicate SITE BOUNDARY dose >100 mR TEDE or >500 mR thyroid CDE for the actual or projected duration of the release
<u>OR</u>
3. Field survey results indicate closed windows dose rates >100mR/hr expected to continue for more than one hour; or analyses of field survey samples indicate thyroid CDE of 500mR for one hour of inhalation, at or beyond SITE BOUNDARY</td>
<td>1.4 MODES: ALL
(1 or 2 or 3)
1. VALID RM-A1 or RM-A2 Accident Range monitor reading exceeds the values in the following table for 15 minutes or longer:
<table>
<tr><th>On-Line Operations (μCi/cc)</th><th>Extended Shutdown or SF Pools (μCi/cc)</th></tr>
<tr><td>3.0E+0</td><td>4.5E-1</td></tr>
</table>
<u>OR</u>
2. Dose Assessment results indicate SITE BOUNDARY dose >1000 mR TEDE or >5000 mR thyroid CDE for the actual or projected duration of the release AND core damage is suspected or has occurred
<u>OR</u>
3. Field survey results indicate closed windows dose rates >1000mR/hr expected to continue for more than one hour; or analyses of field survey samples indicate thyroid CDE of 5000 mR for one hour of inhalation, at or beyond SITE BOUNDARY</td>
</tr>
<tr>
<td>Liquid Effluents
MODES: ALL</td>
<td>1.5 MODES: ALL
(1 or 2)
1. A VALID reading on RM-L2, RM-L7, or sample analysis confirms the release exceeds 2 times the ODCM release setpoint for 60 minutes or longer
<u>OR</u>
2. Release continued for 60 minutes or longer with no dilution flow</td>
<td>1.6 MODES: ALL
A VALID reading on RM-L2, RM-L7, or sample analysis confirms the release exceeds 200 times the ODCM release setpoint for 15 minutes or longer</td>
<td>Not Applicable</td>
<td>Not Applicable</td>
</tr>
</table>

EMERGENCY CLASSIFICATION TABLE
ACCIDENT CONDITION:
ABNORMAL RAD LEVELS/RADIOLOGICAL EFFLUENT (Continued)

CATEGORY	UNUSUAL EVENT	ALERT	SITE AREA EMERGENCY	GENERAL EMERGENCY
Unexpected Radiation Levels MODES: ALL	1.7 MODES: ALL One or more VALID radiation monitor readings unexpectedly exceed the values below for 15 minutes or longer: RM-G3 = 400 mR/hr RM-G4 = 600 mR/hr RM-G5 = 3,000 mR/hr RM-G9 = 100 mR/hr RM-G10 = 800 mR/hr RM-G14 = 1,000 mR/hr RM-G17 = 800 mR/hr	1.8 MODES: ALL (1 or 2) 1. VALID radiation reading greater than 15 mR/hr for 15 minutes or longer in the Control Room (RM-G1) or the Central Alarm Station (CAS) OR 2. One or more VALID radiation monitor readings unexpectedly exceed the values below for 15 minutes or longer: RM-G3 = 5,000 mR/hr RM-G4 = 5,000 mR/hr RM-G9 = 5,000 mR/hr RM-G10 = 5,000 mR/hr RM-G17 = 5,000 mR/hr	*Refer to Fission Product Barrier Matrix, Gaseous Effluents, or Emergency Coordinator Judgment*	*Refer to Fission Product Barrier Matrix, Gaseous Effluents, or Emergency Coordinator Judgment*
Irradiated Fuel Damage Due to Mechanical Damage or Uncontrolled Loss of Water Level Outside the Reactor Vessel MODES: ALL	1.9 MODES: ALL (1 and 2) 1. (a or b) a. Uncontrolled level decrease resulting in indications of -2.5 feet in spent fuel pool OR b. Confirmed Plant personnel report of uncontrolled significant water level drop in spent fuel pool or transfer canal when Spent Fuel transfer tubes are open AND 2. Fuel remains covered with water	1.10 MODES: ALL (1 or 2) 1. (a and b) a. Plant personnel report damage of irradiated fuel AND b. VALID high alarm as indicated on RM-G15 or RM-G16 OR 2. Plant personnel report spent fuel pool or transfer canal water level drop has or will exceed makeup capacity such that irradiated fuel will be uncovered	*Refer to Gaseous Effluents or Emergency Coordinator Judgment*	*Refer to Gaseous Effluents or Emergency Coordinator Judgment*

EMERGENCY CLASSIFICATION TABLE
ACCIDENT CONDITION:
NATURAL / MANMADE HAZARDS AND EC JUDGEMENT

CATEGORY	UNUSUAL EVENT	ALERT	SITE AREA EMERGENCY	GENERAL EMERGENCY
Earthquake Experienced [NOCS 24320] **MODES: ALL**	2.1 **MODES: ALL** **(1 and 2)** 1. Ground motion sensed by Plant personnel **AND** 2. Confirmed earthquake causing Annunciator C-3-14 "Seismic System Trouble" alarm	2.2 **MODES: ALL** **(1 and 2)** 1. Ground motion sensed by Plant personnel or confirmed Annunciator C-3-14 "Seismic System Trouble" alarm **AND** 2. **(a or b)** a. Analysis confirms the earthquake at >0.05g **OR** b. Indications show degraded SAFE SHUTDOWN EQUIPMENT performance due to the earthquake	*Refer to Fission Product Barrier Matrix or Emergency Coordinator Judgment*	*Refer to Fission Product Barrier Matrix or Emergency Coordinator Judgment*
External Flooding **MODES: ALL**	2.3 **MODES: ALL** Intake canal level or visual observation indicates flood water level ≥ 98 feet	2.4 **MODES: ALL** **(1 and 2)** 1. Intake canal level or visual observation indicates flood water level ≥ 98 feet **AND** 2. Indications show degraded SAFE SHUTDOWN EQUIPMENT performance due to the flooding	*Refer to Fission Product Barrier Matrix or Emergency Coordinator Judgment*	*Refer to Fission Product Barrier Matrix or Emergency Coordinator Judgment*
Hurricane **MODES: ALL**	2.5 **MODES: ALL** The Plant is within a Hurricane Warning area	*Refer to Fission Product Barrier Matrix, Tornado/High Winds, or Emergency Coordinator Judgment*	*Refer to Fission Product Barrier Matrix or Emergency Coordinator Judgment*	*Refer to Fission Product Barrier Matrix or Emergency Coordinator Judgment*

EMERGENCY CLASSIFICATION TABLE
ACCIDENT CONDITION:
NATURAL / MANMADE HAZARDS AND EC JUDGEMENT (Continued)

CATEGORY	UNUSUAL EVENT	ALERT	SITE AREA EMERGENCY	GENERAL EMERGENCY
Tornado/High Winds MODES: ALL	**2.6 MODES: ALL** Report by Plant personnel of a Tornado striking within the PROTECTED AREA	**2.7 MODES: ALL** **(1 or 2)** 1. Tornado <u>or</u> High Winds <u>or</u> windborne object cause significant VISIBLE DAMAGE to any of the following structures: - Auxiliary Building, - BWST, - Control Complex, - Diesel Generator Building (EGDG-1A/1B) - EFT-2 Building, - Intermediate Building, - Reactor Building - EFP-3 Building <u>OR</u> 2. Indications show degraded SAFE SHUTDOWN EQUIPMENT performance due to the tornado or high winds or windborne objects	*Refer to Fission Product Barrier Matrix or Emergency Coordinator Judgment*	*Refer to Fission Product Barrier Matrix or Emergency Coordinator Judgment*
Accidental Aircraft / Vehicle Crash MODES: ALL	**2.8 MODES: ALL** Report by Plant personnel of Aircraft <u>or</u> Vehicle Crash involving the following structures: - Auxiliary Building, - BWST - Control Complex - Diesel Generator Building (EGDG-1A/1B) - EFT-2 Building - Intermediate Building - Reactor Building - EFP-3 Building	**2.9 MODES: ALL** **(1 or 2)** 1. Confirmed report of significant VISIBLE DAMAGE to any of the following structures: - Auxiliary Building - BWST - Control Complex - Diesel Generator Building (EGDG-1A/1B) - EFT-2 Building - Intermediate Building - Reactor Building - EFP-3 Building <u>OR</u> 2. Indications show degraded SAFE SHUTDOWN EQUIPMENT performance due to the Aircraft <u>or</u> Vehicle Crash	*Refer to Fission Product Barrier Matrix or Emergency Coordinator Judgment*	*Refer to Fission Product Barrier Matrix or Emergency Coordinator Judgment*

EMERGENCY CLASSIFICATION TABLE
ACCIDENT CONDITION:
NATURAL / MANMADE HAZARDS AND EC JUDGEMENT (Continued)

CATEGORY	UNUSUAL EVENT	ALERT	SITE AREA EMERGENCY	GENERAL EMERGENCY
Toxic or Flammable Gases MODES: ALL	2.10 MODES: ALL (1 or 2) 1. Report or detection of Toxic or Flammable Gas within the SITE BOUNDARY that could enter the Protected Area at levels > IDLH or > 25% Lower Explosive Limits affecting normal operation of the Plant. OR 2. Confirmed notification by PE, County, or State personnel to evacuate or shelter site personnel based on an offsite event	2.11 MODES: ALL (1 or 2 or 3) 1. Flammable Gas levels > 25% Lower Explosive Limit in areas required to maintain safe operations or establish and maintain cold shutdown OR 2. Toxic Gas levels ≥ IDLH levels in areas that require continuous occupancy to maintain safe operation or establish or maintain cold shutdown OR 3. Toxic Gas levels ≥ IDLH levels within the PROTECTED AREA such that Plant personnel are unable to perform actions necessary to maintain safe operations or establish and maintain cold shutdown using protective equipment	*Refer to Fission Product Barrier Matrix, System Malfunction, or Emergency Coordinator Judgment*	*Refer to Fission Product Barrier Matrix, System Malfunction, or Emergency Coordinator Judgment*

EMERGENCY CLASSIFICATION TABLE

ACCIDENT CONDITION:

NATURAL / MANMADE HAZARDS AND EC JUDGEMENT (Continued)

CATEGORY	UNUSUAL EVENT	ALERT	SITE AREA EMERGENCY	GENERAL EMERGENCY
Explosions/ Catastrophic Pressurized Equipment Failure **MODES: ALL**	**2.12 MODES: ALL** Report by Plant personnel of VISIBLE DAMAGE to permanent structures or equipment within the PROTECTED AREA due to an EXPLOSION or catastrophic failure of pressurized equipment *Refer to Security Event*	**2.13 MODES: ALL** **(1 or 2)** 1. EXPLOSION or catastrophic failure of pressurized equipment causes significant VISIBLE DAMAGE to any of the following structures: - Auxiliary Building - BWST - Control Complex - Diesel Generator Building (EGDG-1A/1B) - EFT-2 Building, - Intermediate Building - Reactor Building - EFP-3 Building **OR** 2. Indications show degraded SAFE SHUTDOWN EQUIPMENT performance due to the EXPLOSION or pressurized equipment failure	*Refer to Fission Product Barrier Matrix, System Malfunction, or Emergency Coordinator Judgment*	*Refer to Fission Product Barrier Matrix, System Malfunction, or Emergency Coordinator Judgment*
Fire **MODES: ALL**	**2.14 MODES: ALL** **(1 and 2)** 1. FIRE in or threatening one of the following structures: - Auxiliary Building - BWST - Control Complex, - Diesel Generator Building (EGDG-1A/1B) - EFT-2 Building - Intermediate Building - Reactor Building - EFP-3 Building **AND** 2. FIRE not extinguished within 15 minutes from either Control Room notification or receipt of a VALID fire alarm in the Control Room	**2.15 MODES: ALL** **(1 or 2)** 1. Report by Plant personnel of VISIBLE DAMAGE to SAFE SHUTDOWN EQUIPMENT due to the FIRE **OR** 2. Indications show degraded SAFE SHUTDOWN EQUIPMENT performance due to the FIRE	*Refer to Fission Product Barrier Matrix, Control Room Evacuation, System Malfunctions, or Emergency Coordinator Judgment*	*Refer to Fission Product Barrier Matrix, Control Room Evacuation, System Malfunctions, or Emergency Coordinator Judgment*

EMERGENCY CLASSIFICATION TABLE

ACCIDENT CONDITION:

NATURAL / MANMADE HAZARDS AND EC JUDGEMENT (Continued)

CATEGORY	UNUSUAL EVENT	ALERT	SITE AREA EMERGENCY	GENERAL EMERGENCY
Control Room Evacuation MODES: ALL	*Not Applicable*	**2.16 MODES: ALL** Control Room evacuation is required per AP-990, "Shutdown From Outside the Control Room"	**2.17 MODES: ALL** **(1 and 2)** **1.** Control Room evacuation is required per AP-990, "Shutdown From Outside the Control Room" **AND** **2.** Control of the necessary equipment not established per AP-990 within 15 minutes	*Refer to Fission Product Barrier Matrix, System Malfunction, or Emergency Coordinator Judgment*
Security Event MODES: ALL	**2.18 MODES: ALL** **(1 or 2 or 3)** Report by Security Shift Supervisor or NRC of one or more of the following events: **1.** A validated notification from NRC providing information of an AIRCRAFT or AIRLINER threat. **OR** 2. A CREDIBLE SITE-SPECIFIC SECURITY THREAT NOTIFICATION **OR** **3.** A SECURITY CONDITION that does NOT involve a HOSTILE ACTION as reported by the Security Shift Supervisor.	**2.19 MODES: ALL** **(1 or 2)** **1.** A validated notification from NRC of an AIRLINER attack threat less than 30 minutes away **OR** **2.** A HOSTILE ACTION is occurring or has occurred within the OWNER CONTROLLED AREA as reported by the Security Shift Supervisor.	**2.20 MODES: ALL** A HOSTILE ACTION is occurring or has occurred within the PROTECTED AREA as reported by the Security Shift Supervisor.	**2.21 MODES: ALL** **(1 or 2)** 1. A HOSTILE ACTION has occurred such that plant personnel are unable to operate equipment required to maintain safety functions. **OR** 2. A HOSTILE ACTION has caused failure of Spent Fuel Cooling Systems and imminent fuel damage is likely.

EMERGENCY CLASSIFICATION TABLE

ACCIDENT CONDITION:

NATURAL / MANMADE HAZARDS AND EC JUDGEMENT (Continued)

CATEGORY	UNUSUAL EVENT	ALERT	SITE AREA EMERGENCY	GENERAL EMERGENCY
Internal Flooding **Modes: ALL**	2.22 **MODES: ALL** **(1 and 2)** **1.** Indication of uncontrolled flooding in the Auxiliary Building or Intermediate Building **AND** **2.** Water level/flooding has the potential to affect or immerse SAFE SHUTDOWN EQUIPMENT	2.23 **MODES: ALL** **(1 and 2)** **1.** Water level exceeds 5 inches in the Auxiliary Building or Intermediate Building **AND** **2.** **(a or b)** **a.** Indications show degraded SAFE SHUTDOWN EQUIPMENT due to the flooding **OR** **b.** Electrical hazards prevent Plant personnel normal access to areas of Plant containing SAFE SHUTDOWN EQUIPMENT	*Refer to Fission Product Barrier Matrix or Emergency Coordinator Judgment*	*Refer to Fission Product Barrier Matrix or Emergency Coordinator Judgment*

EMERGENCY CLASSIFICATION TABLE
ACCIDENT CONDITION:
NATURAL / MANMADE HAZARDS AND EC JUDGEMENT (Continued)

CATEGORY	UNUSUAL EVENT	ALERT	SITE AREA EMERGENCY	GENERAL EMERGENCY
Emergency Coordinator Judgment MODES: ALL	2.24 MODES: ALL Other conditions exist which indicate a potential degradation of the level of safety of the Plant	2.25 MODES: ALL Other conditions exist which indicate that events are in process or have occurred which involve potential or actual substantial degradation of the level of safety of the Plant	2.26 MODES: ALL Other conditions exist which indicate actual or likely major failures of Plant functions needed for the protection of the public	2.27 MODES: ALL **(1 or 2)** Other conditions exist which indicate: **1.** Actual or imminent substantial core degradation with potential loss of containment integrity OR **2.** The potential for uncontrolled radionuclide releases that can be expected to exceed EPA Protective Action Guidelines Plume Exposure Levels beyond the SITE BOUNDARY (see EAL 1.4)

EMERGENCY CLASSIFICATION TABLE
ACCIDENT CONDITION:
SYSTEM MALFUNCTION

CATEGORY	UNUSUAL EVENT	ALERT	SITE AREA EMERGENCY	GENERAL EMERGENCY
Loss of Communication MODES: ALL	3.1 **MODES: ALL** **(1 or 2)** **1.** Loss of <u>all</u> the following in-Plant communications capability: **a.** PE Internal Telephone System **b.** PAX **c.** Portable UHF Radios **<u>OR</u>** **2.** Loss of <u>all</u> of the following Offsite Communication capability: **a.** PE Telephone System **b.** State Hot Ringdown (SHRD) **c.** All FTS 2001 NRC phones (ENS, HPN, etc.) **d.** Emergency Management Network (EMnet) **e.** Cellular Telephones	*Not Applicable*	*Not Applicable*	*Not Applicable*
Failure of Reactor Protection **MODES: 1,2,3 for ALERT** **MODES: 1,2 for SITE AREA and GENERAL Emergencies**	***Not Applicable***	3.2 **MODES: 1,2,3** **(1 and 2)** **1.** RPS Trip setpoint exceeded and no Reactor trip occurred **<u>AND</u>** **2.** Manual Reactor trip from Control Room was successful and reactor is shutdown	3.3 **MODES: 1,2** **(1 and 2)** **1.** RPS Trip setpoint exceeded and no Reactor trip occurred **<u>AND</u>** **2.** Manual Reactor trip from Control Room was <u>not</u> successful in shutting down the reactor	3.4 **MODES: 1,2** **(1 and 2 and 3)** **1.** RPS Trip setpoint exceeded and no Reactor trip occurred **<u>AND</u>** **2.** Manual Reactor trip from Control Room was <u>not</u> successful in shutting down the reactor **<u>AND</u>** **3.** **(a or b)** **a.** Core exit thermocouple temperatures > 700°F, as indicated on SPDS. **<u>OR</u>** **b.** Adequate Secondary Cooling not available

EMERGENCY CLASSIFICATION TABLE
ACCIDENT CONDITION:
SYSTEM MALFUNCTION (Continued))

CATEGORY	UNUSUAL EVENT	ALERT	SITE AREA EMERGENCY	GENERAL EMERGENCY
Inability to reach required mode within Improved Technical Specification time limits MODES: 1,2,3,4	3.5 **MODES: 1,2,3,4** **(1 and 2)** 1. Entry into an Improved Technical Specification LCO statement requiring a mode reduction **AND** 2. The Plant is not in the required operating mode within the time prescribed by the LCO required action	*Not Applicable*	*Not Applicable*	*Not Applicable*
Loss of Alarms/Indications MODES: 1,2,3,4	3.6 **MODES: 1,2,3,4** **(1 or 2)** 1. UNPLANNED loss of Annunciator panels A-L and Annunciator printer for 15 minutes or longer **OR** 2. UNPLANNED loss of NNI-X and NNI-Y for 15 minutes or longer	3.7 **MODES: 1,2,3,4** **(1 and 2)** 1. **(a or b)** a. UNPLANNED loss of Annunciator panels A-L and Annunciator printer for 15 minutes or longer **OR** b. UNPLANNED loss of NNI-X and NNI-Y for 15 minutes or longer **AND** 2. **(a or b)** a. SIGNIFICANT TRANSIENT in progress **OR** b. Loss of Plant Computer and SPDS	3.8 **MODES: 1,2,3,4** **(1 and 2 and 3 and 4)** 1. **(a or b)** a. Loss of Annunciator panels A-L and Annunciator printer for 15 minutes or longer **OR** b. Loss of NNI-X and NNI-Y for 15 minutes or longer **AND** 2. SIGNIFICANT TRANSIENT in progress **AND** 3. Loss of Plant Computer and SPDS **AND** 4. Inability to directly monitor any one of the following: Subcriticality Core Cooling Containment RCS Inventory	*Refer to Fission Product Barrier Matrix or Emergency Coordinator Judgment*

EMERGENCY CLASSIFICATION TABLE

ACCIDENT CONDITION:

SYSTEM MALFUNCTION (Continued)

CATEGORY	UNUSUAL EVENT	ALERT	SITE AREA EMERGENCY	GENERAL EMERGENCY
Fuel Clad Degradation **MODES: 1,2,3,4,5**	**3.9 MODES: 1,2,3,4,5** **(a or b)** Radiochemistry analysis indicates: **a.** Dose Equivalent Iodine (I-131) > 1.0 μCi/gm for 48 hours or longer OR **b.** Specific activity >100/E-bar for 48 hours or longer	*Refer to Fission Product Barrier Matrix*	*Refer to Fission Product Barrier Matrix*	*Refer to Fission Product Barrier Matrix*
Turbine Failure **MODES: 1,2,3**	**3.10 MODES: 1,2,3** Report by Plant personnel of main turbine failure causing penetration of the turbine casing or damage to main generator seals	**3.11 MODES: 1,2,3** **(1 or 2)** **1.** Report by Plant personnel of projectiles generated by a main turbine failure causing significant VISIBLE DAMAGE any of the following structures: - Auxiliary Building - BWST - Control Complex - Diesel Generator Building (EGDG-1A/1B) - EFT-2 Building - Intermediate Building - Reactor Building - EFP-3 Building OR **2.** Indications show degraded SAFE SHUTDOWN EQUIPMENT performance due to turbine generated projectiles	*Refer to Fission Product Barrier Matrix*	*Refer to Fission Product Barrier Matrix*

EMERGENCY CLASSIFICATION TABLE
ACCIDENT CONDITION:
SYSTEM MALFUNCTION (Continued)

CATEGORY	UNUSUAL EVENT	ALERT	SITE AREA EMERGENCY	GENERAL EMERGENCY
RCS Leakage [NOCS 40503] **MODES: 1,2,3,4**	3.12 **MODES: 1,2,3,4** **(1 or 2)** 1. Unidentified Leakage ≥ 10 gpm <u>or</u> Pressure Boundary Leakage ≥ 10 gpm **<u>OR</u>** 2. Identified leakage ≥ 25 gpm	*Refer to Fission Product Barrier Matrix or Emergency Coordinator Judgment*	*Refer to Fission Product Barrier Matrix or Emergency Coordinator Judgment*	*Refer to Fission Product Barrier Matrix or Emergency Coordinator Judgment*
Inability to Maintain Hot Shutdown **MODES: 1,2,3,4**	*Not Applicable*	*Not Applicable*	3.13 **MODES: 1,2,3,4** **(1 and 2)** 1. Complete loss of Main, Emergency, and Auxiliary Feedwater and unable to establish HPI cooling **<u>AND</u>** 2. Loss of subcooling margin	*Refer to Fission Product Barrier Matrix or Emergency Coordinator Judgment*
Inadvertent Criticality **MODES: 2,3,4,5,6**	3.14 **MODES: 2,3,4,5,6** An extended and unplanned sustained positive startup rate monitored by nuclear instrumentation	*Not Applicable*	*Not Applicable*	*Not Applicable*
Inability to Maintain Plant in Cold Shutdown **MODES: 5,6**	*Not Applicable*	3.15 **MODES: 5,6** **(1 or 2)** 1. Inability to maintain reactor coolant temperature below 200°F **<u>OR</u>** 2. Uncontrolled reactor coolant temperature approaching 200°F	*Refer to Loss of Water in Reactor Vessel that has uncovered or will uncover fuel*	*Not Applicable*

EMERGENCY CLASSIFICATION TABLE
ACCIDENT CONDITION:
SYSTEM MALFUNCTION (Continued)

CATEGORY	UNUSUAL EVENT	ALERT	SITE AREA EMERGENCY	GENERAL EMERGENCY
Loss of Water Level in Reactor Vessel that Has Uncovered or Will Uncover Fuel **MODES: 5, 6**	*Not Applicable*	*Not Applicable*	**3.16 MODES 5,6** **(1 and 2)** **1.** Loss of decay heat removal per AP-404 **AND** **2.** (a or b) **a.** Incores indicating superheated conditions **OR** **b.** Incores unavailable and time to uncovery exceeded as specified in OP-103H	*Not Applicable*

EMERGENCY CLASSIFICATION TABLE

ACCIDENT CONDITION:

LOSS OF POWER

CATEGORY	UNUSUAL EVENT	ALERT	SITE AREA EMERGENCY	GENERAL EMERGENCY
Loss of AC Power **MODES: ALL for UNUSUAL EVENT** **MODES: 1,2,3,4 for ALERT, SITE AREA and GENERAL Emergencies**	**4.1 MODES: ALL** **(1 and 2)** **1.** Offsite Power Transformer (OPT) <u>and</u> Backup ES Transformer (BEST) <u>and</u> Auxiliary Transformer not available for 15 minutes or longer **<u>AND</u>** **2.** EDGs supplying power to 4160V ES Busses	**4.2 MODES: 1,2,3,4** AC power capability to the 4160V ES busses reduced to a single power source for 15 minutes or longer such that **only** one of the following is available: - "A" EDG - "B" EDG - Offsite Power Transformer (OPT) - Backup ES Transformer (BEST)	**4.3 MODES: 1,2,3,4** Neither 4160V ES bus is capable of being energized within 15 minutes	**4.4 MODES: 1,2,3,4** **(1 and 2)** **1.** Neither 4160V ES bus is capable of being energized **<u>AND</u>** **2. (a or b)** **a.** Restoration of 4160V ES Bus A <u>or</u> 4160V ES Bus B is not likely within 4 hours **<u>OR</u>** **b.** Core exit thermocouples > 700°F as indicated on SPDS
Loss of AC Power (Shutdown) **MODES: 5,6, No Mode (defueled)**	*Not Applicable*	**4.5 MODES: 5,6, No Mode** Neither 4160V ES bus is capable of being energized within 15 minutes	*Not Applicable*	*Not Applicable*
Loss of Vital DC Power **MODES: 1,2,3,4**	*Not Applicable*	*Not Applicable*	**4.6 MODES: 1,2,3,4** Standby Power Status Lights for BUS A1, A2, and BUS B1, B2 on the Main Control Board (SSF Panel) are out for 15 minutes or longer	*Refer to Fission Product Barrier Matrix*
Loss of Vital DC Power (Shutdown) **MODES: 5,6, No Mode (defueled)**	**4.7 MODES: 5,6, No Mode** Standby Power Status Lights for BUS A1, A2, and BUS B1, B2 on the Main Control Board (SSF Panel) are out for 15 minutes or longer	*Not Applicable*	*Not Applicable*	*Not Applicable*

EVACUATION PLANNING GUIDE

1.0 ENERGY COMPLEX PROTECTIVE ACTIONS

1. DETERMINE protective actions for the Energy Complex using B or C or D or E below. (USE information in the tables and map on the following pages of this enclosure as necessary.)

   A. UNUSUAL EVENT **OR** ALERT: **NO** protective actions. (Actions may be required in some security events as determined by Security.)

   B. SITE AREA EMERGENCY:

   - PERFORM assembly and accountability **AND** INSTRUCT Fossil Control Rooms to report results to Nuclear Security at extension 3258 or 795-5078.
   - CONSIDER discretionary evacuation of non-essential personnel if plant conditions are likely to degrade or conditions exist that could impede site evacuation.
   - CONSIDER sheltering for releases lasting less than two hours.
   - For releases lasting greater than two hours or for planned releases, EVACUATE non-essential personnel

   C. GENERAL EMERGENCY:
   (Release has **NOT** occurred and release **NOT** likely within 3 hours.)

   - PERFORM assembly and accountability **AND** INSTRUCT Fossil Control Rooms to report results to Nuclear Security at extension 3258 or 795-5078.
   - EVACUATE non-essential personnel (including Main Assembly Area personnel).
   - NOTIFY Fossil Control Rooms to standby for instructions.
   - CONSIDER supplying dosimetry to remaining personnel.

   D. GENERAL EMERGENCY:
   (Release has occurred or is imminent **AND** RELEASE duration projected less than 2 hours.)

   - NOTIFY Fossil Control Rooms to direct all personnel to take shelter in closest building and standby for further instructions.

EVACUATION PLANNING GUIDE (Cont'd)

1.0 **ENERGY COMPLEX PROTECTIVE ACTIONS** (Cont'd)

E. GENERAL EMERGENCY:
(Release has occurred or is likely within 3 hours AND release duration unknown.)

- NOTIFY Fossil Control Rooms to secure their Plants.
- EVACUATE the Energy Complex even if a release has already started (including Main Assembly Area personnel).
- EVACUATE without performing assembly.

2. NOTIFY Units 1/2 & 4/5 using Attachment 4, Emergency Notification for Units 1/2 & 4/5.
3. ENSURE Nuclear Security coordinates these protective action instructions to all areas of the Energy Complex, per the EC Guide.

EVACUATION PLANNING GUIDE (Cont'd)

2.0 EVACUATION CONSIDERATIONS

1. **IF** evacuation is likely, **THEN** CONSIDER the following measures. UTILIZE Security, other CREC facilities, or local law enforcement agencies (LLEA) as needed.

> **NOTE:** Evacuation of non-essential CREC personnel can be accomplished in 90-165 minutes depending upon onsite population size and weather conditions. "Contra-flow" is the establishment of outbound traffic utilizing both lanes of the access road.

	CREC Worker Vehicles (Day Shift)	**Evacuation Times (Low) (good weather w/contra-flow)**	**Evacuation Times (High) (adverse weather w/o contra-flow)**
Normal operations (all units)	Up to ~850	~90 minutes	~110 minutes
Typical Unit 3 refuel outage	Up to ~1100	~110 minutes	~135 minutes
Concurrent CREC outages and/or major projects, etc.	Up to ~1600	~125 minutes	~165 minutes

- SUSPEND inbound traffic of non-essential personnel.
- SUSPEND inbound and outbound train traffic by calling (407) 880-8500
- SUSPEND barge traffic in the intake canal by calling (352) 302-2189.
- IMPLEMENT a staggered evacuation sequence of CREC facilities to reduce traffic congestion.
- OPEN the exit lane vehicle barrier at the Access Control Point.

> **CAUTION**
>
> Contra-flow should not be established if buses are used to evacuate personnel or if emergency vehicles are required onsite

- ESTABLISH contra-flow (outbound traffic utilizing both lanes of the access road) if practicable in the following order:
  - a) From the North Access Road east to US Highway 19
  - b) From the southeast corner of the main CR3 parking lot (3-way intersection) east to US Highway 19

EVACUATION PLANNING GUIDE (Cont'd)

2.0 **EVACUATION CONSIDERATIONS** (Cont'd)

- **IF** buses are in use but inadequate to evacuate contractor personnel, **THEN** IMPLEMENT car-pooling to transport them to the bus staging area.
- EVACUATE non-essential CR3 personnel directly from Local Assembly Areas, bypassing the Main Assembly Area. Essential personnel can report to the Main Assembly Area if deemed appropriate.
- ESTABLISH traffic control points at the following intersections (in order of priority):

  a) US Highway 19 and Power Line Road

  b) Power Line Road and the North Access Road

  c) Power Line Road and the southeast corner of the main CR3 parking lot (3-way intersection)

  d) Power Line Road and North Tallahassee Road

EVACUATION PLANNING GUIDE (Cont'd)

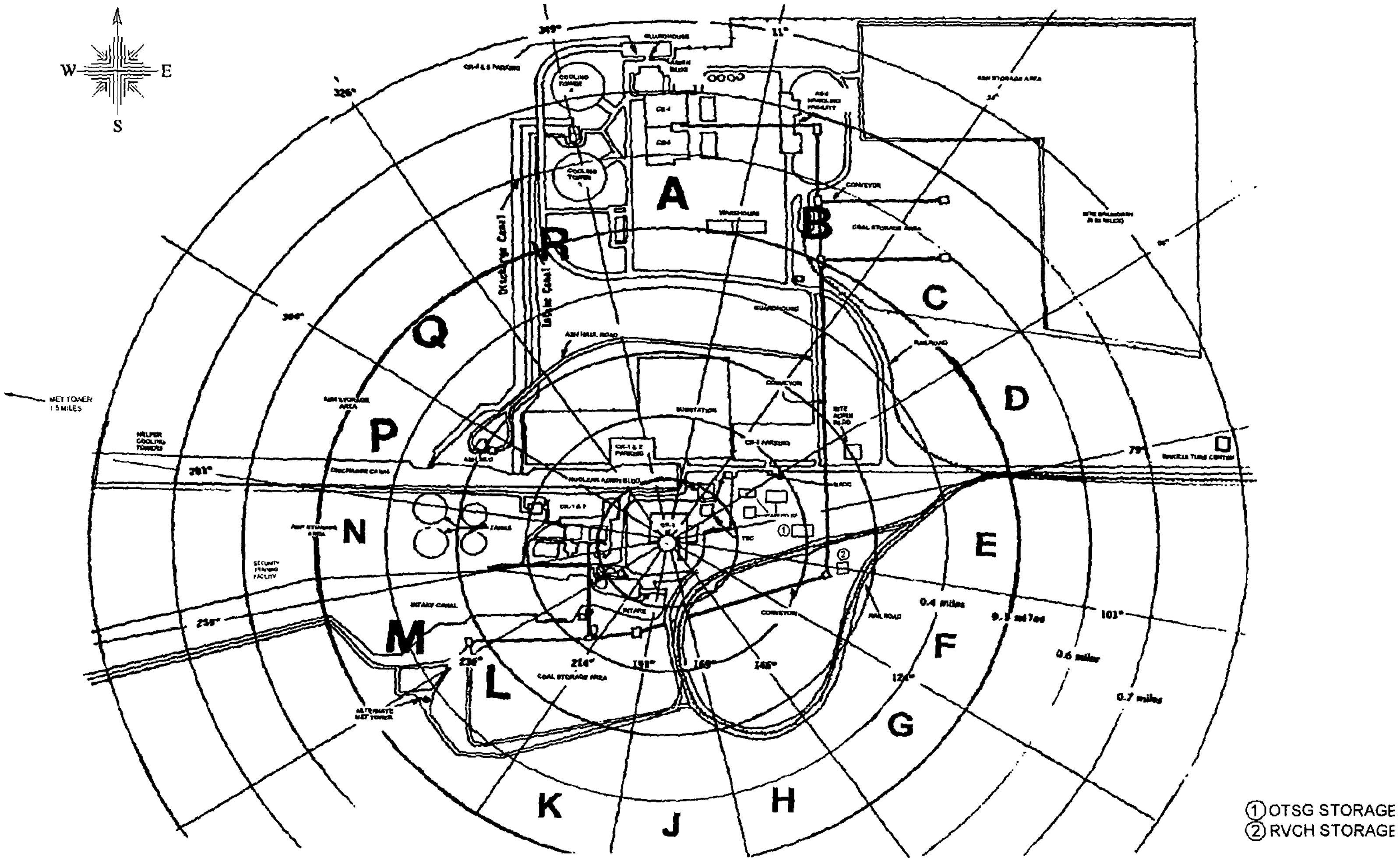
3.0 WIND DIRECTION DATA

WIND FROM DIRECTION	WIND FROM DEGREES	SECTORS AFFECTED
N	349-11 (349-371)	H J K
NNE	12-33 (372-393)	J K L
NE	34-56 (394-416)	K L M
ENE	57-78 (417-438)	L M N
E	79-101 (439-461)	M N P
ESE	102-123 (462-483)	N P Q
SE	124-146 (484-506)	P Q R
SSE	147-168 (507-528)	Q R A
S	169-191 (529-540)	R A B
SSW	192-213	A B C
SW	214-236	B C D
WSW	237-258	C D E
W	259-281	D E F
WNW	282-303	E F G
NW	304-326	F G H
NNW	327-348	G H J

EVACUATION PLANNING GUIDE (Cont'd)

4.0 CONTACTS FOR PERSONNEL ASSEMBLY

SECTOR	AREA	CONTACT
A	Units 4 & 5	Units 4 & 5 Control Room
B / C	Nuclear Administration Building	Public Address System
B / C	North Coal Yard	Units 4 & 5 Control Room
D / E	CR3 Warehouse Area Site Administration Building	Nuclear Security
D / E	Mariculture Center	Nuclear Security
E / F / G / H	Coal Train Yard	Units 4 & 5 Control Room
J / K / L	South Coal Yard	Units 1 & 2 Control Room
N	Units 1 & 2	Units 1 & 2 Control Room
N	Security Training Building	Nuclear Security


W
E
S
A
B
C
D
E
F
G
H
J
K
L
M
N
P
Q
R
Discharge Canal
Intake Canal
0.4 miles
0.6 miles
0.7 miles
①OTSG STORAGE
②RVCH STORAGE

GUIDELINES FOR PROTECTIVE ACTION RECOMMENDATIONS FOR NON-ESSENTIAL ENERGY COMPLEX PERSONNEL AND GENERAL POPULATION

[NOCS 1128, 1592]

<table>
<tr><th>PLANT CONDITIONS / OFFSITE DOSE ESTIMATES</th><th>RECOMMENDED ACTIONS</th></tr>
<tr><td>1. <u>CONDITION:</u>
GENERAL EMERGENCY DECLARED.
<u>NO</u> APPARENT CORE DAMAGE.

<u>CORE DAMAGE INDICATIONS:</u>
a. RCS pressure vs. temperature in Region 1 or 2 (REFER TO EOP-07); or
b. RM-G29/30 reading < 100 R/hr; or
c. RCS chemistry results.</td><td>Evacuate Zone 1
(See Notes 1 and 2)
<u>Note 1</u>:
Relocate/evacuate population in any zone affected by ground contamination after plume passage.

Relocate/evacuate population in Zones 2 & 3 at any time projected dose from actual release is ≥ 1.0 REM TEDE or 5.0 REM Thyroid CDE in either zone.

<u>Note 2</u>:
Sheltering should be recommended for the following conditions for those areas that cannot be evacuated before plume arrival:
• Core damage is in progress and,
• Containment failure or a controlled release is imminent and,
• The release duration is known to be less than 2 hours.
Known impediments to evacuation should also be considered in the decision to evacuate or shelter.

CONSIDER issuance of Potassium Iodide (KI).</td></tr>
<tr><td>2. <u>CONDITION:</u>
GENERAL EMERGENCY DECLARED.
CLAD DAMAGE/GAS GAP RELEASE (<u>NO</u> CORE MELT).

<u>CORE DAMAGE INDICATIONS:</u>
a. RCS pressure vs. temperature in Region 3 (REFER TO EOP-07); or
b. Core uncovered for 15-30 minutes; or
c. RM-G29/30 reading of 100-75,000 R/hr (RB spray off) OR 100-25,000 R/hr (RB spray on); or
d. RCS chemistry results.
<u>OR</u>:
* Dose at the 0.83 mile Site Boundary is projected to be:
a) TEDE: ≥ 1.0 Rem
b) Thyroid CDE: ≥ 5.0 Rem

* PARs within the first hour of an event should be based on PLANT CONDITIONS ONLY until the Dose Assessment Team is operational.</td><td>Evacuate Zone 1
(See Note 2)
<u>Note 2</u>:
Sheltering should be recommended for the following conditions for those areas that cannot be evacuated before plume arrival:
• Core damage is in progress and,
• Containment failure or a controlled release is imminent and,
• The release duration is known to be less than 2 hours.
Known impediments to evacuation should also be considered in the decision to evacuate or shelter.

Shelter Zones 2 & 3
(See Note 1)
<u>Note 1</u>:
Relocate/evacuate population in any zone affected by ground contamination after plume passage.

Relocate/evacuate population in Zones 2 & 3 at any time projected dose from actual release is ≥ 1.0 REM TEDE or 5.0 REM Thyroid CDE in either zone.

CONSIDER issuance of Potassium Iodide (KI).</td></tr>
</table>

(Continued on next page)

GUIDELINES FOR PROTECTIVE ACTION RECOMMENDATIONS FOR NON-ESSENTIAL ENERGY COMPLEX PERSONNEL AND GENERAL POPULATION
[NOCS 1128, 1592]

PLANT CONDITIONS / OFFSITE DOSE ESTIMATES	RECOMMENDED ACTIONS
3. **CONDITION:** **GENERAL EMERGENCY DECLARED.** **CORE MELT OCCURRING OR LIKELY.** CORE DAMAGE INDICATIONS: a. RCS pressure vs. temperature in the Severe Accident Region (REFER TO EOP-07); or b. Core uncovered for > 30 minutes; or c. RM-G29/30 reading > 75,000 R/hr (RB spray off) or > 25,000 R/hr (RB spray on). **WITH:** **NO** projected containment failure and **NO** release underway.	**Evacuate Zone 1** (See Note 2) Note 2: Sheltering should be recommended for the following conditions for those areas that cannot be evacuated before plume arrival: • Core damage is in progress and, • Containment failure or a controlled release is imminent and, • The release duration is known to be less than 2 hours. Known impediments to evacuation should also be considered in the decision to evacuate or shelter. **Shelter Zones 2 & 3** (See Note 1) Note 1: Relocate/evacuate population in any zone affected by ground contamination after plume passage. Relocate/evacuate population in Zones 2 & 3 at any time projected dose from actual release is ≥ 1.0 REM TEDE or 5.0 REM Thyroid CDE in either zone. **CONSIDER Issuance of Potassium Iodide (KI).**
4. **CONDITION:** **GENERAL EMERGENCY DECLARED.** **CORE MELT OCCURRING OR LIKELY.** CORE DAMAGE INDICATIONS: a. RCS pressure vs. temperature in the Severe Accident Region (REFER TO EOP-07); or b. Core uncovered for > 30 minutes; or c. RM-G29/30 reading > 75,000 R/hr (RB spray off) or > 25,000 R/hr (RB spray on). **WITH:** Projected containment failure and/or release underway.	**Evacuate Zones 1 and 2 and 3** (See Notes 2 and 3) Note 2: Sheltering should be recommended for the following conditions for those areas that cannot be evacuated before plume arrival: • Core damage is in progress and, • Containment failure or a controlled release is imminent and, • The release duration is known to be less than 2 hours. Known impediments to evacuation should also be considered in the decision to evacuate or shelter. Note 3: IF projected dose from an actual release is >1.0 REM TEDE or 5.0 REM Thyroid beyond 10 miles, THEN RECOMMEND evacuation to State and Local government by distance in miles, OR by subdivision and geographic boundaries. **CONSIDER Issuance of Potassium Iodide (KI).**

ZONE DESCRIPTIONS	EVACUATION TIME ESTIMATES: Indicates the greatest amount of time to clear 95% of the affected population, but DOES NOT include notification or preparation time for evacuees. (Per 2007 ETE.)
Zone 1: 0-5 miles 360 degrees and out to 10 miles in Gulf	Zone 1: 2 hours 40 minutes (CREC: 90 – 165 minutes (Per 2008 CREC ETE.))
Zones 2 / 3: 5-10 miles in Citrus & Levy Counties	Zones 1, 2, and 3: 3 hours 40 minutes

GUIDELINES FOR PROTECTIVE ACTION RECOMMENDATIONS FOR NON-ESSENTIAL ENERGY COMPLEX PERSONNEL AND GENERAL POPULATION

GUIDELINES FOR PE EMERGENCY WORKER EXPOSURE

CONDITION	DOSE LIMIT (REM TEDE)	GUIDANCE
1. Emergency conditions **NOT** requiring actions to prevent serious injury or protect valuable property.	5	Emergency worker exposure should **NOT** exceed 5 REM TEDE.
2. Emergency conditions requiring actions to prevent serious injury or protect valuable property.	10	Exposure greater than 5 REM TEDE should receive approval of the Emergency Coordinator. Appropriate controls for emergency workers include time limitations and respirators.
3. Emergency conditions requiring lifesaving actions or actions to protect large populations.	25	Exposure greater than 5 REM TEDE should receive approval of the Emergency Coordinator. Appropriate controls for emergency workers include time limitations, respirators, and thyroid blocking.
4. Emergency conditions requiring lifesaving actions or actions to protect large populations.	> 25	Exposure greater than 5 REM TEDE receive approval of the Emergency Coordinator. Exposure at this level should be to volunteers who are healthy, above the age of 45, have an understanding of the health risks involved, and, preferably, be those whose normal duties have trained them for such missions. Appropriate controls for emergency workers include time limitations, respirators, and thyroid blocking.

NOTE: Reference for this table is Table 2.2 in the Manual of Protective Action Guides and Protective Actions for Nuclear Incidents (EPA 400-R/92-001).

NOTE: The dose limits listed above are in addition to any annual occupational dose already received.

EAL DESCRIPTIONS FOR FLORIDA NUCLEAR PLANT EMERGENCY NOTIFICATION FORM

ABNORMAL RADIATION LEVELS / RADIOLOGICAL EFFLUENTS	1.1	Release of *gaseous* radioactivity exceeds the Unusual Event threshold
	1.2	Release of *gaseous* radioactivity exceeds the Alert threshold
	1.3	Site boundary dose from airborne radioactivity > 100 mREM total dose or 500 mREM thyroid dose
	1.4	Site boundary dose from airborne radioactivity > 1000 mREM total dose or 5000 mREM thyroid dose
	1.5	Release of *liquid* radioactivity exceeds the Unusual Event threshold
	1.6	Release of *liquid* radioactivity exceeds the Alert threshold
	1.7	Unexpected increase in radiation levels within the Plant **NOT** impeding necessary access to Plant systems
	1.8	Unexpected increase in radiation levels within the Plant impeding necessary access to Plant systems
	1.9	An uncontrolled water level decrease in spent fuel pool **or** fuel transfer canal with fuel remaining covered
	1.10	Damage to irradiated fuel or loss of water level resulting in uncovering irradiated fuel outside the reactor vessel
NATURAL / MAN-MADE HAZARDS AND EC JUDGMENT	2.1	Earthquake detected by seismic instrumentation and sensed by Control Room personnel
	2.2	Earthquake at a magnitude greater than the limit for continued Plant operation
	2.3	Flooding due to natural phenomena NOT affecting Plant vital equipment
	2.4	Flooding due to natural phenomena affecting Plant vital equipment
	2.5	The Plant is within a Hurricane Warning area
	2.6	Tornado within the Protected Area
	2.7	Tornado or High Winds or windborne object(s) strike within Protected Area and results in significant damage to structures or equipment
	2.8	Accidental Aircraft or vehicle crash within the Protected Area damaging vital structures or equipment
	2.9	Accidental Aircraft or vehicle strikes Plant and results in significant damage to structures or equipment
	2.10	Toxic or flammable gases within or potentially affecting the Protected Area
	2.11	Toxic or flammable gases within the Plant affecting the safe operation of the Plant or the ability to shutdown the Plant
	2.12	Explosion or catastrophic failure of pressurized equipment within the Protected Area
	2.13	Explosion or catastrophic failure of pressurized equipment resulting in damage to vital structures or equipment
	2.14	Fire within the Protected Area that could affect Plant vital equipment
	2.15	Fire affecting the operability of Plant vital equipment
	2.16	Evacuation of Control Room is required and Plant control is established
	2.17	Evacuation of Control Room is required and Plant control CANNOT be established
	2.18	Security Event which indicates a potential degradation in the level of safety of the Plant
	2.19	Security Event in the Owner Controlled Area
	2.20	Security Event in the Protected Area
	2.21	Security Event resulting in loss of physical control of the facility to intruders.
	2.22	Internal flooding affecting areas containing Plant vital equipment
	2.23	Internal flooding affecting Plant vital equipment
	2.24	Conditions exist indicating a potential degradation of the level of safety of the Plant
	2.25	Conditions exist indicating potential or actual substantial degradation of the level of safety of the Plant
	2.26	Conditions exist indicating actual or likely major failures of Plant functions needed for the protection of the public
	2.27	Actual or imminent substantial core degradation with potential loss of containment integrity

EAL DESCRIPTIONS FOR FLORIDA NUCLEAR PLANT EMERGENCY NOTIFICATION FORM (Continued)

SYSTEM MALFUNCTION	3.1	Unplanned loss of all in-Plant or all offsite communication capability
	3.2	Failure of instrumentation to complete an automatic reactor shutdown when required **and** manual reactor shutdown was successful
	3.3	Failure of instrumentation to complete an automatic reactor shutdown when required **and** manual reactor shutdown was **NOT** successful
	3.4	Failure to complete an automatic reactor shutdown **and** manual reactor shutdown was **NOT** successful with indications of an extreme challenge of the ability to cool the Reactor core
	3.5	Inability to shutdown the Plant to comply with Technical Specification limits
	3.6	Unplanned loss of Control Room alarms
	3.7	Unplanned loss of Control Room alarms with a significant Plant status change in progress
	3.8	Inability to monitor a significant Plant status change in progress
	3.9	Chemistry sample indicates fuel clad degradation
	3.10	Turbine failure results in casing penetration or damage to main generator seals
	3.11	Turbine failure generated projectiles cause significant damage to Plant structures or vital equipment
	3.12	Reactor Coolant System leakage
	3.13	Complete loss of core heat removal capability
	3.14	Inadvertent Plant startup
	3.15	Complete loss of core cooling functions during refueling and cold shutdown conditions
	3.16	Loss of water level in the reactor vessel resulting in uncovering fuel
LOSS OF POWER	4.1	Loss of Plant electrical power from all offsite sources
	4.2	AC power capability reduced to a single source
	4.3	Loss of all AC power
	4.4	Loss of all AC power for greater than 4 hours
	4.5	Loss of all AC power during Cold Shutdown or Refueling conditions
	4.6	Loss of all vital Plant batteries during *operational* conditions
	4.7	Loss of all vital Plant batteries during *shutdown* conditions
FISSION PRODUCT BARRIERS	5.1	Loss of Fuel Clad
	5.2	Potential Loss of Fuel Clad
	6.1	Loss of Reactor Coolant System
	6.2	Potential Loss of Reactor Coolant System
	7.1	Loss of Containment
	7.2	Potential Loss of Containment

EMERGENCY RESPONSE FACILITY ACTIVATION SCENARIOS

[NOCS 100521, 100533]

Scenario No.	Scenario Title	Applicability
1	Notification Error	Retraction of any activation message sent in error
2	Unusual Event – ERO Standby	Unusual Event declared. Notify ERO to assume a heightened state of awareness in anticipation of emergency escalation.
3	Discretionary – TSC/OSC	At the discretion of the Emergency Coordinator, activate the following facilities: • Technical Support Center • Operational Support Center
4	Discretionary – TSC/OSC/EOF/ENC	At the discretion of the Emergency Coordinator, activate the following facilities: • Technical Support Center • Operational Support Center • Emergency Operations Facility • Emergency News Center
5	NOTE: Refer to Scenario 8 if activation of remote facilities is required. Alert	Alert declared. Activate the following facilities: • Technical Support Center • Operational Support Center • Emergency Operations Facility • Emergency News Center
6	NOTE: Refer to Scenario 9 if activation of remote facilities is required. Site Area Emergency	Site Area Emergency declared. Activate the following facilities: • Technical Support Center • Operational Support Center • Emergency Operations Facility • Emergency News Center

EMERGENCY RESPONSE FACILITY ACTIVATION SCENARIOS

[NOCS 100521, 100533]

Scenario No.	Scenario Title	Applicability
7	NOTE: Refer to Scenario 10 if activation of remote facilities is required. General Emergency	General Emergency declared. Activate the following facilities: • Technical Support Center • Operational Support Center • Emergency Operations Facility • Emergency News Center
8	Alert (Remote Facilities)	Alert declared. Activate the following facilities: • Remote Technical Support Center • Remote Operational Support Center • Emergency Operations Facility • Emergency News Center
9	Site Area Emergency (Remote Facilities)	Site Area Emergency declared. Activate the following facilities: • Remote Technical Support Center • Remote Operational Support Center • Emergency Operations Facility • Emergency News Center
10	General Emergency (Remote Facilities)	General Emergency declared. Activate the following facilities: • Remote Technical Support Center • Remote Operational Support Center • Emergency Operations Facility • Emergency News Center
11	Fire Brigade Support to SAB	Event requiring off-shift Fire Brigade support to report to the Site Admin Building. (Example: Large-area fire)
12	Fire Brigade Support to EOF	Event requiring off-shift Fire Brigade support to report to the Emergency Operations Facility. (Example: Large-area fire)
13	Event Termination	Plant conditions no longer require ERO to stand by or to report as determined by Emergency Coordinator or EOF Director.

COMMUNICATION WITH NRC MANAGEMENT DURING AN EVENT

Communication with the NRC Executive Team Director

During an incident that is serious enough to potentially require onsite or offsite protective actions, or that involves a significant security event, it is likely that the NRC Executive Team (ET) Director (NRC Chairman or designated Commissioner) will desire to speak periodically with the licensee's management representative.

The ET Director receives information from the NRC staff responding to the incident. However, the ET Director may wish to receive a periodic executive summary from the licensee's management representative before passing it on to other stakeholders such as other Federal agencies, Congress, or the White House. Generally, it is not necessary for the ET Director to be briefed on the detailed sequence of events, but rather on key issues for which the NRC may be able to provide assistance.

Some questions that the ET Director is likely to ask include:

- What are the licensee's current top priorities for the station?
- Are their significant uncertainties about any aspect of the event (e.g., is the situation improving or degrading)?
- Does the licensee need help from the NRC or other Federal agencies?
- Is the licensee having any communication or staffing problems?

The primary responsibilities of the licensee during an event is to mitigate the accident, secure the facility, classify the event, and make notification and protective action recommendations to State and local officials. Meeting those primary responsibilities takes precedence over discussions with ET Director regarding the event. If taking time to talk to the ET Director would interfere with those primary responsibilities, the NRC expects that the licensee's designated manager will direct a subordinate to take the call. If this is not feasible, the NRC will inform the licensee when the ET Director would subsequently like to speak with the licensee's designated representative.

Communication of Security-Related Information

The Security Bridge is placed on the same conferencing system that hosts other NRC communications bridges such as the Emergency Notification System and the Health Physics Network. During a security-related incident, the NRC Safeguards Team will continuously monitor the Security Bridge so that the licensee can readily re-establish communication for situational updates or for other important security-related communications. Following the initial discussions and evaluation, the Safeguards Team will coordinate periodic, scheduled update conversations so that licensee personnel can return to other essential duties between scheduled updates to the NRC.

The Security Bridge is recorded, but it is not a secure line and is not approved for routine discussions involving classified or Safeguards Information (SGI). The NRC Resident Inspector's secure telephone should be used for discussing and transmitting such information unless extraordinary conditions exist, such as an ongoing attack.

The type of information of interest to the Safeguards Team includes:

- Has the facility sustained significant damage (including the central and secondary alarm stations), damage to the physical security features or security force, or loss of licensed materials?
- What are the sources and status of offsite emergency assistance (e.g., local law enforcement, State, Federal (especially Federal Bureau of Investigation), National Guard)?
- Is additional Federal assistance required (e.g., personnel, material, communications)?
- What compensatory measures have been implemented (e.g., temporary barriers, relocation of responders)?

FLORIDA NUCLEAR PLANT EMERGENCY NOTIFICATION FORM

1. THIS IS CRYSTAL RIVER UNIT 3. A. ☐ This Is A Drill B. ☐ This Is An Emergency
   ENSURE: ☐ STATE ☐ CITRUS ☐ LEVY ☐ RADIATION CONTROL-ORLANDO (M-F ONLY) ARE ON LINE.
   I HAVE A(N) ☐ UNUSUAL EVENT ☐ ALERT ☐ SITE AREA EMERGENCY ☐ GENERAL EMERGENCY MESSAGE.

2. A. Date: ___/___/____ B. Contact Time: ________ C. Reported By: (Name) ____________________
   D. Message Number: ___________ E. Reported From: ☐ Control Room ☐ TSC ☐ EOF
   F. ☐ Initial / New Classification OR ☐ Update

3. SITE: A. ☒ CR UNIT 3 B. ☐ SL UNIT 1 C. ☐ SL UNIT 2 D. ☐ TP UNIT 3 E. ☐ TP UNIT 4

4. EMERGENCY CLASSIFICATION: A. ☐ Notification Of Unusual Event B. ☐ Alert
   C. ☐ Site Area Emergency D. ☐ General Emergency

5. A. ☐ EMERGENCY DECLARATION: B. ☐ EMERGENCY TERMINATION: Date: ____/____/____ Time: ________

6. REASON FOR EMERGENCY DECLARATION: A. ☐ EAL Number(s): ___________ OR B. ☐ Description:
   ____
   ____

7. ADDITIONAL INFORMATION OR UPDATE: A. ☐ None OR B. ☐ Description:
   ____
   ____

8. WEATHER DATA: A. Wind direction from ________ degrees B. Downwind Sectors affected ____________________

9. RELEASE STATUS: A. ☐ None (Go to Item 11) B. ☐ In Progress C. ☐ Has occurred, but stopped (Go to Item 11)

10. RELEASE SIGNIFICANCE CATEGORY: (at the Site Boundary)
    A. ☐ Under evaluation B. ☐ Release is within Normal Operating Limits
    C. ☐ Non-Significant (Fraction of PAG Range) D. ☐ Protective Action Guide range
    E. ☐ Liquid release (no actions required)

11. UTILITY RECOMMENDED PROTECTIVE ACTIONS FOR THE PUBLIC:
    A. ☐ No utility recommended actions at this time. B. ☐ Utility recommends the following protective actions:
    EVACUATE ZONES: ________________
    SHELTER ZONES: ________________
    AND consider issuance of Potassium Iodide (KI).

If form is completed in the Control Room, go to item 15. If completed in the TSC or EOF, CONTINUE with item 12.

12. PLANT CONDITIONS:
    A. Reactor Shutdown? ☐ YES ☐ NO B. Core Adequately Cooled? ☐ YES ☐ NO
    C. Containment Intact? ☐ YES ☐ NO D. Core Condition: ☐ Stable ☐ Degrading

13. WEATHER DATA: A. Wind Speed ________ MPH (m/sec x 2.24 = MPH) B. Stability Class _________

14. ADDITIONAL RELEASE INFORMATION: A. ☐ Not Applicable (Go to Item 15)

Distance	Projected Thyroid Dose (CDE) for _______ Hour(s)	Projected Total Dose (TEDE) for _______ Hour(s)
1 Mile (Site Boundary)	B.______________ mrem	C.______________ mrem
2 Miles	D.______________ mrem	E.______________ mrem
5 Miles	F.______________ mrem	G.______________ mrem
10 Miles	H.______________ mrem	I.______________ mrem

15. MESSAGE RECEIVED BY: (Name) ________________________ Date ___/___/____ Time __________
    THIS IS CRYSTAL RIVER UNIT 3. ☐ This Is A Drill ☐ This Is An Emergency END OF MESSAGE.

☐ Form FAXED EC / EOFD INITIALS: ______________

CONDENSED INSTRUCTIONS FOR COMPLETION OF THE FLORIDA NUCLEAR PLANT EMERGENCY NOTIFICATION FORM (FOR CONTROL ROOM USE ONLY)

The purpose of these instructions is to provide succinct guidance for the completion of the Florida Nuclear Plant Emergency Notification Form in the **Control Room only**. Its use assumes that the user is familiar with the form and the more detailed instructions found elsewhere in Attachment 2 and will refer to those instructions if needed.

Item 1 – CHECK "This is a Drill" OR "This is an Emergency" as appropriate. CHECK the applicable classification box.

Item 2.A – ENTER today's date.
Item 2.B – ENTER time of contact with SWO, Citrus County, and Levy County.
Item 2.C – ENTER name of person making notification. Typically, this is the CNO or the SM.
Item 2.D – ENTER the sequential message number.
Item 2.E – CHECK "Control Room".
Item 2.F – CHECK "Initial / New Classification" unless the notification is for the update of an existing classification. IF this is an update notification, THEN CHECK "Update".

Item 3 – No action. Pre-selected for CR3.

Item 4 – CHECK the emergency classification being declared.

Item 5 – CHECK "Emergency Declaration" and ENTER the date and time of the declaration. IF terminating an emergency, THEN REFER to detailed guidance in Attachment 2.

Item 6.A – ENTER the applicable EAL number(s).
Item 6.B – Not needed when contacting SWO. May be used when notifying NRC via ENS.

Item 7.A – IF no additional information or update is needed, THEN CHECK "None"
Item 7.B – CHECK "Description" to provide additional information. Examples include:
- Conditions briefly warranting a higher classification but no longer exist.
- Conditions independently warranting a lower or equal classification. An example would be a fire in the Protected Area during a Site Area Emergency or General Emergency.

Item 8.A – ENTER 33' wind direction from the primary tower **IF** available. Alternate sources are the 175' primary tower and the 33' alternate tower.
Item 8.B – ENTER a minimum of 3 downwind sectors from the table below:

DEGREES	SECTORS	DEGREES	SECTORS	DEGREES	SECTORS
349-11 (349-371)	H J K	102-123 (462-483)	N P Q	214-236	B C D
12-33 (372-393)	J K L	124-146 (484-506)	P Q R	237-258	C D E
34-56 (394-416)	K L M	147-168 (507-528)	Q R A	259-281	D E F
57-78 (417-438)	L M N	169-191 (529-540)	R A B	282-303	E F G
79-101 (439-461)	M N P	192-213	A B C	304-326	F G H
				327-348	G H J

NOTE: A release is **any** of the following:
- Exceeding the <u>warning</u> setpoint of an effluent monitor (e.g. RM-A2) as a direct result of the emergency initiating event condition.
- Radioactivity detected by environmental monitoring.
- OTSG tube rupture > 10 gpm with either: (1) prolonged steaming from the affected OTSG, **OR** (2) an unisolable steam leak outside RB from the affected OTSG.
- Radioactivity escaping unmonitored from the plant.

Item 9.A – **IF** a release is **NOT** occurring, **THEN** CHECK "None" **AND** GO TO Item 11.
Item 9.B – **IF** a release is occurring, **THEN** CHECK "In Progress" **AND** GO TO Item 10.
Item 9.C – **IF** a release occurred but has terminated, CHECK "Has occurred, but stopped" **AND** GO TO Item 11.

Item 10.A – **IF** core condition or release status cannot be determined, **THEN** CHECK "Under evaluation".
Item 10.B – **IF** the release is monitored by RM-A1 or RM-A2 **AND** the low range gas channel is below its high alarm setpoint, **THEN** CHECK "Release is within Normal Operating Limits".
Item 10.C – REFER to Attachment 2 for instructions.
Item 10.D – REFER to Attachment 2 for instructions.
Item 10.E – **IF** a liquid release exceeding limits is occurring, **THEN** CHECK "Liquid release (no actions required)".

Item 11.A – **IF** no protective action recommendations are necessary, **THEN** CHECK "No utility recommended actions at this time".
Item 11.B – **IF** protective action recommendations are necessary, **THEN** CHECK "Utility recommends the following protective actions:" **AND** REFER to Enclosure 3.

GO TO Item 15.

Item 15 – ENTER name of person receiving notification and date / time **AND** CHECK "This is a Drill" **OR** "This is an Emergency" as appropriate.

FLORIDA NUCLEAR PLANT EMERGENCY NOTIFICATION FORM ASSOCIATED INFORMATION AND PROTOCOL

1.0 GENERAL INFORMATION AND PROTOCOL

1. When communicating information to State and Risk Counties, ENUNCIATE properly, READ off the information by line number, TRANSMIT numbers digit by digit, AVOID sound alike action statements, SPELL difficult words, as appropriate, USE three word phrases for descriptions / narratives, and do **NOT** use technical jargon. Be prepared to answer questions from the State and Risk Counties.
2. If the emergency is terminated or reclassified before all contacts are made, or if the emergency is the result of an Emergency Action Level(s) indicating a higher classification that after a brief period is downgraded to a lower classification, PERFORM the following:
   - STATE the current emergency classification
   - STATE the highest classification status and when it was achieved
   - STATE the period of time that the higher classification existed and the mitigating conditions that caused the emergency classification to be downgraded.
3. In long-lasting events caused by natural phenomena, regular update notifications to the State and Counties can be suspended or the frequency reduced (4 hours, per shift, etc.) if both the following criteria are met:
   - State and Risk Counties agree to the suspension or reduction in frequency.
   - There is **NO** significant change in Plant status.

**FLORIDA NUCLEAR PLANT EMERGENCY NOTIFICATION FORM
ASSOCIATED INFORMATION AND PROTOCOL**

1.0 GENERAL INFORMATION AND PROTOCOL (Cont'd)

4. If during a notification, a change in classification occurs, COMPLETE the current notification in progress and within 15 minutes PROVIDE the State and Risk Counties with an update notification **OR** PERFORM the following as appropriate. REFER TO initial notification protocols for when a classification is briefly met:
   - If a higher classification is met:
     - SUSPEND notification of the lower classification.
     - INFORM off-site agencies to stand-by for classification upgrade.
     - TRANSMIT the higher classification verbally **AND** FAX the lower classification form to the agencies.
     - COMPLETE a new form with the upgraded classification.
   - If a lower classification is met:
     - COMPLETE the current communication in progress.
     - INFORM off-site agencies to stand-by for classification downgrade.
5. COMPLETE items 12, 13, and 14 when the EOF is operational. READ the form information as part of the emergency notification.
6. After the EOF Director or designee approves the FLORIDA NUCLEAR PLANT EMERGENCY NOTIFICATION FORM, any information added to or any changes to existing information requires re-approval before transmittal off-site.
7. To correct an error on the form, DRAW a single line through the error, ENTER the correct information, and initial and date.
8. USE the completion time of the last notification transmittal (Item 15) as the start time of the 60-minute clock for update notifications.
9. If the FLORIDA NUCLEAR PLANT EMERGENCY NOTIFICATION FORM is used for the initial notification from the Control Room, RECORD the name of the Headquarters Operations Officer and Event Notification Number on Attachment 3, Reactor Plant Event Notification Worksheet or the Control Room logbook. Do **NOT** write any NRC-type information on the FLORIDA NUCLEAR PLANT EMERGENCY NOTIFICATION FORM.
10. **IF** abbreviations / acronyms are used on the FLORIDA NUCLEAR PLANT EMERGENCY NOTIFICATION FORM, **THEN** state what the abbreviation /acronym stand for when verbally communicating to the State and Risk Counties.

FLORIDA NUCLEAR PLANT EMERGENCY NOTIFICATION FORM ASSOCIATED INFORMATION AND PROTOCOL

1.0 **GENERAL INFORMATION AND PROTOCOL** (Cont'd)

11. In the event where situations affect the entire State of Florida (hurricane, terrorist threat, etc.) and multiple notifications are being received at SWO, and the Duty Officer **CANNOT** take the notification from CR3, PROVIDE at least the emergency classification level and time of declaration and ask for a more suitable time to callback with the remainder of the information. If PARs are being recommended or changed during a similar event, INFORM SWO within 15 minutes of the recommendation or change. Ensure the Risk Counties are provided a separate notification using SHRD or Commercial telephone.

12. When EAL number(s) are used on the form, either have the Duty Officer CONFIRM the paraphrased EAL or state the paraphrased EAL using Enclosure 4 to PROVIDE confirmation of offsite agencies understanding of the event.

FLORIDA NUCLEAR PLANT EMERGENCY NOTIFICATION FORM ASSOCIATED INFORMATION AND PROTOCOL

1.2 **Initial Notification** [NOCS 100521]

1. NOTIFY State Watch Office. This also notifies Citrus and Levy counties and the Department of Health, Bureau of Radiation Control (DHBRC)-Orlando. ENSURE offsite agencies are on-line by checking each box as station roll call is completed. If offsite agencies do **NOT** respond to roll call, separate notifications using Commercial telephones are required to Citrus (746-2555) and Levy County (1-352-486-5212 or 1-352-486-5111 after hours). SWO will contact DHBRC. If information is **NOT** available, do **NOT** delay notification to State Watch Office. Item 2.B of the form is the official time for the 15 minute notification time limit and update notifications and is considered completed when the roll call is completed and the emergency classification has been announced. If the roll call cannot be completed due to lack of response from an offsite agency, the classification announcement should not be delayed.
2. USE one of the following communications networks listed by priority:
   - STATE Hot Ringdown (SHRD) - Station 120 or 121
   - Commercial Telephone System - 1-850-413-9911 or 1-800-320-0519 or 1-850-413-9900
   - Florida Emergency Management Network (EMnet)
   - Portable Satellite Phone (Located in TSC cabinet ONLY)
3. When making the initial notification of an emergency condition to SWO, REPORT the current emergency classification declared at the time the notification is made. If before initial notification or since the previous notification conditions were briefly met for a higher classification, EXPLAIN in Additional Information or Update section using guidance from item 7 on Attachment 1.
4. Once communications is established with the SWO Duty Officer and the station roll call is complete, READ the message in its entirety, REPEAT information **AND** ANSWER questions as requested
5. After the notification is completed, FAX the FLORIDA NUCLEAR PLANT EMERGENCY NOTIFICATION FORM by using Group 1 from the FAX machine. Group 1 consists of SWO, Citrus County EOC, Levy County EOC, Department of Health, Bureau of Radiation Control (DHBRC)-Orlando, and Progress Energy Emergency Response Facilities.

FLORIDA NUCLEAR PLANT EMERGENCY NOTIFICATION FORM AND ASSOCIATED INFORMATION AND PROTOCOL

1.3 Update Notification

1. UPDATE SWO every 60 minutes after initial notification and upgrades of emergency classification.
2. The use of the FLORIDA NUCLEAR PLANT EMERGENCY NOTIFICATION FORM is required for:
   b. Initial notification that an emergency condition exists (Item 4)
   c. Any change in emergency classification (Item 4)
   d. Any change in Protective Action Recommendations (Item 11)
   e. Termination of an emergency classification (Item 5B)
3. Other updated information **NOT** meeting the above criteria does **NOT** require the use of the FLORIDA NUCLEAR PLANT EMERGENCY NOTIFICATION FORM.
4. The 60 minute update notification is still required with a statement there is **NO** change from last update, unless the SWO agrees to less frequent updates.
5. If the update notification will be delayed because of current Plant conditions and Control Room activities, INFORM the SWO Duty Officer.

**FLORIDA NUCLEAR PLANT EMERGENCY NOTIFICATION FORM
ASSOCIATED INFORMATION AND PROTOCOL**

2.0 GUIDANCE FOR COMPLETING THE FLORIDA NUCLEAR PLANT EMERGENCY NOTIFICATION FORM

1. CHECK Item "A" for a drill **OR** Item "B" for an emergency and the applicable classification box.

NOTE:	Items 2.A, 2.A, and 2.C are completed at the time contact is made with the SWO and the Risk Counties.

2. A. ENTER the date (MM/DD/YY) contact is made with the State Watch Office and the Risk Counties.

   B. ENTER the time (24-hour clock) contact is made with the State Watch Office and the Risk Counties. This time must be within 15 minutes of Item 5 for initial and upgrade notifications and within 60 minutes of Item 15 from the previous notification message form for update notifications.

   C. ENTER the name of the person making the notification.

   D. ENTER message number (beginning with #1 and following through sequentially in the TSC and EOF).

   E. CHECK the facility location box from which the notification is made. If the notification is made from the Remote TSC, CHECK the TSC box.

   F. CHECK whether notification is a new classification or an hourly update when classification or PARs have **NOT** changed.

3. ENSURE the CR3 box is checked and report to the State Watch Office or the Risk County during notification.

4. CHECK the appropriate emergency classification box corresponding to the current Plant conditions. REFER TO Item 7 guidance for when conditions briefly exist for a higher classification.

5. CHECK Item "A" and ENTER the declaration date (MM/DD/YY) and the time (24 hour clock) of the current emergency classification. CHECK Item "B" if the emergency is terminated or when the transition from the "Emergency Phase" to the "Recovery Phase" has taken place and ENTER the date (MM/DD/YY) and the time (24-hour clock) of emergency termination. Termination notification messages do **NOT** require Items 6 through 14 to be completed; however, ENTER the bases for the termination in Item 7. If classifying and terminating an emergency in the same notification message CHECK both Item "A" and Item "B," ENTER the declaration date (MM/DD/YY) and time (24-hour clock) in Item 5 **AND** COMPLETE Item 6 to PROVIDE EAL information. Items 7 through 14 can be skipped.

FLORIDA NUCLEAR PLANT EMERGENCY NOTIFICATION FORM ASSOCIATED INFORMATION AND PROTOCOL

2.0 GUIDANCE FOR COMPLETING THE FLORIDA NUCLEAR PLANT EMERGENCY NOTIFICATION FORM (Cont'd)

CAUTION

When completed during a Security threat, the description should **NOT** contain Safeguards Information.

6. CHECK Item "A" **AND** ENTER the EAL Number corresponding to the EAL table **or** EAL Numbers if using the FPB Matrix **OR** CHECK Item "B" and ENTER a short description of the current event in layman's terms to indicate the accident condition Emergency Action Level (paraphrased) using Enclosure 4 or the status of the Fission Product Barriers used to declare the event (e.g., Loss of Reactor Coolant System Barrier, Potential Loss of Fuel Clad Barrier, etc.) from the FPB Matrix. Each EAL has one number (e.g., 2.13) therefore do **NOT** use any EAL sub-numbers on the form. When the classification is upgraded, include all applicable FPB EAL numbers, **NOT** just the EAL number causing the upgrade. Do **NOT** use the enclosure title (e.g., FPB Matrix, System Malfunction, etc.) as a description of the emergency. This information should remain the same throughout update messages unless there is a classification change. Avoid using Plant-specific acronyms or abbreviations.

CAUTION

When completed during a Security threat, the description should **NOT** contain Safeguards Information.

7. CHECK Item "A" for **NO** additional information or UPDATE **OR** CHECK Item "B" and ENTER 1) additional significant events, including if conditions briefly existed for a higher emergency classification but **NO** longer exist, or 2) conditions that would have independently warranted declaration of an equal or lower classification (e.g., a fire within the Protected Area during a SITE AREA **OR** GENERAL EMERGENCY). CONSIDER including emergency response actions underway, any requests for offsite assistance, the bases for termination of the emergency, and facility activation status. AVOID using Plant-specific acronyms or abbreviations. [NOCS 96024].

FLORIDA NUCLEAR PLANT EMERGENCY NOTIFICATION FORM ASSOCIATED INFORMATION AND PROTOCOL

2.0 GUIDANCE FOR COMPLETING THE FLORIDA NUCLEAR PLANT EMERGENCY NOTIFICATION FORM (Cont'd)

NOTE: The preferred source of meteorological data is the 33' Primary Tower (MMP-5). Alternate sources are the 175' Primary Tower and the 33' Alternate Tower (MMP-1).

If the wind direction or wind speed recorders are **NOT** in service, the appropriate meter may be observed for a brief period (approximately 30 seconds) to obtain an estimate.

8. ENTER the wind direction in degrees in Item "A." ENTER a minimum of 3 downwind sectors from the Sectors Affected table below in Item "B." The downwind sectors confirm wind direction because of potential confusion with degrees "from" versus degrees "to."

Sectors Affected

DEGREES	SECTORS	DEGREES	SECTORS	DEGREES	SECTORS
349-11 (349-371)	H J K	102-123 (462-483)	N P Q	214-236	B C D
12-33 (372-393)	J K L	124-146 (484-506)	P Q R	237-258	C D E
34-56 (394-416)	K L M	147-168 (507-528)	Q R A	259-281	D E F
57-78 (417-438)	L M N	169-191 (529-540)	R A B	282-303	E F G
79-101 (439-461)	M N P	192-213	A B C	304-326	F G H
				327-348	G H J

9. CHECK Item "A" if there are **NO** indications of a release and go to Item 11. CHECK Item "B" if a release is occurring, even though it may be less than normal operating limits. CHECK Item "C" if a release has occurred but stopped and go to Item 11. (REFER TO release definition).

10. CHECK applicable Release Significance Category box based on the table on page 12 of this attachment, **OR** CHECK Item "E" if it is a liquid effluent release exceeding limits. If the PAG category is selected, INFORM the Emergency Coordinator that EM-202, Enclosure 1, Emergency Classification Table should be consulted for applicable EALs. Item "A" should be selected only if core condition or release status **CANNOT** be determined.

**FLORIDA NUCLEAR PLANT EMERGENCY NOTIFICATION FORM
ASSOCIATED INFORMATION AND PROTOCOL**

2.0 **GUIDANCE FOR COMPLETING THE FLORIDA NUCLEAR PLANT EMERGENCY NOTIFICATION FORM** (Cont'd)

11. CHECK Item "A" if **NO** Protective Actions Recommendations (PARs) are necessary. **IF** Item "A" is checked, do **NOT** check Item "B" as this is a protective action recommendation. CHECK Item "B" if PARs are necessary **AND** ENTER the Zone designation(s) for evacuation and shelter. If Item A is checked, Item B does **NOT** need completing.

NOTES: 1. If the form is completed in the Main Control Room, completion of Items 12-14 is not required. If the form is completed in the TSC, completion of Items 12-15 is required.

2. The Accident Assessment Coordinator PROVIDES information for Item 12 for all classification levels **OR** this information (A, B, and C) may be obtained from the Critical Safety Function Status Board.

12. CHECK the appropriate status boxes for Item "A" Reactor Shutdown, Item "B" Core Adequately Cooled, Item "C" Containment Intact, and "D" Core Condition based on current Plant conditions.

NOTE: The Radiation Controls Coordinator PROVIDES information for Item 13 for all classification levels. This item information may be obtained from a current dose assessment printout or the status board.

13. ENTER the wind speed in mph *(mph = m/sec x 2.24)* in Item "A." ENTER the Stability Class in Item "B" based on the Sigma Theta, Wind Range, or Delta T from table below.

Stability Class Determination

Sigma Theta (Degrees)	Wind Range (Degrees)	Delta T (Degrees)	Stability Class
≥ 22.5	≥ 135	≤ -1.46	A (Most Dispersed Plume)
< 22.5 to 17.5	134 to 105	-1.45 to –1.31	B
< 17.5 to 12.5	104 to 75	-1.30 to –1.16	C
< 12.5 to 7.5	74 to 45	-1.15 to –0.39	D
< 7.5 to 3.8	44 to 23	-0.38 to 1.15	E
< 3.8 to 2.1	22 to 13	1.16 to 3.07	F
< 2.1	≤ 12	≥ 3.08	G (most concentrated plume)

FLORIDA NUCLEAR PLANT EMERGENCY NOTIFICATION FORM ASSOCIATED INFORMATION AND PROTOCOL

2.0 **GUIDANCE FOR COMPLETING THE FLORIDA NUCLEAR PLANT EMERGENCY NOTIFICATION FORM** (Cont'd)

NOTE:	The Radiation Controls Coordinator only completes Item 14 if a release is occurring or occurred, but stopped. Otherwise, this item may be "N/A." This item information may be obtained from a current dose assessment printout or the status board.

14. CHECK Item "A" (**NOT** Applicable) if no release **AND GO TO** Item 15. If a RASCAL dose projection is available, ENTER the calculation duration in the blanks in the headings of the CDE and TEDE columns (normally 6 hrs.). ENTER the projected thyroid (CDE) and Total Dose (TEDE) for each distance location in Items "B through I".

NOTE:	Item 15 is completed after the message has been read to the offsite agencies.

15. ENTER the name of the SWO Duty Officer or individual receiving the notification. ENTER the date (MM/DD/YY) and time (24 hour clock) provided by the SWO Duty Officer or individual receiving the notification.
    - CHECK the Form FAXED box when the action (FAX, email, copies distributed to positions, etc.) is completed.
    - ENSURE the Emergency Coordinator initials the Form before it is communicated to the SWO.

FLORIDA NUCLEAR PLANT EMERGENCY NOTIFICATION FORM ASSOCIATED INFORMATION AND PROTOCOL

3.0 RELEASE SIGNIFICANCE CATEGORIES

CORE CONDITION	RELEASE STATUS	RELEASE SIGNIFICANCE CATEGORY	FLORIDA NUCLEAR PLANT EMERGENCY NOTIFICATION FORM REFERENCE
NO Core Damage	NO release	NR	9.A
	Release in progress	<NOL NS	10.B 10.C
Clad Barrier Lost or Potentially Lost	NO release	NR	9.A
	Release in progress	PAG	10.D
Core Melt	NO release	NR	9.A
	Release in progress	PAG	10.D

NR: NO RELEASE

This category indicates **NO** release is occurring. This category is appropriate regardless of core status, if there are **NO** indications of a release (e.g., unexplained containment pressure decrease, unexplained abnormal radiation levels in Auxiliary Building or Intermediate Building, on the berm, or in the field). Do **NOT** assume Design Basis Leakage is occurring if it has **NOT** been detected. If a release occurred but has now stopped, MAINTAIN the appropriate category below until EPZ doses have dissipated.

<NOL: RELEASE WITHIN NORMAL OPERATING LIMITS (ITS/ODCM)

This category indicates releases that are monitored by RM-A1 or RM-A2, occurring when the fuel is intact (does **NOT** meet potential loss or loss criteria). These releases are within normal operating limits if the low-range gas channel is below its high alarm setpoint. Do **NOT** make this selection for releases **NOT** monitored by RM-A1 or RM-A2 unless they have been evaluated per the ODCM.

NS: NON-SIGNIFICANT (FRACTION OF PROTECTIVE ACTION GUIDELINE VALUES)

This category indicates releases that are occurring when the fuel is intact (does **NOT** meet potential loss or loss criteria). It includes releases exceeding RM-A1 or RM-A2 high alarm setpoint (e.g., LOCA, Waste Gas System failures). It also includes releases **NOT** monitored by RM-A1 or RM-A2 (e.g., Steam Generator Tube Rupture with safeties lifting). These releases will **NOT** produce site boundary doses that approach the EPA Protective Action Guideline values of 1 REM TEDE and/or 5 REM thyroid, **NO** Protective Action Recommendations are necessary.

PAG: AT OR NEAR PROTECTIVE ACTION GUIDELINE VALUES

This category indicates releases that are occurring after the fuel clad barrier has been lost or potentially lost. Site Boundary doses greater than the EPA Protective Action Guideline of 1 REM TEDE and/or 5 REM thyroid are possible. The category is appropriate with fuel cladding failure even if only minor offsite doses are detected. Shelter or evacuation beyond 5 miles should be determined based on Plant status and dose projections. This category addresses fuel damage in the core only. Spent fuel damage is addressed on a case-by-case basis.

The PAG category also includes Early Health Effects (EHE) which is not listed as a separate category on the Florida Nuclear Plant Emergency Notification Form because the State of Florida implements protective actions based upon the overall PAG classification. EHE indicates releases that are occurring after severe core damage has taken place and where containment has failed early in the event. Doses of 25 REM TEDE and/or 2500 RADS thyroid could cause early health effects and these doses are easily possible within three miles from the Plant. Evacuation of the Energy Complex should be performed and evacuation of the 10-mile EPZ (Zones 1, 2, 3) should be recommended (never sheltering) even if evacuees are exposed to the plume.

REACTOR PLANT EVENT NOTIFICATION WORKSHEET

NRC FORM 361
(12-2000)

REACTOR PLANT EVENT NOTIFICATION WORKSHEET

NRC OPS CENTER COMMUNICATOR________________________ EN #____________

1) Use ENS phone sticker number for NRC direct. IF ENS IS OUT OF SERVICE, use 2) Commercial 1-301-816-5100 or 1-301-951-0550 or 1-301-415-0550 or 1-301-415-0553

NOTIFICATION TIME	FACILITY	UNIT	NAME OF CALLER	CALL BACK:
	CRYSTAL RIVER	3		ENS # 700-821-0027 Or # 1-352-795-6958

EVENT TIME & ZONE	EVENT DATE	POWER / MODE BEFORE	POWER / MODE AFTER
		% /	% /

EVENT CLASSIFICATIONS	1-HOUR NON-EMERGENCY 50.72(b)(1)	☐ (v)(A) Safe S/D Capability
☐ GENERAL EMERGENCY	☐ TS Deviation	☐ (v)(B) RHR Capability
☐ SITE AREA EMERGENCY	4-HOUR NON-EMERGENCY 50.72(b)(2)	☐ (v)(C) Control of Radiological Release
☐ ALERT	☐ (i) TS Required S/D	☐ (v)(D) Accident Mitigation
☐ UNUSUAL EVENT	☐ (iv)(A) ECCS Discharge to RCS	☐ (xii) Offsite Medical
☐ 50.72 NON-EMERGENCY (see next column)	☐ (iv)(B) RPS Actuation	☐ (xiii) Loss Comm/Asmt/Resp
☐ PHYSICAL SECURITY (73.71)	☐ (xi) Offsite Notification	60-DAY OPTIONAL 50.73(a)(1)
☐ MATERIAL/EXPOSURE (20.2202)	8-HOUR NON-EMERGENCY 50.72(b)(3)	☐ Invalid Specified System Actuation
☐ FITNESS FOR DUTY	☐ (ii)(A) Degraded Condition	Other Unspecified Requirement (Identify)
☐OTHER UNSPECIFIED REQMT (see last column)	☐ (ii)(B) Unanalyzed Condition	☐
☐ INFORMATION ONLY	☐ Specified System Actuation	☐

EVENT DESCRIPTION

Include: Systems affected, actuations & their initiating signals, causes, effect of event on Plant, actions taken or planned, etc. *(Continue on back)*

NOTIFICATIONS	YES	NO	WILL BE
NRC RESIDENT			
STATE			
LOCAL			
OTHER GOVT AGENCIES			
MEDIA/PRESS RELEASE			

ANYTHING UNUSUAL OR NOT UNDERSTOOD? ☐ YES (Explain above) ☐ NO

DID ALL SYSTEMS FUNCTION AS REQUIRED? ☐ YES ☐ NO (Explain above)

MODE OF OPERATION UNTIL CORRECTED:	ESTIMATED RESTART DATE:________	ADDITIONAL INFO ON BACK ☐ YES ☐ NO

REACTOR PLANT EVENT NOTIFICATION WORKSHEET (Cont'd)

ADDITIONAL INFORMATION

NOTIFICATION TIME ____________

RADIOLOGICAL RELEASES Check or Fill in Applicable Items (specific details/explanations should be covered in event description)

☐ LIQUID RELEASE ☐ GASEOUS RELEASE ☐ UNPLANNED RELEASE ☐ PLANNED RELEASE ☐ ONGOING ☐ TERMINATED
☐ MONITORED ☐ UNMONITORED ☐ OFFSITE RELEASE ☐ TS EXCEEDED ☐ RM ALARMS ☐ AREAS EVACUATED
☐ PERSONNEL EXPOSED OR CONTAMINATED ☐ OFFSITE PROTECTIVE ACTIONS RECOMMENDED * State release path in description

Release Rates/Limits (From Dose Assessment Team)	Release Rate (Ci/sec)	% ODCM Limit	Total Activity (Ci)	% ODCM Limit
Noble Gas				
Iodine				
Particulate				
Liquid *(excluding tritium & dissolved noble gases)*				
Liquid (tritium)				
Total Activity				

RAD MONITOR READINGS	Plant Stack (RMA-2)	Condenser/Air Ejector (RMA-12)	Main Steam Line (RMG-25,26,27,28)	Other (List)
RAD MONITOR READINGS:				
ALARM SETPOINTS:				
% ODCM LIMIT *(IF APPLICABLE)*				

RCS OR SG TUBE LEAKS CHECK OR FILL IN APPLICABLE ITEMS: *(SPECIFIC DETAILS/EXPLANATIONS SHOULD BE COVERED IN EVENT DESCRIPTION)*

LOCATION OF THE LEAK (E.G., SG#, VALVE, PIPE, ETC.)

LEAK RATE:	UNITS: GPM/GPD	T.S. Limits:	☐ Sudden or ☐ Long Term Development

LEAK START DATE: TIME: | COOLANT ACTIVITY: PRIMARY ________ µCI/ML SECONDARY ________ µCI/ML

LIST OF SAFETY RELATED EQUIPMENT NOT OPERATIONAL:

EVENT DESCRIPTION *(Continued from front)*

EC INITIALS __________ **DATE:**_________

REACTOR PLANT EVENT NOTIFICATION WORKSHEET (Cont'd)

NRC Operations Center Notification Protocols

NOTE: The **initial** NRC notification may be performed using the information from Items 4 through 7 and Item 11 of the FLORIDA NUCLEAR PLANT EMERGENCY NOTIFICATION FORM, in order to expedite notification from the Control Room. Use the form for subsequent notifications, unless continuous communication is established. Do **NOT** use EAL number(s) from Item 6 and instead use the paraphrased EAL from Enclosure 4 or provide a status of the Fission Product Barriers.

NOTIFY the NRC as soon as practicable after the State of Florida, but **NO** later than sixty minutes from declaration of an emergency condition.

NOTE: The NRC automatically records communications on the NRC Event Notification System (ENS).

USE the ENS telephone as primary means of communication and Commercial telephones as secondary means of communication. The ENS number is located on a sticker affixed to the telephone. The Commercial numbers are located on the REACTOR PLANT EVENT NOTIFICATION WORKSHEET.

ENSURE the appropriate sections of the REACTOR PLANT EVENT NOTIFICATION WORKSHEET are completed and SM / EC approval has been granted before making the notification.

The communicator making the notification ensures the person receiving the report has an adequate understanding of the event and the related safety significance to ensure appropriate NRC response.

Include insight if known to the following: (information source)
- Is there any change to the classification of the event? If so, what is the reason? (Accident Assessment Coordinator)
- What is the ongoing / imminent damage to the facility, including affected equipment and safety features? (Accident Assessment Coordinator or Repairs Coordinator)
- Have toxic or radiological releases occurred or been projected, including changes in the release rate? If so, what are the projected on-site and off-site releases, and what is the basis of assessment? (Radiation Controls Coordinator)
- What are the health effect / consequences to on-site / off-site people? How many on-site / off-site people are / will be affected and to what extent? (Radiation Controls Coordinator)
- Is the event under control? When was control established or what is the planned action to bring the event under control? What is the mitigative action underway or planned? (Accident Assessment Coordinator)
- What on-site protective measures have been taken or planned? (Florida Nuclear Plant Emergency Notification Form)
- What off-site protective actions have been recommended to State / County officials? (Florida Nuclear Plant Emergency Notification Form)
- What is the status of State / County / other Federal agencies responses, if known? (EOF Staff)
- If applicable, what is the status of public information activities, such as siren, broadcast, or press releases? Has the Emergency News Center been activated? (ENC Staff)

RESPOND to any request for additional information that you can answer; otherwise, state that the information is **NOT** yet available and will be provided in a follow-up message. Any questions asked by the NRC and the associated responses given should be documented in writing and attached to the REACTOR PLANT EVENT NOTIFICATION WORKSHEET.

NOTE: For Alert or higher classifications, the Headquarters Operations Officer will be attempting to patch the Region II Administrator and other Region II personnel into the call concurrent with recording your message. You may be interrupted by patch-ins and / or requested to repeat information, and you should comply with these requests. If the Regional Administrator or deputy has **NOT** been patched in by the time you have completed your message, the Headquarters Operations Officer will probably request additional information.

Upon declaration of an Alert or higher, the NRC Operations Center will most likely request the communicator stay on the line. If the notification originates from the Control Room, tell the NRC you are signing-off ENS. If requested to maintain an open communications line, notify the SM / EC to provide an alternative communicator or take other action.

REPORT any lower classification(s) declared before the initial notification to ensure the NRC is aware of previous Plant conditions and implementation of the E-Plan.

Upon arrival of the NRC Site Response Team and with the concurrence of or at the request of the Headquarters Operations Officer, face-to-face communication begins between the PE NRC Liaison and the lead NRC representative at the TSC. This information includes emergency classification changes, Protective Action Recommendations changes, and the non-emergency reporting requirements including invoking 10CFR50.54 (x) (y).

REACTOR PLANT EVENT NOTIFICATION WORKSHEET (Cont'd)

Guidance for Completing the Reactor Plant Event Notification Worksheet

If an open communications channel is established, routine use of the form is **NOT** required, provided that verified changes in Plant / equipment status are communicated to the NRC verbally and a summary of the communications with the NRC is maintained in the log.

NOTE: The following items are completed at the time of the notification.

- Print the first and last name of the Headquarters Operations Officer (HOO) in the NRC Operations Center Communicator space provided.
- Print the Event Notification number (EN #) provided by the HOO in the space provided.
- Enter the notification time (24 hour clock) provided by the HOO when communication is established.
- Provide the Call Back numbers as applicable.

NOTE: The following items are completed before the notification.

- Enter the first and last name of the person making the notification in the "Name of Caller" space.
- Enter the "Event Time and Zone (24 hour clock and Eastern Time)," "Event Date (MM/DD/YY)," "Power / Mode Before," and "Power / Mode After."
- Enter the "Mode of Operation Until Corrected (numeric)" and "Estimated Restart Date (MM/DD/YY)" at the bottom of page 1.

<u>Event Classifications Section</u>

- Check the applicable block for the current emergency classification.

<u>1-Hour, 4-Hour, and 8-Hour Non-Emergency Sections</u>

- Check all blocks that apply and are separate reportability items from the reason CR3 has declared an emergency condition. The determination of these items is the responsibility of the TSC Accident Assessment Team Operations Support Representative when the TSC is operational.

<u>Event Description Section</u> **(additional space is provided on page 2 of form)**

- Provide a clear and concise description of the event. Avoid using Plant-specific acronyms or abbreviations.
- Discuss each reportable event, as necessary.
- Report the failure of significant components.
- Include those Plant specific systems or components, which were available to perform the same function as any system or component that failed during the event.
- Include information which will promote understanding of the report, such as any extenuating circumstances (good or bad) or any related generic concerns within the industry.
- If the "Other" block was checked in the Event Classifications Section, provide amplifying information to explain this choice.

<u>Notifications Section</u>

- Check the appropriate box based on the notifications made before the notification to the NRC Operations Center. All are normally checked "Yes", except "Media / Press Release" for the initial notification.
- "Other Govt Agencies" is Department of Health – Bureau of Radiation Control during a declared emergency.

<u>Questions Posed by the Headquarters Operations Officer Section</u>

- Be prepared to answer these questions based on the event and provide explanations in the "Description" section as applicable.
- Check appropriate box for "Additional Information on Back."

<u>Radiological Releases Subsection</u>

- Check or fill-in applicable items based on information from the Radiation Controls Coordinator, logs, SWO notification forms, and status boards and provide specific details / explanations in "Description" section.

<u>RCS or SG Tube Leak Subsection</u>

- Check or fill-in applicable items based on information from the Accident Assessment Coordinator, logs, SWO notification forms, and status boards and provide specific details / explanations in "Description" section.
- After the Emergency Coordinator or designee approves the REACTOR PLANT EVENT NOTIFICATION WORKSHEET, any information added to or any changes to existing information requires re-approval before transmittal off-site unless continuous communication is established where the form is **NOT** required.
- To correct an error on the form, draw a single line through the error, enter the correct information, and initial and date.
- Obtain SM / EC approval of the NRC form before transmittal of the information unless continuous communication is established when the form is **NOT** required.

EMERGENCY NOTIFICATION FOR UNITS 1/2 & 4/5

USE Enclosure 2, Evacuation Planning Guide to determine Protective Action Recommendations for Energy Complex personnel. (NONE for Unusual Event or Alert)

Unit 1/2 (extension 2120 or 563-4454) Contact _______________ Time ______

Unit 4/5 (extension 8-245-5283 or 352-501-5283) Contact _______________ Time ______

GIVE THE FOLLOWING INFORMATION TO THE FOSSIL UNITS:

1. Your name and position: ____
2. State that CR3 is in a(n) ☐ Emergency ☐ Drill
3. State CR3 has declared a(n) ☐ Unusual Event ☐ Alert ☐ Site Area Emergency ☐ General Emergency
4. Briefly explain Plant conditions using basic facts: ____
   ____
   ____
5. STATE if conditions are:
   - ☐ "IMPROVING" (Plant conditions are improving in the direction of a lower emergency classification or termination of the event)
   - ☐ "STABLE" (Plant conditions are **NOT** degrading and the emergency is under control)
   - ☐ "DEGRADING" (Plant conditions continue to degrade and it is evident that the situation will worsen or a higher classification is imminent)
6. STATE one of the following based on Plant conditions:
   - ☐ "NO RADIOACTIVE MATERIAL HAS BEEN RELEASED."
   - ☐ "RADIOACTIVE MATERIAL IS BEING RELEASED AT LOW LEVELS" (when NO fuel is damaged)
   - ☐ "RADIOACTIVE MATERIAL IS BEING RELEASED."
7. STATE one of the following based on declared emergency:
   - ☐ (Unusual Event or Alert) "NO ASSEMBLY OR EVACUATION IS NECESSARY AT THIS TIME." (Unless Security has determined actions are required in a security event.)
   - ☐ (Site Area Emergency; see Enclosure 2) "BEGIN STANDARD ASSEMBLY AND ACCOUNTABILITY. REFER TO THE CRYSTAL RIVER COAL PLANT SITE ACCOUNTABILITY/EVACUATION MANUAL. ONCE ACCOUNTABILITY IS COMPLETE, NOTIFY CR3 SECURITY AT EXTENSION 3258 OR 795-5078, AND STANDBY FOR FURTHER INSTRUCTIONS."
   - ☐ (General Emergency, **NO** release and release **NOT** likely within 3 hrs; see Enclosure 2) "BEGIN STANDARD ASSEMBLY AND ACCOUNTABILITY. REFER TO THE CRYSTAL RIVER COAL PLANT SITE ACCOUNTABILITY/EVACUATION MANUAL. ONCE ACCOUNTABILITY IS COMPLETE, NOTIFY CR3 SECURITY AT EXTENSION 3258 OR 795-5078, AND EVACUATE NON-ESSENTIAL PERSONNEL. STANDBY FOR FURTHER INSTRUCTION."
   - ☐ (General Emergency, release has occurred or is imminent AND the release duration projected to be less than 2 hours; see Enclosure 2) "ALL PERSONNEL TAKE SHELTER IN CLOSEST BUILDING. STANDBY FOR FURTHER INSTRUCTIONS."
   - ☐ (General Emergency, release has occurred or is likely to occur within 3 hours AND the release duration is unknown; see Enclosure 2) "SECURE THE PLANT AND EVACUATE.", "DO **NOT** PERFORM ASSEMBLY."
8. If time permits and you feel qualified, ask for questions.
9. STATE: "WE WILL KEEP YOU INFORMED."

INITIATION OF THE EMERGENCY RESPONSE DATA SYSTEM (ERDS)

1. LAUNCH the ERDS activation application by one of the following methods:

   - SELECT Start > ERDS Activation from a NGG Standard Desktop.......... N/A ☐ ☐

   OR

   - SELECT Start > NGG OSI-PI Displays > CR3 Qualified > Operations tab > ERDS Activation from a NGG Standard Desktop.......... N/A ☐ ☐

   OR

   - SELECT Start >Programs >Business Apps >PI System >CR3 QPIM >Operations tab > ERDS Activation from a NGG Standard Desktop.......... N/A ☐ ☐

> **NOTES:** 1. The ERDS window will display a series of messages such as "Waiting for Connect" and "Waiting for Accept". Once a connection with the NRC has been established, the message will indicate "Transmitting". If connection is broken ERDS will attempt to reconnect automatically and the same series of messages will be displayed.......... ☐
>
> 2. A "wait" period of approximately one (1) minute is required before an ERDS reconnect attempt will be successful.......... ☐
>
> 3. After ERDS is transmitting data, buttons at the bottom of the activation screen may be used to close the window or transition to the ERDS Data sheet to view the data.......... ☐

2. SELECT the "Click to Activate" button on the ERDS Status Control Screen.......... ☐

3. SELECT "Yes" to activate ERDS.......... ☐

> **NOTE:** NCS should be contacted if either the Mode or ERDS Status lights remain yellow for a period exceeding five (5) minutes.

4. VERIFY the following:

   a. The light beside the Mode selection transitions from red to green.......... ☐

   b. The ERDS status transitions to "Transmitting Data" with the initiation time and date stamp with a green indicating light.......... ☐

   c. The "Messages Sent" parameter begins to increment within a minute of the transition to the "Transmitting Data" status (indicating data sets are being sent to the NRC.).......... ☐

INITIATION OF THE EMERGENCY RESPONSE DATA SYSTEM (ERDS) (Cont'd)

5. IF either Mode or ERDS Status lights remain yellow for a period exceeding 5 minutes, **THEN** PERFORM the following .... N/A ☐ ☐
   a. SELECT the "Click to Deactivate" button .... ☐
   b. DECLARE ERDS to be inoperable .... ☐
   c. NOTIFY NIT .... ☐
6. INFORM SM/CRS that the ERDS transmission has been activated....................... ☐

NOTE:	Closing the ProcessBook display does **NOT** terminate the connection with the NRC. Termination will only occur if the NRC disconnects or if the "Click to Deactivate" button on the display is selected.

7. **WHEN** informed by the NRC that data transmission is no longer required, **THEN** TERMINATE the connection as follows:
   a. **IF** the NRC Operations Center has terminated the connection, **THEN** VERIFY that "Disconnected" is displayed in the status window.. N/A ☐ ☐
   b. **IF** CR3 will terminate the connection, **THEN** PERFORM the following:... N/A ☐ ☐
      1) SELECT the "Click to Deactivate" button. .... ☐
      2) SELECT "Yes" to reaffirm ERDS deactivation. .... ☐
      3) VERIFY that "Disconnected" is displayed in the status window. .... ☐
8. INFORM SM/CRS that the ERDS transmission has been terminated .... ☐

SUMMARY OF CHANGES
Rev. 98 (PRR 513740)

NOTE

Writers and Reviewers: Changes to this procedure may impact EPIPs listed below.

EM-202	EAL Bases Manual
Section 3.0	Section 3.1
Enclosure 1	Attachment 1

Ensure DRRs are initiated as needed.

Section	Changes and Reason
Throughout	Revised footer to reflect Revision 98. **Editorial change.**
Change Summary	Replace "PRRs" with "DRRs". Since recent revisions have eliminated duplication between EM-202 and other POM procedures, reference to PRRs no longer applies. **Editorial change.**
9.1.3.a 9.2.3.a 9.3.3.a 9.4.3.b	Added Citrus and Levy Counties as agencies (along with SWO) to be contacted within 15 minutes of declaration. This addition supports the change in notification timeliness made in Revision 97. Refer to EREG 542232. **Editorial change.**
9.1.4.a (old) 9.2.4.a (old) 9.3.4.c (old) 9.4.4.c (old)	Deleted these steps referencing the LMHVC operation, which no longer apply due to RM-A1/A2 monitor replacement. The need to promptly switch scales is no longer a concern. (PRR 513740) The portion of the steps recommending completion of EM-204A or EMG-NGGC-0002 within 15 minutes was moved to the "recommended within 30" section. In practice, the Control Room Emergency Coordinator typically defers dose projections to the Dose Assessment Team. EM-202 does require that Release Significance Category is determined and reported it to the State within 15 minutes. Determination of Release Significance Categories is a basic form of dose assessment as it characterizes the release potential for the State.
Enclosure 1 EALs 1.1, 1.2, 1.3, 1.4	Replaced current threshold values with new values resulting from replacement of RM-A1/-A2 and impact of extended shutdown isotopic mix. (PRR 513740)
Enclosure 3	Move notes to within body of PAR table for clarity. Minor formatting changes. **Editorial change.**
Attachment 2 Section 1.2	Added reference to NOCS 100521. **Editorial change.** (PRR 545775)

Progress Energy

R
Reference
Use

CRYSTAL RIVER UNIT 3

PLANT OPERATING MANUAL

EM-204A

EMERGENCY PLAN IMPLEMENTING PROCEDURE

OFF-SITE DOSE ASSESSMENT DURING RADIOLOGICAL EMERGENCIES (CONTROL ROOM METHOD)

REVISION 25

TABLE OF CONTENTS

1.0 PURPOSE

1. This procedure provides a timely method for Control Room personnel to estimate the radiation exposure at the Site Boundary (0.83 miles) during a radiological emergency. The procedure was developed for use during the first hour of a monitored, filtered release through the Auxiliary Building Vent. **[R1, R2, R3, R4, R5, R6, R7, R8]**
2. This procedure is an Emergency Plan Implementing Procedure. Any revisions must be carefully considered for Emergency Plan impact.

2.0 REFERENCES

2.1 Developmental References

1. Radiological Emergency Response Plan (RERP), Progress Energy Crystal River Unit 3.
2. Manual of Protective Action Guides and Protective Actions for Nuclear Incidents, EPA-400-R-92-001, Environmental Protection Agency (October, 1991).
3. Nuclear Regulatory Commission – Response Technical Manual RTM-96, Vol. 1 Rev. 4.
4. Radiation Monitor Sensitivity Curve Log, Crystal River 3.
5. Engineering Evaluation EEF-00-009, Rev. 1 – Radiological Monitor Response Factors
6. Calculation N12-0001, Calculation of RM-A1 and RM-A2 Threshold Values for Emergency Action Levels.

2.2 Implementing References

1. **[R1]** NOCS 00389
2. **[R2]** NOCS 100422
3. **[R3]** NOCS 01062
4. **[R4]** NOCS 01128
5. **[R5]** NOCS 01582
6. **[R6]** NOCS 01589
7. **[R7]** NOCS 01592
8. **[R8]** NOCS 09815
9. EM-202, Duties of the Emergency Coordinator

3.0 DEFINITIONS

1. **Affected Sectors -** As a minimum, the downwind sector(s) and the adjacent sectors, as indicated Enclosure 2, Determination of Affected Sectors.
2. **Committed Dose Equivalent (CDE) -** Dose to an organ due to the intake of radioactive materials. For initial emergency dose assessment, only dose to the thyroid is considered when calculating CDE (CDE dose = Thyroid dose).
3. **Deep Dose Equivalent (DDE) -** External whole body dose.
4. **Delta T –** The temperature differential between the 175' level and 33' level of the primary meteorological tower.
5. **Extended Shutdown -** As used in Enclosure 1 Table 1, refers to all releases that occur in the unique period that began 9/26/09 (Refuel 16) and continuing through plant restart. Also refers to releases that occur from the Spent Fuel Pools during the fuel cycle beginning at restart from the extended shutdown and continuing until additional irradiated fuel is off-loaded into the Spent Fuel Pools.
6. **Sigma-Theta -** The standard deviation of a set of wind range measurements. The Sigma-Theta meter automatically calculates and displays the standard deviation of wind range for the previous 15 minutes.
7. **Thyroid (THY) Dose -** Dose to the thyroid due to intake of radioactive iodine. For initial emergency dose assessment, only dose to the thyroid is considered when calculating Committed Dose Equivalent (see Step 3.0.2).
8. **Total Dose (TEDE) -** The Total Effective Dose Equivalent is the sum of external Dose (DDE) and the equivalent amount of whole body dose due to the individual organ uptakes.
9. **Wind Direction -** Direction the wind is coming from, where north = 0°, east = 90°, south = 180°, and west = 270°.

4.0 RESPONSIBILITIES

1. The Emergency Coordinator (EC) is responsible during all emergencies for the implementation of procedures. The EC is required to initially evaluate the situation by EM-202, Duties of the Emergency Coordinator. Responsibility for performing this procedure will be assigned by the EC.

5.0 PREREQUISITES

1. None

6.0 PRECAUTIONS, LIMITATIONS, AND NOTES

1. Stability classes have been grouped into three categories: (A,B,C), (D,E), (F,G). The most stable dispersion factor is applied to all the classes in each category except that F was used because of the low probability of G conditions. Table dose rates will be more conservative for classes A, B, and D (i.e., actual dose rates will be lower).
2. **NO** credit is taken for radioactive decay beyond the first 30 minutes. Table dose rates become more conservative as time progresses (i.e., actual dose rates will be lower).
3. For monitor readings that lie between two table values, the higher table value should be used.

7.0 SPECIAL TOOLS AND EQUIPMENT

None

8.0 ACCEPTANCE CRITERIA

None

9.0 INSTRUCTIONS

1. RECORD all input parameters and dose estimates on the data sheet in Attachment 1 **AND** on the Florida Nuclear Plant Emergency Notification Form as indicated by the number in the right margin of the data sheet ☐

9.1 Radiological Data

1. **IF** RM-A2 is **NOT** operating as designed, or otherwise not monitoring the release, **THEN PERFORM** the following N/A ☐ ☐

   a. INFORM the State Watch Office that dose information will be provided in follow-up notifications ☐

   b. EXIT this procedure ☐

2. **IF** the release is being monitored by RM-A2 normal range or accident range, **THEN** READ the μCi/cc **AND** RECORD on the data sheet N/A ☐ ☐

9.2 Meteorological Data

NOTES: 1. The preferred source of meteorological data is the 33' Primary Tower (MMP-5). Alternate sources are the 175' Primary Tower and the 33' Alternate Tower (MMP-1) .... ☐
2. If OSI PI or the wind direction, or wind speed recorders are **NOT** in service, the appropriate meter may be observed for a brief period (possibly 30 seconds) to obtain an estimate .... ☐

1. READ the Sigma-Theta from OSI PI or the meter (MM-7-SI) on the alternate tower panel (or computer point W208) **AND** RECORD on the data sheet. .... ☐

2. **IF** Sigma-Theta is **NOT** available, **THEN** READ Delta T from OSI PI OR ESTIMATE the previous 15 minute average primary tower Delta T from the recorder (MM-21-TR) **AND** RECORD on the data sheet........ N/A ☐ ☐

NOTE: Alternate wind direction sources: Weather.com, Accuweather.com, visual indicators such as clouds, stacks, cooling towers, etc. may be used to determine wind direction........ ☐

3. READ average wind direction from OSI PI OR ESTIMATE the previous 15 minute average wind direction (direction wind is coming from) from the 33' recorder (MM-19-TR) **AND** RECORD on the data sheet .... ☐

NOTE: If the wind speed instrumentation is **NOT** available, the default wind speeds are 2 meters per second for a day release or 1 meter per second for a night release. (Meters per second x 2.24=mph.)........ ☐

4. READ average wind speed miles per hour from OSI PI OR ESTIMATE the previous 15 minute average wind speed meters per second from the 33' recorder (MM-19-TR) **AND** RECORD on the data sheet........ ☐

   a. **IF** meters per second is used, **THEN** MULTIPLY the meters per second from Step 9.2.4 by 2.24 to calculate miles per hour (mph) **AND** RECORD on the data sheet........ ☐

NOTE: If stability class **CANNOT** be determined, the default is the D & E group	...... ☐

5. Determine the stability class from the table below.

SIGMA-THETA (Degrees) (Primary method)	**DELTA T (Degrees F)** (Secondary method)	**STABILITY CLASS**
≥ 22.5	≤ -1.46	A (most dispersed plume)
< 22.5 to 17.5	-1.45 to –1.31	B
< 17.5 to 12.5	-1.30 to –1.16	C
< 12.5 to 7.5	-1.15 to –0.39	D
< 7.5 to 3.8	-0.38 to 1.15	E
< 3.8 to 2.1	1.16 to 3.07	F
< 2.1	≥3.08	G (most concentrated plume)

9.3 Dose Rate Tables

NOTE: For monitor readings that lie between two table values, the higher table value should be used.	...... ☐

1. **IF** in Extended Shutdown, **THEN** FIND the following doses on ENCLOSURE 1
2. Page 1 OF 2 (Table 1), corresponding to the μCi/cc and the stability class (from Step 9.2.5) **AND** RECORD on the data sheet .... N/A ☐ ☐
   a. Thyroid mR/hr .... __________
   b. Total (TEDE) mR/hr .... __________

3. **IF** in On-line Operations, **THEN** FIND the following doses on ENCLOSURE 1
4. Page 1 OF 2 (Table 2) corresponding to the μCi/cc and the stability class (from Step 9.2.5) **AND** RECORD on the data sheet. .... N/A ☐ ☐
   a. Thyroid mR/hr .... __________
   b. Total (TEDE) mR/hr .... __________

9.4 **Dose Calculation**

> **NOTE:** If the release duration **CANNOT** be estimated or determined, the default is one hour .... ☐

1. ESTIMATE the release/exposure duration **AND** RECORD on the data sheet .... ☐

> **CAUTIONS**
>
> 1. EM-202 Attachment 1 indicates a Site Area Emergency if dose assessment results indicate site boundary dose >100 mR TEDE or >500 mR Thyroid CDE for the actual or projected duration of the release.
> 2. EM-202 Attachment 1 indicates a General Emergency if dose assessment results indicate a site boundary dose of >1000 mR TEDE or >5000 mR Thyroid CDE for actual or projected duration of the release.

2. DETERMINE the Thyroid dose and Total (TEDE) dose by multiplying the dose for each by the release/exposure duration. RECORD on the data sheet .... ☐

9.5 **Additional Florida Nuclear Plant Emergency Notification Form Information**

1. DETERMINE the affected sectors on Enclosure 2, Determination of Affected Sectors corresponding to wind direction from 9.2.3 **AND** RECORD on the data sheet **AND** on the Florida Nuclear Plant Emergency Notification Form as indicated by the number in the right margin of the data sheet .... ☐

10.0 **RECORDS**

1. RECORD person's name performing the procedure, and the date and time initial assessment was made.
2. SUBMIT this procedure to the Emergency Coordinator for review and signature.
3. TRANSMIT all documentation and calculations created by this procedure to Records Management after appropriate reviews.

TABLE 1
EXTENDED SHUTDOWN: RM-A2 MONITOR

SITE BOUNDARY (0.83-MILE)
mREM PER 1-HOUR RELEASE & 6-HOUR EXPOSURE

	STABILITY CLASS A,B,C		STABILITY CLASS D,E		STABILITY CLASS F,G	
NG uCi/cc	THY mRem	TEDE mRem	THY mRem	TEDE mRem	THY mRem	TEDE mRem
1E-04	9.4E-02	1.3E-01	1.3E-01	1.9E-01	1.6E-01	2.2E-01
2E-04	1.9E-01	2.7E-01	2.6E-01	3.7E-01	3.3E-01	4.5E-01
4E-04	3.8E-01	5.3E-01	5.3E-01	7.4E-01	6.5E-01	8.9E-01
6E-04	5.7E-01	8.0E-01	7.9E-01	1.1E+00	9.8E-01	1.3E+00
8E-04	7.5E-01	1.1E+00	1.1E+00	1.5E+00	1.3E+00	1.8E+00
1E-03	9.4E-01	1.3E+00	1.3E+00	1.9E+00	1.6E+00	2.2E+00
2E-03	1.9E+00	2.7E+00	2.6E+00	3.7E+00	3.3E+00	4.5E+00
4E-03	3.8E+00	5.3E+00	5.3E+00	7.4E+00	6.5E+00	8.9E+00
6E-03	5.7E+00	8.0E+00	7.9E+00	1.1E+01	9.8E+00	1.3E+01
8E-03	7.5E+00	1.1E+01	1.1E+01	1.5E+01	1.3E+01	1.8E+01
1E-02	9.4E+00	1.3E+01	1.3E+01	1.9E+01	1.6E+01	2.2E+01
2E-02	1.9E+01	2.7E+01	2.6E+01	3.7E+01	3.3E+01	4.5E+01
4E-02	3.8E+01	5.3E+01	5.3E+01	7.4E+01	6.5E+01	8.9E+01
6E-02	5.7E+01	8.0E+01	7.9E+01	1.1E+02	9.8E+01	1.3E+02
8E-02	7.5E+01	1.1E+02	1.1E+02	1.5E+02	1.3E+02	1.8E+02
1E-01	9.4E+01	1.3E+02	1.3E+02	1.9E+02	1.6E+02	2.2E+02
2E-01	1.9E+02	2.7E+02	2.6E+02	3.7E+02	3.3E+02	4.5E+02
4E-01	3.8E+02	5.3E+02	5.3E+02	7.4E+02	6.5E+02	8.9E+02
6E-01	5.7E+02	8.0E+02	7.9E+02	1.1E+03	9.8E+02	1.3E+03
8E-01	7.5E+02	1.1E+03	1.1E+03	1.5E+03	1.3E+03	1.8E+03
1E+00	9.4E+02	1.3E+03	1.3E+03	1.9E+03	1.6E+03	2.2E+03
2E+00	1.9E+03	2.7E+03	2.6E+03	3.7E+03	3.3E+03	4.5E+03
4E+00	3.8E+03	5.3E+03	5.3E+03	7.4E+03	6.5E+03	8.9E+03
6E+00	5.7E+03	8.0E+03	7.9E+03	1.1E+04	9.8E+03	1.3E+04
8E+00	7.5E+03	1.1E+04	1.1E+04	1.5E+04	1.3E+04	1.8E+04
1E+01	9.4E+03	1.3E+04	1.3E+04	1.9E+04	1.6E+04	2.2E+04
2E+01	1.9E+04	2.7E+04	2.6E+04	3.7E+04	3.3E+04	4.5E+04
4E+01	3.8E+04	5.3E+04	5.3E+04	7.4E+04	6.5E+04	8.9E+04
6E+01	5.7E+04	8.0E+04	7.9E+04	1.1E+05	9.8E+04	1.3E+05
8E+01	7.5E+04	1.1E+05	1.1E+05	1.5E+05	1.3E+05	1.8E+05
1E+02	9.4E+04	1.3E+05	1.3E+05	1.9E+05	1.6E+05	2.2E+05
2E+02	1.9E+05	2.7E+05	2.6E+05	3.7E+05	3.3E+05	4.5E+05
4E+02	3.8E+05	5.3E+05	5.3E+05	7.4E+05	6.5E+05	8.9E+05
6E+02	5.7E+05	8.0E+05	7.9E+05	1.1E+06	9.8E+05	1.3E+06
8E+02	7.5E+05	1.1E+06	1.1E+06	1.5E+06	1.3E+06	1.8E+06
1E+03	9.4E+05	1.3E+06	1.3E+06	1.9E+06	1.6E+06	2.2E+06
2E+03	1.9E+06	2.7E+06	2.6E+06	3.7E+06	3.3E+06	4.5E+06
4E+03	3.8E+06	5.3E+06	5.3E+06	7.4E+06	6.5E+06	8.9E+06
6E+03	5.7E+06	8.0E+06	7.9E+06	1.1E+07	9.8E+06	1.3E+07
8E+03	7.5E+06	1.1E+07	1.1E+07	1.5E+07	1.3E+07	1.8E+07
1E+04	9.4E+06	1.3E+07	1.3E+07	1.9E+07	1.6E+07	2.2E+07
2E+04	1.9E+07	2.7E+07	2.6E+07	3.7E+07	3.3E+07	4.5E+07
4E+04	3.8E+07	5.3E+07	5.3E+07	7.4E+07	6.5E+07	8.9E+07
6E+04	5.7E+07	8.0E+07	7.9E+07	1.1E+08	9.8E+07	1.3E+08
8E+04	7.5E+07	1.1E+08	1.1E+08	1.5E+08	1.3E+08	1.8E+08
1E+05	9.4E+07	1.3E+08	1.3E+08	1.9E+08	1.6E+08	2.2E+08

Based on Calculation N12-0001.

TABLE 2
ON-LINE OPERATION: RM-A2 MONITOR

SITE BOUNDARY (0.83-MILE)
mREM PER 1-HOUR RELEASE & 6-HOUR EXPOSURE

	STABILITY CLASS A,B,C		STABILITY CLASS D,E		STABILITY CLASS F,G	
NG uCi/cc	THY mRem	TEDE mRem	THY mRem	TEDE mRem	THY mRem	TEDE mRem
1E-04	9.7E-02	1.7E-02	1.5E-01	2.3E-02	1.7E-01	2.4E-02
2E-04	1.9E-01	3.4E-02	3.0E-01	4.6E-02	3.3E-01	4.9E-02
4E-04	3.9E-01	6.8E-02	6.1E-01	9.3E-02	6.7E-01	9.7E-02
6E-04	5.8E-01	1.0E-01	9.1E-01	1.4E-01	1.0E+00	1.5E-01
8E-04	7.8E-01	1.4E-01	1.2E+00	1.9E-01	1.3E+00	1.9E-01
1E-03	9.7E-01	1.7E-01	1.5E+00	2.3E-01	1.7E+00	2.4E-01
2E-03	1.9E+00	3.4E-01	3.0E+00	4.6E-01	3.3E+00	4.9E-01
4E-03	3.9E+00	6.8E-01	6.1E+00	9.3E-01	6.7E+00	9.7E-01
6E-03	5.8E+00	1.0E+00	9.1E+00	1.4E+00	1.0E+01	1.5E+00
8E-03	7.8E+00	1.4E+00	1.2E+01	1.9E+00	1.3E+01	1.9E+00
1E-02	9.7E+00	1.7E+00	1.5E+01	2.3E+00	1.7E+01	2.4E+00
2E-02	1.9E+01	3.4E+00	3.0E+01	4.6E+00	3.3E+01	4.9E+00
4E-02	3.9E+01	6.8E+00	6.1E+01	9.3E+00	6.7E+01	9.7E+00
6E-02	5.8E+01	1.0E+01	9.1E+01	1.4E+01	1.0E+02	1.5E+01
8E-02	7.8E+01	1.4E+01	1.2E+02	1.9E+01	1.3E+02	1.9E+01
1E-01	9.7E+01	1.7E+01	1.5E+02	2.3E+01	1.7E+02	2.4E+01
2E-01	1.9E+02	3.4E+01	3.0E+02	4.6E+01	3.3E+02	4.9E+01
4E-01	3.9E+02	6.8E+01	6.1E+02	9.3E+01	6.7E+02	9.7E+01
6E-01	5.8E+02	1.0E+02	9.1E+02	1.4E+02	1.0E+03	1.5E+02
8E-01	7.8E+02	1.4E+02	1.2E+03	1.9E+02	1.3E+03	1.9E+02
1E+00	9.7E+02	1.7E+02	1.5E+03	2.3E+02	1.7E+03	2.4E+02
2E+00	1.9E+03	3.4E+02	3.0E+03	4.6E+02	3.3E+03	4.9E+02
4E+00	3.9E+03	6.8E+02	6.1E+03	9.3E+02	6.7E+03	9.7E+02
6E+00	5.8E+03	1.0E+03	9.1E+03	1.4E+03	1.0E+04	1.5E+03
8E+00	7.8E+03	1.4E+03	1.2E+04	1.9E+03	1.3E+04	1.9E+03
1E+01	9.7E+03	1.7E+03	1.5E+04	2.3E+03	1.7E+04	2.4E+03
2E+01	1.9E+04	3.4E+03	3.0E+04	4.6E+03	3.3E+04	4.9E+03
4E+01	3.9E+04	6.8E+03	6.1E+04	9.3E+03	6.7E+04	9.7E+03
6E+01	5.8E+04	1.0E+04	9.1E+04	1.4E+04	1.0E+05	1.5E+04
8E+01	7.8E+04	1.4E+04	1.2E+05	1.9E+04	1.3E+05	1.9E+04
1E+02	9.7E+04	1.7E+04	1.5E+05	2.3E+04	1.7E+05	2.4E+04
2E+02	1.9E+05	3.4E+04	3.0E+05	4.6E+04	3.3E+05	4.9E+04
4E+02	3.9E+05	6.8E+04	6.1E+05	9.3E+04	6.7E+05	9.7E+04
6E+02	5.8E+05	1.0E+05	9.1E+05	1.4E+05	1.0E+06	1.5E+05
8E+02	7.8E+05	1.4E+05	1.2E+06	1.9E+05	1.3E+06	1.9E+05
1E+03	9.7E+05	1.7E+05	1.5E+06	2.3E+05	1.7E+06	2.4E+05
2E+03	1.9E+06	3.4E+05	3.0E+06	4.6E+05	3.3E+06	4.9E+05
4E+03	3.9E+06	6.8E+05	6.1E+06	9.3E+05	6.7E+06	9.7E+05
6E+03	5.8E+06	1.0E+06	9.1E+06	1.4E+06	1.0E+07	1.5E+06
8E+03	7.8E+06	1.4E+06	1.2E+07	1.9E+06	1.3E+07	1.9E+06
1E+04	9.7E+06	1.7E+06	1.5E+07	2.3E+06	1.7E+07	2.4E+06
2E+04	1.9E+07	3.4E+06	3.0E+07	4.6E+06	3.3E+07	4.9E+06
4E+04	3.9E+07	6.8E+06	6.1E+07	9.3E+06	6.7E+07	9.7E+06
6E+04	5.8E+07	1.0E+07	9.1E+07	1.4E+07	1.0E+08	1.5E+07
8E+04	7.8E+07	1.4E+07	1.2E+08	1.9E+07	1.3E+08	1.9E+07
1E+05	9.7E+07	1.7E+07	1.5E+08	2.3E+07	1.7E+08	2.4E+07

Based on Calculation N-12-0001.

DETERMINATION OF AFFECTED SECTORS

WIND DIRECTION (°FROM)	AFFECTED SECTORS
349-11	H J K
12-33	J K L
34-56	K L M
57-78	L M N
79-101	M N P
102-123	N P Q
124-146	P Q R
147-168	Q R A
169-191	R A B
192-213	A B C
214-236	B C D
237-258	C D E
259-281	D E F
282-303	E F G
304-326	F G H
327-348	G H J

DATA SHEET

STEP #	RAD AND MET MONITOR DATA			FLORIDA NUCLEAR PLANT EMERGENCY NOTIFICATION FORM
9.1.2	RM-A2 NORMAL RANGE GAS CHANNEL		µCi/cc	
	RM-A2 ACCIDENT RANGE GAS CHANNEL		µCi/cc	
9.2.1 or 9.2.2	SIGMA-THETA (from OSI PI or meter)		DEGREES (1)	
	DELTA T (from OSI PI or recorder)		DEGREES F (2)	
9.2.3	AVERAGE WIND FROM (33') (from OSI PI or recorder)		DEGREES (2)	8A
9.2.4 or 9.2.4	AVERAGE WIND SPEED (33') (from OSI PI or U3W720)		MPH	13A
	AVERAGE WIND SPEED (33') (from recorder)		M/SEC (2)	
9.2.4.a	WIND SPEED MPH=M/SEC X 2.24		MPH	13A
9.2.5	STABILITY CLASS			13B
	The source for OSI PI data is the CR3 Qualified Displays (QPIM), EP Tab, Met Tower page. (1) Meter displays a rolling 15 minute average, so the current instantaneous value should be used. (2) 15 minute average from chart recorder on meteorological panel.			

DATA SHEET

STEP#	SITE BOUNDARY DOSE INFORMATION		FLORIDA NUCLEAR PLANT EMERGENCY NOTIFICATION FORM
9.3.1 or 9.3.2	THYROID mRem	TEDE mRem	
9.44.1	PROJECTED RELEASE DURATION ________________ HOURS. (If duration can't be estimated, assume 1 hour.)		
9.44.2	DOSE = (TABLE DOSE X DURATION HOURS)		14B, 14C
	THYROID mRem	TEDE mRem	
9.5.1	AFFECTED SECTORS ________, ________, ________, ________ (three minimum)		8B
10.0.1	____________________________________ Performed by Date/Time		
10.0.2	____________________________________ Emergency Coordinator Date/Time		

Summary of Changes

PRR 513744 (October, 2012)

NOTES:	1. Procedure Sponsor: Ensure that any changes to EM-204A that affect information contained in ERF posters, Enclosures, briefing cards, guidelines, etc. are made to those items as well. 2. Procedure Sponsor: Changes to certain parts of EM-204A may impact other EPIPs. Specifically, if any changes are made to section 3.0 definitions, review EM-204B. Ensure appropriate PRRs are initiated as needed.

SECTION/STEP	CHANGE
Throughout	Reformatted per PRO-NGGC-0201.
Throughout	NOCS Commitment 1029, 13140 have been superseded by NOCS Commitment 100442 changed references accordingly.
1.0.2	Added new section stating that this is an Emergency Plan Implementing Procedure. Any revisions must be carefully considered for Emergency Plan impact. (PRR 321575)
2.1.6	Added new development reference Calculation N12-0001, Calculation of RM-A1 and RM-A2 Threshold Values for Emergency Action Levels.
2.2	New Implementing References section
3.0.5	Added new definition of extended shutdown. Renumbered the remaining definitions.
6.0.1	Clarified that due to the low probability of G stability class, F dispersion factor was used for the F, G category.
6.0.2	Clarified that per assumptions of Calculation N12-0001, a 30-minute decay was assumed.
9.0.1	Changed to make one step for clarification
9.1.1	Deleted conditional "IF RM-A2 low-range gas channel is not in on-scale..." and changed to "IF RM-A2 is not operating as designed..."
9.1.1.b	Replaced State Warning Point with State Watch Office.

9.1.3	Deleted step about manually switching range controller to auto as this is unnecessary with the new monitor.
9.2.1	In NOTE 1, changed MMP-3 to MMP-5 for the current met tower. (PRR 400545) In NOTE 2, acknowledged that OSI PI is a source for Sigma Theta. Added OSI PI as an option for Sigma Theta. Deleted "The Sigma-Theta meter reading is based on the past 15 minutes of data. Therefore, the instantaneous reading can be used." as this information was already in the definition of Sigma-Theta.
9.2.2	Added OSI PI as an option for Delta T.
9.2.3 Note	Added internet sources for wind direction.
9.2.3	Added OSI PI option for reading wind direction.
9.2.4	Added OSI PI option for reading wind speed miles per hour and clarified recorder provides wind speed meters per second.
9.2.4.a	Made this step a conditional to convert meters per second to miles per hour.
9.3	Revised conditional steps to address plant condition rather than monitor range. The new normal and accident ranges both read in µCi/cc and so both ranges are covered in each table. Table 1 now address Extended Shutdown and Table 2 now addresses On-line Operations.
9.4	Deleted step and renumbered remaining in the section.
9.4.2 New	Removed reference to DDE dose and dose rate as it's no longer required for Emergency Notification Form. Removed reference to correcting for wind speed as the dose tables now assume the wind speed that produces the maximum dose.
9.5.1	Deleted step and renumbered remaining step in this section. This information is no longer required for Emergency Notification Form. (PRR 303053)
9.5.1 New	Added to also record affected sectors on the Florida Nuclear Plant Emergency Notification Form as indicated by the number in the right margin of the data sheet.
Encl 1 old	Old Enclosure 1 (Data Sheet) became Attachment 1 with

	reformatting. Both pages of the data sheet were significantly revised due to reformatting, characteristics of the new monitors, adding references to OSI PI (PRR 236109), and changes to data required on the Emergency Notification Form (PRR 455853).
Encl 2 old	Old Enclosure 2 (dose tables) became Enclosure 1 with reformatting. RM-A2 has been replaced (EC 76363). The tables now use the new monitor units µCi/cc (vs. cpm and mR/hr) and ranges (PRRs 513744, 514424). The tables doses are now in mRem (vs. mR/hr). The dose tables now reflect the methodology in Calculation N12-0001 for determining EAL 1.4 radiation monitor threshold values. Factors in the dose tables that convert noble gas µCi/cc to TEDE dose and Thyroid CDE dose have been revised based on using RASCAL 3.0.5 to determine the noble gas concentration required to yield PAG doses (PRR 219403). The dose tables were previously based on the RADDOSE-IV dose model.
Encl 3 old	Old Enclosure 3 became Enclosure 2 with reformatting.
Summary of Changes	Added standard notes for Emergency Plan Implementing Procedures. (PRR 411898)

Progress Energy

R
Reference
Use

CRYSTAL RIVER UNIT 3

PLANT OPERATING MANUAL

EMERGENCY PLAN IMPLEMENTING PROCEDURE

EM-204B

OFF-SITE DOSE ASSESSMENT DURING RADIOLOGICAL EMERGENCIES FOR MONITORED RELEASES – MIXTURES

(USER INSTRUCTIONS FOR RASCAL)

TABLE OF CONTENTS

1.0 PURPOSE

1.1 The RASCAL Computer model provides a method to evaluate the magnitude of a radiological release from CR3 and to estimate offsite exposure. This procedure contains instructions for RASCAL use for the Monitored Releases – Mixtures Source Term Option. EMG-NGGC-0002, Off-Site Dose Assessment, should be used when dose estimates are to be performed using any other RASCAL option.

1.2 In addition this procedure also:
- Provides worksheets for determination of the Curie/sec release rate inputs to RASCAL
- Provides methods for obtaining meteorological information

1.3 This procedure is an Emergency Plan Implementing Procedure. Any revisions must be carefully considered for Emergency Plan impact.

2.0 REFERENCES

2.1 Developmental References

2.1.1 NUREG–1889 RASCAL 3.0.5 Workbook

2.1.2 NUREG/CR-5247/PNL-8454, Vol. 1, Rev. 1, RASCAL User's Guide

2.1.3 CR3 Radiological Emergency Response Plan (RERP)

2.1.4 EM-202, Duties of Emergency Coordinator

2.1.5 EM-219, Duties of the Dose Assessment Team

2.1.6 EPA-400-R-92-001, Manual of Protective Action Guides and Protective Actions for Nuclear Incidents, Environmental Protection Agency (October, 1991)

2.1.7 NUREG/BR-0150, Vol. 1, Rev. 4, RTM-96 Response Technical Manual

2.1.8 Final Safety Analysis Report

2.1.9 Engineering Evaluation EEF-00-009, Rev. 1 – Radiation Monitor Response Factors

2.1.10 EMG-NGGC-0002, Off-Site Dose Assessment

3.0 PERSONNEL INDOCTRINATION

3.1 Definitions

3.1.1 **Delta T** – A measurement of the difference in air temperature between two different elevations above ground level. The value provides a measure of the atmospheric stability.

3.1.2 **Deposition** - A means of puff depletion that deposits particulate radioactive material on the ground.

3.1.3 **Partitioning** – Reduction of non-noble gases when the steam generator tube leak is below the water level of the generator. This is highly unlikely at CR3 because most of the tube length in a once-through steam generator is above the secondary level.

3.1.4 **Sigma-Theta** - The standard deviation of a set of wind range measurements. The Sigma-Theta meter automatically calculates and displays the standard deviation of wind range for the previous 15 minutes.

3.1.5 **Stability Class** - A lettering system from A to G to designate certain atmospheric conditions which affect the dispersion of the plume. Class A indicates rapid dispersion, less concentrated plume (unstable conditions). Class G indicates slow dispersion, more concentrated plume (stable conditions).

3.2 Responsibilities

3.2.1 The TSC Accident Assessment Team and the EOF Technical Support Team are responsible for ensuring the Dose Assessment Team is aware of plant conditions related to offsite dose projections. The EOF Radiation Controls Manager augments this responsibility.

3.2.2 The Dose Assessment Team is responsible for the implementation of this procedure.

3.3 Limits & Precautions

3.3.1 Protective Action Guideline (PAG) doses from the Environmental Protection Agency are 1 REM TEDE and 5 REM Thyroid at the site boundary (0.83 miles) or beyond. The risk verses benefit of public protective actions is based on the assumption that protective action will prevent members of the public from receiving the full PAG doses. Public dose previous to the protective action should NOT be considered. The protective action should be based on members of the public avoiding doses of 1 REM TEDE and 5 REM Thyroid. EM-202 Enclosure 1 specifies conditions in which site boundary dose or dose rate may require declaration of a Site Area Emergency or General Emergency.

3.3.2 Detailed instructions, notes, and cautions are provided on various screens depending on input and parameters. Help screens are also available in various options in RASCAL.

3.3.3 In a station blackout, the following instrumentation is available:

- RM-Gs 1, 3, 5, 7, 9, 11, 25, 26, 27, 28, 29, 30.
- RM-Ls 2, 7.
- Primary Meteorological Tower local (at the tower) readouts only.
- RM-A1, RM-A2 control room display units are powered, but pumps, skids, and detectors are NOT.

3.3.4 Recorder AH-1003-TIR Channel 4 indicates total Reactor Building stack flow. AH-294-FT measures Reactor Building purge flow rate only and does NOT include make up flow.

3.3.5 Enclosure 1 provides dose estimates that may be compared with dose projections as a credibility check. However, actual dose rates could vary by orders of magnitude depending on plant conditions.

3.3.6 The option of "Not an Isolated Stack" is normally chosen because the release points at CR3 do NOT meet the height and separation distance requirements of an isolated release point.

4.0 INSTRUCTIONS

4.1 Dose Projection Based on RASCAL Monitored Releases – Mixtures

4.1.1 **IF** the dose assessment computer fails,
THEN CONSIDER the following alternatives:

- **USE** another computer with the standard desktop.
- **OBTAIN** dose projection data from the other facility (TSC or EOF) as appropriate.
- **USE** EM-204(A) as backup dose assessment.
- **CONTACT** Nuclear Computing Services personnel.

NOTE

The TSC and EOF dose assessment computers have an icon installed on the Start Bar. On other computers, the program is located at Start, Programs, Emergency Preparedness, RASCAL.

4.1.2 **LOG ON** to computer using your user name and password.

4.1.3 **START** RASCAL by double clicking on the RASCAL icon on the desktop or accessing through Start Programs.

4.1.4 **SELECT** OK to acknowledge the initial screen showing the version and vendor information.

4.1.5 **SELECT** Source Term to Dose (STDose).

4.1.6 **SELECT** Event Type. The Nuclear Power Plant option radio button will be shown as selected.

4.1.6.1 **SELECT** the "OK" button and a ✓mark will appear in front of Event Type when this step is complete.

4.1.7 **SELECT** Event Location. SELECT "Crystal River – Unit 3", using the drop down menu.

4.1.7.1 **SELECT** the "OK" button and a ✓mark will appear in front of Event Location when this step is complete.

4.1.8 **SELECT** "Source Term" and the "Source Term Options for Nuclear Power Plant screen will be displayed.

4.1.9 **SELECT** "Monitored Releases – Mixtures" AND OK. The Monitored Release – Mixtures (Source Term) screen will be displayed. If any other Source Term option is selected, use EMG-NGGC-0002.

4.1.10 **SELECT** Yes or No for Reactor shutdown status.

4.1.10.1 IF the Reactor is shutdown, THEN ENTER date AND time of shutdown, otherwise SELECT NO.

4.1.11 **ENTER** the current date and time the monitor reading was taken as the sample taken date and time. This will be the release start time.

4.1.12 **DETERMINE** appropriate radiation monitor reading to use for duration of the release.

4.1.12.1 CONSULT with EOF members as necessary to determine appropriate radiation monitor value to use

4.1.13 USE either the worksheets in Enclosure 1, Release Rate Worksheets, or an approved software application (Release Rate EXCEL spreadsheet located in L:\Shared\RASCAL\CR3 Release Rate Worksheets.xlsx) to calculate the noble gas, iodine, and particulate release rates based on radiation monitor reading.

4.1.13.1 **IF** approved software application is used, **THEN**:

- **PRINT** output
- **ENTER** name
- **ENTER** date **AND KEEP** with dose projection that will be generated for validation purposes.

4.1.14 **ENTER** the release rates in Ci/sec for:

4.1.14.1 Noble gases

4.1.14.2 Iodines

4.1.14.3 Particulates

4.1.15 **SELECT** "OK" button on the screen to return to the primary screen for building a dose projection. A ✓mark will appear in front of Source Term when this step is complete.

4.1.16 **SELECT** "Release Path" and the "Release Direct to Atmosphere" screen will be displayed.

4.1.17 **SELECT** the following Release point characterizations:

- Not an isolated stack
- Release height - 0
- Consider building wake effects - Yes

4.1.18 **ENTER** date and time for start of release to atmosphere. This should be the same as the sample taken time, which is the time the monitor reading was taken.

4.1.19 **CONSULT** with Technical Support or Accident Assessment team members as necessary to determine expected duration for the release. If no better information is available a default release time of 6 hours may be used to be consistent with the default calculation time.

4.1.20 **ENTER** date and time for end of release to atmosphere OR ENTER release duration. At this time, RASCAL stops the release from the plant, but will continue to calculate dose from radioactive material already released, up to the calculation end time.

4.1.21 **SELECT** OK to complete Release Direct to Atmosphere release path entry. A ✓mark will appear in front of Release Path when this step is complete.

NOTE

OSI PI EP tab average wind speed and average wind direction is the preferred source.

4.1.22 **OBTAIN** meteorological data from the Control Room or by using the plant computer. **IF** unavailable from these sources, **THEN REFER** to Enclosure 2 for alternate means of obtaining meteorological information.

4.1.23 **USE** the following priority when collecting wind speed and wind direction:

1. 33' Primary Tower
2. 175' Primary Tower
3. 33' Alternate Tower (only source for Sigma-Theta, precipitation rate).

4.1.24 **IF** Sigma-Theta is **NOT** available,
THEN USE Delta T or the wind range to establish the stability class.
REFER to Enclosure 2, for use of wind range.

4.1.25 **IF** Control Room instrumentation is used to obtain meteorological data,
THEN ENSURE that values for wind speed, wind direction, and wind range are determined using the average of the previous 15 minutes.

4.1.26 **RECORD** meteorological data on Enclosure 3 or similar data recording form.

4.1.27 **SELECT** the "Meteorology" button on the RASCAL screen.

4.1.28 **SELECT** "Actual Observations and Forecasts" for Data set type **IF** available, **OTHERWISE PROCEED** to step 4.1.40.

4.1.29 **SELECT** "Create New." The Meteorological Data Processor screen is displayed.

4.1.30 **SELECT** "Enter Data." The Data Entry screen is displayed.

NOTE

Precipitation, air temperature, air pressure, and dew point are optional parameters.

4.1.31 **ENTER** at least the following data:

- **SELECT** OBS for observed **OR** FCST for forecast data type as appropriate.
- Date

NOTE
RASCAL rounds meteorological data times to the nearest quarter hour. If the meteorological data time is the same as the release time and both are in the second half of a quarter hour (e.g., 0008-0014), RASCAL rounds the meteorological data time to 0015 which is after the release time and causes an error. To prevent this error, meteorological time should be adjusted to the first half of the quarter hour (e.g., 0000-0007).

- Time
- 15-minute average wind direction 'from' in degrees
- 15-minute average wind speed in mph
- Stability class
- Precipitation (in/hr categories on dropdown: no precip<0.01, nlgt rain 0.01 to 0.04, rain >0.04 to 0.2, hvy rain >0.2)

4.1.32 **IF** data from additional times will be used,
THEN CLICK the "Add Records" button **AND ENTER** the data on the new row.

4.1.33 **IF** optional data from additional locations will be used,
THEN CLICK the "Next Station" button **AND ENTER** the data.

4.1.34 **SELECT** OK.

4.1.35 **SELECT** "Save and Process Data."

4.1.36 **ENTER** a name for the met data set (e.g., 0800, projection #1). The other options on this screen may be left as is.

4.1.37 **SELECT** OK.

4.1.38 **SELECT** "Return" (button on screen, **NOT** the keyboard Enter key).

4.1.39 **ENSURE** the data set that was just named is highlighted in the available data sets box **AND SELECT** OK to complete Meteorology entry. A ✓mark will appear in front of Meteorology when this step is complete.

4.1.40 **IF** no actual observation or forecast data is available **SELECT** Predefined Data (Non Site-specific), **THEN,**
SELECT the available data set that most resembles current weather conditions at the site.

4.1.41 **SELECT** OK and a ✓ mark will appear in front of Meteorology when this step is complete

4.1.42 **SELECT** the "Calculate Doses" button and the Start the Calculations screen will be displayed.

4.1.43 **SELECT** "Close-in + out to 10 miles" AND "User defined" for Distance of Calculations.

4.1.44 **SELECT** "Set Close Distances."

4.1.45 **REPLACE** 1.000 mile with 0.83 miles for Non-UF6 releases (CR3 site boundary).

4.1.46 **SELECT** OK

NOTE
Item 14 of the Florida Nuclear Plant Emergency Notification Form has blanks for the projection duration and CDE and TEDE dose at the site boundary, 2, 5, and 10 miles. A projection duration that will advance the plume past 10 miles is required. The RASCAL default of six hours is sufficient to estimate the 10 mile dose for most wind speeds, but not wind speeds less than 1.7 mph.

4.1.47 ENTER the number of hours (from 1 to 48 starting at release time) that is desired for this projection OR ENTER date and time to end the projection for "End Calculations at."

4.1.48 **ENTER** a case description (e.g., SGTR, 0800, projection #1).

4.1.49 **SELECT** OK to start STDose calculations.

4.2 Displayed and Printed Reports

4.2.1 **SELECT** "Dose to 10 miles" for Value displayed to view the Maximum Dose Values (rem) – To 10 miles (normally 3 to 10 miles).

4.2.2 **SELECT** "Close-in dose" for Value displayed to view to Maximum Dose Values (rem) – Close in (normally 0.1 to 2 miles and will include the CR3 site boundary dose at 0.83 miles).

NOTE
The appropriate printer must be selected in the Print Setup option under the RASCAL File menu. This is in addition to the Windows default printer selection.

4.2.3 With the Maximum Dose Values Tab selected, **SELECT** the "Print" button to print the Summary Report which includes the Close-in dose and Dose to 10 miles for TEDE and thyroid CDE (and other dose types) and the Case Summary. This should be sufficient for the EC/EOF Director to determine EAL and PAR impacts as discussed in section 4.5.

4.2.4 **SELECT** the "Detailed Results" button for the menu of numerous options for display formats and result types. **REFER TO** section 4.4 for more information on some of these options that can be used for field measurement comparison including calculating results for user-defined locations (Special Receptors).

4.3 Saving the Case

4.3.1 **SELECT** the "Save Case" button to save the projection for later review or revision. **CONSIDER** the following:

- The user creates the file name and RASCAL saves the projection file with a .STD extension.
- The default folder for saved cases is C:\Program Files\Rascal3\Save Case\.
- The user may create a new subfolder for a group of projections for the current event. The recommended location for a new subfolder on the C: drive is under the default Save Case folder.
- The case may be saved to a shared folder to allow both the TSC and EOF access. The recommended location for a shared folder is L:\Shared\RASCAL\. The folder may have to be created. Subfolders may be created here as well to organize projections.

4.4 Emergency Action Levels (EALs) and Protective Actions Recommendations (PARs)

> **NOTE**
>
> RASCAL does **NOT** accumulate doses from previous projections. Site boundary dose EALs are accumulated dose. Previous dose projections may have to be added to the current projection to determine if an EAL has been met.

4.4.1 **IF** dose projections equal or exceed 100 mREM TEDE or 500 mREM Thyroid at the site boundary (0.83 miles),
THEN NOTIFY the facility lead (EC or EOF Director) to review EALs.

> **NOTE**
>
> Offsite doses for accidents with **NO** fuel damage are **NOT** likely to exceed 1 REM TEDE or 5 REM Thyroid. Section 3.3.1 provides additional information on these dose limits.

4.4.2 **IF** dose projections equal or exceed 1 REM TEDE or 5 REM Thyroid at the site boundary (0.83 miles),
THEN NOTIFY the facility lead (EC or EOF Director) to review EALs and PARs.

4.5 Documentation

4.5.1 **FORWARD** all documentation to the EOF Radiation Controls Manager for review as time permits.

4.5.2 **TRANSMIT** the documentation to Document Services under EM-204(B).

RELEASE RATE WORKSHEETS

1.0 METHODS FOR DETERMINING RELEASE RATES

This Enclosure contains worksheets used in conjunction with the Monitored Release – Mixtures source term for calculating the release rate in Ci/sec from the sources listed below. Worksheet data, assumptions and calculations should be verified by a second person.

Worksheet 1: Noble Gas Release Rate from RB Purge Exhaust Duct Based on RM-A1
This worksheet is NOT necessary if Ci/sec has been read directly from OSI PI. The use of Worksheets 6 & 7 to calculate iodine and particulate release rates is required.

Worksheet 2: Noble Gas Release Rate from AB/FH Exhaust Duct Based on RM-A2.
This worksheet is NOT necessary if Ci/sec has been read directly from OSI PI. The use of Worksheets 6 & 7 to calculate iodine and particulate release rates is required.

Worksheet 2A: Noble Gas Release Rate from AB/FH Exhaust Duct based on RM-A4 or RM-A8 or the AB vent temporary monitor. This worksheet is intended for use during the extended shutdown that began in 2009 for periods when RM-A2 is not available. Also requires the use of Worksheet 7 to calculate particulate release rates. Worksheet 6 is NOT necessary. All iodines have decayed.

- Use of the temporary monitor also requires RM-A4 or RM-A8 data to establish its μCi/cc per mR/hr conversion factor.
- Use of OSI PI trending may be needed to find on-scale RM-A4 or RM-A8 data concurrent with temporary monitor.
- After the temporary monitor conversion factor is established, carry it forward to subsequent uses of the worksheet after RM-A4 and/or RM-A8 are offscale.
- To access temporary monitor: NGG OSI-PI Displays, CR3 General, O&S tab, All GEDDS Tags on PI – Trend 1, scroll to 143FT AB EXH DUCT LOCATION 1 and 2 (U3GRP_OCB 101 and U3GRP_OCB 102).

Worksheet 3: Noble Gas Release Rate from Containment Based on RM-G29/30.
Also requires the use of Worksheets 6 & 7 to calculate iodine and particulate release rates. Containment Radiation Monitor source term may be a better choice.

Worksheet 4: Manual Source Term Calculation

Worksheet 5: Noble Gas Release Rate from Main Steam Safeties or ADV's Based on RM-G25 or RM-G28.
Also requires the use of Worksheets 6 & 7 to calculate iodine and particulate release rates. Ultimate Core Damage source term may be a better choice.

Worksheet 6: Iodine Release Rate Based on Iodine/Noble Gas Ratios
(I/NG ratio and recommended Iodine decontamination factors) This worksheet is not necessary in extended shutdown conditions as all iodines have decayed.

Worksheet 7: Particulate Release Rate Based on Particulate/Noble Gas Ratios
(Part/NG ratio and recommended particulate decontamination factors)

Worksheet 8: Noble Gas Release Rate Based on Onsite Plume Measurement
(Uses release elevation, wind speed, and decay time to covert plume dose rate to release rate)

Worksheet 1
Noble Gas release rate from RB Purge Exhaust Duct based on RM-A1

1.1 **IF** Ci/sec is already known, **THEN** GO TO Worksheets 6 and 7 to determine the Iodine and Particulate Release Rates.

1.1.1 **DETERMINE** Noble Gas release rate from AB/FH Exhaust Duct based on RM-A1.
Excel spreadsheet L:\Shared\RASCAL\ CR3 Release Rate Worksheets.xlsx also performs these calculations.

INPUT DATA				
A. Rx Shutdown Date:		Time:		
B. Met/Rad Data: Date:		Time:		
C. Projection Time Period:	From:		To:	
D. RM-A1 Normal Range Gas Reading: or			µCi/cc	
E. RM-A1 Accident Range Gas Reading: ~~or~~			µCi/cc	
F. RB Exhaust Flow - from AH-1003-TIR Channel 4 or 50,000 CFM default:				CFM
RELEASE RATE ESTIMATE				
H. Time since Rx shutdown (B-A)			hours	

Noble Gas
Release Rate = ______________ x ________________ x 4.7E-4* = ___________
Ci/sec D, E F Ci/sec

* 4.7E-4 = 472 cc/sec per cfm x 1E-6 Ci/µCi

Go to Worksheets 6 and 7 to determine the Iodine and Particulate Release Rates

Completed by: ______________________ Date/Time: __________________

Verified by: ________________________ Date/Time: __________________

Worksheet 2
Noble Gas release rate from AB/FH Exhaust Duct based on RM-A2

1.2 **IF** Ci/sec is already known, **THEN** GO TO Worksheets 6 and 7 to determine the Iodine and Particulate Release Rates.

1.2.1 **DETERMINE** Noble Gas release rate from AB/FH Exhaust Duct based on RM-A2.
Excel spreadsheet L:\Shared\RASCAL\ CR3 Release Rate Worksheets.xlsx also performs these calculations.

INPUT DATA				
A. Rx Shutdown Date:		Time:		
C. Met/Rad Data: Date:		Time:		
C. Projection Time Period:	From:		To:	
D. RM-A2 Normal Range Gas Reading: or			µCi/cc	
E. RM-A2 Accident Range Gas Reading: ~~or~~			µCi/cc	
F. AB Exhaust Flow - from AH-1003-TIR Channel 3 or 156,000 CFM default:				CFM
RELEASE RATE ESTIMATE				
H. Time since Rx shutdown (B-A)			hours	

Noble Gas
Release Rate = ______________ x ________________ x 4.7E-4* = ____________
Ci/sec D, E F Ci/sec

* 4.7E-4 = 472 cc/sec per cfm x 1E-6 Ci/µCi

Go to Worksheets 6 and 7 to determine the Iodine and Particulate Release Rates

Completed by: ______________________ Date/Time: __________________

Verified by: ________________________ Date/Time: __________________

Worksheet 2A
Noble Gas release rate from AB/FH Exhaust Duct based on RM-A4, RM-A8, Temporary Monitor

Excel spreadsheet L:\Shared\RASCAL\ CR3 Release Rate Worksheets.xlsx also performs these calculations.

<table>
<tr><th colspan="5">INPUT DATA</th></tr>
<tr><td>A. Rx Shutdown Date:</td><td></td><td>Time:</td><td colspan="2"></td></tr>
<tr><td>B. Met/Rad Data: Date:</td><td></td><td>Time:</td><td colspan="2"></td></tr>
<tr><td>C. Projection Time Period:</td><td colspan="2">From:</td><td colspan="2">To:</td></tr>
<tr><td>D. RM-A4 Gas Reading: or</td><td colspan="2"></td><td colspan="2">cpm</td></tr>
<tr><td>E. RM-A8 Gas Reading: or</td><td colspan="2"></td><td colspan="2">cpm</td></tr>
<tr><td>F. Temporary Monitor Reading:</td><td colspan="2"></td><td colspan="2">mR/hr</td></tr>
<tr><td colspan="3">G. AB Exhaust Flow - from AH-1003-TIR Channel 3 or 156,000 CFM default:</td><td></td><td>CFM</td></tr>
<tr><th colspan="5">RELEASE RATE ESTIMATE</th></tr>
<tr><td colspan="5">H. Enter conversion factor:</td></tr>
<tr><td>RM-A4 or RM-A8 Gas - from Eff. Curve Slope = __________; calculate the inverse of slope or use 3.4E-8 μCi/cc per cpm as the default inverse→</td><td colspan="2">Inverse of slope</td><td colspan="2">μCi/cc per cpm</td></tr>
<tr><td colspan="3">Temporary Monitor: Refer to Worksheet 2A information on the first page of Enclosure 1 for guidance.

RM-A4 or RM-A8 μCi/cc from Eff. Curve _____________

______ ÷ ____________________ = _____________
μCi/cc Temporary Monitor mR/hr</td><td colspan="2">μCi/cc per mR/hr</td></tr>
</table>

Noble Gas
Release Rate = ____________ x ____________ x ___________ x 4.7E-4* = _____________
Ci/sec D, E, F G H Ci/sec

*4.7E-4 = 472 cc/sec per CFM x 1E-6 Ci/μCi

Go to Worksheet 7 to determine the Particulate Release Rate

Completed by: __________________________ Date/Time: ____________________

Verified by: _____________________________ Date/Time: ____________________

Worksheet 3
Noble Gas release rate from Containment based on RM-G29/G30

1.3 **DETERMINE** Noble Gas release rate from Containment based on RM-AG29/G30.

INPUT DATA			
A. Release Date:		Time:	
B. Projection Time Period	From:		To:
C. *RM-G29 Reading: or			R/hr
D. *RM-G30 Reading:			R/hr
E. RB Pressure:			psig
F. Estimated RB hole size:			in^2
G. RB Sprays: Circle one:	ON OFF		
RELEASE RATE ESTIMATE			
H. Circle Noble Gas Factor:	Sprays On - 0.02 µCi/cc per R/hr Sprays Off - 0.007 µCi/cc per R/hr		
I. Flow =	145 x (______ psig $)^{1/2}$ x ______ in^2 = ______ CFM*** (E) (F) (I)		

Noble Gas
Release Rate = ____________ x ____________ x ____________ x 4.7E-4* = ____________
Ci/sec (C or D) (H) (I) Ci/sec

Notes: * - The lower of the two RM-G readings is the preferred reading.
** - 4.7E-4 = 472 cc/sec per CFM x 1E-6 Ci/µCi
*** - If for a projected RB purge, directly enter the projected purge rate CFM
If for a "What if" the RB inventory is all released in 1 hr, enter 3E4 CFM

Go to Worksheets 6 and 7 to determine the Iodine and Particulate Release Rates

Completed by: ______________________ Date/Time: ________________

Verified by: ______________________ Date/Time: ________________

1.4 **DETERMINE** Steam Generator Tube Rupture release rates from Condenser based on RM-A2 OR from MSSV/ADV based on RCS Activity.

NOTE

Emergency Operating Procedures direct operators to continue to use both steam generators for RCS cooling until mode 5 is reached unless specific parameters are exceeded. These parameters are part of the Tube Rupture Alternate Control Criteria (TRACC) and involve RCS activity, BWST level, and OTSG level.

Steam will be directed to the condenser if available (vacuum established). Noble gases will be discharged from the condenser through the Auxiliary Building Ventilation and RM-A2.

If the condenser is NOT available, steam will be discharged through the Atmospheric Dump Valves.

1.4.1 **DETERMINE** whether the leaking OTSG is steaming to the condenser **OR** to the atmosphere during the release period.

1.4.2 **REFER** to the appropriate step below to develop source terms.

NOTE

Periodic steam releases through the Main Steam Safety Valves may occur immediately after a reactor trip.

The Control Room Dose Assessment Communicator may be able to track times the valves are open.

Computer points W354, W355, RECL114, RECL115 track ADVs percent open.

Downloading intervals of 1 minute or less over the period of the time step may be useful in determining minutes that the ADVs are open.

1.4.3 **IF** the leaking OTSG is steaming to the condenser **OR** for normal intermittent releases from MSSVs **PERFORM** the following steps as applicable.

1. **OBTAIN** RM-A2 Ci/sec **OR CALCULATE** noble gas Ci/sec using Worksheet 2 (Noble Gas release rate from AB/FH Exhaust Duct Based on RM-A2) of this enclosure entering RM-A2 μCi/cc and the Auxiliary Building Vent CFM.

2. **CALCULATE** iodine in Ci/sec using Worksheet 6 (Iodine Release Rates) of this enclosure.

3. **USE** the Monitored Release - Mixtures source term.

1.4.4 **IF** the leaking OTSG is steaming directly to the atmosphere **OR** for continuous releases from the ADVs or MSSVs **PERFORM** the following steps as applicable.

1. **USE** the Ultimate Core Damage State source term and Steam Generator Tube Rupture release path (quickest method).

2. **USE** the Coolant Sample source term and Steam Generator Tube Rupture release path if possible.

3. **USE** Radiation Monitors for releases via the ADV's with RM-G25/28 available and on-scale.

   a. **ESTIMATE** the Ci/sec of noble gases **USING** Worksheet 5.

   b. **DETERMINE** the Ci/sec iodine **USING** Worksheet 6 **AND** the Ci/sec particulate **USING** Worksheet 7.

   c. **ENTER** using the Monitored Release – Mixtures source term.

1.4.5 **CALCULATE** the source term manually.

1. **IF** RCS isotopic data is available **THEN** step 1.4.7 of this enclosure may be used to manually calculate the source term.

1.4.6 USE the additional information provided below as applicable.

- RM-G26 and RM-G27 are N-16 monitors calibrated to read in gallons per day at 100% power.
- RM-G25 and RM-G28 monitor release from the ADVs only, NOT from the MSSVs. The monitors will probably NOT detect normal reactor coolant.
- It is assumed that all noble gas activity leaking into the OTSG will be released via the AB stack (RM-A2), MSSVs / ADVs, or EFP-2.
- If core integrity is maintained, activity is based on the most recent RCS activity. RM-L1 may be used to scale this value as transients cause spikes in RCS activity.
- 1 gpm = 63 cc/s
- Maximum Leak Rate = 400 gpm (for one tube)
- Default Flow Rate through stuck open MSSV/ADV = 3E7 cc/sec = 6E4 cfm

Worksheet 4
Manual Source Term Calculation

1.4.7 The following worksheet can be used to manually calculate the source term. The release rate equals the primary-to-secondary Ci/sec.

1. For each nuclide identified in the RCS, **ENTER** the µCi/cc **AND CALCULATE** the Ci/sec using the equation in block G.

2. **ENTER** into RASCAL using the Effluent Isotopic Release Rate source term.

INPUT DATA			
A. Rx Shutdown Date:		Time:	
B. RCS sample: Date:		Time:	
C. P→S Leak Rate		Gallons per minute	
D. Fraction of time there are releases directly to Atmosphere on the affected OTSG (0 - 1).			
E. Release to atmosphere period:	From:	To:	

RCS SAMPLE DATA					
Nuclide	(F.) µCi/cc	Ci/sec	Nuclide	(F.) µCi/cc	Ci/sec

RELEASE RATE ESTIMATE

G.

Release Rate = ______ x ________ x ________ x 6.3E-5* = __________ Ci/sec
Ci/sec C D F

$$^{*}6.3E-5 = \left[\frac{1Ci}{1E6\mu Ci} \times \frac{3780\ cc}{1\ Gal} \times \frac{1\ min}{60\ sec}\right]$$

Completed by:	Date/Time:
Verified by:	Date/Time:

Worksheet 5
Noble Gas Release Rate from Main Steam Safeties or ADV's Based on RM-G25 or RM-G28

1.5 **DETERMINE** SGTR Noble Gas release rate from Main Steam Safeties or ADV's based on RM-G25 or RM-G28.

INPUT DATA			
A. Rx Shutdown Date:		Time:	
B. Met/Rad Data: Date:		Time:	
C. Projection Period:	From:	To:	
D. RM-G25 Reading: or		mR/hr - monitors A OTSG ADV line	
E. RM-G28 Reading:		mR/hr - monitors B OTSG ADV line	
F. Number of ADV/Safeties open on affected SG (1 - 9):			
G. Fraction of time releases in progress on affected OTSG (0 - 1):			
RELEASE RATE ESTIMATE			
H. Time since RX Shutdown (B - A):		hours	
I. Circle conversion factor:			
From 0 to 4 hours post Rx shutdown:	0.03 µCi/cc per mR/hr		
From 4 to 12 hours post Rx shutdown:	0.1 µCi/cc per mR/hr		
For ≥ 12 hours post Rx shutdown:	0.3 µCi/cc per mR/hr		

Noble Gas
Release Rate = ________ x ________ x ________ x _______ x 30* = _________ Ci/sec
D or E F G I

* 30 = Estimated flow of 3E7 cc/sec per open valve x 1E-6 Ci/µCi

Go to Worksheets 6 and 7 to determine the Iodine and Particulate Release Rates

Completed by: ______________________ Date/Time: __________________

Verified by: ________________________ Date/Time: __________________

Worksheet 6
Iodine Release Rates

1.6 **IF** in extended shutdown, **THEN** GO TO Worksheet 7 (all iodines are decayed).

1.6.1 **DETERMINE** Iodine release rate based on Iodine/Noble Gas Ratio. [NOCS 100442]

INPUT DATA		
Time Period:	From:	To:
Assumed Release Path from source (e.g., RCS) to release point:		
A. Noble Gas Release Rate:		Ci/sec
RELEASE RATE ESTIMATE		
B. Base Iodine/Noble Gas Ratio (circle the most appropriate value) 1. Underwater FHA (ratio at pool/cavity water surface) 1E-3 2. WGDTR (ratio at tank release) 1E-4 3. Normal Coolant (0-3 hours post shutdown) 1 4. Normal Coolant (>3 hrs post shutdown) 10 5. Gap Activity Released from Fuel 0.5 6. Melt Activity Released from Fuel 0.2		
C. Iodine Decontamination Factors (DF) - Enter DF for each removal mechanism that exists and **multiply** together to obtain overall DF:		
Partitioning (SGTR's: steam flashing under water) Unlikely with OTSGs as most tube length is above secondary level. Obtain AAT or TST advice. Default DF: Above water = 2 OR Below water = 50		______
Plateout (LOCA's or SGTR's in containment or OTSG): Always assume Default DF = 3		X ______
RB Sprays: Default DF = 10 for 0-2 hrs or 100 for >2 hrs of spray time		X ______
SGTR Release via condenser: Default DF = 20 (The condenser will not be available during loss of offsite power.)		X ______
RB/AB Exhaust filters: Default DF = 20 (Obtain AAT or TST advice for effectiveness under adverse conditions.). Min Aux Bldg efficiency threshold DF =8.		X ______
Product of all above DFs→		**Product =**______(C)

Iodine Release Rate

Ci/sec = ____________ X ____________ ÷ ____________ = ____________
A B C Ci/sec

Completed by: ______________________________ Date/Time: ______________________

Verified by: ______________________________ Date/Time: ______________________

Worksheet 7
Particulate Release Rates

1.7 DETERMINE Particulate release rate based on Particulate/Noble Gas Ratio.

INPUT DATA		
Time Period:	From:	To:
Assumed Release Path from source (e.g., RCS) to release point:		
A. Noble Gas Release Rate:		Ci/sec
RELEASE RATE ESTIMATE		
B. Base Particulate/Noble Gas Ratio (circle the most appropriate value) 1. Extended shutdown: pool drained, filtered release 2. Extended shutdown: pool drained, unfiltered release 3. Underwater FHA (ratio at pool/cavity water surface) 4. WGDTR (ratio at tank release) 5. Normal Coolant 6. Gap Activity Released from Fuel 7. Melt Activity Released from Fuel		0.03* 7* 1E-8 1E-8 0.1 0.2 0.1
C. Particulate Decontamination Factors (DF) - Enter DF for each removal mechanism that exists and **multiply** together to obtain overall DF:		
Extended shutdown: section C not applicable, product of DFs = 1		
Partitioning (SGTR's: steam flashing under water) Unlikely with OTSGs as most tube length is above secondary level. Obtain AAT or TST advice. Default DF: Above water = 2 OR Below water = 100		X ______
Plateout (LOCA's or SGTR's in containment or OTSG): Always assume Default DF = 10		X ______
RB Sprays: Default DF = 10 for 0-2 hrs or 100 for >2 hrs of spray time		X ______
SGTR Release via condenser: Default DF = 1000 (The condenser will not be available during loss of offsite power.)		X ______
RB/AB Exhaust filters: Default DF = 100		X ______
Product of all above DFs→		**Product =**______(C)

Particulate Release Rate

Ci/sec = ____________ X ____________ ÷ ____________ = ____________
A B C Ci/sec

* Extended shutdown assumed isotopic mix and base ratio factors determined by RASCAL 3.0.5.

Completed by: ______________________ Date/Time: ____________________

Verified by: ______________________ Date/Time: ____________________

Worksheet 8
Noble Gas Release Rate based on Onsite Plume Measurement

1.8 **DETERMINE** Noble Gas release rate based on Onsite Plume Measurement.

NOTE: This method assumes reasonable assurance that measured dose rates represent near maximum plume levels.

Date of Measurement: ______________ Time of measurement: ______________

____________ x ____________ x ____________ x ____________ = ____________

Measured Dose Rate (mR/hr)*	Conv. Factor	Wind Speed (m/sec)	Decay Corr.	**Noble Gas Release Rate (Ci/sec)**

Conv. Factor:

```
0.18 for Elevated **
Release (e.g. ADV)
0.03 for Ground Release
```

Decay Corr.:

Time Since Rx Shutdown	Corr.Factor
0 hr	1
1 hr	2
2 hr	4
4 hr	6
≥8 hr	10

* Measured dose rate is the maximum closed window reading found while traversing the plume within 400 meters of the release point.

**Elevated factor based on assumed effective release height of 400 ft. This requires a thermal buoyant plume such as from the ADV's. A release from the RB/AB vent would lie somewhere between the 2 factors and would depend on the ratio of the wind speed to the vent exit velocity. For conservatism use the elevated factor for a vent release. Factors are based on RASCAL 3.0.1 runs.

Go to Worksheets 6 and 7 to determine the Iodine and Particulate Release Rates

Completed by: ______________________ Date/Time: __________________

Verified by: ______________________ Date/Time: __________________

ALTERNATE METHODS FOR DETERMINING METEOROLOGICAL DATA

1. Wind direction, wind speed, and wind range can be estimated by observing cooling tower vapor, flags, fossil stack smoke, etc.

2. Stability class can be determined using Sigma Theta, Delta T, or wind range. Wind range is the difference (in degrees) between the highest and lowest wind direction tracing on the recorder for a 15 minute period.

STABILITY CLASS DETERMINATION

SIGMA THETA (degrees)	DELTA T (DEGREES)	WIND RANGE (degrees)	STABILITY CLASS
≥ 22.5	≤ -1.46	≥ 135	A (most dispersed plume)
< 22.5 to 17.5	-1.45 to –1.31	134 to 105	B
< 17.5 to 12.5	-1.30 to –1.16	104 to 75	C
< 12.5 to 7.5	-1.15 to –0.39	74 to 45	D
< 7.5 to 3.8	-0.38 to 1.15	44 to 23	E
< 3.8 to 2.1	1.16 to 3.07	22 to 13	F
< 2.1	≥ 3.08	≤ 12	G (most concentrated plume)

3. Wind direction is determined by estimating the average value of the tracing for a 15 minute period.

4. Weather data is available via the intra/internet. Example sites follow:

   AccuWeather.com: enter zip code 34429 and hour-by-hour.

   Weather.com: enter zip code 34429 and select hourly.

   Energy Control Center: Progress Net, Business Units and Departments, Energy Control Center Florida, Forecasts, Real Time Weather, Zone City Forecasts, click west central Florida area on map.

5. Meteorological data may also be obtained from the following sources; however, non-local backup sources may NOT be representative. Phone numbers are in the Off-site Support Directory.

   Primary Backup - FAA Flight Service Station in Gainesville, FL.
   Secondary Backup - Tampa Weather Service in Ruskin, FL.

6. Refer to the following table to determine sectors affected based on the wind from direction.

DEGREES	SECTORS	DEGREES	SECTORS	DEGREES	SECTORS
349-11 (349-371)	H J K	102-123 (462-483)	N P Q	214-236	B C D
12-33 (372-393)	J K L	124-146 (484-506)	P Q R	237-258	C D E
34-56 (394-416)	K L M	147-168 (507-528)	Q R A	259-281	D E F
57-78 (417-438)	L M N	169-191 (529-540)	R A B	282-303	E F G
79-101 (439-461)	M N P	192-213	A B C	304-326	F G H
				327-348	G H J

METEOROLOGICAL INPUT SHEET

Sources listed by priority – enter number of source data used in each column heading

1. 33 ft Primary Tower
2. 175 ft Primary Tower
3. 33 ft Alternate Tower
4. Other_______________

METEOROLOGICAL DATA

Rec.	Obs. or Fcst.	Date	Data Time	Wind Direction From Deg.	Wind Speed mph	Sigma Theta Deg. or Wind Range Deg. or Delta T °F	Stability Class	Rain Inches Per 15 min	Air Temp ° F

Summary of Changes
PRR 513748 (October, 2012)

NOTES:	1. Procedure Sponsor: Ensure that any changes to EM-204B that affect information contained in ERF posters, Enclosures, briefing cards, guidelines, etc. are made to those items as well. 2. Procedure Sponsor: Changes to certain parts of EM-204B may impact other EPIPs. Specifically, if any changes are made to section 3.0 definitions, review EM-204A. Ensure appropriate PRRs are initiated as needed.

SECTION/STEP	CHANGE
Throughout	Changed revision to 42. This revision implements changes necessary for EC 76363 replacement of RM-A1 and RM-A2. (PRR 513748)
1.3	Added new section stating that this is an Emergency Plan Implementing Procedure. Any revisions must be carefully considered for Emergency Plan impact.
3.3.3	Updated the impact of a station blackout (SBO) on the new RM-A1 and RM-A2 being installed by EC 76363. The skids, pumps and detectors will be powered from ES MCC 3A1 and will lose power in a SBO. With the old system, the detectors and meters remained powered. The new control room display units will be energized, but will have no process information. Like the old system, the new system will lose the capability of monitoring releases in a SBO. Unlike the old system, the new system will not be able to monitor area background levels around the skid. However, background levels around the skid have no impact on the intent of this procedure.
3.3.7	Deleted step and caution and instructions about the low/medium/high valve controller as it does not apply to the new system.
Encl 1 all pages	Added "Page x of 13" to the top of each page in the Enclosure. Corrected spacing of letter labels in the equations as necessary.
Encl 1 page 1	Worksheets 1 and 2 descriptions: Added that the worksheet is not necessary if Ci/sec is already known as the new monitors feed Ci/sec to OSI PI. Worksheet 2A description: Added that the worksheet is intended for use during the extended shutdown that began in 2009. Worksheet 6 description: Added that this worksheet is not necessary in extended shutdown conditions as all iodines have decayed.
Encl 1 Worksheets 1, 2	Added instruction above the worksheets that if Ci/sec is known to go to Worksheets 6 and 7. Worksheets 1 and 2 will now be used for converting µCi/cc to Ci/sec when Ci/sec is not directly available. Replaced low-range and mid-range with normal range and accident range. Replaced cpm and mR/hr with µCi/cc. Deleted high-range row. Deleted section I for conversion factor determination because it is no longer necessary.
Encl 1 Worksheet 2A	Deleted the Particulate Ci/sec equation as Worksheet 7 has been revised to calculate particulate Ci/sec in extended shutdown.
Encl 1 section 1.4.3.	In step 1, deleted reference to RM-A2 low, mid, and high ranges as the new system does not have those ranges. Revised the monitor readout units and the units needed by Worksheet 2 as a result of the new monitors.

	In step 2, revised the units needed by Worksheet 6 as a result of the new monitors.
Encl 1 Worksheet 6	Added instruction that if in extended shutdown, go to Worksheet 7 as all iodines are decayed.
Encl 1 Worksheet 7	Added to section B extended shutdown base ratios for pool drained with filtered and unfiltered releases. Added a note that these ratios and the assumed isotopic are based on RASCAL 3.0.5. Added to section C that decontamination factor section does not apply in extended shutdown and to enter a factor of 1.

PROGRESS ENERGY

CRYSTAL RIVER UNIT 3

PLANT OPERATING MANUAL

EMERGENCY PLAN IMPLEMENTING PROCEDURE

EM-219

DUTIES OF THE DOSE ASSESSMENT TEAM

TABLE OF CONTENTS

1.0 PURPOSE

1.1 The primary purpose of the Dose Assessment Team (DAT) is to provide dose assessment information for the Emergency Coordinator (EC) and the Emergency Operations Facility (EOF) Director.

Dose assessment is a component of determining both emergency classification and protective action recommendations.

This procedure provides guidance to the DAT for setting up operations in the EOF (in conjunction with EMG-NGGC-0002), interfacing with the TSC Radiation Controls Coordinator (RCC) and the EC in the TSC and with the EOF Radiation Controls Manager (RCM) and EOF Director in the EOF, and comparing dose projections with actual data collected by the Off-site Radiation Monitoring Team. [NOCS 00387, 01582]

This procedure is an Emergency Plan Implementing Procedure. Any revisions must be carefully considered for Emergency Plan impact.

2.0 REFERENCES

2.1 Developmental References

2.1.1 Radiological Emergency Response Plan (RERP)

2.1.2 EMG-NGGC-0002, Offsite Dose Assessment

2.1.3 Manual of Protective Action Guides and Protective Actions for Nuclear Incidents, EPA-400-R-92-001, Environmental Protection Agency (October, 1991)

3.0 PERSONNEL INDOCTRINATION

3.1 Definitions

3.1.1 **Committed Dose Equivalent (CDE)** - Dose to an organ due to the intake of radioactive materials.

3.1.2 **Deep Dose Equivalent (DDE)** - External whole body dose.

3.1.3 **Off-Site RMT** - The portion of the Radiation Monitoring Team (RMT) that performs environmental sampling within the Crystal River Energy Complex and within the 10 mile Emergency Planning Zone (EPZ). The Off-Site RMT is also referred to as the Environmental Survey Team (EST).

3.1.4 **Plume Tracking** - Locating, tracking, and monitoring radiological characteristics of an off-site release.

3.1.5 **Replacement Emergency Dose Assessment System (REDAS)** - System to retrieve and archive the meteorological, radiological, and operational data required for emergency dose assessment purposes.

3.1.6 **Thyroid Dose** - Dose to the thyroid due to intake of radioactive iodine.

3.1.7 **Total Dose (TEDE)** - The sum of external dose (DDE) and the equivalent amount of whole body dose due to individual organ uptakes.

3.2 Responsibilities

3.2.1 The DAT staffs at the EOF at an Alert and assumes primary responsibility for dose assessment. Initially, the EOF DAT reports to the TSC Radiation Controls Coordinator and supplies dose assessment information to the TSC. When the EOF is operational, the DAT reports to the EOF Radiation Controls Manager.

3.2.2 The DAT provides the TSC Radiation Controls Coordinator with the appropriate dose assessment information necessary for the EC to determine emergency classifications and protective action recommendations (PARs).

3.2.3 When the EOF is operational, responsibility for PARs transfers from the EC to the EOF Director.

3.2.4 **Radiation Controls Manager [NOCS 9817]:**

3.2.4.1 As applicable; DETERMINE release status, and Release Significance Category.

3.2.4.2 NOTIFY the following positions at an ALERT to STAFF the EOF and PROVIDE coordination and support (unless assured that qualified individual for each position have already been contacted and are responding):

- EOF Dose Assessment Team Members (3)
- PE Field Team Liaison

3.2.4.3 ASSIGN Dose Assessment Team members as EOF Team Leader, Dose Assessment Computer Operator, and Plant Computer Data Operator.

3.2.4.4 PROVIDE guidance to EOF Dose Assessment personnel performing dose calculations.

3.2.4.5 INTERFACE with the Technical Support Team to ASSURE that EOF and Department of Health (DOH) Dose Assessment personnel are provided information necessary for generating off site dose projections.

3.2.4.6 ENSURE the results of PE and DOH dose assessment and field monitoring activities are compared.

3.2.4.7 PROVIDE off site dose projection summaries to the EOF Director and/or Assistant EOF Director.

3.2.4.8 PROVIDE PARs (on the basis of dose assessment projections) to the EOF Director and/or Assistant EOF Director.

3.2.4.9 PROVIDE information to complete Items 8 through 11 and 14 of the Florida Nuclear Plant Emergency Notification Form.

3.2.4.10 PROVIDE information (for PE briefings, ENC briefings, etc.) concerning the radiological condition within the Crystal River Energy Complex.

3.2.4.11 DIRECT activities of the PE Field Team Liaison.

3.2.4.12 DESIGNATE a Dose Assessment Team member to ASSIST OR PERFORM the duties of the PE Field Team Liaison if necessary.

3.2.4.13 ENSURE the EOF Director is informed when the EOF Dose Assessment Team assumes responsibility for dose projections.

3.2.4.14 ENSURE radiological monitoring is set-up at the EOF, if needed. An air sampler is typically available at the EOF.

3.2.4.15 PROVIDE comparison of PE/DOH dose projection models with PE/DOH field survey results.

3.2.4.16 If desired in the EOF; DIRECT the display of the dose assessment computer onto one of the EOF screens.

3.2 Responsibilities (Continued)

3.2.5 EOF Dose Assessment Team Leader:

3.2.5.1 As applicable; DETERMINE release status, and Release Significance Category.

3.2.5.2 ENSURE equipment is operational and work area is ready for use.

3.2.5.3 UTILIZE and REINFORCE various Human Performance tools and techniques (i.e., pre-job brief, two-minute rule, place-keeping, etc.) as appropriate throughout an event to ensure DAT performance is timely and projections are credible.

3.2.5.4 COMMUNICATE with the EOF Technical Support Team to determine and verify the parameters/source term and release pathway to use for dose assessment.

3.2.5.5 REQUEST and REVIEW dose projection printouts from TSC.

3.2.5.6 COORDINATE DAT activities and ENSURE the EOF Radiation Controls Manager and TSC Radiation Controls Coordinator are aware when the DAT has taken responsibility for dose projections.

3.2.5.7 COMPARE dose projection results with the State and NRC results and ATTEMPT to resolve any significant discrepancies between results.

3.2.5.8 DIRECT PE Field Team Liaison to compare dose projections with field data.

3.2.5.9 COMPARE dose projection data with Dose Assessment Credibility Table found in EMG-NGGC-0002.

3.2.5.10 COORDINATE PE and State Field Team monitoring locations.

3.2.5.11 PROVIDE dose projection and field monitoring results to the EOF Radiation Controls Manager and REQUEST peer checks by the RCM of these results.

3.2.5.12 ENSURE Dose Assessment Status Board is updated.

3.2.6 Dose Assessment Computer Operator:

3.2.6.1 ASSIST in the set-up of the Dose Assessment Work Area.

3.2.6.2 OPERATE dose assessment computer to GENERATE off site dose projections.

3.2.6.3 PROVIDE dose projection results to the Dose Assessment Team Leader.

3.2.7 Plant Data Computer Operator:

3.2.7.1 ASSIST in the set-up of the Dose Assessment Work Area.

3.2.7.2 GENERATE meteorological and radiation monitoring data, AND PROVIDE it to the Dose Assessment Computer Operator.

3.2.7.3 COMPARES data with that provided in the TSC/OSC and the Control Room.

3.2.8 PE Field Team Liaison:

3.2.8.1 COMMUNICATE/COORDINATE with the PE Off Site Radiation Monitoring Team(s) either directly or through the TSC Environmental Survey Team Dispatcher.

3.2.8.2 ESTABLISH contact with the DOH Field Team Coordinator, upon arrival.

3.2.8.3 COMPARE PE Environmental Survey Team data with DOH Field Team data.

3.2.8.4 VERIFY that the Radiation Controls Manager has requested Health Physics support and radiological monitoring equipment from the TSC, if conditions warrant or are expected.

3.2.8.5 ASSIST with monitoring of radiological conditions in the EOF, if needed. This will be directed by the Radiation Controls Manager.

3.2.8.6 UPDATE the Dose Assessment Status Board as needed.

3.3 Limits and Precautions

3.3.1 The estimated dose rates and measured dose rates will probably not be equal due to the numerous sources of uncertainty. All available data should be analyzed for credibility and considered in making informed decisions.

3.4 Equipment & Materials

3.4.1 Hand-held calculator (as needed).

3.4.2 Equipment identified in EMG-NGGC-0002, Off-Site Dose Assessment.

4.0 INSTRUCTIONS

4.1 Radiation Controls Manager [NOCS 9817]

4.1.1 DETERMINE Release Significance Category and ensure the electronic status board is correct. ☐

4.1.2 PERFORM notifications (unless assured that qualified individual for each position have already been contacted and are responding):
- Three (3) Dose Assessment Team Members ☐
- PE Field Team Liaison ☐

4.1.3 ASSIGN Dose Assessment Team members:
- Team Leader ☐
- Dose Assessment Computer Operator ☐
- Plant Data Computer Operator (e.g. SPDS, OSI PI) ☐

4.1.4 BRIEF Dose Assessment Team members and PE Field Team Liaison on plant/release status. ☐

4.1.5 CONSIDER calling in an additional Radiation Controls Manager to assist, if needed. ☐

NOTE:	Consideration should be given to getting radiological monitoring equipment/instruments sent to the EOF before a possible radioactive release that may affect the EOF.

4.1.6 CONTACT the TSC Radiation Controls Coordinator to request radiation monitoring assistance and radiological monitoring equipment/instruments, if needed. [NOCS: 24180] ☐

4.1.7 ENSURE radiological monitoring and area TLD is set-up at EOF, if needed. An air sampler/constant air monitor is typically available at the EOF. ☐

4.1.8 INFORM EOF Director or Assistant EOF Director when the Dose Assessment Team assumes responsibility for dose projections. ☐

4.1.9 PROVIDE impacts to PARs and EALs based dose projections to the TSC Radiation Controls Coordinator and EOF Director or Assistant EOF Director. ☐

4.1.10 If desired, DIRECT the display of the dose assessment computer onto one of the EOF screens ☐

4.1.11 BRIEF DOH Dose Assessment personnel upon arrival. ☐

4.1.12 ENSURE PE and DOH dose projection results are compared. ☐

4.1.13 ENSURE PE and DOH field monitoring results are compared. ☐

4.1.14 ENSURE PE and DOH field survey results are compared to PE and DOH dose projections results ☐

4.1.15 INFORM Technical Support Coordinator of any significant changes in radiation monitors. ☐

4.1.16 COMPLETE the radiological/meteorological data on the Florida Nuclear Plant Emergency Notification Form if needed. ☐

NOTE:	ENC Technical Advisors/Spokespersons should be available to attend ENC briefings on your behalf.

4.1.17 PARTICIPATE in ENC briefings, if time permits, and PROVIDE information concerning the radiological conditions within the Crystal River Energy Complex. ☐

4.2 Dose Assessment Team Leader

4.2.1 OBTAIN Release Significance Category from Radiation Controls Manager and VERIFY assumptions. ☐

4.2.2 ENSURE work area is set-up and functional per EM-400. ☐

4.2.3 CONDUCT a pre-job brief (e.g., Two-Minute Rule) to ensure that Team members are aware of their objectives, roles and responsibilities ☐

4.2.4 EMPHASIZE the use Human Performance tools and techniques (e.g., peer checks, questioning attitude, three-way communications) as appropriate to ensure DAT performance is timely and credible. ☐

4.2.5 ESTABLISH communications with TSC Dose Assessment Communicator and ENSURE the Communicator is available for interface with the Environment Survey Team Dispatcher and the Accident Assessment Team. REFER to Enclosure 1 for conference call instructions if necessary [NOCS 00387]. If necessary, REQUEST the Radiation Controls Manager ask the TSC Radiation Controls Coordinator to assign a Control Room Dose Assessment Communicator ☐

4.2.6 ESTABLISH an interface with the Technical Support Team with expectations to receive the EM-402 Dose Assessment Team Notification Form with updates as conditions change and to confirm source term and release path assumptions. ☐

4.2.7 REVIEW dose projection data from the Control Room and/or TSC if available. ☐

4.2.8 BEGIN dose assessment using EMG-NGGC-0002. Ensure Attachment 6 table "CR3 Accident Types with RASCAL SourceTerm/Release Path Options" is referenced. ☐

4.2.9 NOTIFY Radiation Controls Manager and TSC Dose Assessment Communicator when the EOF Dose Assessment Team has started. ☐

4.2.10 COMPARE dose projection results with DOH and NRC results and with EMG-NGGC-0002 Attachment 6 table "Dose Assessment Credibility." ☐

4.2.11 ENSURE the PE Field Team Liaison compares dose projections with PE and/or DOH field team results. Enclosure 3 provides instructions for accessing the specific RASCAL parameters needed for each of the comparison methods. Enclosure 4 provides supplemental information on RASCAL Detailed Results Options. ☐

4.2.12 ENSURE dose assessment status board is updated, as needed. ☐

4.2.13 PROVIDE Radiation Controls Manager with dose projection and field monitoring results for approval. ☐

4.2.14 PROVIDE all approved dose projections results to TSC. ☐

4.3 Dose Assessment Computer Operator

4.3.1 ENSURE work area is set-up and functional. ☐

4.3.2 ENSURE dose assessment computer is operational. ☐

4.3.3 PERFORM dose projections per EMG-NGGC-0002 and as directed by the Dose Assessment Team Leader. ☐

4.3.4. PROVIDE RASCAL data to compare with field team results, as requested by the Field Team Liaison. Enclosure 3 provides instructions for accessing the specific RASCAL parameters needed for each of the comparison methods. Enclosure 4 provides supplemental information on RASCAL Detailed Results Options. ☐

4.4 Plant Data Computer Operator

4.4.1 ENSURE work area is set-up and functional. ☐

4.4.2 ENSURE Plant Data Computer is operational. (e.g., SPDS, OSI PI) ☐

4.4.3 PROVIDE meteorological/radiation monitoring data per Enclosure 2 to the Dose Assessment Computer Operator. ☐

4.4.4 REVIEW data for credibility based on TSC and Control Room data. ☐

4.5 PE Field Team Liaison

4.5.1 NOTIFY Radiation Controls Manager of arrival. ☐

4.5.2 ESTABLISH contact with TSC Environmental Survey Team Dispatcher or PE Off Site Radiation Monitoring Team. PROVIDE information to the EST Dispatcher to aid in plume location and tracking as necessary. Enclosure 7 provides guidelines for EST deployment. ☐

4.5.3 REFERENCE EM-210B for Off Site Radiation Monitoring Team coordination and control. ☐

NOTE:	Comparison may be performed by the Field Team Liaison, the Dispatcher, or an available DAT member. Spreadsheets are available to assist with Enclosure 3 calculations at: L:\Shared\RASCAL\EM-219 Enclosure 3 Spreadsheets.xlsx

4.5.4 COMPARE dose projections with field team results (PE and/or DOH) per Enclosure 3. (Enclosure 5 may be used to record additional field team data as necessary for review or documentation purposes). ☐

4.5.5 IF the calculated values seem inconsistent with the field data,

THEN INFORM the EOF Radiation Controls Manager immediately,

AND VERIFY all calculations and Off-Site RMT data. REFER TO Enclosure 6 for potential reasons for inconsistencies. ☐

4.5.6. UPDATE dose assessment status board, as needed. ☐

4.5.7 ESTABLISH contact with DOH Field Team Coordinator, upon team's arrival. ☐

4.5.8 COMPARE TSC Environmental Survey Team data with DOH Field Team data. ☐

ESTABLISHING DOSE ASSESSMENT COMMUNICATIONS

The following are two method of establishing dose assessment three-way communications among the EOF, TSC, and the Control Room.

1.0 Dose Assessment Ringdown Telephone:

1.1 LIFT the receiver of the Dose Assessment Ringdown Telephone to establish communications among the TSC Dose Assessment Communicator, the EOF Dose Assessment Team and, if desirable, a communicator in the Control Room monitoring radiological and meteorological data. [NOCS 00387]

2.0 EOF/Plant Extension (EOF should initiate the call):

NOTE:	Note: If the phone has a "Flash" button, press it instead of the "hook flash" below.

2.1 HOOK FLASH by quickly depressing and releasing the connection button, to receive a stutter dial tone, THEN DIAL the third extension.

2.2 HOOK FLASH to receive the feature dial tone.

2.3 DIAL access code 4 to establish the conference.

2.4 IF the other extension CANNOT be reached, HOOK FLASH again and communication with the TSC will be re-established.

DATA FROM THE PLANT COMPUTER [NOCS 00387, 40188]

This Enclosure contains six methods for obtaining data from the CR3 plant computer. Select the most appropriate method. **NOT** all methods may be available. Data can also be obtained directly from the Control Room.

1.0 **OSI PI EMERGENCY PREPAREDNESS SCREENS** – Contains selected live and archived operational, radiation monitor, and meteorological data in tables and graphs related to dose assessment, accident assessment, and Emergency Action Levels. Also available here are Emergency Plan implementing procedures, EAL Bases Manual, phone directories, and the on-call roster.

1.1 **DOUBLE-CLICK** the OSI PI CR3 QPIM icon **OR GO** to Start, Programs, Business Apps, PI Systems, CR3 QPIM.

1.2 **SELECT** the EP Tab.

1.3 **REVIEW** the available selections using the Page Up and Page Down keys (may have to click on any selection first).

1.4 **DOUBLE-CLICK** on the selection **OR CLICK** once **AND THEN SELECT** the Open button to open the desired selection.

1.5 **DOUBLE-CLICK** the graph icons to view a graph of recent history of the parameter.

- **IF** the Trend Scale box opens, **THEN SELECT** Cancel, **AND DOUBLE-CLICK** the graph icon again on the center or right side of the icon.
- **DOUBLE-CLICK** on the graph to close it.

1.6 **CLICK** the Close button to return to the EP Tab menu.

1.7 **CLICK** the Full-Screen icon to toggle between full-screen format and displays with menus and icons at the top and bottom of the screen.

2.0 **DYNAMIC DATA EXCHANGE SPREADSHEET** – Contains real-time data from radiation monitors and meteorological instruments displayed in an Excel spreadsheet.

2.1 **DOUBLE-CLICK** on the PICS icon **OR GO TO** Start Programs, Engineering, CR3, CR3 PICS.

2.2 In the Access Control Client window:

1. **SELECT** CR3 PPCS in the Choose a system box.
2. **TYPE** either tsc **OR** eof in the User Name box.
3. **TYPE** either tsc **OR** eof in the Password box.
4. **CLICK** LogOn.

2.3 **MINIMIZE** the PICS Access Control Client window.

2.4 **GO TO** the c:\PICS\RtdbDde directory in Windows Explorer **AND THEN double- click** on RtdbDde.exe file.

- **WHEN** the hourglass disappears (takes < 1 second), **THEN GO TO** the next step.

2.5 **START** Excel.

2.6 **OPEN** the file L:\Shared\RASCAL\RADMET.xls.

2.7 **SELECT** "Read Only" to update all linked information.

3.0 **SPDS DISPLAYS** – Contains real-time operational data, graphs, and selected radiation monitors.

3.1 **DOUBLE-CLICK** on the PICS icon **OR GO TO** Start Programs, Engineering, CR3, CR3 PICS.

3.2 In the Access Control Client window:

1. **SELECT** CR3 PPCS in the Choose a system box.
2. **TYPE** either tsc **OR** eof in the User Name box.
3. **TYPE** either tsc **OR** eof in the Password box.
4. **CLICK** LogOn.

3.3 **DOUBLE-CLICK** on the SPDS Display icon in the PICS Access Control Client window.

3.4 **WHEN** the SPDS graphic screen is displayed, **THEN PRESS** the "A" key to display the Alpha pages. Page 7 of 8 displays; RM-G29/30, RM-A6, RM-L1, RM-A1 Normal-range, RM-A2 Normal-range, RM-A12, RM-Gs25-28, RM-L2, RM-L7, RM-G1, and RM-A5.

4.0 PICS ARCHIVE RETRIEVAL – Contains data from any point recorded in the PICS Real Time Database downloaded per the user specifications of point selection, time selection, and time intervals.

4.1 **DOUBLE-CLICK** on the PICS icon **OR GO** to Start Programs, Engineering, CR3, CR3 PICS.

4.2 In the Access Control Client window:

1. **SELECT** CR3 PPCS in the Choose a system box.
2. **TYPE** either tsc **OR** eof in the User Name box.
3. **TYPE** either tsc **OR** eof in the Password box.
4. **CLICK** LogOn.

4.3 **DOUBLE-CLICK** on the Retrieval icon in the PICS Access Control Client window.

4.4 **SELECT** File, New Retrieval in the PDRSrtrv box.

4.5 **PERFORM** the following to submit the Simple Retrieval Query Form:

1. **ENTER** start and stop times of desired data.
2. **SELECT** Fixed Width Text.
3. **ENTER** file name **AND** path for output file.
4. **ENTER** Snapshot interval (time between data points).
5. **HIGHLIGHT** point to read **THEN CLICK** Select. **REPEAT** as needed.
6. **ADD** point EVI-1 to the point selection list.
7. **CLICK** Submit.

4.6 **START** Excel.

4.7 **OPEN** the output file from 4.5.3 above.

4.8 **SELECT** Fixed Width in the Text Import Wizard box.

4.9 **CLICK** Finish in the Text Import Wizard box.

5.0 **PICS RECALL DISPLAY PROGRAM** – Displays real-time data either in tabular or graphic format. 50 points can be displayed in one alphanumeric group. Multiple groups file can be opened at one time. Multiple group files already exist.

5.1 **DOUBLE-CLICK** on the PICS icon **OR GO TO** Start Programs, Engineering, CR3, CR3 PICS.

5.2 In the Access Control Client window:

1. **SELECT** CR3 PPCS in the Choose a system box.
2. **TYPE** either tsc **OR** eof in the User Name box.
3. **TYPE** either tsc **OR** eof in the Password box.
4. **CLICK** LogOn.

5.3 **DOUBLE-CLICK** the Recall Display Program icon in the PICS Access Control Client window.

5.4 **SELECT** File in the Recall Display box, **THEN SELECT** Open **AND SELECT** desired group.

6.0 **REDAS –** Allows time blocks of archived operational, radiation monitor, and meteorological data from pre-designated groups to be downloaded into an Excel spreadsheet.

6.1 **LOG** on using personal OT number and password.

6.2 **ACCESS** REDAS **AND PERFORM** initial setup as follows:

1. **DOUBLE CLICK** on the REDAS icon **OR GO TO** Start Programs, Engineering, CR3, CR3 REDAS.
2. **SELECT OK** on the REDAS Network Accessor box.
3. **SELECT** Request, **THEN SELECT** Request Group.
4. **SELECT** the following:
5. Group List is Standard
6. Sort By is Name, and
7. File Format is ASCII Tabular.
8. **ENTER** Start & End Dates & Times.
9. **CLICK** on the box to change parameters, **THEN ENTER** dates and times.
10. **SPECIFY** at least one hour for time.

6.3 **SELECT AND DOWNLOAD** the following REDAS Groups. The order in which groups are selected is NOT important and the following steps may be performed interchangeably for Group selection.

- **AA_ENG** Engineering Data
- **AA_MET** Meteorological Data
- **AA_RADAL** Air and Liquid Radiation Detecto (SIC)
- **AA_RADG** General Area Radiation Detectors

6.3.1 **CLICK** on AA_ENG.

1. **VERIFY** Frequency is 15 minutes **AND SELECT** Average box.
2. **Click** on OK. All download parameters will be displayed in a "Group Confirmation" window.
3. **IF** data are correct, **CLICK** on Yes, **OTHERWISE CLICK** on No to return to previous screen and correct. Downloading will start, and should take less than 1 minute. While downloading is taking place, the "Data Request Status" window will be active.
4. **WHEN** the "REDAS-NIS" window is displayed, downloading is complete **THEN NOTE** the file name and location.
5. **CLICK** on OK in the "REDAS-NIS" screen.
6. **SELECT** Request, **THEN SELECT** Request Group.

6.3.2 **CLICK** on AA_MET.

1. **VERIFY** Frequency is 15 minutes **AND SELECT** Average box.
2. **CLICK** on OK to accept download settings.
3. **VERIFY** settings in "Group Confirmation" window **THEN CLICK** on Yes to accept and begin download.
4. **WHEN** the "REDAS-NIS" window is displayed, downloading is complete **THEN NOTE** the file name and location.
5. **CLICK** on OK in the "REDAS-NIS" screen.
6. **SELECT** Request, **THEN SELECT** Request Group.

6.3.3 **CLICK** on AA-RADAL.

1. **VERIFY** Frequency is 15 minutes **AND SELECT** Average box.
2. **CLICK** on OK to accept download settings.
3. **VERIFY** settings in "Group Confirmation" window **THEN CLICK** on Yes to accept and begin download.
4. **WHEN** the "REDAS-NIS" window is displayed, downloading is complete **THEN NOTE** the file name and location.
5. **CLICK** on OK in the "REDAS-NIS" screen.
6. **SELECT** Request, **THEN SELECT** Request Group.

6.3.4 **CLICK** on AA_RADG.

1. **VERIFY** Frequency is 15 minutes **AND SELECT** Average box.
2. **CLICK** on OK to accept download settings.
3. **VERIFY** settings in "Group Confirmation" window **THEN CLICK** on Yes to accept and begin download.
4. **WHEN** the "REDAS-NIS" window is displayed, downloading is complete **THEN NOTE** the file name and location.
5. **CLICK** on OK in the "REDAS-NIS" screen.

6.4 **START** Excel.

6.5 **OPEN** the output file recorded earlier (normally C:\My Documents\Aa_eng.txt, etc.).

6.6 **SELECT** the following in the Text Import Wizard:

1. **SELECT** Delimited, in the Original Data Type.
2. **CLICK** Finish.

COMPARISON OF DOSE PROJECTIONS WITH FIELD MEASUREMENTS

INTRODUCTION:

Comparison of field measurements with dose projection are made to assess the validity of the estimates and determine whether the source term being used should be adjusted. This comparison is done to assist in validating dose projections that would be considered when making emergency classifications and/or protective action recommendations.

The results obtained from this enclosure should be considered guidance. Revisions to the calculated source term should be made only after careful consideration of all factors involved with the release. A listing of factors to consider and precautions in making conclusions are given in Enclosure 6.

In the first few hours following the start of a release, the need for rapid information transfer and decision making may have to be performed on a qualitative basis without the benefit of the completed forms provided in this enclosure. The forms are provided as a tool as time permits for a more quantitative assessment.

COMPARISON AND ADJUSTMENT METHODS:

There are three types of comparisons presented in this enclosure:

Method A may be used for both noble gas and iodine source terms and determines the ratios between the field measurements and projected doses rates.

Method B is for use on Iodine source term only and determines the ratio of Noble Gas µCi/cc to Iodine µCi/cc measured in the field and compares it to the ratio of Noble Gas curies and Iodine curies calculated by RASCAL.

Method C compares deposition data measured in the field with the levels calculated by RASCAL.

Section D of this Enclosure provides a discussion on the use of these comparisons in adjusting estimated source terms or dose consequences.

Method A -Noble Gas and Iodine Source Terms:

A.1. a. Record the noble gas gamma dose rate (window closed) measurements from the Off-Site RMT on Table 1. The location (distance and sector) and time are also recorded.

b. Enter the RASCAL External Dose Rate – Closed Window estimate[NOTE 1] for the corresponding location (distance and sector) and time.

c. Divide the noble gas field measurement value (mRem/hr) by the RASCAL External Dose Rate – Closed Window (mRem/hr) and record this ratio in Table 1.

d. Perform A.1.a through A.1.c for each location.

A.2. a. Record the air concentrations for total iodine from the field measurements on Table 2. The location (distance and sector) and time are also recorded.

b. Convert iodine air concentration to a thyroid dose rate by multiplying the iodine air concentration (µCi/cc) by the appropriate thyroid dose conversion factor (DFI, mRem/hr/µCi/cc) given at the bottom of Table 2. Calculate the thyroid dose rate for each measured iodine air concentration given and record on Table 2.

c. Enter the RASCAL thyroid dose rate estimate[NOTE 1] for the corresponding location (distance and sector) and time. CONVERT REM/HR TO MREM/HR.

d. Divide the thyroid dose rate based on field measurement air concentrations by the RASCAL thyroid dose rate estimates and record on Table 2.

e. Perform A.2.a through A.2.d for each location.

A.3 Determine the median of the ratios of measured to calculated DDE dose rates. Enter the median of the ratios in the box provided below Table 1.

A.4. Repeat A.3 above for the Thyroid dose rate ratios in Table 2. Enter the ratio median in the box below Table 2.

A.5. All data should be provided to the EOF Radiation Controls Manager, who will review the data in accordance with Section D of this Enclosure.

[NOTE 1] To obtain RASCAL dose rate:

1. When calculations are complete, select "Detailed Results" button.

2. Under Display Format, select "Numeric table" from either the "From 10-mile calculation" option or the "From Close-in Calculation" option. Or select the "Special Receptors" option. The Close-in option provides data out to two miles. The Special Receptor option provides data at user-entered locations (a pre-defined set is also available).

3. Under Result Type, select "Thyroid" (for iodine) or "External Dose Rate -- Closed Window"

4. Under Time Period, select "Single Time" and enter the time the field results were obtained.

5. Select the "Display Selected Results" button.

6. The Close-in table displays the dose rates by distance and bearing degrees. The 10-mile table displays the dose rates by distance North or South and distance East or West. To obtain the distance and bearing degrees, place the cursor over a cell (do NOT click).

Method B - Noble Gas to Iodine Ratio Comparison:

B.1. The noble gas to iodine ratio should be fairly consistent throughout the plume. However, at the edges of the plume, or to the side or below the plume, there can be measurable gamma dose rates from shine, but no measured iodine concentration. Comparisons should only be done if it is known the team was in the plume (Window open > 2 times window closed dose rates.)
Record on Table 3:
- time of field measurements
- the location (distance/sector)
- the Noble Gas (gamma) dose rate (DDE) measured in the field
- the Iodine µCi/cc measured in the field (total iodine).

B.2. Convert the DDE to Noble Gas µCi/cc by dividing by the Noble Gas Dose Conversion Factor (DFNG) given at the bottom of Table 3.

B.3. Calculate the field measurement (Noble Gas to Iodine) ratio by dividing the Noble Gas µCi/cc by the Iodine µCi/cc and record in the right hand column of Table 3.

B.4. Perform steps B.1, B.2, and B.3 for each location.

B.5. Calculate the average of the ratios by summing the ratios and dividing by the number of ratios. Enter into the appropriate formula below the Table 3.

B.6. Determine the RASCAL ratio by dividing the Noble Gas curies[NOTE 2] by the Iodine curies[NOTE 2] from the dose projection corresponding to the field measurement time. Enter into the appropriate formula below Table 3.

B.7. The Noble Gas to Iodine ratio in the field can now be compared to the Noble Gas to Iodine ratio used in RASCAL. All data should be provided to the EOF Radiation Controls Manager, who will review the data in accordance with Section D of this Enclosure.

[NOTE 2] To obtain RASCAL curies:

1. When calculations are complete, select "Source Term Summary" tab to display the total curies released.

2. Select the "Details" button to display a table of curies of each nuclide released in each 15-minute increment of the release period.

3. Select the 15-minute increment that best corresponds to the field data based on distance and wind speed. For example, if the field data was obtained 1 mile from the plant and the wind speed is 2 mph, then select the 15-minute increment released 30 minutes before the field reading. Selecting the 15-minute increment becomes more important if the release contains status changes in building spray or filters which could alter Nobel Gas to Iodine ratios by orders of magnitude. With steady-state releases, selection is less important.

4. Sum the curies of the Iodine isotopes. Sum the curies of the Xenon and Krypton isotopes. (If desired, the entire table can be exported as a comma-delimited text file and opened in Excel for summing by selecting the "Export" button.)

Method C – Deposition Comparisons

C.1 Deposition modeling is extremely uncertain. The source term for particulate releases will likely not be known, especially as a function of time. The nuclide mix can be highly variable from the mix assumed. Deposition levels within the first few hours of an event will likely be dominated by the noble gas particulate daughters Rb-88 and Cs-138. Meteorological deposition models, particularly the deposition rate factor assumed, are highly variable and uncertain. Deposition levels can be highly variable within a short distance due to factors such as surface roughness, overhead tree covers, washout of activity to lower lying areas, etc. Models cannot predict this local variability. Therefore, once field team deposition data is available, it is recommended that any information or decisions based on deposition be based solely on the field measurements and not the model predictions. Therefore, there is no need for detailed forms for performance of comparisons between field team results and model results.

C.2 However, if it is desired to make a comparison of the measured field team deposition results to the RASCAL predictions[NOTE 3], the following conversion factor should be applied to the RASCAL results:

RASCAL Ci/m^2 x 2.22E10 = dpm/100 cm^2

[NOTE 3] To obtain RASCAL deposition:

1. When calculations are complete, select "Detailed Results" button.
2. Under Display Format, select "Numeric table" from either the "From 10-mile calculation" option or the "From Close-in Calculation" option. Or select the "Special Receptors" option. The Close-in option provides data out to two miles. The Special Receptor option provides data at user-entered locations (a pre-defined set is also available).
3. Under Result Type, select "Deposition of" and Rb-88 from the drop-down menu.
4. Under Time Period, select "Single Time" and enter the time the field results were obtained.
5. Select the "Display Selected Results" button.
6. Repeat for Cs-138 and add to Rb-88 deposition.

D. Review of Data Comparisons

D.1 It is recommended that if results compare within a factor of 3, the agreement should be considered good, and no adjustments should be made. If the results do not agree within a factor of 3, then adjustments to model inputs are recommended if the field team results are considered credible.

D.2 The EOF Radiation Controls Manager should review the comparison data. It is not expected that the model and field results will be consistent. Enclosure 6 provides a discussion of some of the more likely reasons there will be differences and uncertainties in the various results. The uncertainties in Enclosure 6 should be considered in trying to confirm the validity of any of the results and in making conclusions concerning offsite radiological conditions.

D.3 After review of the applicable data, the EOF Radiation Controls Manager will determine if adjustments should be made to the assumed model inputs to RASCAL. Instructions will be provided to the Dose Assessment Team members running RASCAL.

TABLE 1

COMPARISON OF NOBLE GAS (GAMMA) FIELD MEASUREMENTS AND CALCULATED DEEP DOSE EQUIVALENT (DDE) DOSE RATE ESTIMATES

Excel spreadsheet L:\Shared\RASCAL\EM-219 Enclosure 3 Spreadsheets.xlsx also performs these calculations.

	TIME	LOCATION DISTANCE/ SECTOR	NOBLE GAS (GAMMA) FIELD MEASUREMENT mRem/hr.	RASCAL External Dose Rate Closed Window mRem/hr.	FIELD / RASCAL
1.					
2.					
3.					
4.					
5.					
6.					
7.					
8.					
9.					
10.					

Median*of Ratios =

*- To determine the median of the ratios, rank the ratios in ascending order. If there is an odd number of values, the middle value will be the median. If there is an even number of values, the median will be the average of the two values around the middle point.

Performed by: ____________________________ Verified by: ________________________________

TABLE 2

COMPARISON OF FIELD MEASUREMENTS AND CALCULATED THYROID DOSE RATE ESTIMATES

Excel spreadsheet L:\Shared\RASCAL\EM-219 Enclosure 3 Spreadsheets.xlsx also performs these calculations.

	TIME	LOCATION DISTANCE/ SECTOR	FIELD MEASUREMENT		RASCAL THYROID DOSE RATE mRem/hr.	FIELD / RASCAL
			IODINE µCi/cc	THYROID DOSE RATE* mRem/hr		
1.						
2.						
3.						
4.						
5.						
6.						
7.						
8.						
9.						
10.						

* THYROID mRem/HR = (IODINE µCi/CC) X (DFI**)

Accident Type	**DFI (mRem/HR PER µCi/CC)
FHA	1.3E9
WGDTR	1.3E9
LOCAN	5E8
LOCAG	1E9
LOCAC	1E9
SGTRN	5E8
SGTRG	1E9
SGTRC	1E9

**DFI (DOSE FACTORS FOR IODINE) The DFI is a weighted average for total Iodine based on the distribution of Iodine isotopes in each accident type (The individual nuclide dose factors are based on Table 5.2 of EPA 400. DFI I-131 = 1.3E+9).

Median*** of Ratios = [ ]

***- To determine the median of the ratios, rank the ratios in ascending order. If there is an odd number of values, the middle value will be the median. If there is an even number of values, the median will be the average of the two values around the middle point.

Performed by: ______________________ Verified by: ______________________

TABLE 3

COMPARISON OF NOBLE GAS TO IODINE RATIOS
FIELD AND RASCAL

Excel spreadsheet L:\Shared\RASCAL\EM-219 Enclosure 3 Spreadsheets.xlsx also performs these calculations.

	TIME	LOCATION DISTANCE/ SECTOR	FIELD MEASUREMENT			
			DDE mRem/HR	NG µCi/CC*	I µCi/CC	NG TO I RATIO
1						
2						
3						
4						
5						
6						
7						
8						
9						
10						

AVERAGE FIELD TEAM RATIO = ______________ ÷ ______________ = ______________
sum of the ratios   number of ratios   avg. field ratio

RASCAL RATIO = ______ NG CURIES ÷ ______ I CURIES = ______________
RASCAL RATIO

Accident Type	DFNG** (mRem/HR PER µCi/CC)
FHA	2E4
WGDTR	5E4
LOCAN	7E5
LOCAG	1E6
LOCAC	1E6
SGTRN	7E5
SGTRG	1E6
SGTRC	1E6

*NG µCi/CC = (DDE mRem/HR) ÷ (DFNG**)
**DFNG (DOSE FACTORS FOR NOBLE GAS) CALCULATED FROM EPA-400 TABLE 5.3

Performed by: ______________________________ Verified by: ______________________________

ADDITIONAL INFORMATION ON RASCAL DETAILED RESULTS OPTIONS

To facilitate comparison of dose projection data to field team data, the following information is provided to supplement instructions in Enclosure 3 for accessing the specific RASCAL parameters needed for each of the comparison methods.

1. SELECT the "Detailed Results" button to access the menu of numerous options for result types and display formats. The following are among the selections are available:

2. From 10-mile calculation: Displays data from typically 0.5 to 10 miles.
From Close-in calculation: Displays data from typically 0.1 to 2 miles.
Footprint: Displays data on a sector grid with color-coded ranges.
Numeric Table: Displays data in tabular format.

3. Special Receptor data:
SELECT "Define Receptors"
ENTER receptor name (location), bearing degrees, and distance OR
SELECT "Load" and "Special Receptors.txt" and "Open" for a standard set.
SELECT "OK"
SELECT "Special receptors"
SELECT Result Type and Time Period as noted in the bullets below.
SELECT "Display Selected Result"

4. External dose rate – Open Window (units are mR/hr)
Display Format = Numeric Table
Time Period = Single Time with time of field reading selected.
SELECT "Display Selected Result"

5. External dose rate – Closed Window (units are mR/hr)
Display Format = Numeric Table
Time Period = Single Time with time of field reading selected.
SELECT "Display Selected Result"

5. I-131 Air Concentration (units are Ci/m3 which equals µCi/cc)
Display Format = Numeric Table
Time Period = Single Time with time of field reading selected.
SELECT "Display Selected Result"

7. Field readings are probably total iodine concentration. RASCAL I-131 concentration can be converted to total by comparing the deposition rate of I-131 to the sum of deposition rates of the all the iodine isotopes. SELECT as follows:
Deposition of (select isotopes, units will be Ci/m2/sec)
Display Format = Numeric Table
Time Period = Single Time with time of field reading selected.
DETERMINE each iodine isotope deposition rate.
SUM deposition rates and DETERMINE fraction of I-131
Total iodine concentration = I-131 concentration / fraction of I-131

8. Deposition of (select isotope, units will be Ci/m2 accumulated)
(Cs-138 and Rb-88 are likely significant contributors.)
Display Format = Numeric Table
Time Period = Cumulative over interval with time of field reading selected.
SELECT "Display Selected Result"
SUM deposition for all isotopes selected.
DPM/100 cm2 = Ci/m2 X 2.22E10

Date:

FIELD TEAM DATA mR/hr											
Miles	Sector	Time:		Time:		Time:		Time:		Time:	
		DDE	THY	DDE	THY	DDE	THY	DDE	THY	DDE	THY

Uncertainties in Offsite Dose Assessment

A. General

A dose model is made up of 3 primary parts: an estimate of the source term, an estimate of the meteorological dispersion, and an estimate of the dose given a calculated concentration. All 3 of these parts are subject to many uncertainties, as listed below.

A field team measurement eliminates the first two parts of the dose model and hence all the associated uncertainties. However, it adds other uncertainties such as the use of an instantaneous reading as being representative of a time averaged dose.

The importance of the various uncertainties is very dependent on the exact conditions of the event in progress. There are no fixed rules that can be applied. Decisions based on radiological conditions will not be an exact science, but will be a subjective decision based on all available information and judgment.

Given the numerous significant uncertainties involved, it is not recommended that valuable time be spent trying to resolve differences between results (model vs. field team, utility compared to State or NRC, etc.) that are within the same order of magnitude. In most cases, such agreement should be considered good enough for decision-making.

B. Model – Source Term Uncertainties

The following are some of the uncertainties associated with estimating the source term for model input:

1. Unmonitored releases – it is very likely that for the more significant release events (when dose model results are more important) that the release will be unmonitored. This could result from a loss of all power to the radiation monitors or release paths that bypass the monitored pathways. In such cases there will be no measure of the magnitude of the radioactivity concentration, nor of the flow rates from the release pathway. Estimates based on default assumptions as to what the release might be for a given accident can be many orders of magnitude high or low as they are based on one set of conditions (e.g. – assumed amount of fuel failure or RCS concentration, assumed containment leak rate or tube rupture leak rate, etc.).

2. Radiation Monitor Uncertainties – if the release path is monitored, there are many uncertainties associated with the use of the radiation monitor results. These include:
- High background from direct sources resulting from the accident giving unknown detector response to the general area dose rates.
- Particulate daughter interference – the noble gas particulate daughters Rb-88 and Cs-138 will be 2 significant contributors to dose rates during the first few hours post trip, assuming some core damage. These particulates will plate out in the gas sample chamber or on the charcoal filter and mask any contribution from the noble gases or iodines the monitor is attempting to quantify.
- Degraded conditions – degraded conditions during an accident could include loss of power or degraded voltage, high temperature and humidity, or monitor saturation. These conditions can significantly affect the monitor response.
- Conversion factor dependence on mix – the monitor conversion factor is based on an assumed mix of nuclides, or even a single nuclide calibration. The mix of nuclides in an accident can be significantly different than that assumed resulting in an inappropriate conversion factor for that event.
- Noble gas interference on iodine channels – for monitored pathways, the noble gas activity should be much higher than the iodine activity. Therefore, any response on the iodine monitor is likely due to noble gases being delayed in the charcoal cartridge (plus particulate daughter buildup). Iodine monitor results would likely significantly overestimate the iodine releases.

3. Highly Variable Release Rates – an accident will usually be associated with many transient conditions – more fuel will fail over time, radionuclide concentrations will build up in a building, flows from the source_volume will change as pressures change, or fans are brought back into service. Hence, the release rate is expected to be highly variable over short periods of time. One release rate must be chosen to represent the entire 15 or 30 minute model step. Should that be the peak release rate, the current release rate, or some estimated average, which may be difficult to make.

4. Iodine and particulate removal mechanisms -there are numerous mechanisms for the removal of iodines and particulates. These include partitioning, water scrubbing, washout, deposition, and filters. Each of these is highly dependent on the release path and conditions. For example, an OTSG tube rupture at a location below the secondary side water level will result in an iodine removal factor in the OTSG of approximately 100, compared to a DF of approximately 3 if the break is above the water level. Although default removal factors for iodine are incorporated into the dose procedures/models, they can be orders of magnitude off from actual conditions. Iodine levels will spike following a reactor trip, but noble gases will not. Hence, a constantly changing iodine to noble gas ratio with time can be expected.

5. Nuclide mix – the model has assumed mixes of nuclides for the various accident types. The accident mix could be significantly different, which will affect the dose per curie. For example, the normal coolant mix (LOCAN, SGTRN) of iodine has I-131 at 1% based on the past few cycles of actual RCS coolant sample results. However, what if the accident doesn't cause gross fuel clad damage, but causes 12 pins to start leaking. I-131 may now increase to 20% of the RCS mix. This would result in a 10 factor higher dose per Ci/sec, but would be uncompensated for in the model unless we entered grab sample isotopic results. If an isotopic analysis was obtained, that isotopic represents one isolated period of time in an ever changing nuclide mix period.

6. Sample representativeness – accident conditions could cause normal sample uncertainty issues to be amplified. For example, if the flow rate from the vent is not the normal flow rate the sample will not be isokinetic. If there is more water vapor in the sample due to leakage into the Auxiliary Building, it could affect sample line losses.

C. Model – Dispersion Uncertainties

1. Single Point Measurement/Straight Line Model – the model assumes that the plume goes in a straight line for the entire advection period based on a single location meteorological measurement. A wind field model, based on multiple meteorological data location inputs would in many cases predict a much different plume location, more representative of actual conditions. Hence the current simplified model may predict high doses where there is no dose, and no doses where there are high doses. This would include phenomenon such as the sea-breeze effect, where, for example, the indicated wind direction is from the West, but 3 miles inland, the westerly sea-breeze ends and the plume takes the direction of the prevailing local winds, even back to the east where there could be some reconcentration.

2. Release elevation – Releases are assumed at ground level. The recommended meteorological data to be used is the 33' data, representative of ground dispersion conditions. In most cases, the releases will be partially elevated. For AB/RB vent releases, the fraction that is ground and the fraction elevated depends on the ratio of the wind speed to the vent exit velocity. For containment failure events, even though the release point may be near ground, the containment air leaking out may be near 200 degrees and hence have thermal buoyancy. Releases from the ADV's and MSSV's will have both momentum and thermal plume rise and be essentially totally elevated. The wind direction for the elevated portion of the plume may be significantly different than the ground wind direction. Wind speeds could be a factor of 2 or 3 different. The plume touchdown point would be important as discussed below.

3. Plume touchdown point – If releases are assumed to be at ground level, RASCAL always predicts a concentration at ground level. For elevated releases, plume touchdown may not be for several miles. For the DDE, this would only result in a difference of a factor of approximately 5 between an assumed ground release and a plume that is still 400 feet above the ground, as there will still be a gamma ray flux (shine) from the overhead cloud. However, for iodine, the code would predict ground level iodine concentrations resulting in a calculated thyroid dose when there would be no thyroid inhalation dose until the plume touched down.

4. Complicating factors – there are a number of real-life factors that affect the plume such that it does not behave as a straight-line gaussian function. These factors include building wake effects, low-wind speed plume meander, terrain effects, fumigation effects, lid reflection with variable lid height, buoyant gases or heavier than air gases, plume depletion due to decay and deposition, rainfall. Corrections are made from some of these factors, but each correction is a simplification that does not match reality and each adds more uncertainty.

D. Model – Dose Estimation

Compared to the significant uncertainties in the source term and dispersion estimates, the dose estimation part is more accurate. However, it is still prone to a number of uncertainties, including:

1. Time of exposure – since the EPA PAG's are in integrated dose, the calculated dose rate must be multiplied by an exposure time. That could be 1 hour or 5 hours before the release stops or the wind direction changes significantly.

2. Receptor – the dose could be calculated to the adult, or another age group such as the child or infant, who may not be the limiting individual depending on the dose pathways and mix of nuclides.

3. Standard man assumptions – built into the dose factors are many assumptions such as breathing rates and organ sizes. These factors vary for each individual. A jogger who is breathing heavy is going to receive a higher dose than that based on a standard man assumed breathing rate.

4. Finite cloud corrections – the model may assume that the plume is semi-infinite. This would overestimate the DDE, particularly for very stable conditions at the site boundary, where the actual plume may be very narrow. Finite cloud correction factors can be applied, but again this adds uncertainty and they are typically based on a fix nuclide mix.

E. Field Team Uncertainties

1. Finding the plume – as noted above, the plume may not be in the down wind sector as indicated by the single point meteorological tower. Hence, the field team may not be in the right location to monitor the plume. (Note however that a zero dose rate reading does provide important information – it provides the actual radiological conditions at that location, indicates that the model is not accurately predicting the plume if it had shown measurable dose rates at that location (at least as far as plume location), and combined with zero readings from other locations is part of a demonstration of a lack of a significant release.) If the field team is detecting activity, then it can be uncertain as to whether the team has found the maximum dose rates or is on the fringe of the plume.

2. Instantaneous Readings – the model provides time averaged dose rates over a 15 or 30 minute period. Field team readings are instantaneous. The plume will meander. It will not always be over the averaged plume centerline. Dose rates at what will be the time averaged plume centerline at any one instant could be close to zero. Dose rates 15 degrees off plume centerline could be higher than the calculated plume centerline dose rate for any one instant (For example - if the time averaged centerline dose rate is 1 R/hr, then for the periods of time that the plume is over the centerline, the dose rate right at the plume centerline must be greater than 1 R/hr to compensate for the times when the plume is off centerline and the dose rate at that location is close to 0. Hence, at a particular instance the plume centerline may be 15 degrees off of the time-averaged centerline and the dose there may be 3 R/hr.) Uncertainties associated with instantaneous data can be reduced by having the field team determine a time averaged dose rate over 5-15 minutes at the same location. However, this adds the uncertainty of trying to eyeball an average without a recorder on a varying dose rate meter.

3. Contamination – with the deposition of noble gas particulate daughters, there is a good possibility that once in the plume, the field team's survey meter, as well as their clothing and vehicle may become contaminated to the point that subsequent surveys are detecting dose rates from the contamination and not the plume. It could also become difficult to distinguish plume dose rates from shine dose rates from the contaminated ground.

4. Plume arrival time – the model, which is a 15 or 30-minute projection, may predict a dose rate at some distance downwind, yet the actual plume may not have traveled that far yet. Hence a field team may detect no dose rate, when there will be one in a few minutes. Likewise the field team may never catch up with a short-term puff release, where the release has stopped, but there is still a plume within the Emergency Planning Zone.

RADIOLOGICAL MONITORING TEAM DEPLOYMENT STRATEGY

Every situation is different. No set of rules on monitoring team deployment will work for all situations. However, the following provides some general principles that could be employed in most cases. The guidance is limited to priorities within the first few hours of an event.

A dose rate survey taken close to the plant in all compass directions (or at least a 180 degree sweep centered on the perceived downwind direction) will provide a rapid indication of the order of magnitude plume of dose rates. Therefore, as long as direct shine dose rates from sources such as the containment do not interfere with accurate plume readings, a walk around the berm would confirm the lack of a measurable plume or would readily find a plume maximum instantaneous dose rate. If sources onsite result in high direct dose rates, then this close-in dose rate survey could be performed at a distance of approximately 1000 feet. There is a system of roads to the fossil plants at this approximate distance that would make it possible to perform a 360 degree survey within a short time period. Note that this survey does not require a team with their full kit of air samplers, etc.

Once a plume is located and the general downwind location is known, two teams (if available) with survey meters should be located in the general downwind direction, close to the plant, approximately 2 or 3 sectors apart. They should continuously observe the plume dose rate. This will help distinguish whether changes in the measured dose rate are due to changes in the wind direction or changes in the release rate.

In most cases it will be important to get at least one air sample in the plume immediately. This is to confirm whether iodine is a significant contributor to the dose compared to the noble gases and establish a more credible iodine to noble gas ratio to be used in future calculations. Therefore, the Environmental Survey Team should find a location that they know is in the plume, by ensuring the window open dose rate reading is at least two times the window closed reading. Once the air sample is obtained, it is recommended that it be immediately transported for a gamma spectrum analysis. This is due to the high potential for noble gas and particulate interference on a gross count rate meter. If a ground release, this sample can likely be obtained on site (within the owner-controlled area). If a vent release, the team may have to search at the approximate site boundary distance to ensure they are in the plume. If an effectively elevated release, such as from the Atmospheric Dump Valves, the plume may not touch down for many miles. Before an Environmental Survey team is dispatched a far distance to look for plume touchdown, it should be confirmed that the plume will have traveled that far by that time and that a second survey team is available to continue to survey close to the plant/site to ensure changing conditions are rapidly identified.

REVISION SUMMARY

PRR # 513751

NOTE:

1. **Writers and Reviewers:** **Ensure that any changes to this procedure that affect information contained in ERF posters, enclosures, briefing cards, guidelines, etc. are made to those items as well.**

2. **Writers and Reviewers:** **Changes to certain parts of this procedure may impact other Emergency Plan-implementing procedures. Specifically, if any changes are made to: EM-219 section 3.2, review EM-400. Ensure appropriate PRRs are initiated as needed.**

PROCEDURE SECTION	CHANGES AND REASONS
Throughout	Increased revision number to 21, editorial (spelling, punctuation, spacing, etc.) corrections
Step 4.2.8 and the Caution preceding step Page 8	Deleted step and Caution because the new Radiation Monitors to be installed under EC76363 do not operate in this manner and do not have this valve controller. Renumbered remaining steps in section 4.2.
Enclosure 2 (page 1 of 6) Page 12 of 33	Deleted Caution and first paragraph because the new Radiation Monitors to be installed under EC76363 do not operate in this manner and do not have this type of valve controller.
Enclosure 2 Step 3.4 (page 2 of 6) Page 13 of 33	Changed RM-A1 and RM-A2 "low" range to "normal" range to represent the function of the replacement Radiation Monitors per EC 76363.

Progress Energy	I Information Use

CRYSTAL RIVER UNIT 3

PLANT OPERATING MANUAL

EM-225

DUTIES OF THE TECHNICAL SUPPORT CENTER ACCIDENT ASSESSMENT TEAM

REVISION 27

TABLE OF CONTENTS

1.0 PURPOSE

1. This procedure provides guidance for the establishment and operation of the Technical Support Center Accident Assessment Team (AAT), for the determination of core and fission product barrier status, and for the interface with the Radiation Controls Coordinator. Information from these assessments will be used in conjunction with other guidance for development of accident mitigation strategies. This procedure also provides guidance to the AAT to perform actions described in the EOPs [NOCS 062718].
2. This procedure is an emergency plan implementing procedure. Any revisions to this procedure must be carefully considered for emergency plan impact.

2.0 REFERENCES

2.1 Developmental References

1. Response Technical Manual (RTM-96); USNRC; Volume 1, Rev. 3
2. Radiological Emergency Response Plan
3. Emergency Operating Procedures (EOPs)
4. NUREG-1228, Source Term Estimation during Incident Response to Severe Nuclear Power Plant Accidents
5. B&W Technical Bases Document
6. FPC IOC CR97-0122, Dated 12/23/97
7. NEI 91-04, Revision 1, Severe Accident Issue Closure Guidelines
8. FPC IOC SE99-0184, Dated 9/14/99
9. EEM-99-018, Rev. 0 Operating Limits for SWP-1A/SWP-1B under Minimum Flow Conditions
10. EM-202, Duties of the Emergency Coordinator
11. EM-102, Operation of Technical Support Center
12. EM-103, Operation and Staffing of the CR-3 Control Room During Emergency Classification
13. CP-151, External Reporting Requirements
14. Generic Letter 2004-02, Potential Impact of Debris Blockage on Emergency Recirculation During Design Basis Accidents at Pressurized-Water Reactors
15. EC 58982, RB Sump Strainer Modification
16. EC 59476, RB Sump Level Instrumentation Modifications
17. EC 55315, Alternate AC Diesel Generator
18. EC 66671, Installation of CR3 Intrusion Detection System (Firewall) and Refinement of the CR3 Cyber Security Defensive Model
19. INPO IER 11-2, Fukushima Daiichi Nuclear Station Sent Fuel Pool Loss of Cooling and Makeup
20. 10CFR50.54(x) or Section 24 of the Physical Security Plan
21. 10CFR50.72(a)(1)(i) and 10CFR50.72(b)(2)(i)

Subsection 2.1, Developmental References (Cont'd)

22. EEM-01-021
23. FSAR Table 4-10
24. IOC SE-99-0184
25. NOCS 62718, 62764, 62767, 96042, 100056, 100408, 100441, and 100483
26. IER 11-46 Extended Emergency Power Operations Following A Loss Of Off-Site Power
27. EC 84553, Minimum Expected EGDG Electrical Loads For R16 Extended Outage
28. Calc M89-0063, Waste Gas Decay Tank Rupture Environmental Condition

2.2 **Implementing References**

1. CR-3 Severe Accident Guideline
2. Emergency Response Personnel Roster
3. AP-770, Emergency Diesel Generator Actuation
4. AP-990, Shutdown from Outside the Control Room
5. CH-632, Post Accident Sampling and Analysis of the Reactor Coolant, Decay Heat, and Reactor Building Sump
6. CP-151, External Reporting Requirements
7. EMG-NGGC-0002, Off-Site Dose Assessment
8. EM-103, Operation and staffing of CR-3 Control Room during Emergency
9. EM-202, Duties of the Emergency Coordinator
10. EM-206, Emergency Response Organization Notification
11. EM-225A, Post Accident RB Hydrogen Control
12. EM-225B, Post-Accident Boron Concentration Management
13. EM-225C, Post Accident Monitoring for Reactor Building Temperature
14. EM-225D, Guidance for Dry OTSG Tube to Shell Delta T Monitoring and Control
15. EM-225E, Guidelines for Long Term Cooling
16. EM-225F, Long Term Emergency Feedwater Management
17. EOP-3, Inadequate Subcooling Margin
18. EOP-5, Excessive Heat Transfer
19. EOP-6, Steam Generator Tube Rupture
20. EOP-7, Inadequate Core Cooling
21. EOP-8A, Loca Cooldown
22. EOP-8B, HPI Cooldown
23. EOP-9, Natural Circulation Cooldown
24. EOP-12, Station Blackout
25. EOP-14, Emergency Operating Procedure Enclosures
26. MP-575, Hydrogen Recombiner Installation
27. MP-815, Installation of Post Accident H2 Purge Flow Instruments
28. OP-103C, Cycle 17 Reactivity Worth Curves
29. OP-417B, Operation of the Post Accident Hydrogen Recombiner
30. SP-306, Routine Surveillance Log

3.0 DEFINITIONS

1. **Accident Assessment Team (AAT):** Consists of Coordinator, TSC Ringdown Communicator, Control Room Ringdown Communicator, Engineer, Operations Support, and NRC Communicator.
2. **Candidate High Level Actions (CHLA):** Actions described in the CR-3 Severe Accident Guideline which could be taken to mitigate a Severe Accident and are deemed appropriate based on Plant Damage Conditions.
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 complex status.
4. **Inadequate Core Cooling:** Accident conditions that result in a loss of core cooling that requires entering EOP-7, Inadequate Core Cooling.
5. **Emergency Action Levels (EALs):** Conditions or indications that may be used as thresholds for initiating specific emergency measures (see EM-202, Duties of the Emergency Coordinator, Enclosure 1).
6. **Plant Damage Conditions (PDC):** Damage conditions used in the CR-3 Severe Accident Guideline to describe the status of the reactor coolant system, reactor core, and the containment during the progression of a Severe Accident.
7. **Protective Action Recommendations (PARs):** Emergency measures recommended for purposes of preventing or minimizing radiological exposures to the Energy Complex personnel or members of the general public.
8. **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 Rx vessel, Reactor Building or the environment.
9. **Full HPI:** The conditions necessary to ensure ≥ the minimum required HPI flow assumed in the plant design basis. These conditions include: at least 1 MUP running with HPI flow through all 4 HPI nozzles (all 4 HPI valves open, or HPI crossties open with 1 train of HPI valves open) with one of the following:
    - HPI recirc to sump, MUP recirc, MU flowpath to the RCS, and RCP seal injection flowpaths isolated.
    - Total HPI flow is in the "Acceptable Region" of the "Minimum Required HPI Flow" figure.
10. **Less Than Full HPI** (Inadequate HPI flow, maximum cooldown in progress per EOP-3, Inadequate Subcooling Margin)**:** Not all portions of the HPI flow path satisfy the independence criteria discussed in the CR3 ITS. Specifically, the HPI flow path downstream of the HPI/Makeup pumps is not separable into two distinct trains, and is therefore, not independent. As such, in the event of a postulated break in the HPI injection piping, injection flow is required through one of the following alignments:
    - A minimum of three (3) intact injection legs, assuming one pump operation
    - A minimum of two (2) intact injection legs, assuming two HPI pump operation.

4.0 RESPONSIBILITIES

1. Control Room Ringdown Communicator:

   - Reports to the Control Room and establishes communication with the TSC Ringdown Communicator on the Accident Assessment Ringdown phone. Brief TSC Ringdown Communicator on operator actions that are in progress.
   - Relays status of overall plant conditions, operator activities and questions to the TSC AAT.
   - Relays instructions to Control Room Operators for mitigating actions as directed by the Emergency Coordinator (EC).
   - If a Severe Accident is occurring, directs Control Room personnel regarding actions to take to mitigate the Severe Accident, based on actions approved by the TSC EC.
   - Relay request for support from the Control Room to OSC teams, via TSC Ringdown Communicator.
   - Once TSC is operational, request extra plant operators (if available) be sent to OSC for in plant support.
   - Inform TSC of in plant operator actions that are being performed.

2. AAT Coordinator:

   - Informs the EC of any developments in plant status that may impact EALs and PARs.
   - Ensures appropriate AAT personnel have staffed the TSC.
   - Ensures additional AAT members are notified as needed.
   - Identifies plant parameters to be tracked.
   - Coordinates AAT activities and ensures that team members remain focused on objectives.
   - Keeps the EC informed of AAT activities.
   - If a Severe Accident is occurring, reviews recommended Candidate High Level Actions and mitigation plans prior to submitting to the Emergency Coordinator. [NOCS 100056]
   - If a Severe Accident is occurring, coordinates efforts of the Accident Assessment team to ensure the development of mitigation strategies using the CR-3 Severe Accident Guideline.
   - If additional resources are needed, coordinates with the EOF Technical Support Team to provide required support.
   - Establishes communications with the Emergency Operating Facility (EOF) Technical Support Team, if the EOF is staffed.
   - Approve Attachment 11, OSC Request Form to request operator actions outside CCHE or maintenance repair activities that have been initiated by the Control Room or AAT. This request should be processed through TSC Repairs Coordinator to the OSC.
   - Notifies Shift Manager in requal training for additional Operations support.
   - Ensures TSC display screen computers are logged in. If computer room door is locked, contact Security for access.

3. TSC Ringdown Communicator:

   - Establishes communications with the Control Room Ringdown Communicator on the Accident Assessment Ringdown phone.
   - Relays information on changing radiological conditions and maintenance activities to the Control Room.
   - Relays plant conditions from the Control Room to the TSC AAT.
   - Maintains the Accident Assessment Team Log.
   - Relays information and directions to the Control Room of actions required to mitigate a Severe Accident based on approved Candidate High Level Actions.
   - Monitors progression through EOPs and APs.
   - Initiate Attachment 11, OSC Request Form to request operator actions outside CCHE or maintenance repair activities for the OSC that is requested by the Control Room or AAT.

4. AAT Engineers:
   - Assesses plant conditions and provides engineering support for developing accident mitigation strategies as needed.
   - Aids in determining additional Engineering resources.
   - Monitors plant parameters for indications of core damage and status of fission product barriers.
   - During Severe Accident conditions, evaluates plant parameters, determines Plant Damage Conditions, and develops Candidate High Level Action recommendations using appropriate calculational aids from the CR-3 Severe Accident Guideline.
5. AAT Operations Support:
   - Monitors overall plant status during an emergency with emphasis on Critical Safety Functions.
   - Functions as a technical resource for Operations in assessing plant conditions and in development of accident mitigation strategies that are outside the scope of Emergency Operating Procedures (EOPs).
   - Maintains the CSF Status Board at the TSC.
   - During Severe Accident Conditions, provides support to the AAT Engineers in determining Plant Damage Conditions and developing mitigation strategies using the CR-3 Severe Accident Guideline.
   - Coordinates/processes requests for operator actions or maintenance support activities through the TSC Repairs Coordinator using Attachment 11, OSC Request Form.
   - Determine emergency and non-emergency notifications to the NRC as defined in CP-151, External Reporting Requirements.
6. NRC Communicator: [NOCS 96042]
   - Maintains an open, continuous communication line on the Emergency Notification System with the NRC Operations Center upon request by the Headquarters Operations Officer.
   - Log times NRC is notified of Emergency Classification changes and Protective Action Recommendations.
   - Make emergency and non-emergency notifications to the NRC as defined in CP-151, External Reporting Requirements.

7. EOF Technical Support Team:
   - Functions as a technical resource for the EOF Director in development of PARs by monitoring plant conditions (particularly the CSFs).
   - Assists the TSC AAT team as needed in development of mitigation strategies and in research of solutions to plant problems.
   - Responsible for the development of long-term recovery plans.
8. Emergency Coordinator (EC) or designee:
   - Controls all activities at CR-3 during activation of the Radiological Emergency Response Plan.
   - Implements EM-202, Duties of the Emergency Coordinator.
   - Determines EAL and PAR changes based on information obtained from the Accident Assessment Team and Radiation Controls Coordinator.
   - Functions as the decision maker during a Severe Accident. The EC will approve all recommended Severe Accident mitigation strategies prior to implementation.
   - Is authorized to declare 10CFR50.54(x and y) to implement emergency actions deemed necessary to protect the health and safety of the public. A separate notification is required to the NRC for each occasion. Once a Severe Accident is declared, only one notification to the NRC is required.
9. Radiation Controls Coordinator:
   - Supports the Accident Assessment team with on-site radiological data and with chemical and radiological analysis of samples as needed to assess the accident.
   - Provides Plant Radiation Monitor readings and assessments.
   - Provides projected radiological data (on-site and off-site doses, dose rates, and deposition) (> 1 hour to obtain).
   - Provides capability to obtain RCS samples for boron concentration.
   - Provides capability to obtain grab samples for RB Atmosphere and RB/AB Vent.
   - Provides in-plant radiological data.
   - Provides chemical and radiological analysis of OTSGs and secondary samples.
   - Provides Reactor Building sump boron concentration (> 1 hour to obtain).

5.0 PREREQUISITES

None

6.0 PRECAUTIONS, LIMITATIONS AND NOTES

Under Severe Accident Conditions, plant instrumentation may provide false or highly inaccurate readings due to harsh environments beyond their qualifications. Several instruments should be monitored along with trends to assess plant conditions.

7.0 SPECIAL TOOLS AND EQUIPMENT

None

8.0 ACCEPTANCE CRITERIA

None

9.0 INSTRUCTIONS

9.1 Accident Assessment Initiation

1. [AAT Coordinator or designee] PERFORM the duties of Attachment 1, AAT Coordinator Checklist.
2. [TSC Ringdown Communicator] PERFORM the duties of Attachment 3, TSC Ringdown Communicator Checklist.
3. [AAT Operations Support member] PERFORM the duties of Attachment 4, AAT Operations Support Checklist.
4. [AAT Engineers] PERFORM the duties of Attachment 5, AAT Engineers Checklist.
5. [Control Room Ringdown Communicator] REPORT to the Control Room **AND** PERFORM the duties of Attachment 6, Control Room Ringdown Communicator Checklist.
6. [NRC Communicator] PERFORM the duties of Attachment 7, NRC Communicator Checklist.

10.0 RECORDS

All attachments are quality records

TSC GUIDANCE FOR EOPS
[NOCS 62718, 62764, 62767]

This enclosure provides the relationship with the EOPs and TSC guidance during emergency events. It is management's expectation that the guidance steps will be implemented, based on the emergency condition of the plant, by either invoking 10 CFR 50.54 (x), (y), formal 10 CFR 50.59 reviews and approvals, or by existing approved procedures.

PARAMETER	EOP	TBD REF.	TSC GUIDANCE
RB Hydrogen Control	EOP-3, EOP-6, EOP-7, EOP-8A, EOP-8B	HPIC, 5.4 III.F, 6.2, 10.0, 12.6b, 13.6b LBLO 4.4, 6.3 SBLO 12.4, 20.3, 9.3	1. Align hydrogen monitoring equipment using EOP-14, Enclosure 2, PPO Post Event Actions. 2. Monitor hydrogen concentrations using EOP-14, Enclosure 21, RB Hydrogen Monitor Log. 3. Purge RB when authorized per EM-225A. [NOCS 62767] Interfacing references are: • EM-206 for telephone number for procurement representative to obtain recombiners • MP-575 for installation of recombiners • OP-417B for operation of recombiners • MP-815 for installing H^2 purge flow indicators
Building Spray Termination Criteria	EOP-3, EOP-8A, EOP-8B EOP-14,Enc 19	None	If RB sump strainer blockage occurs consider alternate criteria for BSP shutdown (See EM-225E, Section 9.6) Verify all of the following before terminating Building Spray: 1. BS has been on for > or equal to 5 hours. 2. RB pressure is < 10 psig. 3. RB pressure is stable or lowering. 4. RB atmosphere is < 13 µci/cc I-131. 5. RB temperature is stable or lowering (also refer to EM-225C). 6. Concurrence is obtained from EC and Dose Assessment to terminate BS.
SFP Level and Temperature Trending	EOP-06, EOP-8A, EOP-8B, EOP-10 EOP-12 EOP-14 Enc 24	V1-IIIE V1-IIIA V1-IVA	Perform EOP-14 Enc 24, Monitoring Spent Fuel Parameters Interfacing references are: 1. AP-406, Loss of SFP Cooling 2. AP-1080, Refueling Canal, SFP level Lowering 3. AAG-05, Contingencies for Loss of SFP Level

TSC GUIDANCE FOR EOPs

PARAMETER	EOP	TBD REF.	TSC GUIDANCE
Continue Cooldown With DHR System	EOP-6, EOP-8A, EOP-8B	FF, 11.5 NC, 11.4	Verify all of the following: 1. Begin establishing a Post Accident Recovery Plan (this can be done during plant cooldown). 2. The reactor is being cooled by DHR. 3. DHR cooling is consistent with maintaining adequate SCM. 4. The RCS is subcooled (use DH cooler outlet temperature for cooldown rates). 5. The RCS is depressurized. 6. Prohibit establishing any flow path that was isolated by the ES system unless the potential for radioactive releases is evaluated and the release path, doses, and methods have been approved by the EC. 7. Control of containment penetrations has been established. 8. Monitor and maintain RCS boron concentration for required shutdown margin.
Steaming an isolated OSTG for TRACC	EOP-6	III.E	Steaming an affected OTSG may be desirable for the following reasons: • Increase cooldown rate • Prevent challenging tube to shell dT limits • Prevent idle loop voiding when in natural circulation. All of the following conditions should be evaluated to determine if steaming an affected OTSG is appropriate: 1) BWST > 35 ft (1) **AND** 2) Affected OTSG Level < 90%(2) **AND** 3) Any of the following conditions exists: • Steaming is required to avoid core damage 1. Estimated OTSG leakage times RCS DE I-131 concentrations is < 0.4 OTSG Leakage (gpm) X Initial RCS DE I-131 (µci/gm) < 0.4 • Wind is blowing off-shore (Off-shore winds originate from NNE to SE sectors 011.2° to 146.3°)

Note (1) - If BWST level is < 35 ft, then determine if adequate BWST level is available for long term cooldown (Ref calc M89-1089) prior to steaming the OTSG.
(2) - If OTSG level is > 90%, then determine if OTSG level is low enough to prevent water carry-over. As long as water level can be ensured to be below the bottom of the main steam outlet nozzles there should not be any carry-over concern.

TSC GUIDANCE FOR EOPs

PARAMETER	EOP	TBD REF.	TSC GUIDANCE
BWST Makeup	EOP-6	III.E	Monitor BWST level trend and evaluate depletion rate. Ensure adequate BWST inventory is available to support RCS cooldown to DHR. Evaluation should include the following: • Primary to secondary leak rate • BWST available inventory • BWST depletion rate • Current RCS temperature • BWST volume required to support cooldown (refer to OP-304) • Potential for leak rate increase (leak before break) **IF** ECCS water supplies are insufficient to support cooldown to DHR, **THEN**, make preparations to initiate BWST makeup from spent fuel pools. • Refer to EM-225E, Enclosure 11, BWST Refill from Spent Fuel Pool
RCS Leakage No Longer Exists	EOP-8A, EOP-8B	None	1. The RCS is capable of being cooled by DHR. 2. Prohibit establishing any flow path that was isolated by the ES system unless the potential for radioactive releases is evaluated and the release path, doses, and methods have been approved by the EC. 3. Begin DHR.
Break size > 1 HPI Pump Capability or Unable to transition to DHR	EOP-8A, EOP-8B	None	1. Establish a Post Accident Recovery Plan. This plan is dependent on the scope of the applicable Emergency Event. 2. The Post Accident Recovery Plan is approved by the PNSC, and applicable regulatory agencies as determined by FPC Management. 3. Prohibit establishing any flow path that was isolated by the ES system unless the potential for radioactive releases is evaluated and the release path, doses, and methods have been approved by the EC. 4. The availability of borated water sources for required shutdown margin is maintained until the actions of the Post Accident Recovery Plan are completed or to the extent that plant and public safety is ensured. 5. Post and label protected train boundaries for the borated water sources and components that are available.

TSC GUIDANCE FOR EOPs

PARAMETER	EOP	TBD REF.	TSC GUIDANCE
Break size < 1 HPI Pump Capability and able to transition to DHR	EOP-8A EOP-8B	None	1. Transition to DHR cooldown. 2. Establish a Post Accident Recovery Plan. This plan is dependent on the scope of the applicable Emergency Event. 3. The Post Accident Recovery Plan is approved by the PNSC, and applicable regulatory agencies as determined by FPC Management. 4. Prohibit establishing any flow path that was isolated by the ES system unless the potential for radioactive releases is evaluated and the release path, doses, and methods have been approved by the EC. 5. The availability of borated water sources for required shutdown margin is maintained until the actions of the Post Accident Recovery Plan are completed or to the extent that plant and public safety is ensured. 6. Post and label protected train boundaries for the borated water sources and components that are available.
Establishing Primary to Secondary Heat Transfer to One or Both OTSGs		SS-2	1. Refer to the entry conditions and recommendations of the Emergency Operating Procedures Technical Basis Document (TBD), Section SS-2 for guidance related to establishing primary to secondary heat transfer to one or both OTSGs. 2. Accident Assessment personnel in the TSC will provide recommended guidance to the EC for when and how to establish heat transfer using one or both OTSGs. 3. The EC will approve any actions recommended.
Termination of HPI and Shutdown of RCPs	EOP-8A EOP-8B	LBLO, 2.2, 3.0	1. Recommended guidance is to stop HPI pumps and trip running RCPs when LPI flow has been in excess of 1400 gpm in each injection line for at least 20 minutes. Accident Assessment personnel will evaluate plant conditions and provide recommendations to the EC. 2. The EC will approve any actions recommended.
Control of Radioactive Release Paths from Containment Penetration Valves	EOP-8A EOP-8B	SBLO 12.0	1. Prohibit establishing any flow path that was isolated by the ES system unless the potential for radioactive releases is evaluated and the release path, doses and methods have been approved by the EC.
Monitoring of RB Sump Level, RB Sump Boron Concentration, RB Sump pH and RB Sump strainer ΔP	EOP-8A EOP-8B	None Other: IOC CR 97-0122	**NOTE:** With the installation of the TSP baskets, pH data is not required but still desired if feasible. 1. Accident Assessment personnel to monitor and trend RB sump level, boron concentration, pH and RB Sump strainer ΔP at intervals recommended by the EC. 2. Data for sump pH and boron concentration to be obtained using CH-632 or other PNSC approved alternate methods dependent on the Emergency Event.

TSC GUIDANCE FOR EOPs

PARAMETER	EOP	TBD REF.	TSC GUIDANCE
Venting of Non-Condensable Gases	EOP-8A EOP-8B	None	1. Once subcooling margin is regained, all of the noncondensable gas production will have ceased. However, as the RCS is depressurized these gases will come out of solution and should be vented. If natural circulation is lost to an available OTSG, Accident Assessment personnel will recommend to the EC when to vent noncondensable gases. 2. The EC will approve any actions recommended.
Reactor is Being Adequately Cooled Using HPI or LPI and OTSG Cooling is No Longer Desired	EOP-8A EOP-8B	SBLO, 17.7	1. Verify TBVs/ADVs are closed. 2. Fill available OTSGs to 90%. 3. Close EFW/AFW/MFW Valves. 4. Stop all EFW/AFW Pumps. 5. Stop MFWPs and MFWBPs.
Boron Concentration Management When Adequate Sub Cooling Margin Does Not Exist (Boron Precipitation)	EOP-8A EOP-8B EOP-14, Enc. 20	None	Refer to EM-225B **NOTE:** If a failure of ES MCC 3AB has occurred, ensure repair efforts are initiated to repower auxiliary pressurizer spray valve RCV-53 prior to the onset of boron precipitation.
RB Temperature Monitoring (To Preserve EQ Standards)			Refer to EM-225C
Feeding a Dry OTSG (Tube to Shell Delta T Monitoring and Control)	EOP-5, EOP-9, EOP-14, Enc. 3	III.D, 12.0 III.E, 17.7 NC, 5.2, 5.3, 6.4	Refer to EM-225D
Long-Term Core Cooling Using the RB Sump	EOP-8A EOP-8B	LBLO, 6.4a, 6.4b, 6.6, 6.7	Refer to EM-225E
EFW or AFW is Operating	EOP-14, Enc. 7 Enc. 22		Refer to EM-225F

TSC GUIDANCE FOR EOPs

PARAMETER	EOP	TBD REF.	TSC GUIDANCE
TBP-3 is Running. TBP-2 is Not Running. Generator Purge Complete	EOP-14, Enc. 14		TBP-3 will drain non-1E battery during LOOP. Stopping TBP-3 before 24 hours may result in Turbine bearing damage. Refer to IOC SE-99-0184
Concentrated BA addition made and flush water not available.	EOP-14, Enc 18	None	If concentrated BA is allowed to remain in the boron injection path piping (letdown/DH purification piping) the BA will eventually cool down and solidify. Timely action is required to preclude this condition. 1. Direct the control room to reestablish a continuous BA injection at a flow rate of 2 - 3 GPM (Batch controller is the preferred method). 2. Monitor RCS boron concentration. **DO NOT** allow RCS boron concentration to exceed the values listed in FSAR Table 4-10. 3. Evaluate the following options. • If plant conditions permit, expedite restoration of RCS letdown (or DH purification). • If plant condition permit expedite restoration of power to at least one source of flush water (DWP-1A, DWP-1B, WDP-5A, WDP-5B, or WDP-5C). • If BA flow rate and AB temperature conditions permit, evaluate securing continuous BA addition and performing periodic batch additions to prevent boron solidification. 4. **IF** letdown or DH purification flow is established, **THEN** direct the control room to STOP concentrated BA additions. 5. **IF** any flush water source becomes available, **THEN** direct the control room to STOP concentrated BA additions and perform a line flush using EOP-14, Enclosure 18. Refer to EEM-01-021, FSAR Table 4-10

TSC GUIDANCE FOR EOPs

PARAMETER	EOP	TBD REF	TSC GUIDANCE
Indications of RB sump strainer blockage have occurred.	EOP-14, Enc. 19	None	ECCS pumps have been aligned to the RB sump and are now showing signs of sump strainer blockage (flow oscillations, pump amp swings, high RB strainer ΔP). Per EOP-14, Enclosure 19, ECCS Suction Transfer, LPI flows should have been reduced to 1400 gpm per pump. At least one BSP should be secured. If both trains of LPI are in service HPI should be secured. If only one train of LPI is in service, one train of HPI must be aligned to the operable LPI pump in piggy back mode. [NOCS 100483 and 100408] 1. Verify proper ECCS pump configuration 2. Closely monitor ECCS pump parameters, Incore temperatures and RB sump strainer ΔP (Ref. Recall Point 79). 3. Expedite BWST refill operation from spent fuel pool using EM-225E Enclosures 11 and 12. 4. Expedite mixing of boric acid for BAST makeup per OP-403B, Section 4.2, Boric Acid Production. 5. Refer to EM-225E, Section 9.6, Contingency Actions for RB Sump Strainer Blockage, for specific guidance.
Non-Vital Battery Hydrogen	EOP-12 AP-770	None	During operation of the Alternate AC Diesel the potential exists for hydrogen accumulation in the Non-vital battery room. A portable fan must be set up within the first 24 hours of continuous operation of the Alternate AC Diesel to promote ventilation and the dilution of any hydrogen gas. A 120 VAC duplex receptacle exists adjacent to ACDP-176.
CFT isolation not closed, RCS depressurizing	EOP-3 EOP-8A EOP-12		EOP-3, EOP-8A and EOP-12 have requirements to maintain RCS pressure above 140 psig if LPI is not available and CFT isolation valves are not closed to prevent the CFT from further being depleted. The continued depletion of the CFT below an RCS pressure of 140 psig can allow nitrogen gas used to pressurize the CFTs to enter the RCS. This nitrogen intrusion may result in voiding and interfere with primary to secondary heat transfer (Natural Circulation). Maintain RCS pressure > 140psig, re-establish LPI or close the CFT isolation valves.

TSC GUIDANCE FOR EOPs

PARAMETER	EOP	TBD REF	TSC GUIDANCE
Station Blackout Guidance with only AAC Diesel supplying power to ES 4160V Bus. (no ES Diesel or offsite power available)	EOP-12	None	The AAC Diesel has more limitations then the ES Diesels based on additional loads on 4160 Rx Aux Bus 3 (i.e. Non-1E battery chargers). The AAC Diesel does not have the capability to load sequence and therefore ES is blocked from actuating when the AAC Diesel is powering the ES 4160V Bus. All actions for mitigating the event must be done manually. Actions outside of EOP-12 or procedures not referenced by EOP-12 may not work for all situations. Based on the situations which are beyond the scope of EOP-12 (i.e. LOCA, inadequate heat transfer, excessive heat transfer, SGTR, etc) the AAT needs to evaluate the specific guidance within other EOPs to develop mitigation strategy for the specific conditions. Based on the duration of the loss of offsite power or ES Diesels the following parameters need to be monitored and additional guidance given: • Monitor EFT depletion rate and heatup rate. Based on the depletion of EFT-2 give guidance to establish alternate EFW sources from the CST, FST or hotwell using cross-connect lines between the tanks if necessary (Ref EOP-14 Enclosures 22, Secondary Inventory Management). Refer to EM-225F. • Monitor BWST depletion rate. Based on BWST depletion rate establish guidance for transferring the ECCS or MUP suction to the RB sump if necessary (Ref EOP-14 Enclosure 19, ECCS Suction Transfer). Refer to EM-225E for guidance for filling BWST or allowable BWST level to support only a makeup pump. • Monitor Containment Temperatures. Ensure adequate RB cooling is being maintained. Refer to EM-225C.
Inadequate HPI flow, maximum cooldown in progress.	EOP-3		If multiple component/equipment failures or plant configuration results in inadequate HPI flow, then an alternate evaluation of less than FULL HPI flow can be considered based on ITS 3.5.2 bases, which states that the following injection flow path options can provide adequate HPI flow without meeting the configuration requirements of FULL HPI, as described in the definition section of this procedure: • A minimum of three (3) intact injection legs, assuming one pump operation • A minimum of two (2) intact injection legs, assuming two HPI pump operation.

EDG SCENARIOS TO ESTABLISH ADEQUATE LOADING

The following pumps and fans combinations will place an electrical loading on the EDGs greater than 600 kW (Reference EC 84553).

Case 1	Case 2
EGDG-1A Operating Loads Potential EDG Loading total – 686.0 KW • 272.9 KW - SWP-1A running • 383.3 KW - RWP-2A running (Note 1) • 29.8 KW - SFP-1A running	EGDG-1A Operating Loads Potential EDG Loading total – 689.4 KW • 63.4 KW - DCP-1A running • 160.1 KW - RWP-3A running • 215.4 KW - DHP-1A in recirculation mode to BWST at 3000 gpm (Note 2) • 155.4 KW - BSP-1A in recirculation mode to BWST at 1500 gpm (Note 2) • 95.1 KW - AHF-1A or AHF-1C running in high speed
EGDG-1B Operating Loads Potential EDG Loading total – 689.4 KW • 63.4 KW - DCP-1B running • 160.1 KW - RWP-3B running • 215.4 KW - DHP-1B in recirculation mode to BWST at 3000 gpm (Note 2) • 155.4 KW - BSP-1B in recirculation mode to BWST at 1500 gpm (Note 2) • 95.1 KW - AHF-1B or AHF-1C running in high speed	EGDG-1B Operating Loads Potential EDG Loading total – 686.0 KW • 272.9 KW - SWP-1B running • 383.3 KW - RWP-2B running (Note 1) • 29.8 KW - SFP-1B running

Note: (1) Operating both redundant pumps together can affect EDG loading in some cases due to load sharing between pumps (example: RWP-2A and RWP-2B).
(2) Monitor BWST water temperature to maintain below 92.5 deg F. Use of DHHE to cool the BWST may be required.

Other identified loads that should be available for operation to further increase the load if required or replace loads (example: BSP-1A/1B secured due to BWST temperature). Reference EC 84553 evaluation.

Load	Nameplate KW/HP	85% Nameplate KW	Comments
CHHE-1A/1B*	194 kW	164.9	Electrical load is dependent on the CC heat load.
AHF-17A/17B AHF-18A/18B	60 HP	38.0	
AHF-19A/19B	20 HP	12.7	
AHF-24A/24B	15 HP	9.5	
AHF-29A/29B	10 HP	6.3	

AAT COORDINATOR CHECKLIST

NOTE:	Attachment steps can be completed in any order..................................... ☐

1. BADGE IN at TSC card reader **AND** PLACE name on TSC Staffing Board .... ☐
2. NOTIFY the EC that the Accident Assessment Team is operational when ALL of the following are accomplished:
   - INITIATE Critical Safety Functions evaluation (Attachment 2, TSC Briefing Guideline) .... ☐
   - ESTABLISHED Communication via phone link with the control room or ability to monitor plant via computer (e.g. SPDS.............................. ☐
3. EVALUATE plant conditions **AND** ASSIST the EC in making timely and proper Emergency Classifications and Protective Action Recommendations.. ☐
4. ENSURE Attachment 2, TSC Briefing Guideline is complete. (normally by AAT Operations support) .... ☐
5. ENSURE Critical Safety Functions Status Board is updated................................. ☐
6. ENSURE phone link between Control Room and TSC Ringdown Communicators .... ☐
7. ENSURE each AAT position is staffed. REQUEST Security to contact additional AAT members as needed. (Refer to "Emergency Response Personnel Roster".)
   - Operations Support: ______________________________
   - TSC Ringdown Communicator:______________________________
   - Control Room Ringdown Communicator:______________________
   - Two Engineers:______________________________
   - NRC Communicator:______________________________
8. ENSURE all AAT members have badged in at TSC Card Reader .... ☐
9. DETERMINE parameters or parameter groups (SPDS and RECALL) to monitor **AND** ENSURE the desired parameters are displayed (Reference Attachment 12) .... ☐
10. ENSURE times and results of significant actions are documented throughout the emergency.. ☐

AAT COORDINATOR CHECKLIST

11. ENSURE AAT performs applicable attachments in EM-225 .... ☐
12. ENSURE OSC repair priorities are appropriate for plant conditions .... ☐
13. ENSURE the EC is informed of significant AAT activities and changes in plant status .... ☐
14. **IF** the EOF is staffed, **THEN** ESTABLISH communication with the EOF Technical Support Team via Accident Assessment Ringdown line, or extensions 6720/6205 .... N/A ☐ ☐
15. NOTIFY Off-Duty Shift Manager (requal crew or OSS crew) for additional Operations support .... ☐
16. APPROVE Attachment 11, OSC Request Form: (This request should go through TSC Repairs Coordinator to the OSC)
    - Requests for operator actions outside CCHE .... ☐

      OR

    - Maintenance repair activities that have been initiated by the Control Room or AAT .... ☐
17. REVIEW Attachment 9, Dose Assessment Team Notification **AND** ENSURE updates are provided as plant conditions change .... ☐

TSC BRIEFING GUIDELINE

NOTE:	REFER TO Attachment 8, Critical Safety Function Checklist and Attachment 10, Core Damage Assessment to aid in this evaluation .... ☐

1.1 **REACTOR SHUTDOWN** .... **Yes** ☐ **No** ☐

1.2 **CORE ADEQUATELY COOLED (1)** .... **Yes** ☐ **No** ☐

1.3 **FISSION PRODUCT BARRIER ASSESSMENT** .... ☐ **N/A (Defueled)** ☐

(Use Attachment 8,Critical Safety Function Checklist, Table 3)

1.	Fuel Clad:	Intact ☐	Potential Loss ☐	Loss ☐
2.	RCS:	Intact ☐	Potential Loss ☐	Loss ☐
3.	Containment:	Intact ☐	Potential Loss ☐	Loss ☐

1.4 **SPENT FUEL POOL STATUS:**

1. SF Pool Cooling Available? .... Yes ☐ No ☐
2. SF Clad Intact? .... Yes ☐ No ☐
3. SF Pool Level Stable (Indication On scale)? .... Yes ☐ No ☐
4. SF Pool Temperature Stable? .... Yes ☐ No ☐

1.5 **EMERGENCY ELECTRICAL POWER STATUS**

1. Off-Site Power Available? .... Yes ☐ No ☐
2. ES Bus Energized? .... Yes ☐ No ☐
3. Emergency Diesel Generator Available? .... Yes ☐ No ☐
4. Alternate AC Diesel Generator Available? .... Yes ☐ No ☐
5. DC Power Available? .... Yes ☐ No ☐

1.6 **CONTROL COMPLEX STATUS**

1. Ventilation / Cooling Available? .... Yes ☐ No ☐
2. Necessary Instrumentation Available? (2) .... Yes ☐ No ☐

1.7 **OTHER CONDITIONS / CHALLENGES**

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Note (1) - Inadequate Core Cooling is accident conditions that result in a loss of core cooling that requires entering EOP-7, Inadequate Core Cooling (Ref step3.0.4)

(2) - Necessary refers to specific instruments and annunciators that are needed to identify, diagnose, and track the problems that are causing the emergency.

TSC RINGDOWN COMMUNICATOR CHECKLIST

NOTE:	Attachment steps can be completed in any order ☐

1. ESTABLISH contact with the Control Room Communicator via the Accident Assessment Ringdown phone (receiver needs to be off the hook to use the headset) ☐
2. ENSURE the Control Room is informed of:
   - changing radiological conditions ☐
   - ongoing TSC maintenance and repair activities ☐
   - accident mitigation priorities ☐
   - operator actions outside the CCHE ☐
3. **IF** the EOF is staffed, **THEN** ESTABLISH communication with the EOF Technical Support Team via Accident Assessment Ringdown line, or extensions 6720/6205 N/A ☐ ☐
4. MAINTAIN the Accident Assessment Team log book with all significant events, changes in plant status, and requests to and from the Control Room ☐
5. RELAY information and directions to the Control Room as appropriate ☐
6. MONITOR progression through EOPs and APs,
   - Anticipate problems created by unavailable equipment or other unusual plant conditions ☐
   - MARK place keeping aids as appropriate to allow other AAT members to determine status of procedure usage. ☐
   - PROVIDE periodic status to AAT Operations Support member ☐
7. INITIATE Attachment 11, OSC Request Form:
   - Requests for operator actions outside CCHE ☐

     OR
   - Maintenance repair activities for the OSC that is requested by the Control Room or AAT ☐

AAT OPERATIONS SUPPORT CHECKLIST
[NOCS 62764]

NOTE:	Attachment steps can be completed in any order ☐

1. BEGIN assessment of Critical Safety Functions to ensure adequate core cooling and fission product barrier preservation, USING Attachment 8, Critical Safety Function Checklist as applicable ☐
2. COMPLETE Attachment 2, TSC Briefing Guideline **AND** PROVIDE the results to the AAT Coordinator. Attachment 2, TSC Briefing Guideline should be completed periodically or as conditions change ☐
3. MAINTAIN the CSF Status Board at the TSC ☐
4. COMPLETE Attachment 9, Dose Assessment Team Notification **AND** PROVIDE the results to the TSC Radiation Controls Coordinator and the EOF Dose Assessment Team Leader. If conditions change, Attachment 9, Dose Assessment Team Notification should be reassessed and submitted to the Radiation Controls Coordinator ☐
5. Coordinates/processes requests for operator actions or maintenance support through the Repairs Coordinator using Attachment 11, OSC Request Form. REFER TO SP-306 for a list of EOB and EOL locations and contents ☐
6. **IF** RCS LOCA conditions exist, **THEN** COORDINATE performance of EM-225A, Post Accident RB Hydrogen Control [NOCS 62767] N/A ☐ ☐
7. **IF** RCS LOCA conditions exist, **THEN** COORDINATE performance of EM-225E, Guidelines For Long Term Cooling N/A ☐ ☐
8. **IF** SGTR exists, **THEN** MONITOR BWST depletion rate **AND** INITIATE BWST MU early in the event if necessary (see Enclosure 1, TSC Guidance for EOPs page 2 of this procedure) N/A ☐ ☐
9. **IF** EFW or AFW is operating, **THEN** COORDINATE performance of EM-225F, Long Term Emergency Feedwater Management N/A ☐ ☐
10. **IF** a Severe Accident is in progress, **THEN** ASSIST engineering in developing appropriate mitigation strategies using the Candidate High Level Actions in the CR-3 Severe Accident Guideline. [NOCS 100056] N/A ☐ ☐
11. PROVIDE appropriate input to the Communication/Report Coordinator to update Florida Nuclear Plant Emergency Notification Form Supplemental Data Sheet ☐

AAT OPERATIONS SUPPORT CHECKLIST
[NOCS 62764]

12. **IF** any diesel operated equipment is running, **THEN** EVALUATE the following parameters (OSC support and local observation might be required to obtain information on support systems and operating parameters) .... N/A ☐ ☐

- Diesel support systems (i.e., ventilation, fuel transfer, cooling, etc.) .... ☐
- Establish periodic monitoring of diesel operating parameters to ensure proper operation. Monitor every 4 hrs and adjust monitoring intervals as required based on trending results. For EDGs refer to OP-707, Operation Of The ES Emergency Diesel Generators and SP-354C, Functional Test Of The Alternate AC Diesel Generator EGDG-1C for operating parameters .... ☐
- Operating EDG load limitation (loaded and unloaded) .... ☐
- Fuel and lube oil supplies .... ☐

NOTE: Low load/no load EDG Operation will accumulate unburned oil in exhaust system due to low exhaust temperatures, and may result in a challenge to EDG availability due to excessive exhaust back-pressure.

A key symptom of excessive exhaust back-pressure is erratic RPM/Frequency control followed by EDG Stall. When load is raised to burn off oil, ignited oil will be present in exhaust system. Dark smoke or possibly flame may be visible from the exhaust stack until excess oil is consumed, or removed........ ☐

13. **IF** any ES diesel loading is < 600 KW, **THEN** perform the following.

1) Ensure EDG is being monitored and Increase monitoring frequency to hourly interval and continue trending EDG parameters to determine if more frequent monitoring will be required. .... ☐

NOTE: EDG Operational issues are expected after 2 hrs if no load or approx 6 hrs if loaded between 400 KW to 600 KW.

2) Determine equipment that can be added to maintain the EDG > 600 KW. Refer to Enclosure 2, EDG Scenarios To Establish Adequate Loading .... ☐
3) Develop strategy for adding EDG loads .... ☐
4) Obtain EC approval on developed strategies .... ☐
5) Coordinate with the MCR on implementing strategies........ ☐

AAT OPERATIONS SUPPORT CHECKLIST
[NOCS 62764]

14. **IF** DHV-3 is required to be manually opened due to a Control Room fire resulting in AP-990, Shutdown from Outside the Control Room, entry, **THEN** COORDINATE RB entry activities .... N/A ☐ ☐

15. DETERMINE emergency and non-emergency notifications to the NRC as defined in CP-151, External Reporting Requirements .... ☐

AAT ENGINEERS CHECKLIST
[NOCS 62764]

NOTE:	Attachment steps can be completed in any order..................... ☐

1. PERFORM Attachment 10, Core Damage Assessment. PERFORM an initial and periodic assessment of core damage and fission product barriers, **AND** PROVIDE the results to the AAT Operations Support Member and the Radiation Controls Coordinator .... ☐
2. **IF** RCS LOCA conditions exist, **THEN** COORDINATE performance of EM-225B, Post-Accident Boron Concentration Management.................. N/A ☐ ☐
3. **IF** RCS LOCA conditions exist, **THEN** OBTAIN RB atmosphere I^{131} concentration **AND** TRANSMIT value to control room (for BS pump shutdown decision making).................. N/A ☐ ☐
4. MAINTAIN the Plant Parameters Status Board (if required). Based on plant conditions, PLACE key parameters on status board for trending .... ☐
5. MONITOR for conditions listed in Enclosure 1, TSC Guidance for EOPs. PROVIDE the AAT Operations Support member with recommended actions.................. ☐
6. **IF** RB temperatures are elevated, **THEN** COORDINATE the performance of EM-225C, Post Accident Monitoring Of Reactor Building Temperature.................. N/A ☐ ☐
7. **IF** any OTSG level is ≤ 12.5 inches (indicating a dry OTSG), **THEN** COORDINATE the performance of EM-225D, Guidance For Dry OTSG Tube To Shell Delta T Monitoring And Control.................. N/A ☐ ☐
8. EVALUATE the effects of proposed maintenance repair activities and operational manipulations on plant equipment.................. ☐
9. DEVELOP contingency plans **AND** SUPPORT emergency repair efforts as applicable .... ☐
10. **IF** a Severe Accident is in progress, **THEN** DEVELOP mitigation strategies using the Candidate High Level Actions in the CR-3 Severe Accident Guideline.................. N/A ☐ ☐
11. Within 7 days, ENSURE SW minimum flow requirements are maintained. **IF** ES or RBIC has actuated and either SWV-353 or 354 has failed closed, **THEN** ESTABLISH flow to the RB coolers **OR** ENSURE only 1 SW pump is running. .... ☐
12. **IF** additional computers are required **THEN** obtain, as needed, from nuclear administrative building (i.e., engineering laptop computers), that can be used to access documentation on the network.................. N/A ☐ ☐
13. IDENTIFY AAT priorities using the AAT priority board in the AAT room .... ☐

CONTROL ROOM RINGDOWN COMMUNICATOR CHECKLIST

NOTE:	Attachment steps can be completed in any order ☐

1. ESTABLISH communication with the TSC Ringdown Communicator on the Accident Assessment Ringdown phone in the Control Room. BRIEF TSC Ringdown Communicator on operator actions that are in progress ☐
2. RELAY status of overall plant conditions, operator activities and questions to the TSC AAT ☐
3. RELAY instructions to Control Room Operators for mitigating actions as directed by the EC ☐
4. INFORM Control Room Operators of the following:
   - Changes in Emergency Classifications ☐
   - TSC repair efforts ☐
   - Operators activities dispatched from the TSC/OSC ☐
   - Changing radiological conditions ☐
   - Mitigation priorities ☐
5. MONITOR EOPs or APs in use by Control Room ☐
6. **IF** a Severe Accident is in progress, **THEN** DIRECT Control Room personnel regarding mitigation strategies, based on actions approved by the TSC Emergency Coordinator N/A ☐ ☐
7. RELAY requests for support from the Control Room to OSC teams, via the TSC Ringdown Communicator ☐
8. Once TSC is operational, REQUEST extra plant operators (if available) be sent to OSC for in plant support. (Ref. EM-103, Enclosure 1, Dispatching of Resources During Emergency Plan Entry) ☐
9. INFORM TSC of operator actions being performed ☐

NRC COMMUNICATOR CHECKLIST

NOTE:	Attachment steps can be completed in any order ☐

1. CONTACT the Communication/Report Coordinator to determine if continuous communication with the NRC is required ☐
2. OBTAIN copies of any previously submitted NRC reports ☐
3. **IF** the NRC has requested continuous communication, **THEN** ESTABLISH communication with the NRC on the Emergency Notification System (ENS) N/A ☐ ☐
4. MAINTAIN a log book of significant communications between the NRC and CR-3, including a summary of responses to NRC questions and transmittal of information ☐
5. MAINTAIN an open line on the ENS until the NRC agrees to terminate communications ☐
6. LOG time(s) when TSC notifies NRC of Emergency Classification changes ☐
7. LOG time(s) when TSC notifies NRC of Protective Action Recommendations ☐
8. **WHEN** communication with the NRC is not required, **THEN** PROVIDE support to other AAT members as needed ☐
9. MAKE emergency and non-emergency notifications to the NRC as defined in CP-151, External Reporting Requirements. Examples include, but are not limited to:
   - Suspension of Safeguards (invoked under 10 CFR50.54(x), or Section 24 of the Physical Security Plan) ☐
   - The declaration of any of the Emergency Classes specified in EM-202, Duties of the Emergency Coordinator {10CFR50.72(a)(1)(i)} ☐
   - The initiation of any nuclear plant shutdown required by CR-3 Technical Specifications {10CFR50.72(b)(2)(i)} ☐
   - The condition of CR-3, including its principal safety barriers, being seriously degraded ☐
   - CR-3 being in an unanalyzed condition that significantly degrades plant safety ☐

CRITICAL SAFETY FUNCTION CHECKLIST

NOTE: 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.

1. MONITOR the parameters associated with the Critical Safety Functions. Qualified OSI PI EP folder has established Accident groups.("Start-Program-Business Apps-PI System-CR3 QPIM") ☐
2. NOTIFY the AAT Coordinator immediately if any of the CSFs cannot be verified ☐

TABLE 1: 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			
Adequate Shutdown Margin	OP-103C Curve 18&19				

CRITICAL SAFETY FUNCTION CHECKLIST

TABLE 2: Core Cooling Status

ECCS/Support Status

PARAMETER	COMPUTER POINT	RECALL POINT			
Subcooling Margin					
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 X064	RECL-207 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			
RWP-1 operating	X060				
RWP-2A operating	X061	RECL-222			
RWP-2B operating	X062	RECL-223			
RWP-3A operating		RECL-217			
RWP-3B operating		RECL-218			
DCP-1A operating (yes/no)		RECL-220			
DCP-1B operating (yes/no)		RECL-221			
SWP-1A operating		RECL-219			
SWP-1B operating					
SWP-1C operating					
RB Sump Strainer ΔP		RECL-79			

CRITICAL SAFETY FUNCTION CHECKLIST

TABLE 2: Core Cooling Status (Cont'd)

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					
EFP-1/3 Flow to A OTSG		RECL-246			
EFP-1/3 Flow to B OTSG		RECL-245			
EFP-2 Flow to A OTSG		RECL-248			
EFP-2 Flow to B OTSG		RECL-247			
Total EFW Flow to A OTSG	S300	RECL-408			
Total EFW Flow to B OTSG	S312	RECL-409			
EFW Tank Level		RECL-236			

CRITICAL SAFETY FUNCTION CHECKLIST

TABLE 3: Fission Product Barrier Assessment

FUEL CLAD [NOCS 100441]		
☐ **INTACT**	☐ **POTENTIAL LOSS**	☐ **LOSS**
• Does **NOT** meet the criteria for "Potential Loss" or "Loss"	• RCS condition warrant entry into EOP-07, Inadequate Core Cooling • Core Exit Thermocouples > 700 degrees F	• RCS conditions in (or previously in) Region 3 or Severe Accident Region • RCS activity >300μCi/gr I^{131} dose equivalent. Additional indication is 100 mR/hr measured on RM-G3 or at one foot from sample lines in Nuclear Sample Room • RM-G29/30 > 100 R/hr for ≥ 15 minutes • Attachment 10, Core Damage Assessment indicates failed fuel
REACTOR COOLANT SYSTEM		
☐ **INTACT**	☐ **POTENTIAL LOSS**	☐ **LOSS**
• Does **NOT** meet the criteria for "Potential Loss" or "Loss"	• RCS leak or OTSG tube leak requiring one or more injection valves to maintain adequate subcooling margin • RCS pressure /Tincore relationship violates NDT limits • RCS leak or OTSG tube leak results in ES actuation on low RCS pressure. • HPI/PORV or HPI/Code Safety valve cooling is in progress	• RCS leak resulting in loss of adequate subcooling margin • OTSG Tube Rupture resulting in loss of adequate subcooling margin • RM-G29/30 >10R/hr for ≥ 15 minutes
CONTAINMENT		
☐ **INTACT**	☐ **POTENTIAL LOSS**	☐ **LOSS**
• Does **NOT** meet the criteria for "Potential Loss" or "Loss"	• RB pressure > 54 psig • RB hydrogen concentration > 4% • RB pressure > 30 psig with **NO** building spray available • RMG-29 or 30 reading > 25,000 R/hr • Core conditions in severe accident region of ICC curves for >15 min	• Containment isolation is incomplete and release path to environment exists. Confirmation may be from elevated radiation readings in areas adjacent to the RB. • OTSG Tube Rupture > 10 gpm exists and prolonged steaming to atmosphere or an unisolable steam leak outside RB from affected OTSG. • Containment pressure or sump level response **NOT** consistent with LOCA conditions • Rapid unexplained RB pressure decrease following an initial increase

CRITICAL SAFETY FUNCTION CHECKLIST

TABLE 4: Spent Fuel Pool Status

PARAMETER		YES	NO
SF Pool Cooling Available	• SF Pump, SWP, RW running OR • DHP aligned for SF Pool Cooling, DCP, RWP Running		
SF Clad Intact	• RM-G14/15 rising dose rate indication OR • Visual Report of Fuel Damage		
		STABLE	UNSTABLE
SF Pool Level	Indication On scale (Available MCB only) (Refer to EOP-14 Enclosure 24)		
SF Pool-Temperature	Ref to EOP-14 Enclosure 24		

TABLE 5: 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 (1)		
B-ES 480V BUS (1)		

Emergency Diesel Generator

PARAMETER	RECALL PT.	LOADED	AVAILABLE	UNAVAILABLE
A-EDG	RECL-133,171			
B-EDG	RECL-134,172			
Alternate AC Diesel	N/A			

DC Electrical

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

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

CRITICAL SAFETY FUNCTION CHECKLIST

TABLE 6: 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:____

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Performed By: ______________________ Date: _____/_____/_____ Time: __________

DOSE ASSESSMENT TEAM NOTIFICATION

NOTES: 1. The TSC/AAT supplies the TSC Radiation Control Coordinator and EOF DAT Leader with Attachment 9 to assist with the projection of off-site doses. .... ☐

2 MARK items N/A or unknown based on information available at the time. .... ☐

3 PROVIDE readily available information promptly **AND** FOLLOW UP with additional forms as more information becomes available. .... ☐

4 Attachment 9 can be completed in any order. .... ☐

- LOSS-OF-COOLANT ACCIDENT: Rx Trip Date/Time: _____/_____......N/A ☐ ☐

a. Rx Fuel Cladding status: (from Attachment 10, Core Damage Assessment)
☐ Normal Activity (Spike Factor _______) (See General Information - Pg 3)
☐ Clad Damage
☐ Fuel melt

b. Rx Core Uncovered? (Rx Core covered is based on RCS NOT superheated)
☐ NO (Not in EOP-07, Inadequate Core Cooling)
☐ YES Uncovered - Date/Time __________/___________
Recovered - Date/Time __________/___________

c. Start of release to containment (start of the LOCA)
☐ Unknown
☐ Date/Time ___________________;

d. Release to atmosphere?
☐ NO
☐ YES
Date/Time _____/_____ Estimated duration _______
Release path (from where to where) ________________________
Release path flow rate (unmonitored releases):
☐ Estimated hole size ______ ☐ Diameter (in) or ☐ Area (in^2)
Containment pressure (RECL-82/83) _________ PSIG **OR**
☐ % RB Volume /day__________ (RB Volume is 2 $X10^6$ cu ft) **OR**
☐ gpm _________ **OR**
☐ Design Basis Leakage

e. Reactor Building Spray Actuated? (RECL-212/213)
☐ NO
☐ YES Dates/Times _____________

f. Rx Bldg Vent flow rate (AH-1003-TIR Channel 4) __________
Charcoal banks in service ☐ YES / ☐ NO

g. Auxiliary Building ventilation (W351): flow rate ____________
Charcoal banks in service ☐ YES / ☐ NO

h. Loose Parts Monitor indications?
☐ Unavailable / ☐ NO
☐ YES Location _________________

DOSE ASSESSMENT TEAM NOTIFICATION

- WASTE GAS DECAY TANK RUPTURE: (See General Information-Pg 3).........N/A ☐ ☐

a. Release pathway:
☐ Tank rupture **OR**
☐ Valve leakage **OR**
☐ Other____________

b. Tank volume _________ (Each WGDT volume = 1753 ft^3)
Tank pressure ________ (RW203/204/205)

c. Release rate ☐ Unknown **OR** ☐ _________ CFM

d. Start of release
☐ Unknown **OR**
☐ Date/Time ________/________ Estimated duration _________

e. Auxiliary Building Ventilation (W351) Flow rate ____________
Charcoal banks in service ☐ YES / ☐ NO

- STEAM GENERATOR TUBE RUPTURE: Rx Trip Date/Time: ______/_____ N/A ☐ ☐

a. Primary-to-secondary leak rate:
☐ Unknown **OR**
☐ _______________ gpm **OR**
☐ Number of tubes _________

b. Rx Fuel Cladding status: (From Attachment 10, Core Damage Assessment)
☐ Normal Activity (Spike Factor _______) (See General Information – Pg 3)
☐ Clad Damage
☐ Fuel melt

c. Rx Core Uncovered? (Rx Core cover is based on RCS <u>NOT</u> superheated)
☐ NO (Not in EOP-07, Inadequate Core Cooling)
☐ YES Uncovered - Date/Time __________/___________
Recovered - Date/Time __________/___________

d. Leaking OTSG isolated? ☐ NO / ☐ YES Date/Time __________/___________

e. Release Point:
☐ MSSV/ADV (intermittent/continuous) **OR**
☐ Condenser

f. Start of leak
☐ Unknown **OR** ☐ Date/Time _______________;

g. OTSG Water Mass (See General Information – Pg 3)
☐ OTSG Water Mass _____________ **OR**
☐ RASCAL default value of 93000 lbm

h. OTSG Steaming Rate (See General Information – Pg 3)
☐ OTSG Steaming rate _______________ **OR**
☐ RASCAL default 75000 lbm/hr

i. Auxiliary Building Ventilation (W351): Flow rate ____________
Charcoal banks in service ☐ YES / ☐ NO

DOSE ASSESSMENT TEAM NOTIFICATION

- SPENT FUEL ACCIDENT: .... N/A ☐ ☐

a. Spent Fuel uncovered?
☐ NO
☐ YES Date/Time _____/_____ Recovered Date/Time _____/_____

b. Spent Fuel Pool Empty?
☐ NO
☐ YES Date/Time _____/_____ Recovered Date/Time _____/_____

c. Fuel assembly damaged by dropped component/handling?
(See General Information below)
☐ NO
☐ YES Number of assemblies damaged ______ Damage Date/Time ____/____
Last irradiation Date __________ (use last refueling outage)
Damaged Fuel assembly underwater ☐ YES / ☐ NO / ☐ UNKNOWN

d. Auxiliary Building Ventilation (W351): Flow rate ____________
Charcoal banks in service ☐ YES / ☐ NO

e. Dry Cask lost cooling?
☒ N/A ☐ NO
☐ YES <24 hrs_____ >24 hrs_____ Cask on fire _____;
Number assemblies in cask _____ Type of dry cask _________________

GENERAL INFORMATION:

1. Initially use a spiking factor of 100. Adjusted spiking factor based on RCS sampling and analyses when accident conditions allow sampling to be performed. The Spiking Factor is used in RASCAL to distinguish the change in concentration of the fission products in the RCS due to a rapid drop in RCS pressure. The sudden RCS pressure drop increases the rate at which the radioactive fission products in the fuel rod cladding gap escape to the RCS.
2. Waste Gas Decay Tank Rupture information does not need to be completed if RM-A2 is in service monitoring release.
3. Use RASCAL default value if data is not known at the time of completing Attachment 9.
4. Fuel assembly damage is associated with damage from a dropped component and not clad failures due to overheating or flaws.

COMMENTS:____

____

____

Performed By: _________________________ Date: _____/_____/_____ Time: __________

Reviewed By Accident Assessment Team Coordinator: ______________________________

CORE DAMAGE ASSESSMENT

1. This attachment does not apply if the core is defueled .... N/A ☐ ☐
2. DETERMINE if core damage has occurred using one or more of the following methods **AND** ESTIMATE the extent of the damage and status of the fission product barriers .... ☐
   - DETERMINE/ESTIMATE core damage based on RM-G29/30 radiation levels .... ☐

NOTES:
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 below table 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

CORE DAMAGE ASSESSMENT

- DETERMINE/ESTIMATE core damage based on iodine ratios..................... ☐

NOTE:	Core damage assessment based on Iodine Ratios will be evaluated by the Radiation Controls Coordinator using EMG-NGGC-0002, Off-Site Dose Assessment. Contact Dose Assessment Team to coordinate the activity of estimating core damage using this method. This method can take several hours based on the requirements to perform a gamma isotopic of a grab sample.

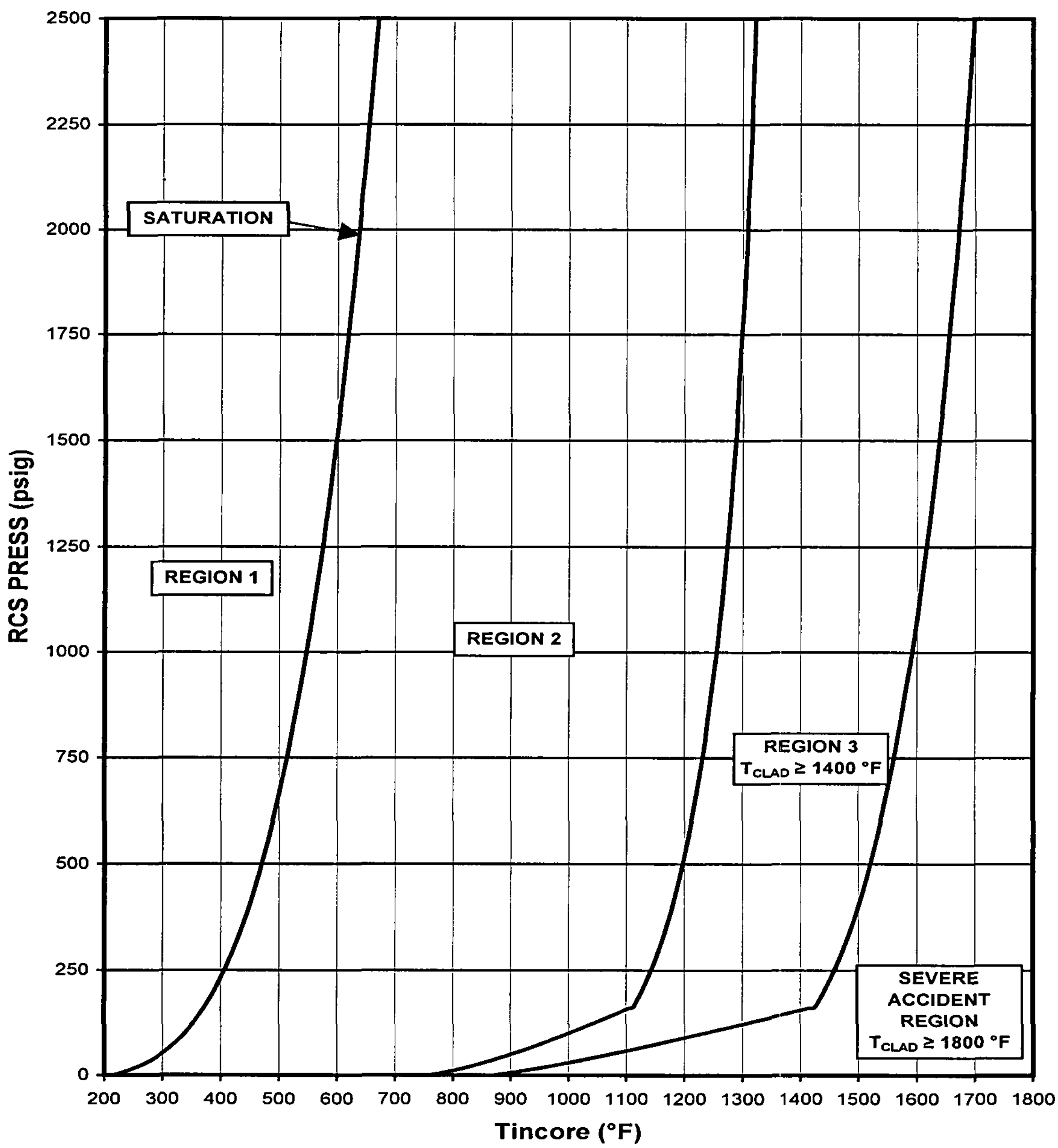
- No core damage.. ☐
  - I-131/Total Iodine < 0.05
- Possible clad failure and gas gap release / possible core melting(There is no way to distinguish between a gap release and a core melt release using iodine ratios) .... ☐
  - I-131/Total Iodine ≥ 0.05

CORE DAMAGE ASSESSMENT

CORE DAMAGE ASSESSMENT BASED ON ICC CURVE

- DETERMINE/ESTIMATE core damage by plotting RCS pressure/Tincore temperature on the ICC curve below .... ☐

NOTES:	1. Regions 1 and 2 indicate no fuel damage (normal RCS activity).
	2. Severe Accident Region indicates possible core melt.
	3. Region 3 indicates possible gas gap failure.



CORE DAMAGE ASSESSMENT

CORE DAMAGE PROGRESSION ONCE UNCOVERED

3. **IF** inadequate subcooling margin exists, **THEN** DETERMINE if the core is uncovered .... N/A ☐ ☐

NOTES:
1. 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.
2. 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.
3. The Reactor Void Trend Subsystem (RC-169-XR) monitors void trends in the RCS when RCPs are running. RCP motor power and T_{cold} are used to infer average density of fluid passing through the pump (liquid or two-phase). A 0% reading infers no voiding, while 100% reading infers complete voiding.
4. 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

CORE DAMAGE ASSESSMENT

CORE DAMAGE PROGRESSION ONCE UNCOVERED (Cont'd)

- Uncovered for 30 to 90 minutes .... ☐
  - Core temperature 2400-4200°F
  - Possible un-coolable 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

4. Report the results of the evaluation to the AAT operations support member and the Radiation Controls Coordinator .... ☐
5. Continue to re-assess core and fission product barrier status as conditions change .... ☐

(SAMPLE)

OSC REQUEST FORM

INSTRUCTIONS:

1. Use this form for each requested action from the Control Room, or Accident Assessment Team (multiple steps of EOPs/APs may be covered by one request)
2. Obtain approval from the AAT Coordinator
3. Obtain acknowledgement from TSC Repairs Coordinator
4. Make copy and give original to TSC Repairs Coordinator
5. Give copy to TSC Ringdown Communicator and OSC Manager
6. Feedback to the Control Room on status of request.

REQUEST NUMBER: (UNIQUE NUMBER)	INITIATED BY: (AAT MEMBER)	TIME	DATE
REQUESTED ACTION(S):			
CONSEQUENCES IF NOT PERFORMED:			
TIME FRAME REQ'D	TAG NO:	TRAIN:	LOCATION:
APPROVAL (AAT COORDINATOR)		TIME:	
RECEIVED BY: (TSC REPAIR COORDINATOR)		TIME:	
FEEDBACK PROVIDED TO MAIN CONTROL ROOM (TSC RINGDOWN COMMUNICATOR)		TIME	

SPDS OR RECALL DISPLAY SETUP FOR TSC PROJECTION SCREENS

NOTE:	TSC plant computers are labeled EMCO-81 and EMCO-50. Rebooting any of the TSC plant computers will result in an alarm in the main control room. Call the control room before rebooting any TSC plant computer. Rebooting an NGG standard desktop computer will not result in an alarm.

1. In the projector room, CONNECT computer monitor to the desired computer interface box along the side of computer rack. **IF** the desired computer is not connected to an interface box, **THEN** CONNECT the video display cable from an interface box to the desired computer .... ☐
2. **IF** the computer is an NGGC standard desktop computer, **THEN** USE the desired computer mouse to select the specific display desired: .... N/A ☐ ☐
   a. LOG into the computer with a corporate ID .... ☐
   b. SELECT start - programs - engineering - CR3 - CR3 PICS .... ☐
   c. LOG into the pics access control client using system "CR3 PPCS" and "TSC" as the username and password .... ☐
   d. For a specific recall display, SELECT "recall display program", SELECT one of the pre-established displays from the "workspaces" drop down menu, **AND** CLICK "open" .... ☐
   e. For a specific SPDS display, SELECT "SPDS display" **AND** CLICK on the desired SPDS display buttons **OR** USE the keyboard (refer to the laminated card for commands). CNTRL H displays or hides the button bar. The button bar mimics the function panel on the MCB (Ref OP-509) .... ☐
   f. For the subcooling margin monitor display, SELECT "t sat" .... ☐
3. **IF** the computer is a TSC plant computer, **THEN** USE the desired computer mouse to select the specific display desired: .... N/A ☐ ☐
   - For a specific recall display, SELECT one of the pre-established displays from the "*workspaces*" drop down menu, **AND** CLICK "open" .... ☐
   - For a specific SPDS display, just CLICK on the desired SPDS display buttons **OR** USE the keyboard (refer to the laminated card for commands) .... ☐
4. GO TO the touch screen (located in the main TSC room) which controls the projection screens. SELECT the desired projector room computer from the associated screen location (left, center, or right screen) .... ☐

SUMMARY OF CHANGES

PRR 527011

NOTES:	1. Writers and Reviewers: Ensure that any changes to this procedure that affect information contained in Emergency Response Facility posters, enclosures, briefing cards, guidelines etc. are made to those items as well. 2. Writers and Reviewers: Changes to certain parts of this procedure may impact other Emergency Plan Implementing Procedures. Ensure appropriate PRRs are initiated as needed.

SECTION	CHANGE
2.28, 2.29	Added Calculation M89-0063 and EC 76363 to the reference section. This captured the information for the waste gas tank volume and the EC for RM-A1/A2 replacement. (Editorial Change)
Attachment 9	Removed reference to RM-A1/A2. Also the attachment was reformatted to improve the human factoring of the attachment. The required information was not changed only the appearance of where the data is entered. The definition for spiking factor and volume of the waste gas decay tank was added. The waste gas decay tank volume was taken from calculation M89-0063. The Radiation Monitor upgrade project will be done in phases. EC 76363 is replacing RM-A1 and RM-A2. The new RM-A1 and RM-A2 radiation monitors do not have a low/mid/hi range. Instead, they have a normal range and an accident range. When the radiation monitors are operable, no manual operator actions are required to shift from normal range to accident range when the hi-hi setpoint is reached on the normal range. The status of RM-A1/A2 was added under EM-225 revision 24 based on drill input. The existing design requires knowledge of the RM-A2 gas low-range high trip setpoint and LMHVC status. The new design has no manual operator actions to shift range. The reformatting improves human performance implementation of the attachment (Ref PRR 527011 and 490832).

Progress Energy

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Information
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CRYSTAL RIVER UNIT 3

PLANT OPERATING MANUAL

EMERGENCY PLAN IMPLEMENTING PROCEDURE

EM-225A

POST ACCIDENT RB HYDROGEN CONTROL

REVISION 11

TABLE OF CONTENTS

1.0 PURPOSE

1. This procedure provides guidance for the Accident Assessment Team (AAT) and other emergency response personnel in developing appropriate actions to monitor and control post-accident hydrogen concentration in the Reactor Building (RB) to protect the health and safety of the general public and Crystal River Energy Complex personnel during an emergency at CR-3.
2. This procedure is an emergency plan implementing procedure. Any revisions to this procedure must be carefully considered for emergency plan impact.

2.0 REFERENCES

2.1 Developmental References

1. FSAR Chapter 14 Appendix B
2. MAR 91-05-03-01, "Hydrogen Purge Redundancy Restoration"
3. MAR 93-05-03-02, "Hydrogen Purge Redundancy Restoration, Elect. & I&C"
4. CALC M-99-0051, "Mission Dose Assessment"
5. CALC I-90-0013, "Post Accident Reactor Building Hydrogen Purge Flow Accuracy"
6. CALC M-90-0056, "Hydrogen Mini Purge Pressure Loss"
7. CALC M-99-0052, "Zone Environmental Radiation Dose for LOCA"
8. CALC N-00-0002, "Public And Control Room Dose From A LOCA Using The Alternative Source Term"
9. CALC M-85-1004, "H2 Generation Rate"
10. CALC I-90-0023, "RB Hydrogen Concentration Loop Accuracy"
11. EC 76363, Radiation Monitors RM-A1/RM-A2 Replacement

2.2 Implementing References

1. EOP-14, Emergency Operating Procedure Enclosures
2. MP-815, Installation of Post Accident Hydrogen Purge Monitors
3. EM-104, Operation of the Operational Support Center

3.0 DEFINITIONS

1. **Off-shore winds:** Winds originating from NNE to ESE sectors (011° to 124°). The most common time for this to occur is midnight.

4.0 RESPONSIBILITIES

1. Emergency Coordinator (EC) or designee:
   - Approves RB purge before initiation (Enclosure 6, Purge Release Authorization Form).
   - Ensures coordination with off-site agencies before initiation of RB purges.
2. Accident Assessment Team:
   - Tracks RB conditions and predicts time for RB purge initiation.
   - Monitors the effectiveness of purge methods in hydrogen removal.
   - Informs the EC of RB conditions and the status of pre-planned releases
   - Assign a Purge Release Authorization Form number (Enclosure 6, Purge Release Authorization Form).
3. Dose Assessment Team:
   - Monitors meteorological conditions and predicts when off-shore winds should exist.
   - Projects off-site doses for proposed RB purges.
4. Procurement Representative:
   - Ensures required air compressors are delivered on-site within the required time.
   - Ensures support materials (fuel, oil, etc,) are available to support portable compressor operations.
5. Emergency Repair Team:
   - Connects temporary air compressors when delivered.
   - Installs LR-82-FE, LR-83-FE, LR-82-FI, and LR-83-FI in accordance with MP-815, Installation of Post Accident Hydrogen Purge Monitors.
6. Radiation Monitoring Team:
   - Evaluates actual plant radiological conditions and determine routes to be used (see Enclosure 9, Access Routes).
7. Operations:
   - Performs RB purge per Enclosure 7, Purging RB.

5.0 PREREQUISITES

None

6.0 PRECAUTIONS, LIMITATIONS, AND NOTES

1. All hydrogen concentration values referenced in this procedure are presented in % by volume as indicated on the hydrogen analyzers.
2. Maintain RB hydrogen concentration < 3.6% to provide adequate margin below the lower flammability limit of 4.1% for hydrogen in air.
3. Travel through radiation areas should be as shown in Enclosure 9, Access Routes, unless otherwise directed by the emergency RWP.
4. Purging should be performed under favorable meteorological conditions (off-shore winds) whenever possible.
5. RB pressure must be carefully controlled during purge evolutions to prevent ES actuations from high RB pressure.
6. The purging criteria established by this procedure is not valid during Severe Accidents.
7. Mission dose calculations credit 10 days of radioactive decay when determining the dose received for performance of local actions. Taking local actions before this time may result in excessive radiation exposure.
8. If a predictable pattern of off-shore winds is identified, consideration should be given to performing a series of intermittent releases during periods when off-shore winds are present.
9. The AAT is responsible for overall implementation of this procedure. TSC teams responsible for performing the specific actions listed in the enclosures of this procedure are denoted at the end of each step as applicable.
10. Hydrogen is a flammable and explosive gas. Care must be taken to ensure ignition source are not in the immediate area where potential for explosive hydrogen concentrations exits. This will minimize the potential for personnel injury and equipment damage.

7.0 SPECIAL TOOLS AND EQUIPMENT

1. Air Compressors (as needed)

8.0 ACCEPTANCE CRITERIA

None

9.0 INSTRUCTIONS

NOTE:	Enclosure 11, Hydrogen Purge System Flow Diagram, depicts the hydrogen Purge flow paths established by this procedure. Enclosure 11, Hydrogen Purge System Flow Diagram, is provided for information only.

1. If RCS LOCA conditions exist, then monitor RB hydrogen concentration in accordance with Enclosure 1, Hydrogen Monitoring, of this procedure.
2. If at any time RB hydrogen concentration ≥ 1%, then perform the following:
   - Perform Enclosure 2, Preparations for RB Hydrogen Purge, in this procedure.
   - Notify Procurement Representative to contact Hydrogen Recombiner vendor to coordinate preliminary transportation plan and schedule for delivery of recombiner. Refer to EM-104, Operation of the Operational Support Center.
   - RMT/AAT evaluate plant conditions and equipment availability to determine if a Hydrogen Recombiner will be required. Notify Procurement Representative if recombiner is required.
3. When at any time RB purge compressors arrive on site, and radiological conditions permit, then perform Enclosure 3, Portable Compressor Installation, of this procedure.
4. When RB hydrogen concentration ≥ 3.3%, and radiological conditions permit, then perform Enclosure 4, Prerequisite Field Actions, of this procedure.
5. When RB hydrogen concentration ≥ 3.4%, then perform Enclosure 5, RB Pressurization for Hydrogen Purge, of this procedure.
6. When RB hydrogen concentration ≥ 3.5%, then begin Enclosure 6, Purge Release Authorization Form, of this procedure.
7. When any of the following conditions exist, then perform Enclosure 7, Purging RB, of this procedure:
   - RB H_2 concentration ≥ 3.5% for ≥ 24 hours
   - RB H_2 concentration ≥ 3.5% and off shore winds exist
   - RB H_2 concentration ≥ 3.6%,
8. When RB purge is stopped, then go to Step 9.0.6 or this procedure.

10.0 RECORDS

All enclosures are quality records.

HYDROGEN MONITORING

STATUS
• LOCA Conditions Exist

	ACTIONS	DETAILS
1.1	___ Ensure one H_2 analyzer is aligned and placed in service (Ops).	• ___ Ensure applicable steps of EOP-14, Enclosure 2, PPO Post Event Actions, have been completed for H_2 analyzers.
1.2	___ Plot RB H_2 concentration on Enclosure 8, RB Hydrogen Concentration Trend of this procedure (AAT).	• Obtain H_2 concentrations from either of the following: ___ EOP-14, Enclosure 21, RB Hydrogen Monitor Log. ___ RECALL
1.3	___ Project when RB H_2 concentration will exceed action levels of this procedure (AAT).	• ___ Use H_2 concentration plotted on Enclosure 8, RB Hydrogen Concentration Trend of this procedure. • ___ Extrapolate to estimate time when H_2 concentration will reach procedure action levels.

Action Level	Date	Time
$H_2 \geq 1\%$	____	_____
$H_2 \geq 3.3\%$	____	_____
$H_2 \geq 3.4\%$	____	_____
$H_2 \geq 3.5\%$	____	_____
$H_2 \geq 3.6\%$	____	_____

HYDROGEN MONITORING

		ACTIONS	DETAILS
1.4	___	IF at anytime H_2 concentration is ≥ an action level of this procedure, THEN immediately notify the Accident Assessment Team Coordinator (AAT).	• Action levels based on RB H_2 concentrations. Action Level — Required Action $H_2 \geq 1\%$ — See step 9.0.2 $H_2 \geq 3.3\%$ — See step 9.0.4 $H_2 \geq 3.4\%$ — See step 9.0.5 $H_2 \geq 3.5\%$ — See step 9.0.6
1.5	___	Continue monitoring RB H_2 concentration (AAT).	___ Plot RB H_2 concentration on Enclosure 8, RB Hydrogen Concentration Trend of this procedure every 8 hours. ___ Perform Step 1.3 of this Enclosure every 8 hours.

PREPARATIONS FOR RB HYDROGEN PURGE

STATUS
• RB H_2 Concentration ≥ 1%

	ACTIONS	DETAILS
1.1	___ Notify the Procurement Representative, Radiation Controls Coordinator, Repairs Coordinator and Control Room to begin preparations for RB purge.	• Review this procedure for: ___ Procurement of tools and equipment. ___ Selection of emergency team personnel. ___ Assigning Operations support to the OSC. ___ Initiation of reentry process per EM-104. ___ Collection of radiological and meteorological data. ___ Review of dose projection process.
1.2	___ Evaluate plant radiological conditions and determine routes to be used to perform Enclosures 2, 3, 4, 5, and 7 (RMT).	• Refer to Enclosure 9, Access Routes for locations of required actions/components and suggested routes.
1.3	___ Notify off-site sources to obtain portable air compressors (Procurement Representative).	• Obtain 3 or more air compressors from one of the following off-site sources: ___ Compressed Air Systems, Telephone (800) 626-8177 OR (813) 626-8177 (Tampa) ___ Air Components & Equipment, Inc., Telephone (813) 621-3087 (Tampa) • ___ Obtain air compressors capable of 225 scfm minimum each for continuous purge (rated exhaust flow) and rated discharge TEMP < 150°F.
1.4	___ Ensure all CCHE habitability breaches are sealed (ERT).	

PREPARATIONS FOR RB HYDROGEN PURGE

ACTIONS	DETAILS
1.5 ___ Monitor meteorological conditions to predict off-shore wind cycle (DAT).	• ___ Off-shore winds originate from NNE to SE sectors (011.2° to 146.3°). • ___ Most common time for off-shore winds is midnight.
1.6 ___ Ensure the purge flow instrumentation cart is properly staged and equipped (ERT).	• ___ Refer to MP-815 for location of equipment. • ___ DO NOT install purge instruments until Enclosure 4, Prerequisite Field Actions is performed.
1.7 ___ Ensure power is available to LR-82-FI and LR-83-FI receptacle (OPS).	• ___ RX MCC 3B2 is energized. • ___ RX MCC 3B2, BKR 8AR closed. • ___ ACDP-20, BKR 12 closed. (143 ft AB near elevator)
1.8 ___ Notify the Accident Assessment Team Coordinator that Enclosure 2, Preparations for RB Hydrogen Purge is complete (AAT).	

PORTABLE COMPRESSOR INSTALLATION

STATUS
• Purge Compressors Are On Site • Hydrogen Concentration ≥ 1%

ACTIONS	DETAILS
1.1 ___ Consult Radiation Monitoring Team to determine routes and precautions to be used during compressor installation (ERT).	• Refer to Enclosure 9, Access Routes for locations of required actions/components and suggested routes.
1.2 ___ Connect portable air compressors (ERT).	• ___ DO NOT open LRVs at this time. • ___ Indicate LRVs to which portable air compressors are connected. • ___ Preferred - RB portable compressor connections (119 ft IB outside west wall):

___ LRV-11	___ LRV-16
___ LRV-12	___ LRV-17
___ LRV-13	___ LRV-18
___ LRV-14	___ LRV-19
___ LRV-15	___ LRV-20

• ___ Alternate - H_2 recombiner connections
(119 ft IB outside west wall):
(adapters in stores – CAT ID # 0001260356)

___ LRV-92 (Pen 125)
___ LRV-90 (Pen 121)
___ LRV-94 (Pen 125)
___ LRV-88 (Pen 122)

PORTABLE COMPRESSOR INSTALLATION

ACTIONS	DETAILS
1.3 ___ Ensure plant personnel are familiar with the operation of the portable compressors (OPS/ERT).	
1.4 ___ Obtain support materials for portable compressors (Procurement Representative).	• ___Determine portable compressor fuel and oil consumption rate from compressor vendor. • ___Ensure sufficient fuel and oil supplies are available to support compressor operation.
1.5 ___ Notify the Accident Assessment Team Coordinator that Enclosure 3, Portable Compressor Installation is complete (OPS/ERT).	

PREREQUISITE FIELD ACTIONS

STATUS
• RB H_2 Concentration ≥ 3.3%

	ACTIONS	DETAILS
1.1	___ Consult Radiation Monitoring Team to determine routes and precautions to be used while performing RB Purge Field Actions (ERT).	• Refer to Enclosure 9, Access Routes for locations of required actions/components and suggested routes.
1.2	___ Defeat all starting interlocks on AHF-7A and 7B (OPS).	1. ___ Obtain key 92 from the Control Room. 2. Select RB exhaust fan permissive bypass switches to the "Emergency" position .(119 ft IB East Door) • ___ AHF-7A, Ventilation MCC 3A-10C • ___ AHF-7B, Ventilation MCC 3B-9C
1.3	___ Open RB exhaust dampers for emergency operation (OPS).	• ___ Select AHV-77 to the "EMERGENCY OPERATION OF AHD-95, AHD-96, AND AHD-94" position. (143 ft AB Ventilation Equipment Area, HVAC-13) • ___ Select AHV-78 to the "EMERGENCY OPERATION OF AHD-97, AHD-98, AND AHD-94" position. (143 ft AB Ventilation Equipment Area, HVAC-13)

PREREQUISITE FIELD ACTIONS

ACTIONS	DETAILS
1.4 ___ Ensure RM-A1 is in service (OPS/DAT).	1 ___ Place RM-A1N-1 Pump Switch in Auto (143 ft AB RM-A1 area)
	2 Ensure the following MCB annunciator links are closed: ___ 1712 ___ 1713 ___ 1714
	3 ___ Place RM-A1 Horn Silence Switch in the OFF Position
	4 ___ Notify chemistry to ensure RM-A1N-F Particulate/Iodine Filter or Filter Holder is installed per SP-731B
	5 VERIFY RM-A1N RDU indication: ___ The Green OPERATE LED is LIT ___ The Yellow TEST LED is NOT LIT ___ The Orange ALERT LED is NOT LIT ___ The Red HIGH LED is NOT LIT ___ The Red HIGH-HIGH LED is NOT LIT
	6 VERIFY RM-A1A RDU indication: ___ The Green OPERATE LED is LIT ___ The Yellow TEST LED is LIT ___ The Orange ALERT LED is NOT LIT ___ The Red HIGH LED is NOT LIT ___ The Red HIGH-HIGH LED is NOT LIT

PREREQUISITE FIELD ACTIONS

ACTIONS	DETAILS
1.5 ___ Bypass RM-A1 auto actuation (OPS).	• ___ Obtain Key 138 and place RM-A1N in "BYPASS"
1.6 ___ Notify Repairs Coordinator to obtain and install flow instrumentation (ERT).	• ___ CONCURRENTLY PERFORM MP-815, Installation of Post Accident H_2 Purge Flow Instruments.
1.7 ___ WHEN H_2 Purge Flow Instruments are installed THEN notify the Accident Assessment Team Coordinator that Enclosure 4, Prerequisite Field Actions is complete (OPS/ERT).	

RB PRESSURIZATION FOR HYDROGEN PURGE

STATUS
• RB H_2 Concentration ≥ 3.4% • Portage Air Compressors are installed.

ACTIONS	DETAILS
1.1 ___ Consult Radiation Monitoring Team to determine routes and precautions to be used while performing RB Pressurization (ERT).	• Refer to Enclosure 9, Access Routes for locations of required actions/components and suggested routes.

1.2 ___ IF portable air compressors were connected to RB portable compressor connections, THEN start air supply to RB and establish and maintain RB PRESS at ≈ 2 psig (ERT/Ops).

1 ___ Start portable air compressors.

2 Open isolation valves for operating air compressors (119 ft IB west door):

___ LRV-11	___ LRV-16
___ LRV-12	___ LRV-17
___ LRV-13	___ LRV-18
___ LRV-14	___ LRV-19
___ LRV-15	___ LRV-20

3 ___ Unlock and open LRV-36 "AIR SUPPLY TO PENETRATION 121 ISO" (119 ft IB south of A MSSVs).

4 ___ Unlock and open LRV-50 "PENETRATION 121 ISO" (119 IB ft south of PZR Htr MCC 3B overhead).

5 ___ Adjust LRV-26 "LRV-24 BYPASS" (119 ft IB south of A MSSVs) to maintain RB PRESS at ≈ 2 psig.

RB PRESSURIZATION FOR HYDROGEN PURGE

ACTIONS	DETAILS
1.3 ___ IF portable air compressors were connected to H_2 recombiner connections, THEN start air supply to RB and establish and maintain RB PRESS at ≈ 2 psig (ERT/Ops).	1 ___ Start portable air compressors. 2 Open H_2 recombiner connection isolations for operating air compressors (119 ft IB):

___ LRV-87 (unlock)	___ LRV-88 (unlock)
___ LRV-89 (unlock)	___ LRV-90 (unlock)
___ LRV-91 (unlock)	___ LRV-92 (unlock)
___ LRV-93 (unlock)	___ LRV-94 (unlock)

3 ___ Adjust the compressor output to establish and maintain RB PRESS at ≈ 2 psig.

1.4 ___ WHEN RB PRESS is being maintained at ≈ 2 psig, THEN notify the Accident Assessment Team Coordinator that Enclosure 5, RB Pressurization for Hydrogen Purge is complete (OPS/ERT).

PURGE RELEASE AUTHORIZATION FORM

PRAF #________________

COMPLETED BY THE ACCIDENT ASSESSMENT TEAM:

1) Date/Time accident started: ____________/____________

2) Projected Date/Time for purge start: ____________/____________

3) Time after accident for purge start: ____________ (hrs) [1 minus 2]

4) Error Corrected Flowrate based on time after accident (see Enclosure 10) ____________ (scfm)

Completed By: ____________________________________ Date: ____________

COMPLETED BY THE DOSE ASSESSMENT TEAM:

Containment Atmosphere Activity (μCi/cc) : ________________________________

Meteorological Conditions used in projection:

Wind Direction _________ Wind Speed _________ Stability Class _________

Projected purge duration = 1440 minutes (1 day)

RADDOSE-IV Projected Dose (REM) based on Error Corrected Flow rate:

Site Boundary ____________ 2 miles ____________ 5 miles ____________10 miles ____________

RADDOSE-IV Projected Curies to be released: Noble Gas ____________ Iodine ____________

Completed By: ____________________________________ Date: ____________

COMPLETED BY EMERGENCY COORDINATOR:

EOF Director notified: ________________________

EOF Director notified: Date/Time ____________/____________

Ensure the EOF Director has coordinated with the State and local government officials before initiating purge.

EMERGENCY COORDINATOR APPROVAL ____________________/____________
Sign/Date

PURGING RB

STATUS
• RB Purge Is Required

ACTIONS	DETAILS
1.1 ___ Ensure Enclosure 2, 3, 4, and 5 of this procedure have been completed (AAT).	• ___ Enclosure 2, Preparations for RB Hydrogen Purge complete • ___ Enclosure 3, Portable Compressor Installation complete • ___ Enclosure 4, Prerequisite Field Actions complete • ___ Enclosure 5, RB Pressurization for Hydrogen Purge complete
1.2 ___ Determine required purge flow rate (AAT/DAT).	• ___ IF H_2 purge has been previously performed, THEN use flows from previous purge. • ___ IF H_2 purge has NOT been previously performed, THEN refer to Enclosure 10, Continuous Purge Flow Rates after a LOCA to determine flows: ___ Required Purge Flow ______ scfm ___ Error Corrected Flow______ scfm • ___ Record Error Corrected Flow on Enclosure 6, Purge Release Authorization Form.
1.3 ___ Consult Radiation Monitoring Team to determine routes and precautions to be used while performing RB Pressurization (ERT).	• Refer to Enclosure 9, ACCESS ROUTES for locations of required actions/components and suggested routes.
1.4 ___ WHEN Enclosure 6, Purge Release Authorization Form is complete and approved by the EC, THEN continue with this enclosure.	

PURGING RB

STATUS
• EC has approved Purge Release Authorization Form, Enclosure 6

		ACTIONS	DETAILS
1.5	___	Notify the EC and the EOF Director that RB hydrogen purge is commencing (AAT).	
1.6	___	Start RB purge Exhaust fan (OPS).	• Start at least one RB Exhaust fan: ___ AHF-7A ___ AHF-7B
1.7	___	IF RB purge has previously been performed, THEN open purge isolation valves associated with the previously adjusted throttle valve (OPS).	• IF LRV-121 was previously throttled THEN Open A Train isolation valves. ___ LRV-70 ___ LRV-71 • IF LRV-123 was previously throttled THEN Open B Train isolation valves. ___ LRV-72 ___ LRV-73

PURGING RB

ACTIONS	DETAILS
1.8 ___ IF purge has NOT previously been performed, THEN establish required RB purge flow (OPS).	1 ___ Record “Required Purge Flow” from Step 1.2 of this enclosure. • Required Purge Flow ___ scfm 2 ___ IF A Train purging is desired, THEN perform the following in order: ___ Open LRV-70 ___ Open LRV-71 ___ Throttle LRV-121 to obtain “Required Purge Flow” on flow indicator LR-82-FI (143 ft AB Ventilation Room). ___ Record reading from LR-82-FI _________ scfm 3 ___ IF B Train purging is desired, THEN perform the following in order: ___ Open LRV-72 ___ Open LRV-73 ___ Throttle LRV-123 to obtain “Required Purge Flow” on flow indicator LR-83-FI (143 ft AB Ventilation Room). ___ Record reading from LR-83-FI _________ scfm
1.9 ___ Maintain RB PRESS constant at ≈ 2 psig (OPS).	• ___ IF portable air compressors were connected to RB portable compressor connections, THEN adjust LRV-26 "AIR SUPPLY TO PENETRATION 121 CONTROL BYPASS" (119 ft IB south of A MSSVs) to maintain RB PRESS at ≈ 2 psig. • ___ IF portable air compressors were connected to H_2 recombiner connections, THEN adjust the compressor output to maintain RB PRESS at ≈ 2 psig.

PURGING RB

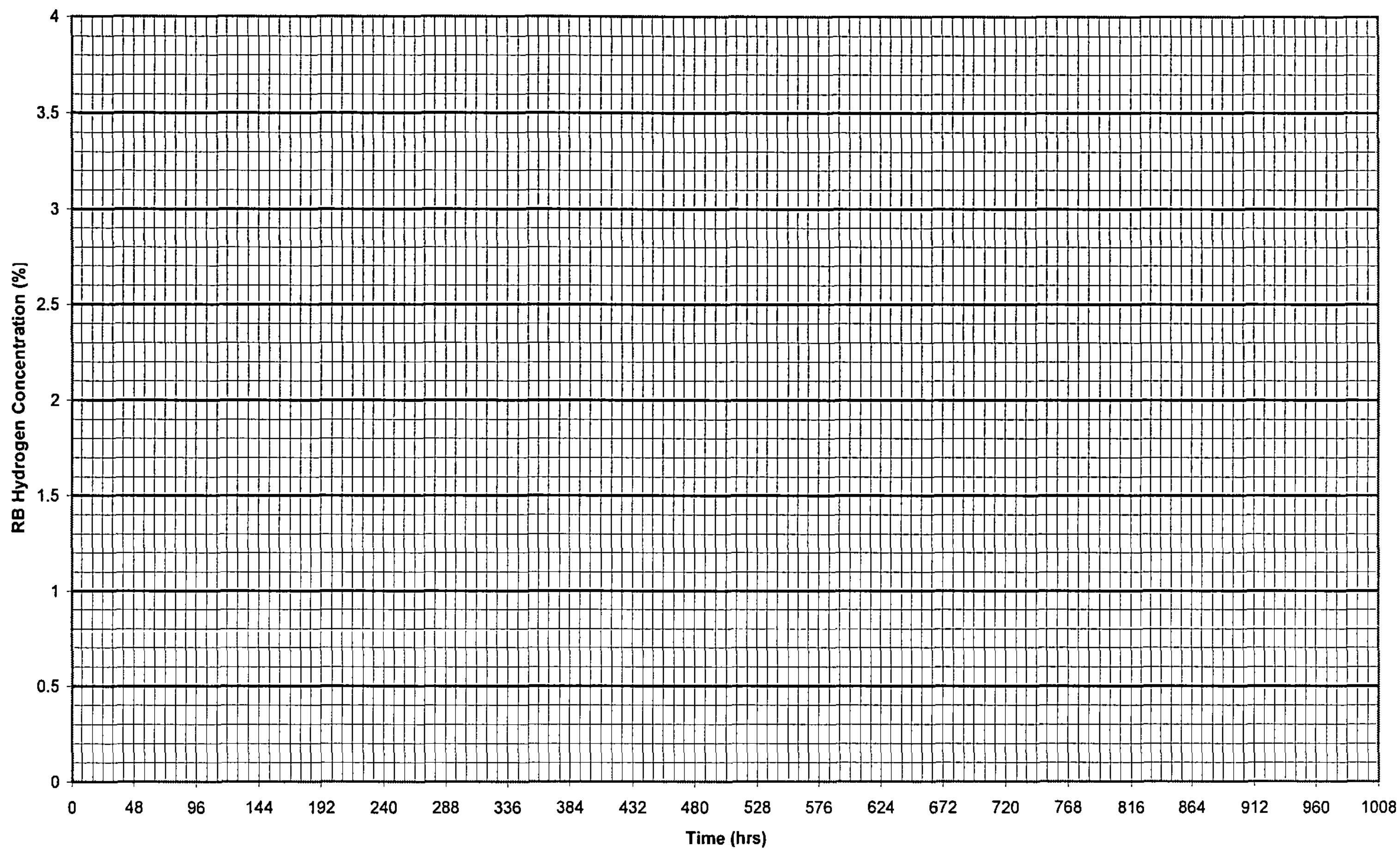
ACTIONS	DETAILS
1.10 ___ WHEN all of the following exist: ___ RB H_2 Concentration is ≤ 3.5% ___ EC approves termination THEN stop RB purge (OPS/ERT).	1 Ensure the following valves are closed: A Train: ___ LRV-70, ___ LRV-71 B Train: ___ LRV-72, ___ LRV-73 2 Ensure RB exhaust fans are stopped: ___ AHF-7A ___ AHF-7B 3 ___ IF portable air compressors are connected to RB portable compressor connections, THEN close the following valves: ___ LRV-50 "PENETRATION 121 ISO" (119 ft IB south of PZR Htr MCC 3B overhead) ___ LRV-36 "AIR SUPPLY TO PENETRATION 121 ISO" (119 ft IB south of A MSSVs) 4 ___ IF portable air compressors are connected to H_2 recombiner connections, THEN close the following valves: ___ LRV-87, ___ LRV-88, ___ LRV-89, ___ LRV-90, ___ LRV-91, ___ LRV-92, ___ LRV-93, ___ LRV-94 5 ___ Stop portable air compressors.

A Train	B Train
___ LRV-70	___ LRV-72
___ LRV-71	___ LRV-73

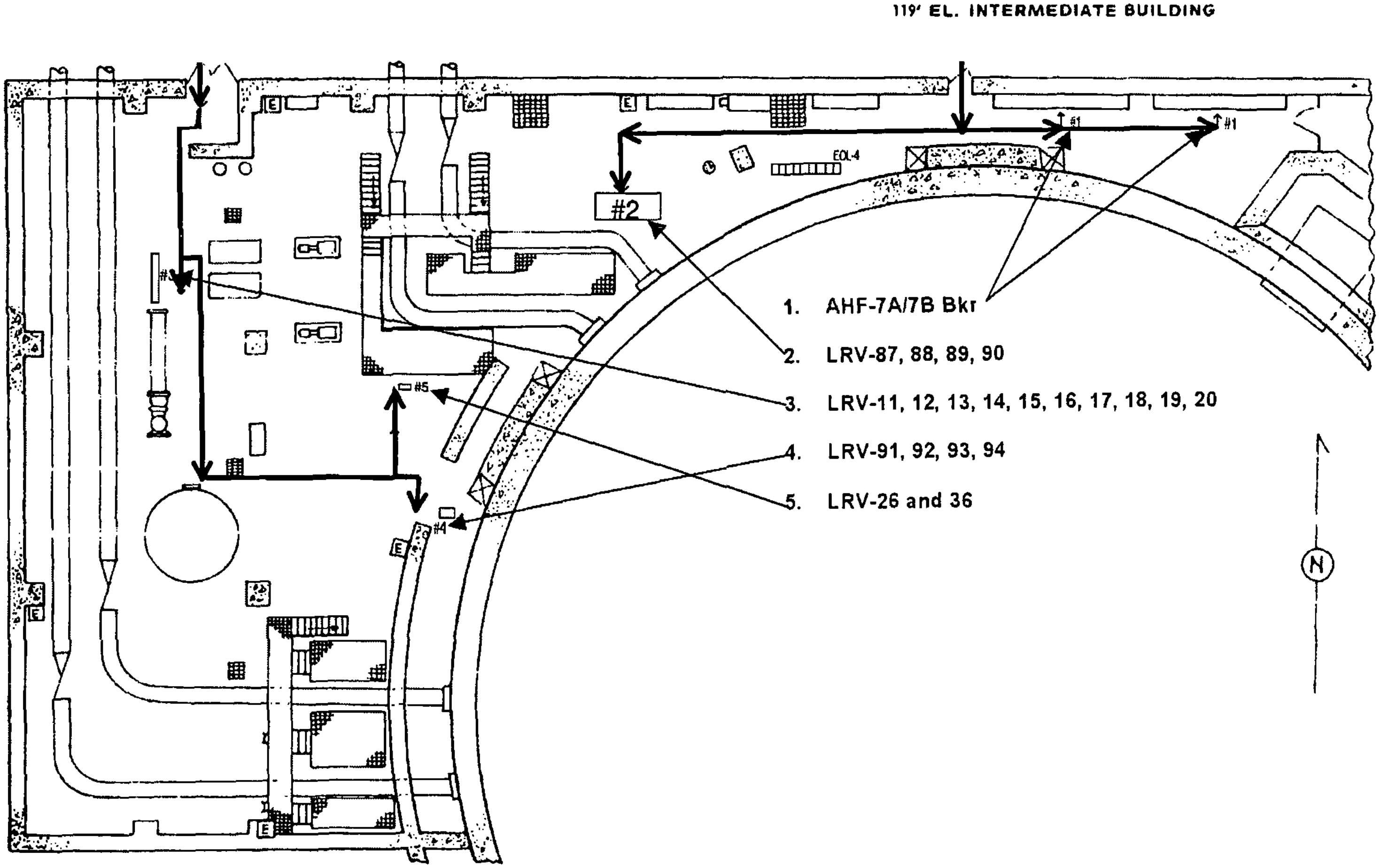
___ LRV-87	___ LRV-88
___ LRV-89	___ LRV-90
___ LRV-91	___ LRV-92
___ LRV-93	___ LRV-94

1.11 ___ Notify the Accident Assessment Team Coordinator that RB purge is secured.

RB HYDROGEN CONCENTRATION TREND



ACCESS ROUTES



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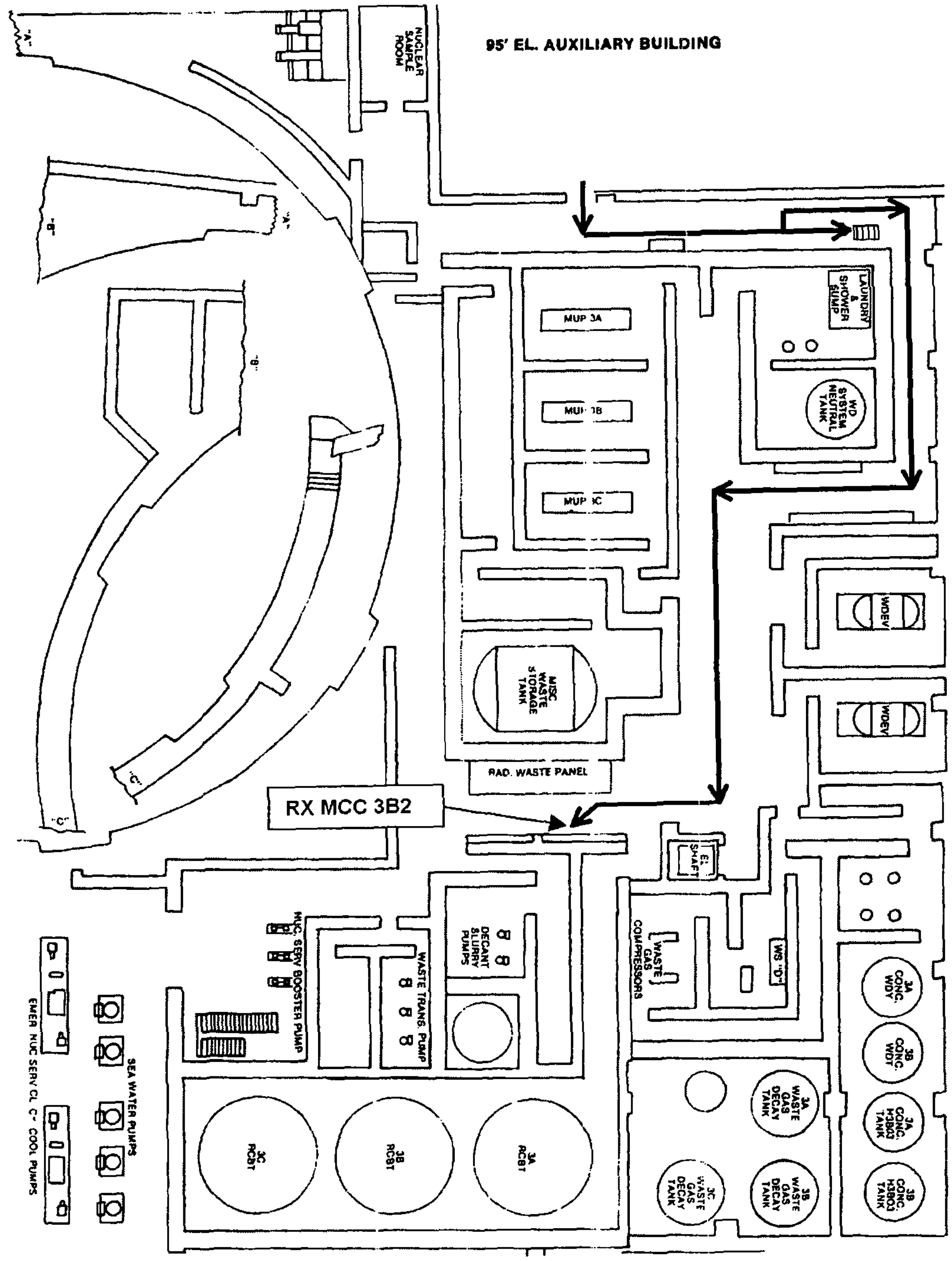

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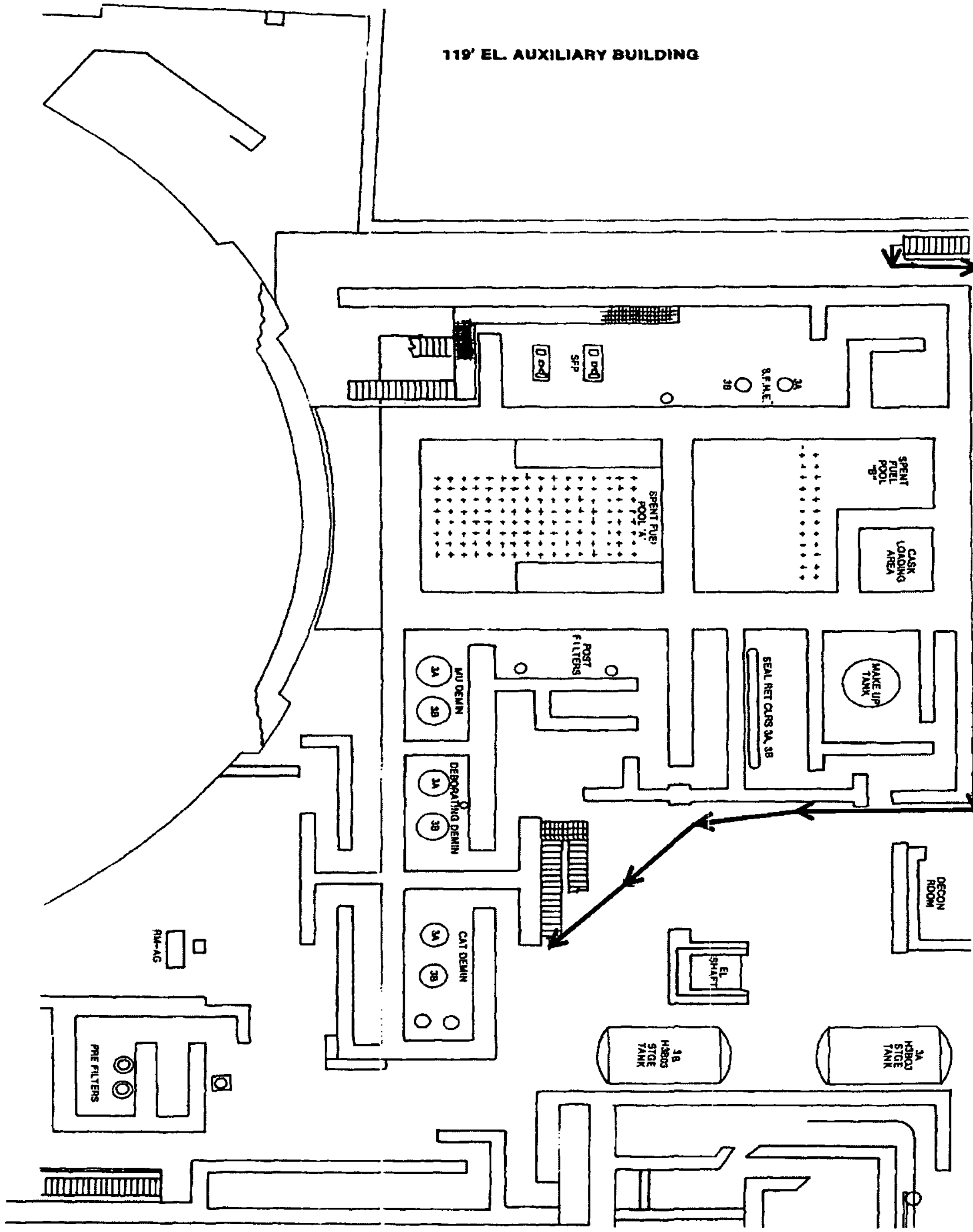

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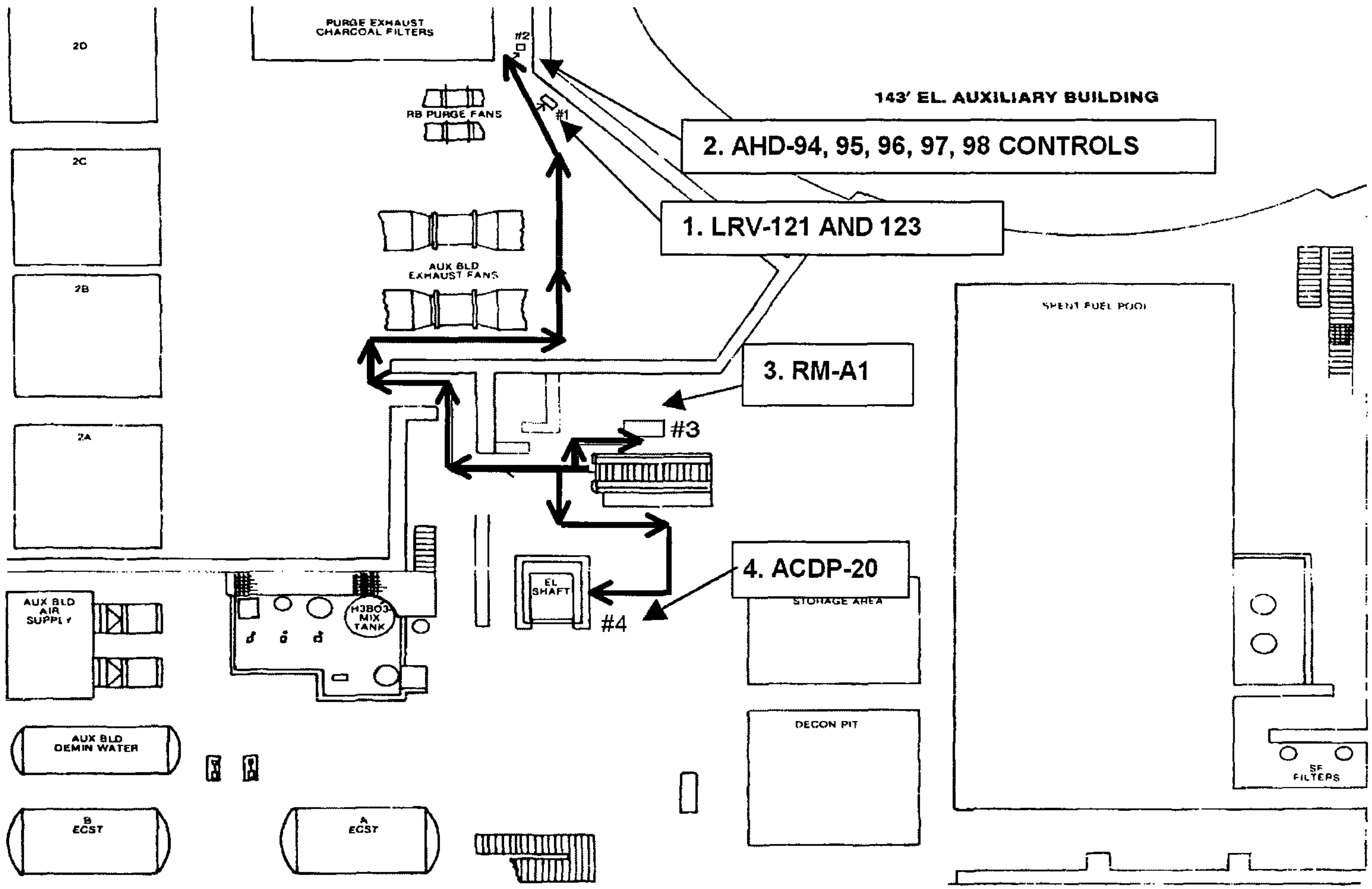
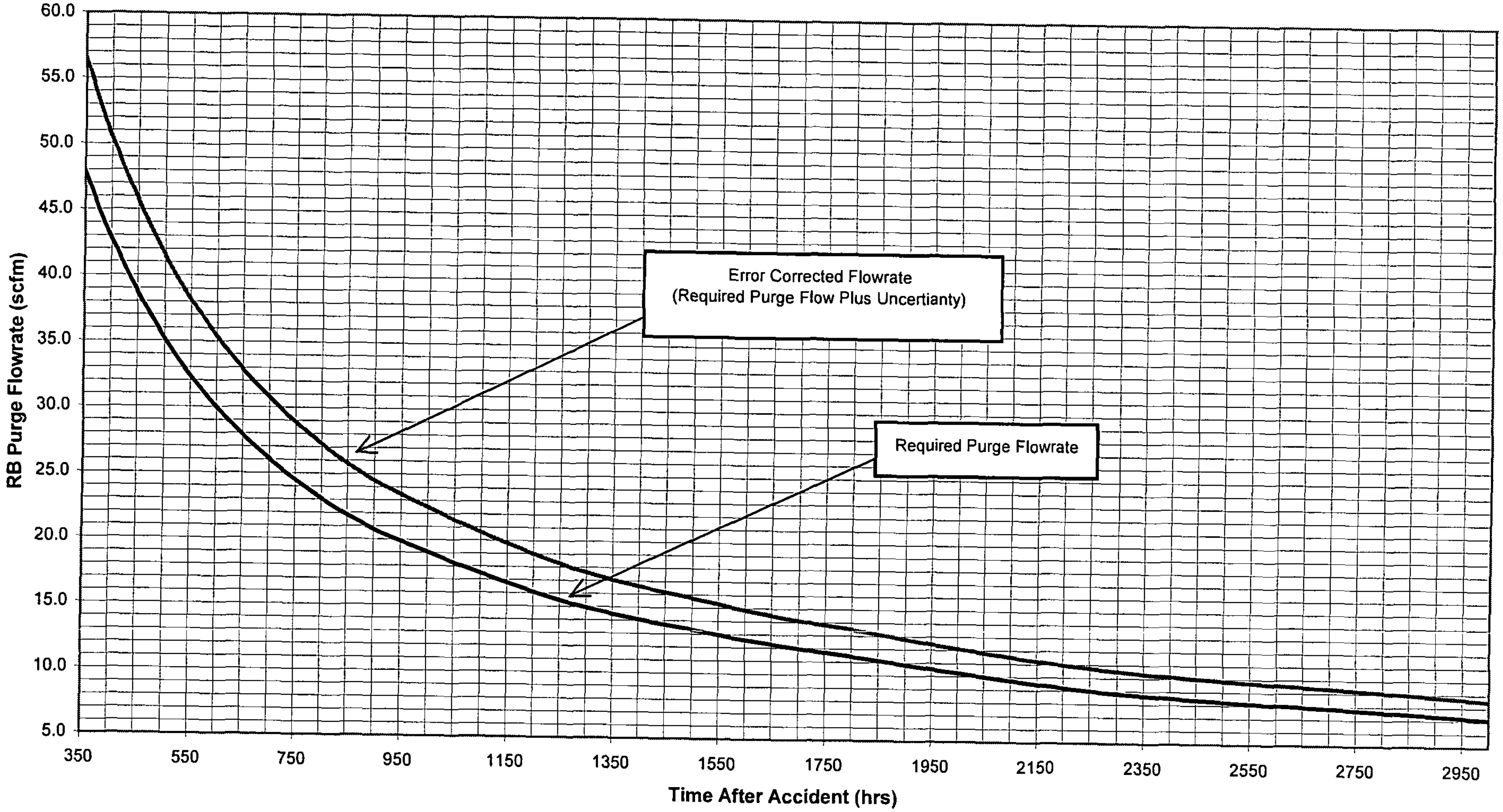
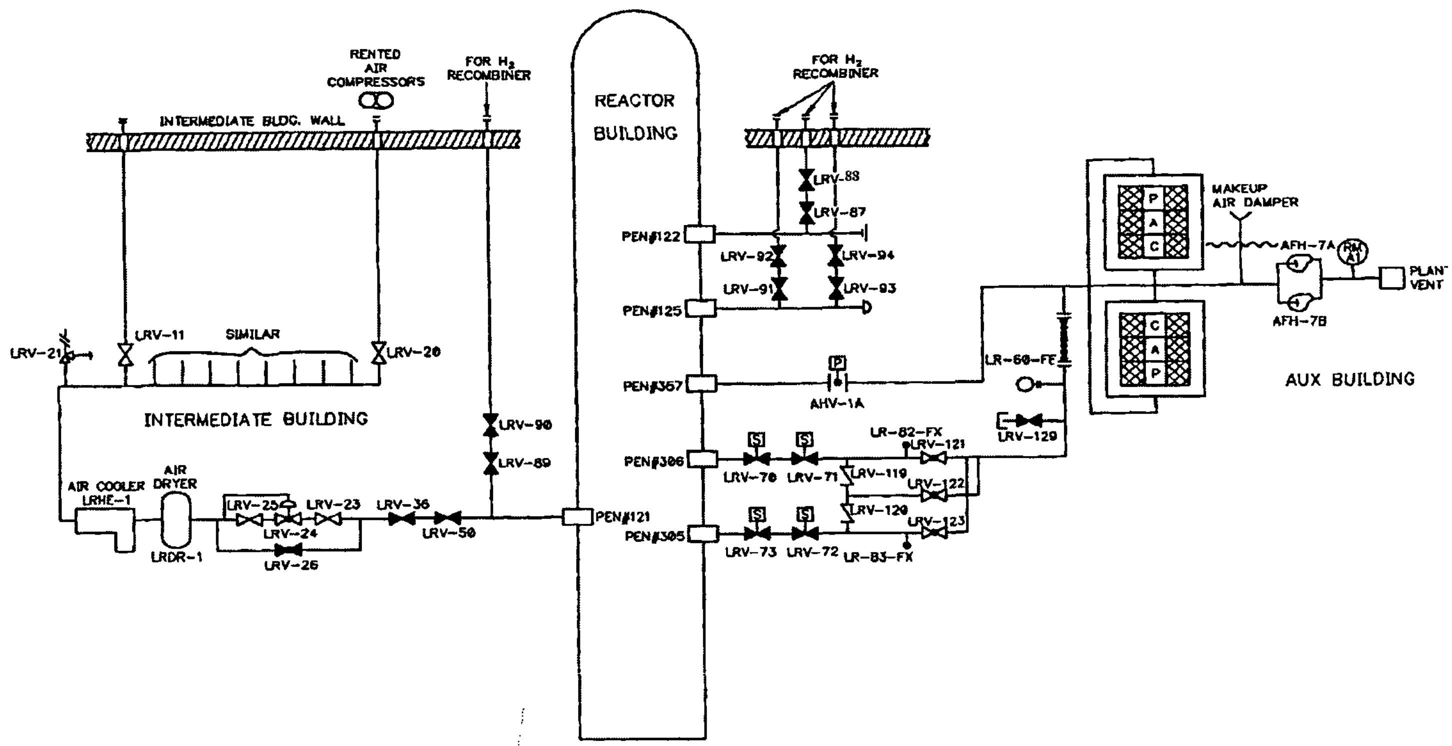


CONTINUOUS PURGE FLOW RATES AFTER A LOCA



HYDROGEN PURGE SYSTEM FLOW DIAGRAM



SUMMARY OF CHANGES
PRR # 553863

NOTES:	1. Writers and Reviewers: Ensure that any changes to this procedure that affect information contained in Emergency Response Facility posters, enclosures, briefing cards, guidelines etc. are made to those items as well. 2. Writers and Reviewers: Changes to certain parts of this procedure may impact other Emergency Preparedness Implementing Procedures. Ensure appropriate PRRs are initiated as needed.

SECTION	CHANGE
2.1.11	Added EC 76363, Radiation Monitors RM-A1/RM-A2 Replacement to the developmental references.
Enclosure 4 Step 1.4 and 1.5	Revised step based on changes from RM-A1/A2 replacement under EC 76363. The new RM-A1 and RM-A2 radiation monitors do not have a low/mid/hi range. Instead, they have a normal range and an accident range. When the radiation monitors are operable, no manual operator actions are required to shift from normal range to accident range when the hi-hi setpoint is reached on the normal range. Added additional guidance for start-up of RM-A1N which was taken for OP-505 Rev. 26 changes. Added a separate step for placing the key switch in the bypass prevent any auto actuation.

Progress Energy

R
Reference
Use

CRYSTAL RIVER UNIT 3

PLANT OPERATING MANUAL

EMERGENCY PLAN IMPLEMENTING PROCEDURE

EM-402

EMERGENCY OPERATIONS FACILITY TECHNICAL SUPPORT TEAM

TABLE OF CONTENTS

ENCLOSURES (Optional Non-Quality Records)

1.0 PURPOSE

1.1 To provide guidance for establishment and operation of the Emergency Operations Facility (EOF) Technical Support Team (TST).

1.2 The Technical Support Team primarily functions as a technical resource for the EOF Director in development of public Protective Action Recommendations (PARs). Technical Support Team duties also include interfacing with the Dose Assessment Team, monitoring Critical Safety Functions, and assisting in the development of recovery plans. The Technical Support Team can assist the TSC Accident Assessment Team in the development of mitigation strategies and in researching solutions to plant problems provided such assistance does NOT interfere with the team's primary responsibilities. Technical Support Team personnel are NOT considered Severe Accident Management evaluators.

1.3 This procedure is an Emergency Plan Implementing Procedure. Any revisions must be carefully considered for Emergency Plan impact.

2.0 DEVELOPMENTAL REFERENCES

2.1 EM-202, Duties of the Emergency Coordinator

2.2 EM-401, Setup of the Emergency Operations Facility

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

2.4 Emergency Operating Procedures (EOPs)

2.5 Response Technical Manual (RTM-96), USNRC, Volume 1, Revision 3

2.6 Radiological Emergency Response Plan

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

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

2.9 CH-632, Post Accident Sampling and Analysis Of Reactor Coolant, Decay Heat, and Reactor Building Sump

2.10 Engineering Evaluation EC 86189

3.0 PERSONNEL INDOCTRINATION

3.1 Definitions

3.1.1 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.

3.1.2 Emergency Action Levels (EALs)

A pre-determined, observable threshold for Plant conditions that places the Plant in a given emergency classification.

3.1.3 Inadequate Core Cooling

Accident conditions that result in a loss of core cooling that requires entering EOP-7, Inadequate Core Cooling.

3.1.4 Protective Action Recommendations (PARs)

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

3.1.5 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.

3.1.6 Technical Support Team (TST)

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

3.2 Responsibilities

3.2.1 Technical Support Coordinator

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

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

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

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

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

f. Provides support to the TSC Accident Assessment Team in determining the causes and consequences of the emergency.

g. Ensures interface is established with the EOF Dose Assessment Team using Section 3.2.4 and Enclosure 7 as guidance.

h. Refers to enclosures for additional accident assessment guidance and information.

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

j. Monitors CSFs and provides status to EOF personnel during briefings, as needed.

k. Supplies input for completion of the Florida Nuclear Plant Emergency Notification Form for plant conditions information.

3.2.2 Technical Support Operations Representative

a. Ensures the SPDS computer is properly set up and operational.

b. Operates and monitors the SPDS computer.

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

d. Obtains CSF assessments from the TSC Accident Assessment Team. Verifies accuracy using Enclosure 5. Performs an independent assessment of CSFs using Enclosure 5 if CSF assessments from the TSC are NOT available.

e. Monitors communications between the Control Room and the TSC Accident Assessment Team via speakerphone in the Technical Support Room.

f. Assists in the setup of the Technical Support Work Area.

3.2.3 Technical Support Engineer

a. Assesses plant conditions.

b. Performs Core Damage Assessment using Enclosure 6.

c. Provides Engineering support to the TSC Accident Assessment Team.

d. Notifies additional Engineering resources when necessary.

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

f. Provides the Dose Assessment Team Leader with an evaluation of the accident type using Enclosure 7.

3.2.4 Dose Assessment Team

a. Supports the TSC Accident Assessment Team with on-site radiological data.

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).

3.3 Limits & Precautions

3.3.1 None

4.0 INSTRUCTIONS

4.1 The Technical Support Coordinator or designee performs the duties of Enclosure 1.

4.2 The Technical Support Operations Representative performs the duties of Enclosure 2.

4.3 The Technical Support Engineer performs the duties of Enclosure 3.

TECHNICAL SUPPORT COORDINATOR CHECKLIST

Check

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

- ☐ Technical Support Operations Representative
- ☐ Technical Support Engineer (3)

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

☐ 3. Ensure the Accident Assessment Ringdown speakerphone is functional. (Refer to Enclosure 9) (Notify the TSC AAT by plant extension as necessary or press the "MUTE" button to remove the mute function.)

☐ 4. Ensure accuracy of the TSC Critical Safety Functions. (Refer to Enclosure 4) Report Critical Safety Functions discrepancies to the TSC Accident Assessment Coordinator and the EOF Director. Upon request of the EOF Director, perform an independent assessment of Critical Safety Functions.

☐ 5. Ensure PARs are appropriate per EM-400, Attachment 31.

☐ 6. Provide input for completion of the Florida Nuclear Plant Emergency Notification Form.

☐ 7. Ensure Room 124 SPDS computer is functional. (Refer to Enclosure 9)

☐ 8. Establish communication with the TSC until the EOF Communicator arrives. (Refer to Enclosure 9)

☐ 9. Retrieve the EOP/AP procedure book from the Operations Briefing Room (217) or Simulator Control Booth (220).

☐ 10. Brief the EOF Director on Critical Safety Functions and plant status.

☐ 11. Provide Critical Safety Functions and plant status reports to the EOF Staff during briefings. Use guidelines on Enclosure 4 and Enclosure 5 as necessary.

☐ 12. Establish interface with EOF Dose Assessment Team Leader. Enclosure 7 lists information needed by the Dose Assessment Team; Section 3.2.4 lists information available from the Dose Assessment Team.

EOF TECHNICAL SUPPORT COORDINATOR CHECKLIST (Continued)

Check

☐ 13. Provide the Status Board Coordinator current Critical Safety Functions and plant status information for the Main Conference Room.

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

☐ 15. Assist the EOF Director with the recovery plan. (Refer to Enclosure 8)

EOF TECHNICAL SUPPORT OPERATIONS REPRESENTATIVE CHECKLIST

Check

☐ 1. Ensure the SPDS computer is operational. (Refer to Enclosure 9)

☐ 2. Ensure the work area is set up and functional. (Refer to EM-401, Enclosure 7)

☐ 3. Ensure the Accident Assessment Ringdown Monitor is functional. (Refer to Enclosure 9) (Notify the TSC AAT by plant extension as necessary or press the "MUTE" button to remove the mute function.)

☐ 4. Obtain Critical Safety Functions assessments from the TSC. Review TSC Critical Safety Functions assessments using Enclosure 5. If TSC Critical Safety Functions assessments are NOT available, perform an independent assessment using Enclosure 5.

☐ 5. Monitor communications between the Control Room and the TSC Accident Assessment Team using the speakerphone in the Technical Support Room.

☐ 6. Provide periodic Critical Safety Functions and plant status updates to the Technical Support Coordinator.

☐ 7. Assist the Status Board Coordinator to ensure the appropriate SPDS screens are displayed in the Main Conference Room. Screens may need to be adjusted as conditions change.

TECHNICAL SUPPORT ENGINEER CHECKLIST

Check

☐ 1. Ensure the work area is set up and functional. (Refer to EM-401, Enclosure 7)

☐ 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 Enclosure 8)

☐ 6. Perform Core Damage Assessments. (Refer to Enclosure 6)

☐ 7. Provide initial core damage assessment to Technical Support Coordinator.

☐ 8. Provide core status updates to Technical Support Coordinator.

☐ 9. Complete Enclosure 7, "Dose Assessment Team Notification." Review results with the Technical Support Coordinator and forward to the EOF Dose Assessment Team Leader.

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

BRIEFING GUIDELINE

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

I. REACTOR SHUTDOWN

Yes ☐ No ☐

II. CORE ADEQUATELY COOLED

Yes ☐ No ☐

III. FISSION PRODUCT BARRIER ASSESSMENT (Use Enclosure 5, Part III)

Fuel Clad:	Intact ☐	Potential Loss ☐	Loss ☐
RCS:	Intact ☐	Potential Loss ☐	Loss ☐
Containment:	Intact ☐	Potential Loss ☐	Loss ☐

IV. EMERGENCY ELECTRICAL POWER STATUS

Off-Site Power Available?	Yes ☐	No ☐
ES Bus Energized?	Yes ☐	No ☐
Emergency Diesel Generator Available?	Yes ☐	No ☐
Alternate AC Diesel Generator 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.

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. Critical Safety Function information is also available from OSI/PI (Start/Programs/Business Apps/PI System/Progress Energy PI Displays/CR3 Qualified/EP).

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			
Adequate Shutdown Margin	OP-103C Curve 18&19				

Performed By: ______________________ Date: ____________ Time: ________

II. CORE COOLING STATUS:

ECCS/SUPPORT STATUS

PARAMETER	COMPUTER POINT	RECALL POINT			
Subcooling Margin					
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 X064	RECL-207 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			
RWP-1 operating	X060				
RWP-2A operating	X061	RECL-222			
RWP-2B operating	X062	RECL-223			
RWP-3A operating		RECL-217			
RWP-3B operating		RECL-218			
DCP-1A operating (yes/no)		RECL-220			
DCP-1B operating (yes/no)		RECL-221			
SWP-1A operating		RECL-219			
SWP-1B operating					
SWP-1C operating					
RB Sump Strainer ΔP		RECL-79			

II. CORE COOLING STATUS:

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					
EFP-1/3 Flow to A OTSG		RECL-246			
EFP-1/3 Flow to B OTSG		RECL-245			
EFP-2 Flow to A OTSG		RECL-248			
EFP-2 Flow to B OTSG		RECL-247			
Total EFW Flow to A OTSG	S300	RECL-408			
Total EFW Flow to B OTSG	S312	RECL-409			
EFW Tank Level		RECL-236			

Performed By: ______________________ Date: ____________ Time: __________

III. FISSION PRODUCT BARRIER ASSESSMENT:

FUEL CLAD		
☐ **INTACT**	☐ **POTENTIAL LOSS**	☐ **LOSS**
• Does NOT meet the criteria for "Potential Loss" or "Loss"	• RCS condition warrant entry into EOP-07 • Core Exit Thermocouples > 700 degrees F	• RCS conditions in (or previously in) Region 3 or Severe Accident Region • Chemistry results indicate increased RCS activity >300μCi/gr I_{131} dose equivalent (refer to CH-632). Additional indication is 100 mR/hr measured on RM-G3 or at one foot from sample lines in Nuclear Sample Room. • RM-G29/30 > 100 R/hr for ≥ 15 minutes • Enclosure 6 indicates failed fuel
REACTOR COOLANT SYSTEM		
☐ **INTACT**	☐ **POTENTIAL LOSS**	☐ **LOSS**
• Does NOT meet the criteria for "Potential Loss" or "Loss"	• RCS leak or OTSG tube leak requiring one or more injection valves to maintain adequate subcooling margin • RCS pressure /Tincore relationship violates NDT limits • RCS leak or OTSG tube leak results in ES actuation on low RCS pressure. • HPI/PORV or HPI/Code Safety valve cooling is in progress	• RCS leak resulting in loss of adequate subcooling margin • OTSG Tube Rupture resulting in loss of adequate subcooling margin • RM-G29/30 >10R/hr for ≥ 15 minutes
CONTAINMENT		
☐ **INTACT**	☐ **POTENTIAL LOSS**	☐ **LOSS**
• Does NOT meet the criteria for "Potential Loss" or "Loss"	• RB pressure > 54 psig • RB hydrogen concentration > 4% • RB pressure > 30 psig with NO building spray available • RMG-29 or 30 reading > 5,000 R/hr • Core conditions in severe accident region of ICC curves for >15 min	• Containment isolation is incomplete and release path to environment exists. Confirmation may be from elevated radiation readings in areas adjacent to the RB. • OTSG Tube Rupture > 10 gpm exists and prolonged steaming to atmosphere or an unisolable steam leak outside RB from affected OTSG. • Containment pressure or sump level response NOT consistent with LOCA conditions • Rapid unexplained RB pressure decrease following an initial increase

Performed By: ____________________ Date: ______________ Time: __________

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			

ALTERNATE AC DIESEL GENERATOR

PARAMETER	RECALL PT.	LOADED	AVAILABLE	UNAVAILABLE

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.

Performed By: ______________________ Date: ______________ Time: __________

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: ____

Performed By: ____________________ Date: ____________ Time: ________

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 Technical Support Operations Representative and the EOF 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 OSC Chemistry Coordinator using CH-632 Enclosure 5. The results are submitted to the TSC Accident Assessment Team. (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

CORE DAMAGE ASSESSMENT (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

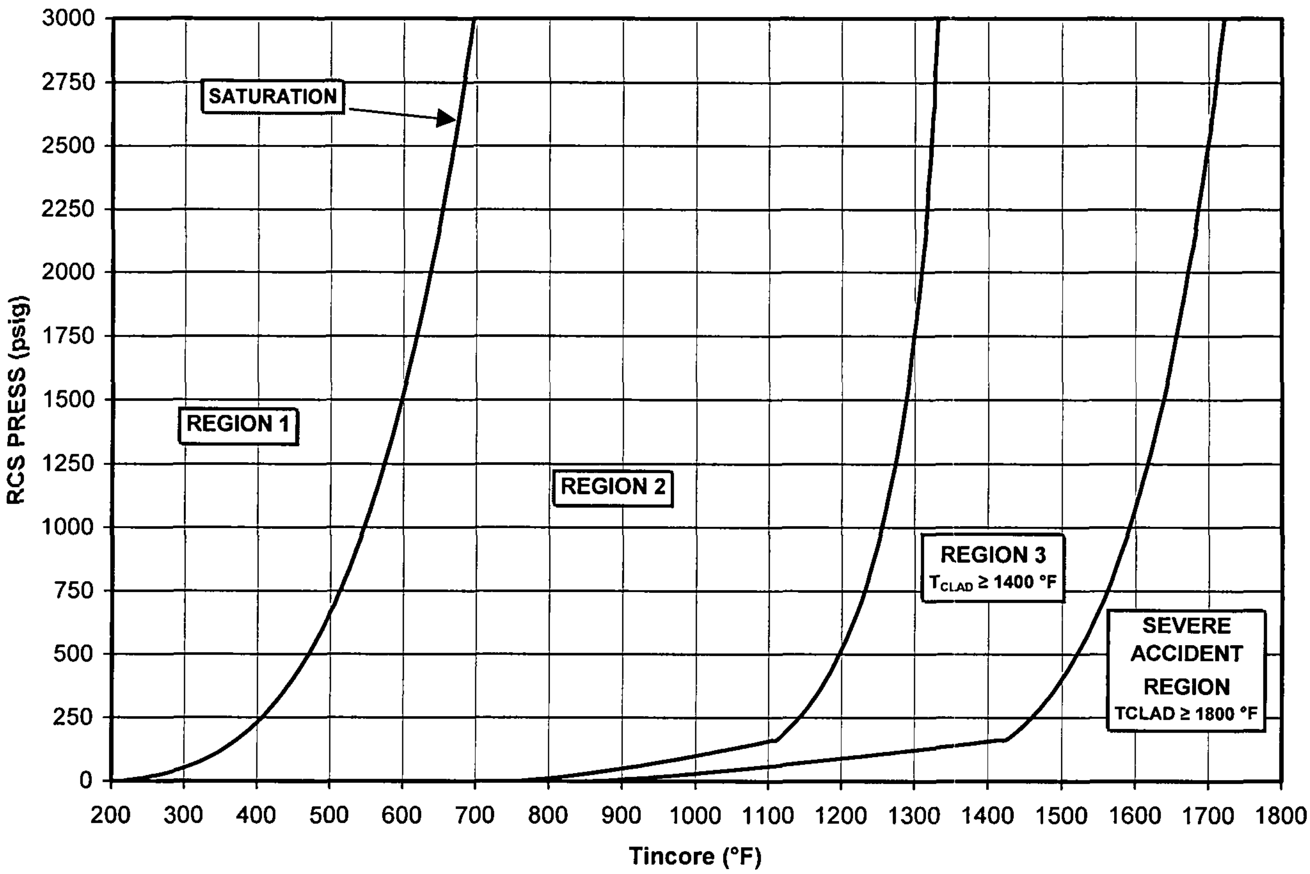
☐ **UNCOVERED FOR 30 TO 90 MINUTES**
Core temperature 2400-4200°F
Possible uncoolable core
Possible slump of molten core
Rapid release of volatile fission products (grain boundary release)

☐ **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

CORE DAMAGE ASSESSMENT (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.



DOSE ASSESSMENT TEAM NOTIFICATION

NOTES:
1. When the EOF is operational, the EOF TSE supplies the EOF DAT Leader with Attachment 7 to assist with the projection of off-site doses. .... ☐
2. MARK items N/A or unknown based on information available at the time. .... ☐
3. PROVIDE readily available information promptly **AND** FOLLOW UP with additional forms as more information becomes available. .... ☐
4. Attachment 7 can be completed in any order. .... ☐

- LOSS-OF-COOLANT ACCIDENT: Rx Trip Date/Time: _____/_____......N/A ☐ ☐

a. Rx Fuel Cladding status: (from Enclosure 6, Core Damage Assessment)
☐ Normal Activity (Spike Factor _______) (See General Information - Pg 3)
☐ Clad Damage
☐ Fuel melt

b. Rx Core Uncovered? (Rx Core covered is based on RCS NOT superheated)
☐ NO (Not in EOP-07, Inadequate Core Cooling)
☐ YES Uncovered - Date/Time __________/___________
Recovered - Date/Time __________/___________

c. Start of release to containment (start of the LOCA)
☐ Unknown
☐ Date/Time ___________________;

d. Release to atmosphere?
☐ NO
☐ YES
Date/Time _____/_____ Estimated duration _______
Release path (from where to where) ________________________
Release path flow rate (unmonitored releases):
☐ Estimated hole size ______ ☐ Diameter (in) or ☐ Area (in^2)
Containment pressure (RECL-82/83) _________ PSIG **OR**
☐ % RB Volume /day__________ (RB Volume is $2 X10^6$ cu ft) **OR**
☐ gpm _________ **OR**
☐ Design Basis Leakage

e. Reactor Building Spray Actuated? (RECL-212/213)
☐ NO
☐ YES Dates/Times ______________

f. Rx Bldg Vent flow rate (AH-1003-TIR Channel 4) __________
Charcoal banks in service ☐ YES ☐ NO

g. Auxiliary Building ventilation (W351): flow rate ____________
Charcoal banks in service ☐ YES ☐ NO

h. Loose Parts Monitor indications?
☐ Unavailable / ☐ NO
☐ YES Location _________________

DOSE ASSESSMENT TEAM NOTIFICATION

- WASTE GAS DECAY TANK RUPTURE: (See General Information-Pg 3).........N/A ☐ ☐

a. Release pathway:
☐ Tank rupture **OR**
☐ Valve leakage **OR**
☐ Other____________

Tank volume _________ (Each WGDT volume = 1753 ft^3)
Tank pressure ________ (RW203/204/205)

b. Release rate ☐ Unknown **OR** ☐ _________ CFM

c. Start of release
☐ Unknown **OR**
☐ Date/Time ________/________ Estimated duration _________

d. Auxiliary Building Ventilation (W351) Flow rate ____________
Charcoal banks in service ☐ YES ☐ NO

- STEAM GENERATOR TUBE RUPTURE: Rx Trip Date/Time: ______/_____ N/A ☐ ☐

a. Primary-to-secondary leak rate:
☐ Unknown **OR**
☐ _______________ gpm **OR**
☐ Number of tubes _________

b. Rx Fuel Cladding status: (from Enclosure 6, Core Damage Assessment)

c. ☐ Normal Activity (Spike Factor _______) (See General Information – Pg 3)
☐ Clad Damage
☐ Fuel melt

d. Rx Core Uncovered? (Rx Core cover is based on RCS NOT superheated)
☐ NO (Not in EOP-07, Inadequate Core Cooling)
☐ YES Uncovered - Date/Time __________/___________
Recovered - Date/Time __________/___________

e. Leaking OTSG isolated? ☐ NO / ☐ YES Date/Time __________/___________

f. Release Point:
☐ MSSV/ADV (intermittent/continuous) **OR**
☐ Condenser

g. Start of leak
☐ Unknown **OR** ☐ Date/Time _______________;

h. OTSG Water Mass (See General Information – Pg 3)
☐ OTSG Water Mass _____________ **OR**
☐ RASCAL default value of 93000 lbm

i. OTSG Steaming Rate (See General Information – Pg 3)
☐ OTSG Steaming rate _______________ **OR**
☐ RASCAL default 75000 lbm/hr

j. Auxiliary Building Ventilation (W351): Flow rate ____________
Charcoal banks in service ☐ YES ☐ NO

DOSE ASSESSMENT TEAM NOTIFICATION

- SPENT FUEL ACCIDENT:.. N/A ☐ ☐

a. Spent Fuel uncovered?
☐ NO
☐ YES Date/Time _____/_____ Recovered Date/Time _____/_____

b. Spent Fuel Pool Empty?
☐ NO
☐ YES Date/Time _____/_____ Recovered Date/Time _____/_____

c. Fuel assembly damaged by dropped component/handling?
(See General Information below)
☐ NO
☐ YES Number of assemblies damaged ______ Damage Date/Time ____/____
Last irradiation Date __________ (use last refueling outage)
Damaged Fuel assembly underwater ☐ YES / ☐ NO / ☐ UNKNOWN

d. Auxiliary Building Ventilation (W351): Flow rate ____________
Charcoal banks in service ☐ YES ☐ NO

e. Dry Cask lost cooling?
☒ N/A ☐ NO
☐ YES <24 hrs_____ >24 hrs_____ Cask on fire _____;
Number assemblies in cask _____ Type of dry cask ________________

GENERAL INFORMATION:

1. Initially use a spiking factor of 100. Adjust spiking factor based on RCS sampling and analyses when accident conditions allow sampling to be performed. The Spiking Factor is used in RASCAL to distinguish the change in concentration of the fission products in the RCS due to a rapid drop in RCS pressure. The sudden RCS pressure drop increases the rate at which the radioactive fission products in the fuel rod cladding gap escape to the RCS.
2. Waste Gas Decay Tank Rupture information does not need to be completed if RM-A2 is in service monitoring release.
3. Use RASCAL default value if data is not known at the time of completing Enclosure 7.
4. Fuel assembly damage is associated with damage from a dropped component and not clad failures due to overheating or flaws.

COMMENTS: ____

____

____

Performed By: __________________________ Date: _____/_____/_____ Time: __________

Reviewed By Technical Support Team Coordinator: ________________________________

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.

SHORT-TERM RECOVERY PLAN GENERIC OUTLINE (Continued)

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 (PE, 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.

EQUIPMENT INSTRUCTIONS

Accident Assessment Ringdown Monitor

1. Ensure the Accident Assessment Ringdown phone in the Simulator Control Room is off the hook. This phone must be off the hook for the Room 124 speakerphone to function.

2. Ensure the Accident Assessment Ringdown phone is connected to phone jack 124-D5.

3. Press the "SP PHONE" button on the Accident Assessment Ringdown phone and press the "MUTE" button. Release the "MUTE" button if conversation with the Accident Assessment Team is desired.

SPDS

1. Ensure the computer and monitor designated for SPDS in northwest corner of Room 124 is turned on.

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

3. Click "LOGON."

4. Double-click SPDS Display.

Summary of Changes
PRR 471764

NOTE

Writers and Reviewers: Changes to certain parts of this procedure may impact other EPIPs. Initiate PRRs as needed.

EM-402	EM-225	ERF Posters
Enclosure 1		
Enclosure 4	Attachment 2	TST Room, PAR Conf. Room
Enclosure 5	Attachment 8	
Enclosure 6	Attachment 10	TST Room, DAT Room
Enclosure 7	Attachment 9	

Ensure that any changes to this procedure that affect information contained in ERF posters, enclosures, briefing cards, guidelines, etc. are made to those items as well.

Section	Changes and Reason
Enclosure 7	The enclosure was reformatted to match revisions being concurrently implemented in EM-225 Attachment 9 to improve the human factoring. The required information was not changed, only the appearance of where the data is entered. The reformatting improves human performance implementation of the enclosure. (This change also addresses PRR 306358.) **(Editorial Change)** The definition for spiking factor and volume of the waste gas decay tank was added. The waste gas decay tank volume was taken from calculation M89-0063. **(Editorial Change)**
Enclosure 2, Step 2	Correct reference to EM-401, Enclosure '6' to Enclosure '7' – Editorial (PRR 471780)
Enclosure 3, Step 1	Correct reference to EM-401, Enclosure '6' to Enclosure '7' – Editorial (PRR 471780)
Enclosure 9	Revised Enclosure 9 to correct nomenclature – Editorial (PRR 471772)

Progress Energy	I Information Use

CRYSTAL RIVER UNIT 3

PLANT OPERATING MANUAL

EM-500

EQUIPMENT IMPORTANT TO EMERGENCY PREPAREDNESS AND RESPONSE

REVISION 00

TABLE OF CONTENTS

TABLE OF CONTENTS (Cont'd)

1.0 PURPOSE

1. The purpose of this procedure is to provide Crystal River 3 specific details for the implementation of EMG-NGGC-0007, Equipment Important to Emergency Preparedness and Response.
2. This procedure includes a description of the plant equipment and emergency response facilities needed to implement the Crystal River 3 Radiological Emergency Response Plan and delineates compensatory measures to be used when the equipment is unavailable.

2.0 REFERENCES

1. 10 CFR 50.47(b), Emergency Plans
2. 10 CFR 50.54(q), Conditions of Licenses, Emergency Plans
3. 10 CFR 50.72, Immediate Notification Requirements for Operating Nuclear Power Reactors
4. 10 CFR 50.73, Licensee Event Report System
5. 10 CFR 50, Appendix A, General Design Criteria for Nuclear Power Plants, GDC 3, Fire Protection
6. 10 CFR 50, Appendix E, Emergency Planning and Preparedness for Production and Utilization Facilities
7. 10 CFR 50, Appendix R, Fire Protection Program for Nuclear Power Facilities Operating Prior to January 1, 1979
8. ADM-NGGC-0104, Work Implementation and Completion
9. EMG-NGGC-0004, Maintenance of the Emergency Response Organization Notification System
10. EMG-NGGC-0005, Activation of the Emergency Response Organization Notification System
11. EMG-NGGC-0007, Equipment Important to Emergency Preparedness and Response.
12. Updated Final Safety Analysis Report (UFSAR)
13. INPO 10-007, Equipment Important to Emergency Response
14. NUREG-0654, Revision 1, "Criteria for Preparation and Evaluation of Radiological Emergency Response Plans and Preparedness in Support of Nuclear Power Plants"
15. NUREG-0696, "Functional Criteria for Emergency Response Facilities"
16. NUREG-0737, Clarification of TMI Action Plan Requirements
17. NUREG-1022, "Event Reporting Guidelines", 10 CFR 50.72 and 50.73

18. NUREG-1394, "Emergency Response Data System (ERDS) Implementation"
19. Offsite Dose Calculation Manual (ODCM)
20. CP-0151 External Reporting Requirements
21. RERP, Radiological Emergency Response Plan (RERP)
22. WCP-NGGC-0300, Work Request Initiation, Screening, Prioritization and Classification
23. EM-204A, Off-site Dose Assessment During Radiological Emergencies(Control Room Method)
24. EM-401, Set-up of the Emergency Operations Facility (Includes Set-up of the Emergency News Center).
25. HPP-409, Inventory and Availability of Emergency Supplies/Equipment
26. EM-211 Duties of the CR3 Nuclear Security Organization
27. OPS-NGG-1000 Fleet Conduct of Operations
28. EM-202 Duties of the Emergency Coordinator

3.0 DEFINITIONS/ABBREVIATIONS

Note
Definitions are detailed in EMG-NGGC-0007 Equipment Important to Emergency Preparedness and Response. Additional plant specific definitions are provided below:

1. **AARD:** Accident Assessment Ringdown
2. **AEF**: Alternate Emergency Facility
3. **Category A (1) Equipment:** Equipment that provides the sole indication, or very little redundancy, for a parameter used to assess an Emergency Action Level (EAL) threshold.
4. **Category A (2) Equipment:** Equipment that provides a sole means of fulfilling an emergency response function.
5. **Category B Equipment:** Equipment that has redundant components or trains that fulfill an emergency response function or redundant indications for a parameter used to assess an Emergency Action Level (EAL) threshold.
6. **Commercial Phone system:** The Commercial phone system connects to offsite lines via transfer codes or use of outside line.
7. **Compensatory Measure**: A temporary means of mitigating the degradation or loss of an emergency response function or of maintaining the emergency response function until the equipment is restored to a fully functional condition.
8. **CR**: Control Room
9. **DARD:** Dose Assessment Ringdown System
10. **EMnet:** Florida Emergency Management Network

11. **Emergency Response Facility (ERF)**: Facilities, buildings, and structures which are identified in the emergency plan and include systems and equipment that are used for emergency response during declared emergency plan events.
12. **ENC**: Emergency News Center
13. **EOF**: Emergency Operations Facility
14. **EP**: Emergency Preparedness (Unit or Staff)
15. **EPZ**: Emergency Planning Zone
16. **Equipment Important to Emergency Response (EIER)**: Refer to EMG-NGGC-0007 Equipment Important to Emergency Preparedness and Response.
17. **ERDS**: Emergency Response Data System
18. **ERFIS**: Emergency Response Facility Information System
19. **ERO**: Emergency Response Organization
20. **Operable / Operability**: As defined in the plant Technical Specifications.
21. **OSC**: Operational Support Center
22. **PAX**: Public Address Exchange System
23. **PE Telephone System:** Consists of equipment utilized to contact on-site extensions. The system is used to communication to on-site specific locations.
24. **SHRD**: State Hot Ring Down System
25. **TRM**: Technical Requirements Manual
26. **TSC**: Technical Support Center

4.0 RESPONSIBILITIES

4.1 Manager – Engineering

Responsible for ensuring that Engineering support is provided in the planning and execution of work on equipment essential to the ERO.

4.2 Supervisor Radiation Control (SRC)

Responsible for implementing the program that tests and inventories Radiological Emergency Kits, and Environmental Survey Vehicles in accordance with HPP-409, Inventory and Availability of Emergency Supplies/Equipment.

4.3 Manager - Maintenance

Responsible for ensuring that Maintenance support is provided for the following: Testing and maintaining the on-site EIER in a timely manner

4.4 Manager - Operations

Responsible for ensuring that Operations support is provided for the following: Ensuring that applicable actions, including identification, tracking, and compensatory measures, are taken when EP equipment or emergency response facilities are degraded or removed from service.

4.5 Manager - Outage and Scheduling

Responsible for ensuring that O&S support is provided for work on EP-related equipment within the scope of the work management program and that work is appropriately prioritized and scheduled (including corrective and preventive maintenance and testing).

4.6 Manager – Nuclear Plant Security

1. Responsible for ensuring that Security support is provided for testing and maintaining operability of the following:
   - Security Fences
   - Security Camera Systems
   - Security Computers
   - Security Communications

4.7 Supervisor – Digital Process Systems

Responsible for ensuring that Engineering support is provided for all process computers related to EIER

4.8 Supervisor - Emergency Preparedness (EP)

Responsible for maintaining oversight of EP facilities and equipment; having an awareness of the operational status of equipment essential to the ERO; and for ensuring that work and change-related processes include appropriate screening requirements to identify impacts on the EP program.

4.9 Supervisor - Licensing/Regulatory Programs

Responsible for providing guidance on compliance with the plant licensing basis and related reportability issues.

4.10 Supervisor – Telecommunications

Responsible for ensuring that Telecommunications support is provided for the communications equipment classified as EIER.

4.11 Superintendent - Chemistry and Environmental

Responsible for maintaining oversight and availability of lab analysis equipment for DEI and Count Room instrumentation; and for ensuring compensatory action are completed for equipment unavailability.

5.0 PREREQUISITES

When conducting a planned loss of an EIER ensure required redundant components are available and any applicable compensatory actions are in place prior to removing the EIER from service.

6.0 PRECAUTIONS, LIMITATIONS, AND NOTES

The Emergency Response Facilities, i.e., Control Room, Technical Support Center/Operational Support Center (TSC/OSC), Nuclear Security Operations Center (NSOC), Emergency Operations Facility (EOF), and the Emergency News Center (ENC), are described in the Emergency Response Plan (RERP) and supporting plant emergency procedures. All emergency facilities, including the AEF, must be maintained in a state of readiness and contain equipment required to respond to an emergency. Due to the broad scope of EP functions conducted from the emergency facilities, the loss of an ERF can have a significant impact on emergency plan implementation. Restoration of nonfunctional or degraded ERF's requires prompt management attention, and degraded or nonfunctioning equipment associated with these facilities will be restored in a timely manner.

7.0 SPECIAL TOOLS AND EQUIPMENT

7.1 State Hot Ring Down System (SHRD)

1. Category – B
2. Compensatory actions-Make notification in accordance with EM-202 Duties of the Emergency Coordinator via Commercial phone system or EMnet.

7.2 Florida Emergency Management Network (EMnet)

1. Category – B
2. Compensatory actions-Use commercial telephone system, SHRD system, or cellular phones

7.3 Public Address Exchange System (PAX)

1. Category – B
2. Compensatory actions – Use PE telephone system, handheld radios, cellular telephones

7.4 Commercial Phone system

1. Category – B
2. Compensatory actions – Use cellular telephones, handheld radios, PAX extensions, satellite phones

7.5 PE Telephone system

1. Category – B
2. Compensatory actions- Use commercial telephone system or cellular phones

7.6 FTS-2001 Phone system including: Health Physics Network (HPN), Reactor Safety Counterpart Link (RSCL), Protective Measures Counterpart Link (PMCL), Management Counterpart Link (MCL)

1. Category – B
2. Compensatory actions-Use commercial telephone system

7.7 Portable UHF Radios

1. Category – B
2. Compensatory actions- Use PE Telephone system, cellular telephones, PAX extensions

7.8 Accident Assessment Ringdown (AARD)

1. Category – B
2. Compensatory actions- Use cellular telephones, handheld radios, PAX extensions

7.9 Dose Assessment Ringdown System (DARD)

1. Category – B
2. Compensatory actions- Use PE telephone system, cellular telephones, handheld radios, PAX extensions, or satellite phones

7.10 **Satellite Phones**

1. Category- B
2. Compensatory action – Use redundant communications equipment available.

7.11 **State Law Enforcement Radio System (SLERS), Citrus County 800 MHz radio system**

1. Category – B
2. Compensatory actions- Use cellular telephones and commercial telephones

7.12 **Crystal River Energy Complex Sirens**

1. Category – A (2)
2. Compensatory action- PAX notifications, Security sweeps of affected areas, notifications to Units 1/2 and 4/5 Control Rooms.

7.13 **Off Site siren notification system**

1. Category B
2. Compensatory actions: Route alerting

7.14 **ERONS 1 and ERONS 2 – ERO notification systems**

1. Category - B
2. Compensatory action- Use redundant system if one of the ERONS fail or are not available. If both systems are not available, ERO notifications can be performed manually using commercial telephone system.

7.15 **Emergency Action Level Instrumentation**

1. RM-A1 Reactor Building Purge Exhaust
    a. Category- A (1)
    b. EAL Supported-1.1, 1.2, 1.3, 1.4
    c. Compensatory actions-Sample analysis, or field survey
2. RM-A2 Auxiliary and Fuel Handling Building Exhaust Duct monitor
    a. Category- A (1)
    b. EAL Supported-1.1, 1.2, 1.3, 1.4
    c. Compensatory actions- Sample analysis, or field survey
3. RM-L2 SW/RW Plant Discharge monitor
    a. Category- A (1)
    b. EAL Supported-1.5, 1.6
    c. Compensatory actions- Use grab sample analysis

7.15 **Emergency Action Level Instrumentation (Cont'd)**

4. RM-L7 Station Drain Tank Discharge monitor.
   a. Category- A (1)
   b. EAL Supported-1.5, 1.6
   c. Compensatory actions- Use grab sample analysis
5. RM-G1 Control Room area radiation monitor
   a. Category- A (1)
   b. EAL Supported-1.8
   c. Compensatory actions- Local Area Survey
6. RM-G3 Auxiliary Building Sample Room
   a. Category- A (1)
   b. EAL Supported-1.7, 1.8
   c. Compensatory actions- Local Area Survey
7. RM-G4 RCA Entrance Corridor
   a. Category- A (1)
   b. EAL Supported-1.7, 1.8
   c. Compensatory actions- Local Area Survey
8. RM-G5 Gas Decay Tank Area
   a. Category- A (1)
   b. EAL Supported-1.7, 1.8
   c. Compensatory actions- Local Area Survey
9. RM-G9 Auxiliary Building near the Personnel Access Hatch
   a. Category- A (1)
   b. EAL Supported-1.7, 1.8
   c. Compensatory actions- Local Area Survey
10. RM-G10 Makeup Pump Area
   a. Category- A (1)
   b. EAL Supported-1.7, 1.8
   c. Compensatory actions- Local Area Survey
11. RM-G14 Fuel Storage Pool area monitor
   a. Category- A (1)
   b. EAL Supported-1.7
   c. Compensatory actions- Local Area Survey

7.15 **Emergency Action Level Instrumentation (Cont'd)**

12. RM-G15 Fuel Handling Bridge Auxiliary Building
    a. Category- A (1)
    b. EAL Supported-1.10
    c. Compensatory actions- Local Area Survey
13. RM-G16 Fuel Handling Bridge Reactor Building
    a. Category- A (1)
    b. EAL Supported-1.10
    c. Compensatory actions- Local Area Survey
14. RM-G17 Reactor Building Near Personnel Hatch
    a. Category- A (1)
    b. EAL Supported-1.7, 1.8
    c. Compensatory actions-Local Area Survey
15. RM-G29, RM-G30 Reactor Building Hi Range Radiation Monitors
    a. Category B
    b. EAL Supported- 5.1, 6.1, 7.2
    c. Compensatory Actions- Use redundant instrument if available, otherwise perform local field surveys.
16. Reactor Coolant System Leakage instrumentation.

RCDT Pressure	U3RECL-68	T-Hot (A loop)	U3R226
RCDT Temperature	U3RECL-69	T-Hot (A loop)	U3R227
RCDT Level	U3X368	T-Hot (B loop)	U3R228
MUT-1Pressure	U3X401	T-Hot (B loop)	U3R229
MUT-1 Temperature	U3X208	T-Cold (A loop)	U3R214
MUT-1 Level	U3X359	T-Cold (A loop)	U3R215
PZR Temperature	U3R203	T-Cold (B loop)	U3R216
PZR Level	U3R874	T-Cold (B loop)	U3R217
RC Pressure (A loop)	U3R222	RCP-1A CBO Flow	U3X922
RC Pressure (A loop)	U3R223	RCP-1B CBO Flow	U3X923
RC Pressure (B loop)	U3R224	RCP-1C CBO Flow	U3X924
RC Pressure (B loop)	U3R225	RCP-1D CBO Flow	U3X925

    a. Category B
    b. EAL Supported – 3.12
    c. Compensatory Action: Calculate RCS leakage using alternate instrumentation as specified in the manual method in accordance with SP-317 RC System Water Inventory Balance

7.15 **Emergency Action Level Instrumentation (Cont'd)**

17. Radio-Chem. Lab analysis equipment for DEI samples 1 of 3 detectors required
    a. Category B
    b. EAL Supported 3.9, 5.1
    c. Compensatory Actions - Use redundant analysis equipment to process samples for DEI determination, Use RM-G29 or 30 reading or ICC curves for Fuel Clad integrity determinations
18. Nuclear Instrumentation NI-1, NI-2, NI-3, NI-4, NI-14, NI-15
    a. Category B
    b. EAL Supported: 3.14
    c. Compensatory Action: Use redundant instrumentation (NI-1, NI-2, NI-3, NI-4, N-14, NI-15)
19. RCS Temperature indication – RC-171-TR, RC-172-TR with 8 inputs each (IM-9H-TE, IM-5G-TE, IM-6C-TE, IM-9E-TE, IM-13G, IM-10O, IM-3L, IM-6O, IM-7F, IM-2G, IM-10C, IM-11G, IM-10M, IM-13L, IM-4N, IM-6L) (Indications also available on SPDS)
    a. Category B
    b. EAL supported: 3.13, 3.15, 3.16, 4.4, 7.2
    c. Compensatory action: Use indications on redundant instrument panel.
20. DH-2-TE1, DH-2-TE1
    a. Category B
    b. EAL supported: 3.15
    c. Compensatory action: Use redundant instrumentation
21. Seismic Instrumentation-(SI-1-MAT, SI-1-MR, SI-2-MAT, SI-2-MR, SI-3-MAT, SI-3-MR)
    a. Category B
    b. EAL Supported- 2.1, 2.2
    c. Compensatory Action- Use redundant instrumentation if available, otherwise contact the US Geological Survey at 303-273-8500
22. Reactor Building Pressure-BS-90-PI, BS-91-PI
    a. Category- B
    b. EAL supported-7.1, 7.2
    c. Compensatory Actions- use redundant instrumentation

7.15 **Emergency Action Level Instrumentation (Cont'd)**

23. Reactor Building Sump Level, WD-303-LI, WD-304-LI
    a. Category –B
    b. EAL supported- 7.1, 7.2
    c. Compensatory Actions- Use redundant instrumentation
24. Reactor Building Hydrogen Monitor, WS-10-CE, WS-11-CE
    a. Category- B
    b. EAL supported-7.2
    c. Compensatory Actions- use redundant instrumentation if available or sample Reactor Building atmosphere.
25. Pressurizer Level: RC-1-LT1, RC-1-LT2, and RC-1-LT3
    a. Category B
    b. EAL Supported: 6.2
    c. Compensatory Action: Use redundant instrumentation
26. Pressurizer Relief Safety Valve Acoustic Elements: RC-160-ME1, RC-160-ME2, RC-160-ME3
    a. Category B
    b. EAL Supported: 6.2
    c. Compensatory Action: Use alternate indications such as Pressurizer Code Safety Valve Temperature and RCDT level
27. RC Wide Range T-Cold Temperature: RC-5A-TE2, RC-5A-TE4, RC-5B-TE2, RC-5B-TE4
    a. Category B
    b. EAL Supported: 6.2
    c. Compensatory Action: Use redundant instrumentation
28. RC Wide Range Pressure: RC-3A-PT3, RC-3A-PT4, RC-3B-PT3
    a. Category B
    b. EAL Supported: 6.2, 7.2
    c. Compensatory Action: Use redundant instrumentation
29. Pressurizer Code Safety Valve Temperature: RC-17-TE1, RC-17-TE2, and RC-17-TE3
    a. Category A (1)
    b. EAL Supported: 6.2
    c. Compensatory Action: Use alternate indications such as RCDT level and acoustic elements

7.16 **Plant Integrated Computer System(PICS) and OSI/PI**

1. Category – B
2. Compensatory actions-manual data collection and posting via communication with CR.

7.17 **ERDS (Emergency Response Data System)**

1. Category – B (Based on alternate communication methods, as described below)
2. Compensatory Actions -Verbally transmit data to the NRC using the following methods:
   - Communicate data collected manually via ENS.
   - Communicate data collected manually via Commercial telephone.
3. Review reportable refer to CP-151, External Reporting Requirements.
4. Basis / EP Function Description
   - 10 CFR 50.47(b)(9); 10 CFR 50, Appendix E (IV.E.2); NUREG-0696 (1.3.5; 6) ERDS is the emergency response data system which transmits selected plant data from the ERFIS computer to the OSI/PI computer, then via VPN connection to the Nuclear Regulatory Commission (NRC), once activated.

7.18 **Safety Parameter Display System (SPDS)**

1. Category – A (2)
2. Compensatory Actions: Manually acquire SPDS data and communicate this information as needed to various ERF
3. Review reportable refer to CP-151, External Reporting Requirements.
4. Basis / EP Function Description
   - NUREG-0696 (1.3.4; 2.9; 4.7; 4.8; 5)
   - The NRC, in NUREG 0737, Supplement 1, states that the SPDS should display a minimum set of plant parameters from which the safety status of the plant can be assessed. SPDS also includes trend plots, drawings, and tables which work together to provide the operator with a broad view of the plant status.

7.19 **Dose Assessment Software (RASCAL)**

1. Category – B (Based on redundant components, equipment, and software, as listed below that fulfils this emergency response function.)
2. Compensatory Actions
   a. Relocate to another Progress Energy computer and perform dose assessment via accessing the RASCAL program.
   b. If Rascal program is unavailable perform offsite dose assessment in accordance with EM-204A Off-site Dose Assessment During Radiological Emergencies (Control Room Method)
3. Basis / EP Function Description
   - 10 CFR 50.47(b) (9); 10 CFR 50, Appendix E (IV.E.2); NUREG-0654 (II.I.9); NUREG-0696 (4.8). RASCAL (Radiological Assessment System for Consequence Analysis) is a set of computer-based tools to estimate the following: source term, atmospheric transport, dose from a radiological accident, dose from field measurements of radiological concentrations, and compute decay of radio-nuclides.

7.20 **TSC facility and support equipment**

1. TSC Emergency Diesel Generator and Transfer Switch. (MEDG-1, MEXS-2)
   a. Category – B
   b. Compensatory Actions- If ERO is activated and a loss of normal power occurs consider relocating the TSC to the Remote TSC location. This decision should be based on existing event conditions; coordinated with the Radiological Controls Director; and approved by the EC.
2. TSC Ventilation System
   a. Category – A (2) (Based on the sole means of fulfilling an emergency response function)
   b. Compensatory action- If ERO is activated, consider relocating the TSC to the Remote TSC location. This decision should be based on existing event conditions; coordinated with the Radiological Controls Director; and approved by the EC.
   c. Review reportable in accordance with 10 CFR 50.72. Refer to CP-151, External Reporting Requirements., for additional information.
   d. Basis / EP Function Description-NUREG-0696 (2.6; 4.2); NUREG-0737 (II.B.2); GDC-19; GL-91-014

7.21 **Meteorological Monitoring System (MET Tower) (MPP-3)**

1. Category – (B)
2. Compensatory Actions-Use data from the alternate meteorological tower (MPP-1).

7.22 **EOF facility including emergency power and ventilation equipment.**

1. Category (B)
2. Compensatory Actions
   a. Emergency Diesel generator: Provide portable temporary power source or consider selecting an alternate location if normal power is lost to EOF.
   b. EOF Ventilation- Use portable ventilation equipment and open doors, or consider alternate location if habitability is not sustainable.

7.23 **Environmental Survey Vehicles**

1. Category (B)
2. Compensatory Actions-Use available Progress Energy vehicles

7.24 **Assembly Areas: NAB, NSOC, PAB, Rusty Building, SAB, Shops**

1. Category (B)
2. Compensatory action- Designate an alternate assembly area if a particular assembly area is not available.

7.25 **ERO Access and Personnel Evacuation including Access roads, vehicle barriers, Personnel turnstiles, and gates(vehicle and personnel)**

1. ACP Equipment
   a. Back-up Diesel Generator
      1) Category (B)
      2) Compensatory actions-ensure normal power available
   b. ACP readers
      1) Category B
      2) Compensatory action: Use phone verification
   c. Patriot Barriers (PSPG) and ACP Wedges GAT-PSVG-OC-2A, GAT-PSVG-OC-2C, GAT-PSVG-OC-5A, GAT-PSVG-OC-7A, GAT-PSVG_OC-8A, GAT-PSVG-OC-2B, GAT-PSVG-OC-2D, GAT_PSVG-OC-5B, GAT-PSVG-OC-7B, GAT-PSVG-OC-8B, GAT_PSPG-OC-1, GAT-PSPG-OC-2, GAT-PSPG-OC-3, GAT-PSPG-OC-4, GAT-PSVG-9
      1) Category B
      2) Compensatory Actions: Take actions in accordance with SS0208 Compensatory Measures (SGI).

7.25 **ERO Access and Personnel Evacuation including Access roads, vehicle barriers, Personnel turnstiles, and gates(vehicle and personnel) (Cont'd)**

2. NSOC Turnstiles Egress (Minimum of 1 required) DOR-AG21A, DOR-AG21A
   a. Category B
   b. Compensatory Actions: Take actions in accordance with EM-211 Duties of the CR3 Nuclear Security Organization
3. NSOC Egress Portal Monitors (Minimum of 1 required)
   a. Category B
   b. Compensatory Actions: Set up manual frisking stations at the direction of Health Physics.
4. NSOC Turnstiles Ingress (Minimum of 2 required) DOR-AG25A, DOR-AG25B, DOR-AG24A, DOR-AG24B,
   a. Category B
   b. Compensatory Actions: Take actions in accordance with SS0208 Compensatory Measures (SGI).
5. Search Train Equipment: ED-PSED-1, ED-PSED-2, ED-PSED-3, ED-PSED-4, MD-PSMD-1, MD-PSMD-2, MD-PSMD-3, MD-PSMD-4, XRA-PSPX-1, XRA-PSPX-2
   a. Category B
   b. Compensatory actions: Take actions in accordance with SS0208 Compensatory Measures (SGI).
6. ALL Security Readers
   a. Category B
   b. Compensatory actions: Take actions in accordance with SS0208 Compensatory Measures (SGI).
7. Primary Gates and Barriers GAT-PSVG-1, GAT-PSVG-5
   a. Category B
   b. Compensatory Action: Take actions in accordance with SS0208 Compensatory Measures (SGI).

7.25 ERO Access and Personnel Evacuation including Access roads, vehicle barriers, Personnel turnstiles, and gates (vehicle and personnel) (Cont'd)

8. Plant Security Computer (PSCS) CPU-PSCS-3C, CPU-PSCS-1S, CS-PSCS-1
    a. Category: A2
    b. Compensatory Action: Take actions in accordance with SS0208 Compensatory Measures (SGI).
9. Security Printers: PRT-PSLP-1, PRT-PSLP-2 (1 of 2 required)
    a. Category: B
    b. Compensatory Action: Use redundant equipment
10. Computer work station SAS/CAS 1 of 2 required
    a. Category: B
    b. Compensatory Action: Use redundant equipment

8.0 ACCEPTANCE CRITERIA

None

9.0 INSTRUCTIONS

9.1 General Information

> Note
>
> Refer to EMG-NGGC-0007, Equipment Important to Emergency Preparedness and Response for all instructions associated with the control of EIER.

9.1.1 Planned Loss of Equipment Important to Emergency Response (EIER)

1. Use attachment 1 as guide for specific EIER listed in Section 7.0

> Note
>
> Planned entry into compensatory action for an EIER removed from service should have a communication strategy based on the impact on ERO response. Changes in evacuation plans or changes in primary ERF locations should be communicated to appropriate personnel prior to implementation of the compensatory action.

2. Contact EP duty manager if compensatory action from Section 7.0 requires alternate ERO response which needs to be communicated to appropriate levels of ERO.
3. Implement compensatory actions from section 7.0 prior to removal of EIER from service.

9.1.2 Unplanned Loss of Equipment Important to Emergency Response

1. Use attachment 2 as a guide for specific EIER listed in section 7.0

> Note
>
> If the EIER is not under the control of the Site work control process, initiate a Condition Report to track and document the timely restoration plan.

2. Initiate appropriate tracking documents (Work Request or Condition Report) to track and drive restoration of EIER.
3. Contact EP duty manager if compensatory action from Section 7.0 requires alternate ERO response which needs to be communicated to appropriate levels of ERO.
4. Implement compensatory actions from section 7.0 for the specific EIER listed.

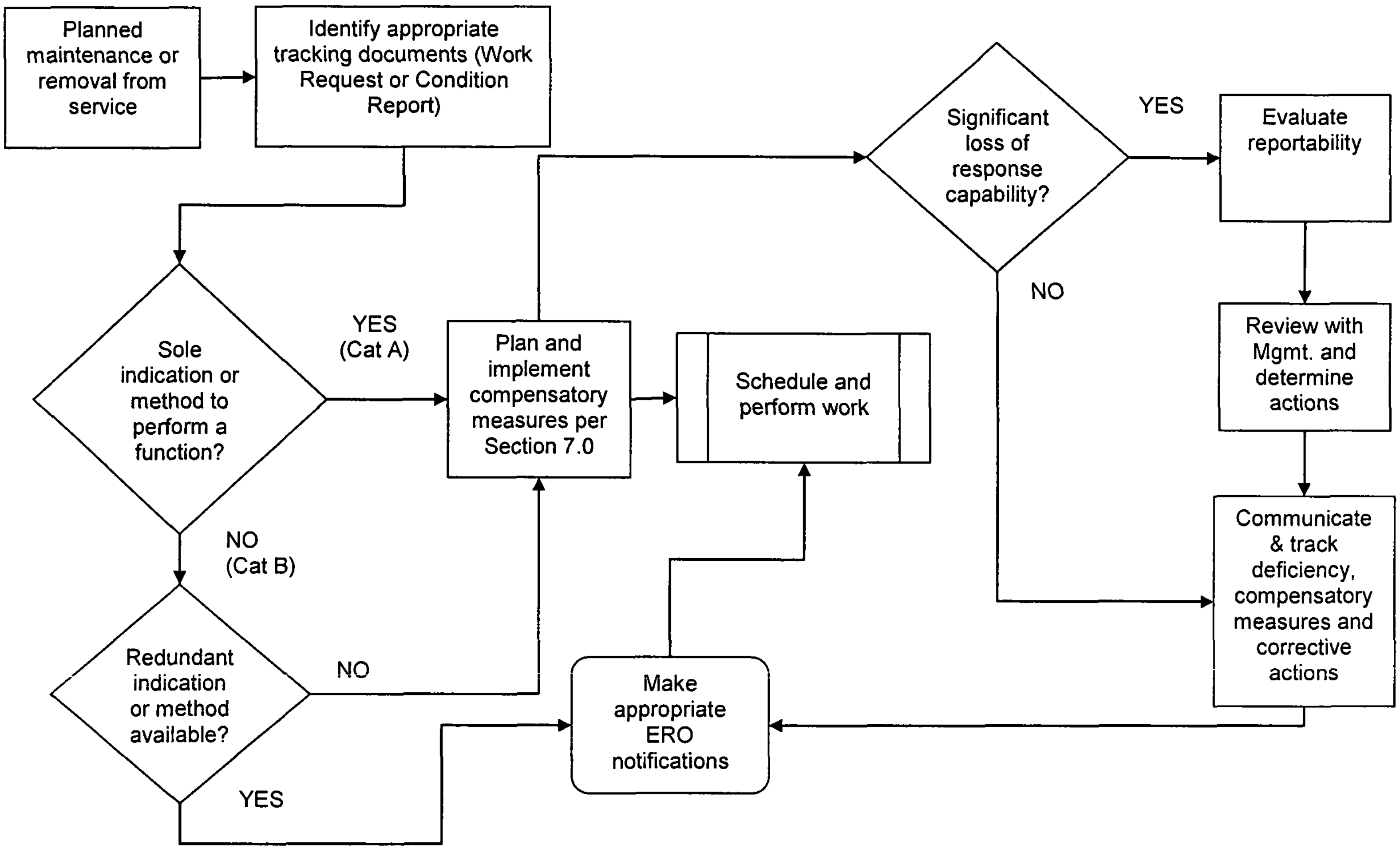
9.1.3 Reportability Determination Guidance

Refer to CP-151 External Reporting Requirements for reportability determination when an unplanned loss of an EIER occurs. Reportability for loss of emergency preparedness capabilities is defined in NUREG-1022, Revision 2, Event Reporting Guidelines, as "Any event that results in a major loss of emergency assessment capability, offsite response capability, or offsite communications capability (e.g., significant portion of Control Room indication, Emergency Notification System, or offsite notification system)."

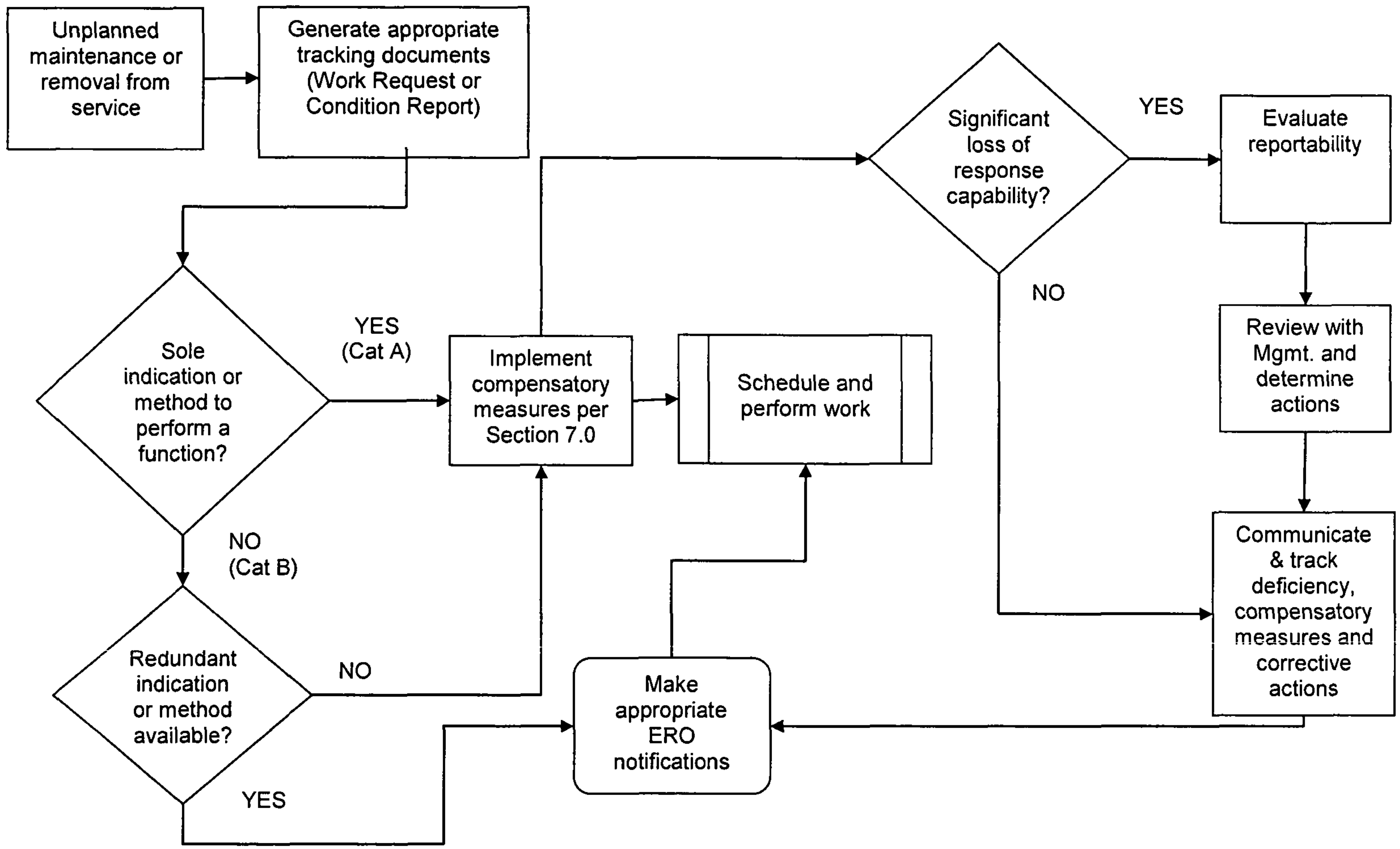
10.0 RECORDS

None

Planned Loss of Equipment Important to EP



Unplanned Loss of Equipment Important to EP



Summary of Changes

PRR 556688

SECTION/STEP	CHANGE
ALL	Revision 00 of EM-500 is a new procedure that incorporates information from EMG-NGGC-007, RERP Radiological Emergency Response Plan and includes information based on guidance in INPO 10-007 (Revision 0), Equipment Important to Emergency Response.